Palisades Nuclear Plant

Application for Renewed Operating License

March 22, 2005

TECHNICAL AND ADMINISTRATIVE INFORMATION

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1.0 Administrative Information

This License Renewal application (LRA) has been prepared to provide the administrative, technical and environmental information required by 10 CFR 54 (Reference 1) and 10 CFR 51 (Reference 2) to support the renewal of the Operating License for the Palisades Nuclear Plant.

This LRA is being submitted on CD-ROM in accordance with 10 CFR 54.7 and 10 CFR 50.4. The FSAR, LR scoping boundary drawings, and other references cited within the application are for information only, and are not incorporated by reference into this application. The LR drawings cited within this application will not necessarily be kept up to date for the life of the plant.

This section of the application provides the following information:

1. Information on the organization of the application (Section 1.1).
2. A general plant description (Section 1.2).
3. The administrative information required by 10 CFR 54.17 and 10 CFR 54.19 (Section 1.3).
4. Summary of abbreviations and passive function code definitions (Section 1.5).
5. A distribution list for written communications related to the application (Section 1.6).

1.1 Application Format and Content

The following discussion describes the content of the Palisades Nuclear Plant License Renewal Application. The Application format and content, including Section 3 tables, have been constructed as described in NEI 95-10, Rev. 4.

Section 1.0 provides the administrative information required by Part 54 of Title 10 of the Code of Federal Regulations, Sections 17 and 19 (10 CFR 54.17 and 10 CFR 54.19).

Section 2.0 provides the scoping and screening methodology. Section 2 also describes and justifies the methodology used to determine the systems, structures, and components within the scope of license renewal and the structures and components subject to an aging management review (AMR). The system groupings in Sections 2 and 3 are organized to be consistent with NUREG-1800 (Reference 5). Table 2.2-1, Plant Level Scoping Results, provides listings of the plant mechanical systems, structures, and electrical/instrumentation and controls (I&C) systems, and identifies those plant systems and structures that are and are not within the scope of license renewal. Section 2.3, Section 2.4 and Section 2.5 provide a description of systems, their intended functions, and for information only, cross references to FSAR sections and license renewal drawings. Each system subsection has a table listing component groups subject to an AMR and their passive intended functions. The drawings and FSAR are provided as separate files for use as review tools.

Section 3.0 describes the results of the AMRs for the components and structures requiring AMR. Section 3 identifies the components and structures subject to aging management review, summarizes the technical results of those reviews, and provides a comparison of the Palisades results with the corresponding results identified in the NRC's "Generic Aging Lessons Learned (GALL) Report," NUREG-1801 (Reference 6). The information is organized into six "super-groups" to be consistent with NUREG-1801 and NUREG-1800.

Within each "super-group", the plant-specific AMR results are provided in a set of system, structure or commodity tables. These tables provide aging management information including component type, intended function, material, environment, aging effect requiring management, and the selected aging management programs for each component type. These tables also include cross-references to NUREG -1801, Volume 2 and Volume 1, line items to facilitate comparison. Finally the last column of each table in Section 3 is set aside for notes or additional explanatory information specific to that line item. In addition these tables have hyperlinked cross references to the aging management program descriptions in Appendix B. A more detailed description of table construction and interrelationships is provided in Section 3.

Section 4.0 includes a list of time-limited aging analyses (TLAAs), as defined by 10 CFR 54.3. It includes the identification of the component or subject, and an explanation of the time dependent aspects of the calculation or analysis. For each TLAA, Section 4 demonstrates that the analyses remain valid for the period of extended operation, the analyses have been projected to the end of the period of extended operation, or the effects of aging on the intended function(s) will be adequately managed for the period of extended operation. Section 4 also states that there are no 10 CFR 50.12 exemptions granted to Palisades that are based on a time-limited aging analysis as defined in 10 CFR 54.3.

Appendix A, Final Safety Analysis Report (FSAR) Supplement, contains a summary description of the programs relied on for managing the effects of aging during the period of extended operation. A summary description of the evaluation of time-limited aging analyses for the period of extended operation is also included.

Appendix B, Aging Management Programs, describes Palisades' aging management programs and demonstrates that the effects of aging on the components and structures within the scope of the license renewal will be managed such that the components and structures will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation. Where a Palisades program is similar to a corresponding program in NUREG-1801, the applicable NUREG-1801 program is referenced.

Appendix C is not used for this application.

Appendix D, Technical Specification Changes, is not used for this application. No technical specification changes were identified as being necessary to manage the effects of aging during the period of extended operation.

Appendix E, Environmental Report, contains an environmental report analyzing the potential environmental impacts of license renewal, as provided for in NRC regulations 10 CFR 51.53(c) and 10 CFR 54.23. The NRC has determined that nuclear power plant license renewal decisions are major federal actions requiring preparation of an environmental impact statement [10 CFR 51.20(a)(2) and 51.95(c)]. In an effort to streamline the license renewal environmental review, the NRC conducted a generic analysis and published the results in NUREG-1437, Generic Environmental Impact Statement for the License Renewal of Nuclear Power Plants (GEIS). To fulfill NEPA requirements, the NRC is required to publish site-specific analyses in the form of a supplemental environmental impact statement to the GEIS. This Palisades Environmental Report provides input to the NRC staff for use in meeting NEPA requirements as they apply to license renewal.

In summary, the information in Section 1 fulfills the requirements of 10 CFR 54.17 and 54.19. The information in Section 2, Section 3, and Appendix B fulfills the requirements of 10 CFR 54.21(a). Section 1.4 discusses how the requirements of 10 CFR 54.21(b) will be met. The information in Section 4 fulfills the requirements in 10 CFR 54.21(c). The information in

Appendix A and Appendix D fulfill the requirements in 10 CFR 54.21(d) and 10 CFR 54.22, respectively. Appendix E provides the Environmental Report, as required by 10 CFR 54.23.

1.2 Plant Description

The site for the Palisades Nuclear Plant consists of approximately 432 acres on the eastern shore of Lake Michigan, in Covert Township, approximately four and one-half miles south of South Haven, Michigan. The Nuclear Steam Supply System consists of a pressurized water reactor with two closed loops. The NSSS was originally expected to have adequate margin to obtain an ultimate output of 2,650 MWt. The steam and power conversion equipment was designed for a maximum expected gross capability of 865 MWe. Palisades is currently licensed for a thermal power of 2565.4 MWt. At present, Consumers Energy does not plan to pursue an increase in the licensed power level.

Descriptions of Palisades' systems and structures can be found in the Final Safety Analysis Report (FSAR). Additional descriptive information about the systems, structures, and components is provided in Sections 2, 3, and 4 of this application, and references to the FSAR are provided where pertinent.

The current license (Facility Operating License No. DPR-20) expires on March 24, 2011. Nuclear Management Company will be named as the exclusive operator licensee on the renewed operating license. Consumers Energy Company will be named as the exclusive owner licensee on the renewed operating license.

1.3 Information Required by 10 CFR 54.17 and 10 CFR 54.19

1.3.1 Name of Applicant

Nuclear Management Company (NMC), the operating licensee, hereby applies for a renewed operating license for the Palisades Nuclear Plant. NMC submits this application individually and as agent for the owner licensee named on the operating license. The owner licensee is Consumers Energy Company.

1.3.2 Address of Applicant

Addresses of Applicant Companies:

Nuclear Management Company, LLC
700 First Street
Hudson, Wisconsin 54016

Consumers Energy Company
One Energy Plaza
Jackson, MI 49201

Address of the Palisades Nuclear Plant:

Palisades Nuclear Plant
27780 Blue Star Hwy
Covert, MI 49043

1.3.3 Description of Business or Occupation of Applicant

Nuclear Management Company, LLC (NMC)

NMC is engaged in the operation of nuclear power plants. NMC operates Palisades Nuclear Plant for Consumers Energy Company, a subsidiary of CMS Energy Corporation; Duane Arnold Energy Center for Interstate Power and Light Company, a subsidiary of Alliant Energy Corporation, Central Iowa Power Cooperative and Corn Belt Power Cooperative; Prairie Island Nuclear Generating Plant and Monticello Nuclear Generating Plant for Northern States Power Company, a subsidiary of Xcel Energy Inc.; Point Beach Nuclear Plant Units 1 and 2 for Wisconsin Electric Power Co, doing business as We Energies; and Kewaunee Nuclear Power Plant for Wisconsin Public Service Corporation and Wisconsin Power and Light Company, a subsidiary of Alliant Energy Corporation. The combined electric generation of the six plants is in excess of 4,500 MW.NMC is the exclusive licensed operator of the Palisades Nuclear Plant, which is the subject of this application. NMC' s corporate purpose is to provide services in connection with the operation and eventual decommissioning of licensed nuclear facilities on behalf of and for the benefit of the owner utilities.

NMC is organized as a Wisconsin limited liability company and is owned by Consumers Nuclear Services, LLC, WEC Nuclear Corporation, WPS Nuclear Corporation, NSP Nuclear Corporation, and Alliant Energy Nuclear, LLC. Consumers Nuclear Services, LLC, is a wholly owned subsidiary of Consumers Energy Company. WEC Nuclear Corporation is a wholly owned subsidiary of Wisconsin Energy Corporation, the parent holding company of Wisconsin Electric Power Co. WPS Nuclear Corporation is a wholly owned subsidiary of WPS Resources, Inc., the parent holding company of Wisconsin Public Service Corporation. NSP Nuclear Corporation is a wholly owned subsidiary of Northern States Power Company. Alliant Energy Nuclear, LLC is a wholly owned subsidiary of Alliant Energy Corporation, the parent holding company of IES Utilities.

Consumers Energy Company

Consumers Energy Company is a combination electric and gas utility company serving Michigan’s Lower Peninsula, and is a wholly owned subsidiary of CMS Energy Corporation. Consumers is the owner licensee of Palisades.

Consumers Energy's electric utility operations include the generation, purchase, distribution and sale of electricity. Consumers Energy serves customers in 61 of the 68 counties of Michigan's Lower Peninsula, including the principal cities of Battle Creek, Flint, Grand Rapids, Jackson, Kalamazoo, Midland, Muskegon, and Saginaw.

Consumers Energy provides natural gas and/or electricity to almost 6 million of Michigan's 10 million residents. To serve electricity to these residents, Consumers Energy operates 44 electric generating facilities in Michigan (including Palisades) with a total owned generation of 6,431 MWs (2003 Summer Net Demonstrated Capability).

Consumers Energy is an electric utility as defined by the NRC regulations at 10 CFR 50.2. Pursuant to the Nuclear Power Plant Operating Services Agreement between Consumers Energy and NMC, all costs, including costs for the operation, maintenance, repair, decontamination and decommissioning of Palisades, and the Independent Spent Fuel Storage Installation at the plant, are liabilities of Consumers Energy when incurred and accrued, and Consumers Energy has committed to provide NMC with funds to pay these costs.

Consumers Energy will be named as the exclusive licensed owner of Palisades on the renewed operating licenses.

1.3.4 Organization and Management of Applicant

Consumers Energy Company is a public utility; it is a wholly owned subsidiary of CMS Energy Corporation. CMS is incorporated under the laws of the State of Michigan, with its principal office located in Jackson, Michigan. NMC is a limited liability company incorporated under the laws of the State of Wisconsin, with its principal office located in Hudson, Wisconsin. Consumers Energy and NMC make this application on their own behalf, and are not acting as agents or representatives of any other person.

CMS Energy and NMC are not owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government. The names and business addresses of CMS Energy and NMC directors and principal officers are listed below. All persons listed are U.S. citizens.

CMS Energy Corporation

CMS Energy Corporation Directors

Name	Address
Kenneth Whipple	One Energy Plaza Jackson, MI 49201
S. Kinnie Smith, Jr.	One Energy Plaza Jackson, MI 49201
Earl D. Holton	One Energy Plaza Jackson, MI 49201
Merribel S. Ayres	One Energy Plaza Jackson, MI 49201
David W. Joos	One Energy Plaza Jackson, MI 49201
Michael T. Monahan	One Energy Plaza Jackson, MI 49201
Joseph F. Paquette, Jr.	One Energy Plaza Jackson, MI 49201
William U. Parfet	One Energy Plaza Jackson, MI 49201
Percy A. Pierre	One Energy Plaza Jackson, MI 49201
Kenneth L. Way	One Energy Plaza Jackson, MI 49201
John B. Yasinsky	One Energy Plaza Jackson, MI 49201

Consumers Energy Company Principal Officers

Name	Address
Kenneth Whipple Chairman of the Board	One Energy Plaza Jackson, MI 49201
S. Kinnie Smith, Jr. Vice Chairman of the Board	One Energy Plaza Jackson, MI 49201
David W. Joos Chief Executive Officer	One Energy Plaza Jackson, MI 49201
John G. Russell President and Chief Operating Officer	One Energy Plaza Jackson, MI 49201
Thomas J. Webb Executive Vice President and Chief Financial Officer	One Energy Plaza Jackson, MI 49201
John F. Drake Senior Vice President, Human Resources and Administrative Services	One Energy Plaza Jackson, MI 49201
Robert A. Fenech Senior Vice President, Nuclear, Fossil, and Hydro Operations	One Energy Plaza Jackson, MI 49201
Frank Johnson Senior Vice President, Electric Transmission and Distribution	One Energy Plaza Jackson, MI 49201
David G. Mengebier Senior Vice President, Governmental and Public Affairs/Community Services	One Energy Plaza Jackson, MI 49201
Paul N. Preketes Senior Vice President, Gas Operations	One Energy Plaza Jackson, MI 49201
Glenn P. Barba Vice President, Controller, and Chief Accounting Officer	One Energy Plaza Jackson, MI 49201
James E. Brunner Vice President and General Counsel	One Energy Plaza Jackson, MI 49201

James R. Coddington Vice President, Fossil Operations	One Energy Plaza Jackson, MI 49201
William E. Garrity Vice President, Electric and Gas Supply	One Energy Plaza Jackson, MI 49201
Laura L. Mountcastle Vice President and Treasurer	One Energy Plaza Jackson, MI 49201
Jon R. Robinson Vice President, Utility Law and Regulation	One Energy Plaza Jackson, MI 49201
Michael J. Shore Vice President and Chief Risk Officer	One Energy Plaza Jackson, MI 49201
Susan C. Swan Vice President, Customer Operations	One Energy Plaza Jackson, MI 49201
Michael D. VanHemert Vice President, Corporate Secretary and Chief Compliance Officer	One Energy Plaza Jackson, MI 49201
Theodore J. Vogel Vice President and Chief Tax Counsel	One Energy Plaza Jackson, MI 49201

Nuclear Management Company, LLC

Nuclear Management Company Directors

Name	Address
David W. Joos	Consumers Energy Corp. 212 West Michigan Ave. Jackson, MI 49201
Frederick Kuester	Wisconsin Electric Power Co. 231 West Michigan Street Milwaukee, WI 53201
Michael B. Sellman	Nuclear Management Company, LLC 700 First Street Hudson, WI 54016
David M. Wilks	Xcel Energy 4653 Table Mountain Dr. Golden, CO 80403
Eliot G. Protsch	Alliant Energy Corp. 200 First Street SE Cedar Rapids, IA 52406
Charles Schrock	Wisconsin Public Service Corp. 700 North Adams St. Green Bay, WI 54307

Nuclear Management Company Principal Officers

Name	Address
Michael B. Sellman President and Chief Executive Officer	700 First Street Hudson, Wisconsin 54016
John Paul Cowan Executive Vice President and Chief Nuclear Officer	700 First Street Hudson, Wisconsin 54016
Greg Palmer Vice President and Chief Financial Officer	700 First Street Hudson, Wisconsin 54016

Lyle H. Bohn Senior Vice President - Nuclear Support Programs	700 First Street Hudson, Wisconsin 54016
Jonathan M. Rogoff Vice President, General Counsel, and Secretary	700 First Street Hudson, Wisconsin 54016
Douglas E. Cooper Senior Vice President - Group Operations	700 First Street Hudson, Wisconsin 54016
Craig G. Anderson Senior Vice President - Group Operations	700 First Street Hudson, Wisconsin 54016

1.3.5 Class of License, Use of Facility, and Period of Time for which the License is Sought

NMC requests renewal of the Class 104b operating license for Palisades Nuclear Plant (license number DPR-20) for a period of 20 years beyond the expiration of the current license. License renewal would extend the operating license from midnight March 24, 2011, until midnight March 24, 2031. This application includes a request for renewal of those NRC source material, special nuclear material, and byproduct material licenses that are included within the current operating license and that were issued pursuant to 10 CFR Parts 30, 40 and 70.

The facility will continue to be known as the Palisades Nuclear Plant.

1.3.6 Earliest and Latest Dates for Alterations, if Proposed

NMC does not propose to construct or alter any production or utilization facility in connection with this renewal application. The current licensing basis will be continued and maintained throughout the period of extended operation.

1.3.7 Listing of Regulatory Agencies Having Jurisdiction and News Publications

The Federal Energy Regulatory Commission (FERC) and the Michigan Public Service Commission are the principal regulators of Consumers' electric operations.

Secretary
Federal Energy Regulatory Commission
888 First Street, NE, Room 1A
Washington, DC 20426

Chairperson
Michigan Public Service Commission
PO Box 30221
Lansing, Michigan 48909-7721

The area news publications and their associated addresses are provided below

The Herald-Palladium
PO Box 128
St. Joseph, Michigan 49085

The Kalamazoo Gazette
401 S. Burdick
Kalamazoo, Michigan 49007

South Haven Tribune
255 Center St.
South Haven, Michigan 49090

1.3.8 Conforming Changes to Standard Indemnity Agreement

The requirements at 10 CFR 54.19(b) state that license renewal applications must include "conforming changes to the standard indemnity agreement, 10 CFR 140.92, Appendix B, to account for the expiration term of the proposed renewed license." The current indemnity agreement No. B-40 for Palisades states, in Article VII, that the agreement shall terminate at the time of expiration of that license specified in Item 3 of the attachment to the agreement, which is the last to expire. Item 3 of the attachment to the indemnity agreement, as revised by Amendment No. 1, lists DPR 20 as the applicable operating license number. Should the operating license number be changed upon issuance of the renewed license, NMC requests that conforming changes be made to Item 3 of the attachment, and any other sections of the indemnity agreement as appropriate.

1.3.9 Restricted Data Agreement

This application does not contain restricted data or other national defense information, nor is it expected that subsequent amendments to the license application will contain such information. However, pursuant to 10 CFR 54.17(g) and 10 CFR 50.37, NMC, as a part of the application for a renewed operating license, hereby agrees that it will not permit any individual to have access to or any facility to possess Restricted Data or classified National Security Information until the individual and/or facility has been approved for such access under the provisions of 10 CFR Parts 25 and/or 95.

1.4 Current Licensing Basis Changes During NRC Review

Each year, following the submittal of the Palisades License Renewal Application and at least three months before the scheduled completion of the NRC review, NMC will submit an amendment to the application pursuant to 10 CFR 54.21(b). This amendment will identify any changes to the Current Licensing Basis of the facility that materially affect the contents of the License Renewal Application, including the FSAR supplement, that have not already been submitted.

1.5 Abbreviations

This section contains the abbreviations that pertain to the administrative and technical information within the license renewal application. The abbreviations that pertain to the environmental information are included as part of Appendix E (Environmental Report).

AAC	Alternate Alternating Current
AC	Alternating Current
ACI	American Concrete Institute
ACSR	Aluminum Conductor Steel Reinforced
AEC	Atomic Energy Commission
AFW	Auxiliary Feedwater
AISC	American Institute of Steel Construction
AMMS	Advanced Maintenance Management System
AMP	Aging Management Program
AMR	Aging Management Review
AMSAC	ATWS Migrating System Actuation Circuit
ANSI	American National Standards Institute
APP R	10 CFR 50 Appendix R
AR	Action Request
ASME	American Society of Mechanical Engineers
ASME III	American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III
ASSY	Assembly
ASTM	American Society for Testing of Materials
ATWS	Anticipated Transients Without a Scram
AUX	Auxiliary
B&W	Babcock and Wilcox
BAW	Babcock and Wilcox
BOP	Balance of Plant

BTP	Branch Technical Position
BWR	Boiling Water Reactor
BWOG	B & W Owners Group
CASS	Cast Austenitic Stainless Steel
CB	Control Building
CBD	Class Boundary Diagram
CCS	Component Cooling System
CCCW	Closed Cycle Cooling Water
CCW	Component Cooling Water
CD-ROM	Compact Disk-Read Only Memory
CDS	Condensate System
CE	Condition Evaluation
CFR	Code of Federal Regulations
CHECWORKS	The suite of products developed to evaluate power plants for the most common forms of corrosion that degrade their performance and shorten the operating life of critical components.
CIV	Containment Isolation Valve
CIS	Containment Isolation System
CLB	Current Licensing Basis
CMAA	Crane Manufactures Association of America
CMS	CMS Energy Corporation
Co	Company
CPCo	Consumers Power Company
CR	Condition Report
CRD	Control Rod Drive
CRDM	Control Rod Drive Mechanism
CS	Carbon Steel

CSR	Cable Spreading Room
CSS	Containment Spray System
CST	Condensate Storage Tank
CSW	Critical Service Water
CUF	Cumulative Usage Factors
CV	Control Valve
CVCS	Chemical and Volume Control System
CWS	Circulating Water System
DBA	Design Basis Accident
DBD	Design Basis Document
DBE	Design Basis Event
DC	Direct Current
DG	Diesel Generator or Design Guide
DI	Deionized
DMW	Demineralized Water
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
EAB	Emergency Breathing Air
ECCS	Emergency Core Cooling System
ECT	Eddy Current Testing
EDG	Emergency Diesel Generator
EEQ	Electrical Equipment Qualification
EFPY	Effective Full Power Years
EIN	Equipment Identification Number
EL	Elevation
EOCI	Electric Overhead Crane Institute
EOLE	End Of Life - Extended

EPA	Electrical Penetration Assemblies
EPDM	Ethylene Propylene Diene Monomer
EPR	Ethylene Propylene Rubber
EPRI	Electric Power Research Institute
EQ	Environmental Qualification
ESF	Engineered Safety Feature
ESFAS	Engineered Safety Features Actuation System
F	Fahrenheit
FAC	Flow-Accelerated Corrosion
FatiguePro	Automated Cycle Counting and Fatigue Monitoring Program
FE	Flow Element
FERC	Federal Energy Regulatory Commission
FHAR	Fire Hazards Analysis Report
FP	Fire Protection
FPER	Fire Protection Evaluation Report
FM	Frequency Modulation
FSAR	Final Safety Analysis Report
GALL	Generic Aging Lessons Learned
GDC	General Design Criterion
GEIS	Generic Environmental Impact Statement
GL	Generic Letter
GSI	Generic Safety Issues
GTR	Generic Technical Report
HAZ	Heat-Affected Zone
HELB	High Energy Line Break

HEPA	High Efficiency Particulate Filter
HMWPE	High Molecular Weight Polyethylene
Hr	Hour
HVAC	Heating, Ventilating and Air Conditioning
HX	Heat Exchanger
I&C	Instrumentation & Controls
I/P	Current to Pressure Converter
IA	Instrument Air
IASCC	Irradiation Assisted Stress Corrosion Cracking
ID	Identification or Inside Diameter
IE	Inspection and Enforcement
IEB	Inspection and Enforcement Bulletin
IEEE	Institute of Electrical and Electronics Engineers, Inc.
IF	Intended Function
IGA	Intergranular Attack
IGSCC	Intergranular Stress Corrosion Cracking
ILRT	Integrated Leak Rate Test
IN	Information Notice or Inch
INPO	Institute of Nuclear Power Operations
IPA	Integrated Plant Assessment
IR	Insulation Resistance or Inspection Report
ISG	Interim Staff Guidance
ISI	Inservice Inspection
ITG	Issues Task Group
IWB	ASME Boiler and Pressure Vessel Code, Section XI, Requirements for Class 1 Components of Light-Water Cooled Power Plants

IWC	ASME Boiler and Pressure Vessel Code, Section XI, Requirements for Class 2 Components of Light-Water Cooled Power Plants
IWD	ASME Boiler and Pressure Vessel Code, Section XI, Requirements for Class 3 Components of Light-Water Cooled Power Plants
IWE	ASME Boiler and Pressure Vessel Code, Section XI, Requirements for Class MC and Metallic Liners of Class CC Components if Light-Water Cooler Power Plants
IWF	ASME Boiler and Pressure Vessel Code, Section XI, Requirements for Class 1, 2, 3, and MC Component Supports of Light-Water Cooled Power Plants
IWL	ASME Boiler and Pressure Vessel Code, Section XI, Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants
K_{IC}	Reference Stress Intensity Factor As A Function Of the Metal Temperature (T) and the Metal References Nil-Ductility Temperature (RT_{NDT})
KIP	1000 lb; or 1 Kilo-pound
K_{IR}	ASME Fracture Toughness Curve
Ksi	One KIP per Square Inch, 1000 psi
KV	Kilovolts
KWe	Kilowatt Electric
Lb	Pound
LBB	Leak-Before-Break
LEFM	Leading Edge Flow Meter
LER	Licensee Event Report
LLC	Limited Liability Company
LO	Lubricating Oil

LOCA	Loss-Of-Coolant-Accident
LR	License Renewal
LRA	License Renewal Application
LTOP	Low-Temperature Overpressure Protection
LWR	Light Water Reactor
Mat'l	Material
MIC	Microbiologically Induced Corrosion
MIRVP	Master Integrated Reactor Vessel Surveillance Program
Misc.	Miscellaneous
MRP	EPRI Materials Reliability Program
MS	Main Steam
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break
MT	Magnetic Particle Test
MW	Megawatts
MWD	Megawatt-Day
MWe	Megawatt Electric
MWt	Megawatt Thermal
NA OR N/A	Not Applicable
NaOH	Sodium Hydroxide
NCR	Non-Conformance Report
NDE	Non-Destructive Examination
NEI	Nuclear Energy Institute
NEPA	National Environmental Policy Act
NES	Nuclear Engineering Services
NFPA	National Fire Protection Association

NMC	Nuclear Management Company
NPS	National Pipe Size
NQAP	Nuclear Quality Assurance Program
NRC	Nuclear Regulatory Commission
NSAC	Nuclear Safety Analysis Center
NSP	Northern States Power
NSR	Non-Safety Related
NSSS	Nuclear Steam Supply System
NUMARC	Nuclear Utility Management and Resource Council
NUREG	Nuclear Regulation Document
OCCW	Open Cycle Cooling Water
ODSCC	Outside Diameter Stress Corrosion Cracking
OEM	Original Equipment Manufacturer
P&ID	Piping and Instrument Diagram
P-T	Pressure - Temperature
PAS	Post Accident Sampling
PBD	Program Basis Document
PCP	Primary Coolant Pump
PCS	Primary Coolant System
PDI	Performance Demonstration Initiative
PM	Preventative Maintenance
PNP	Palisades Nuclear Plant
PORV	Power-Operated Relief Valve
PPAC	Palisades Periodic Activity Control
PPB	Parts Per Billion
PPC	Plant Process Computer
PS	Pipe Support

PSI	Pounds Per Square Inch
PSIA	Pounds per Square Inch Absolute
PSIG	Pounds Per Square Inch Gauge
PT	Penetrant Testing
PTS	Pressurized Thermal Shock
PVC	Poly Vinyl Chloride
PWR	Pressurized Water Reactor
PWSCC	Primary Water Stress Corrosion Cracking
PZR	Pressurizer
Q-List	Quality List
QA	Quality Assurance
RAI	Request for Additional Information
RCS	Reactor Coolant System
RCCA	Rod Control Cluster Assembly
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
REV	Revision
RG	Regulatory Guide
RHR	Residual Heat Removal
RI-ISI	Risk Informed Inservice Inspection Program
RIA	Radiation Monitoring
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RTD	Resistance Temperature Detectors
RT_{NDT}	Reference Temperature for Nil Ductility Transition

RT_{PTS}	Reference Temperature for Pressurized Thermal Shock
RV	Reactor Vessel or Relief Valve
RVI	Reactor Vessel Internals
RWST	Refueling Water Storage Tank
S&PC	Steam and Power Conversion
SA	Service Air
SBO	Station Blackout
SC	Structure and Component
SCBA	Self-Contained Breathing Apparatus
SCC	Stress Corrosion Cracking
SER	Safety Evaluation Reports
SFP	Spent Fuel Pool
SG	Steam Generator
SGBD	Steam Generator Blowdown
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SIRW	Safety Injection and Refueling Water
SIS	Safety Injection System or Safety Injection Signal
SOC	Statement Of Considerations
SOER	INPO Significant Operating Event Report
SV	Solenoid Valve
SR	Safety Related
SRP	Standard Review Plan
SRP-LR	Standard Review Plan for License Renewal
SS	Stainless Steel or Sampling System
SSAR	Safe Shutdown Analysis Report
SSC	System, Structure, or Component

SSE	Safe Shutdown Equipment
SWS	Service Water System
T_{AVG}	PCS Average Temperature
TB	Turbine Building
TID	Total Integrated Dose
TLAA	Time-Limited Aging Analysis
TR	Technical Report
TRM	Training Requirements Manual
TS	Technical Specifications
TSC	Technical Support Center
U.S.	United States
UHS	Ultimate Heat Sink
USE	Upper Shelf Energy
USI	Unresolved Safety Issue
USNRC	United States Nuclear Regulatory Commission
UT	Ultrasonic Testing
UTS	Ultimate Tensile Strength
UV	Ultraviolet
VAC	Volts-Alternating Current
VCT	Volume Control Tank
VDC	Volts-Direct Current
Vol.	Volume
Vs.	Versus
VT	Visual Examination
WCAP	Westinghouse Commercial Atomic Power

WO	Work Order
WOG	Westinghouse Owners Group
XLPE	Cross-Linked Polyethylene
YARD	Yard Structures
Zn	Zinc

1.6 Communications

Written communications on this application should be directed to:

Mr. Daniel J. Malone
Site Vice President
Palisades Nuclear Plant
27780 Blue Star Highway
Covert, MI 49043

With copies to:

Mr. Darrel G. Turner
License Renewal Project Manager
Palisades Nuclear Plant
27780 Blue Star Highway
Covert, MI 49043

And

Mr. Douglas F. Johnson
Director, Plant Life Cycle Issues
Nuclear Management Company, LLC
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Section 1.0 References

1. 10 CFR 54, Requirements for Renewal of Operating Licenses for Nuclear Power Plants, U.S. Nuclear Regulatory Commission.

2. 10 CFR 51, Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions, U.S. Nuclear Regulatory Commission.

3. Not used

4. NEI 95-10, Industry Guideline for Implementing the Requirements of 10 CFR Part 54 -The License Renewal Rule, Rev. 4, Nuclear Energy Institute, October 2004.

5. NUREG-1800, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, U.S. Nuclear Regulatory Commission, July 2001.

6. NUREG-1801, Generic Aging Lessons Learned (GALL) Report, U.S. Nuclear Regulatory Commission, July 2001.

2.0 Scoping and Screening Methodology for Identifying Structures and Components Subject to Aging Management Review, and Implementation Results

This section describes the process for identification of structures and components subject to aging management review in the Palisades Integrated Plant Assessment (IPA). For those systems, structures, and components (SSCs) within the scope of license renewal, 10CFR54.21(a)(1) requires a license renewal applicant to identify and list structures and components subject to aging management review. Furthermore, 10CFR54.21(a)(2) requires that methods used to identify these structures and components be described and justified. Technical information in this chapter serves to satisfy these requirements.

The scoping and screening methodology is described in Section 2.1. The results of the assessment to identify the systems and structures within the scope of license renewal (plant level scoping) are in Section 2.2. The results of the identification of the components and structural components subject to aging management review (screening) are in Section 2.3 for mechanical systems, Section 2.4 for structures, and Section 2.5 for electrical and instrumentation and controls systems.

Table 2.1-1 defines the terms used in the Scoping/Screening results tables to represent the intended functions of components, subcomponents, and structural members. Intended functions are the specific intended functions performed by in-scope passive components in support of system or structure intended functions. Passive components are components that perform an intended function without moving parts or without a change in configuration or properties.

2.1 Scoping and Screening Methodology

2.1.1 Introduction

Palisades conducted scoping and screening using the NEI 95-10, Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - the License Renewal Rule, as a primary guide. The scoping process defined the entire plant in terms of systems and commodity groups and identified their system level functions. These were then evaluated against the scoping criteria set forth in 10 CFR 54.4 (a)(1), (2), and (3), to determine whether they performed or supported a license renewal system intended function. Even if only a portion of a system or commodity group met the scoping criteria of 10 CFR 54.4, the system or commodity group was identified as in scope for license renewal.

An evaluation was performed to determine which components performed or supported the system intended functions identified previously. Not all components within an in-scope system or structure were found to be in-scope for license renewal. The in-scope boundary is depicted on the License Renewal Boundary Drawings (See Palisades Plant Scoping Boundary Drawing Index), which show the in-scope components highlighted in color.

The screening process evaluated the in-scope structures and components to determine which were long-lived and passive. The result was a list of long-lived, passive components that would be subject to an aging management review (AMR). Boundary descriptions, scoping, and screening results are contained in the appropriate sections of this application.

The Aging Management Review methodology is discussed in Section 3.0 of the LRA.

2.1.1.1 Plant Information Sources

A number of different information sources played a role in how scoping and screening was performed at PAL. The principal sources are discussed below.

2.1.1.1.1 Current Licensing Basis

The Current Licensing Basis (CLB) for license renewal purposes is defined in 10 CFR 54.3. The documents that comprise the CLB for PAL have been defined in accordance with this guidance. Therefore, where the term CLB is used in the LRA, its definition is consistent with 10 CFR 54.

2.1.1.1.2 Design Basis Events (DBEs)

The safety classifications for SSCs at Palisades were established based on a set of Design Basis Events (DBE), which include design basis accidents, anticipated operational occurrences, natural phenomena, and external events. The DBEs considered are consistent with the Palisades CLB. Chapter 2 and Chapter 14 of the Palisades FSAR describe these DBEs.

2.1.1.1.3 Safety Classifications

Safety classifications for systems, structures, and components at Palisades, for License Renewal purposes, fall into one of the following:

- Safety Related (SR)
- Non-Safety Related Affecting Safety Related (NSAS)
- Non-Safety Related (NSR)

Safety Related

Safety related structures, systems, and components are those that are relied upon to remain functional during and following design basis events to ensure:

- The integrity of the reactor coolant pressure boundary
- The capability to shut down the reactor and maintain it in a safe shutdown condition

- The capability to prevent or mitigate the consequences of accidents that could result in potential off site exposures comparable to the guideline exposures set forth in 10CFR100.11

An item is considered “relied upon” if failure of the item, in conjunction with a single active failure, results in the inability to perform a safety function (assuming that the item is designed for that purpose, and assuming that offsite power may or may not be available).

The safety related classification includes Electrical/Instrumentation and Control (I&C) systems whose purpose is to initiate automatic safety features or operator actions that are required for accident prevention and mitigation, or to shutdown the reactor and maintain it in a safe shutdown condition.

Non-Safety Related Affecting Safety Related

10 CFR 54.4(a)(2) specifies that the scope of systems, structures, and components (SSC) addressed by the license renewal rule includes “All non-safety related systems, structures, and components whose failure could prevent satisfactory accomplishment of any of the functions identified in paragraphs (a)(1)(i), (ii), or (iii) of this section” (i.e., the functions identified as safety related).

Non-Safety Related

Systems, structures, and components whose failure would not affect the safety of the plant or any safety related SSC.

Within the Palisades equipment database, components are coded as safety related and/or one of several other classifications based on design and safety functions. Components are not coded as non-safety per se, but several of the codes relate to non-safety related design considerations. The combination of codes determines whether the components are subject to the Consumers Energy Quality Program (i.e., Q-listed). All safety related components, and some non-safety related components, are identified as Q-Listed in the equipment database. The portion of the database which contains the various codes is called the Q-List, and is formally controlled. This list was used for the classification of equipment for license renewal.

2.1.1.1.4 Palisades Equipment Database

Component information for systems and equipment at Palisades can be found in the Palisades Equipment Database (EDB). The Palisades Equipment Database is a relational database system that is used by the entire site for maintenance and modifications. Components which have unique equipment identification numbers are identified in the EDB. The EDB does not include commodity items such as cables, raceways, conduits, piping, fireproofing, general construction items (e.g., nuts, bolts), or consumable materials (e.g., boric acid, diesel fuel, resins, etc.).

The EDB is the repository for Palisades Quality List (Q-List) information and contains component information data fields, including design related information and maintenance related information. The EDB was the primary source of component information used for determining the Q-List classification of each component.

2.1.1.1.5 **Design Basis Documents**

Design Basis Documents (DBDs) were available for a number of support and accident mitigation systems, selected licensing issues, and FSAR Chapter 14 Accident Analyses. DBDs are a tool to explain the requirements behind a design rather than describe the design itself. DBDs are not part of the CLB. However, DBDs were used to complement information obtained from other sources and to identify potential reference documents. These reference documents include, as applicable:

- FSAR and Technical Specifications
- Industry Codes, Standards, and Regulations
- Regulatory Correspondence and Documents
- Technical Correspondence, Analyses, and Reports
- Calculations
- Drawings, Specifications, Modifications, and Other Documents
- Vendor Reports, Specifications, and Drawings

2.1.1.1.6 **Drawings**

Reference Drawings

Controlled plant drawings were used as references when performing system, structure, and component evaluations for license renewal. In addition, the color-coded In Service Inspection (ISI) P&IDs were used to aid in identification of safety to non-safety related boundaries, and the color coded Appendix R Safe Shutdown Equipment P&IDs were used to aid identification of SSCs and flowpaths to support Appendix R requirements.

License Renewal Scoping Boundary Drawings

For mechanical, electrical, and civil/structural engineering disciplines, scoping boundary drawings were generated during the Boundary Evaluation phase of the scoping process (See Section 2.1.2.3). Boundary evaluations used controlled plant drawings, along with system/structure/component level scoping information. Color-coded drawings were generated to show the scoping boundaries. (See Palisades Plant Scoping Boundary Drawing Index)

2.1.1.1.7 Maintenance Rule Information

Maintenance Rule Scoping information for Palisades was the primary basis for system and structure level functions. This information included a comprehensive list of system functions originally considered by the Maintenance Rule Expert Panel.

2.1.1.2 License Renewal Tools

2.1.1.2.1 License Renewal Database

A license renewal database was used as the information repository for system, structure, and component evaluations, and to provide a platform for the administration of equipment data and output reports. The database design is consistent with the process guidance in NEI 95-10, and the process requirements of 10 CFR 54.

2.1.1.2.2 License Renewal Database Population

The license renewal database was initially populated with system and component level information from the Palisades Equipment Database, and system function information from the Palisades Maintenance Rule database. Component information within the Palisades Equipment Database relevant to license renewal, such as the equipment identifications, descriptions, and Q List information, was included. Additional system/component/commodity information was added as needed.

2.1.1.2.3 License Renewal Database Outputs

The license renewal database is a tool to assist in performing the license renewal process for Palisades. As such, it is not the official record for documentation. The license renewal database outputs were packaged into reports and then reviewed and approved outside of the database. The approved reports are considered the permanent plant records supporting the application.

2.1.1.3 Interim Staff Guidance Discussions

The NRC staff has identified a number of issues for which additional clarification was considered necessary, and has documented those clarifications in Interim Staff Guidance (ISG) documents. While many of the ISGs do not relate to scoping and screening, all are discussed here for completeness.

The ISGs determined to be in final form are as follows:

1) Staff Guidance on the Position of the GALL Report Presenting One Acceptable Way to Manage Aging Effects for License Renewal (ISG-1)
2) Staff Guidance on Scoping of Equipment Relied on to Meet the Requirements of the Station Blackout (SBO) Rule (10 CFR 50.63) for License Renewal (10 CFR 54.4(a)(3)) (ISG-2)
3) Proposed Revision of Chapters II and III of Generic Aging Lessons Learned (GALL) Report on Aging Management of Concrete Elements (ISG-3)
4) Aging Management of Fire Protection Systems for License Renewal (ISG-4)
5) Addition of Aging Management Program (AMP) for Fuse Holders (ISG-5)
6) Updating the Approved Guidance Documents ISG Process (ISG-8)
7) “Class of 03” Standard License Renewal Application Format (ISG-10)
8) Environmental Assisted Fatigue for Carbon/Low Alloy Steel (ISG-11)

NRC action has not been completed on some ISGs. However, those draft ISGs that were publicly available during preparation of this application, were considered. In addition to the final form ISGs discussed above, the following draft ISGs were also considered:

1) Identification and Treatment of Housing for Active Components (Draft ISG-6)
2) Scoping Guidance for Fire Protection (FP) Systems, Structures and Components (Draft ISG-7)
3) Scoping Criteria 54.4(a)(2) (Draft ISG-9)
4) Addition of GALL Aging Management Program XI.M35, “One-Time Inspection of Small-Bore Piping” (Draft ISG-12)
5) Management of Loss of Preload on Reactor Vessel Internals Bolting Using the Loose Parts Monitoring System (Draft ISG-13)
6) Operating Experience with Cracking in Bolting (Draft ISG-14)
7) Revision to Generic Aging Lessons Learned (GALL) Aging Management Program (AMP) XI.E2 (Draft ISG-15)

8) Time Limited Aging Analyses Supporting Information for license renewal applications (Draft ISG-16)

9) Periodic Inspections of Bus Ducts and Develop GALL AMP XI.E4 for bus ducts (Draft ISG-17)

10) Revision to GALL AMP XI.E3 for Inaccessible Cable (Draft ISG-18)

11) Revision of Aging Management Program XI.M11 (Draft ISG-19)

12) Revision of Aging Management Program XI.M19 (Draft ISG-20)

For the convenience of NR reviewers, discussions are provided in Sections 2.1.1.3.1 through 2.1.1.3.20 to summarize how Palisades considered the ISGs.

2.1.1.3.1 Staff Guidance on the Position of the GALL Report Presenting One Acceptable Way to Manage Aging Effects for License Renewal (ISG-1)

This ISG highlights that the GALL is only one acceptable way to manage aging for license renewal. No response is necessary.

2.1.1.3.2 Staff Guidance on Scoping of Equipment Relied on to Meet the Requirements of the Station Blackout (SBO) Rule (10 CFR 50.63) for License Renewal (10 CFR 54.4(a)(3)) (ISG-2)

ISG-2 establishes the NRC position that the SSCs relied on for recovery from Station Blackout (SBO), in addition to SSCs relied on for coping with an SBO, should be in scope for License Renewal, even if it is outside of the plant’s CLB. The staff position is that the plant system portion of the offsite power system should be included in scope.

For Palisades, including the SBO recovery equipment into the scope of license renewal brings into scope various electrical components and associated structures associated with providing offsite power via the switchyard to the two 2400 VAC safety related buses.

In summary, the following components support plant recovery from an SBO event, and have been included in the scope of license renewal in accordance with ISG-2.

- Switchyard disconnects and associated high voltage conductors, insulators and bus
- Safeguards transformer and direct buried cables to the safeguards bus
- Safeguards bus and building
- Non-segregated bus, isolation breakers and medium voltage cables to the 2400 VAC safety related buses

- High voltage overhead lines and towers from the switchyard to the startup transformers
- Startup transformers and load isolation breakers and associated medium voltage cables to the 2400 VAC safety related buses

The passive, long-lived electrical commodities associated with the restoration path for offsite power that are subject to an aging management review are:

- High-voltage transmission conductors
- High-voltage insulators
- High-voltage switchyard bus and connectors
- Inaccessible medium voltage cables and connections
- Non-segregated phase bus and connections
- Electrical cables and connections

2.1.1.3.3 Proposed Revision of Chapters II and III of Generic Aging Lessons Learned (GALL) Report on Aging Management of Concrete Elements (ISG-03)

ISG-3 summarized the changes as a result of the performance of aging management reviews of in-scope concrete components. Palisades has concluded that many of these components do not require aging management for the period of extended operation. This conclusion is based on a review of the material of construction, the environment, and industry and plant-specific operating experience for these components. However, for accessible concrete portions of the containment (NUREG-1801, Section IIA), the examination requirements and inspection intervals of ASME Section XI, Subsection IWL remain applicable as an aging management program (AMP) for the period of extended operation. NUREG-1801, Section IIIA structures, that are subject to an aging management review, are subject to similar inspections as part of the Structural Monitoring Program.

NUREG-1801 does not require plant specific aging management programs for concrete components in inaccessible areas for which the applicant can demonstrate a non-aggressive environment and the use of high quality concrete. The environment to which inaccessible concrete at Palisades is exposed, is not aggressive; Palisades concrete structures are constructed using ingredients conforming to ACI and ASTM standards and practices, which provide for a good quality, dense, well cured, and low permeability concrete. Therefore, further aging management inspections of normally inaccessible structures will only be considered when aging effects on accessible concrete structures indicate that potential detrimental aging effects could also be occurring in inaccessible areas. In addition, based on a long history of water chemistry data, with test results documenting a non-aggressive environment, a formal program to perform lake or ground water chemistry monitoring is unnecessary.

2.1.1.3.4 Aging Management of Fire Protection Systems for License Renewal (ISG-4)

ISG-04 clarifies the NRC staff position concerning GALL Report Section XI.M26, “Fire Protection,” and XI.M27, “Fire Water System,” with regard to wall thinning of fire protection piping due to internal corrosion, testing of sprinkler heads, and valve line-up inspections of Halon/Carbon Dioxide fire suppression systems. Palisades considered this ISG during the preparation of the LRA, and the Palisades Fire Protection program is consistent with the aging management program elements presented in ISG-04.

2.1.1.3.5 Addition of Aging Management Program (AMP) for Fuse Holders (ISG-5)

The NRC ISG for screening of fuse holders determined that fuse holders that are not part of an active component or assembly, such as switchgear, power supplies, power inverters, battery chargers, and circuit boards, are considered to be passive electrical components and, therefore, require an aging management review. Such fuse holders are evaluated for license renewal in the same manner as terminal blocks and other types of electrical connections. The NRC ISG also determined that fuse holders, that are piece parts of an active assembly, are not subject to aging management review, because they would be subject to periodic inspection and maintenance in accordance with the maintenance and surveillance activities applicable to the active assembly.

Using the criteria of ISG-5, a review of Palisades’ fuse holders was performed. The review determined there are some fuse holders that meet these criteria but there is no aging mechanism that would warrant aging management. Therefore, it was concluded that no aging management activity is required for this commodity group. See Section 3.6 for additional information.

2.1.1.3.6 Identification and Treatment of Housing for Active Components (Draft ISG-6)

ISG-6 reiterates the philosophy that, during the extended period of operation, safety related functions should be maintained in the same manner and to the same extent as during the current licensing term. Examples of structures and components that perform passive functions are listed in 10 CFR 54.21(a)(1)(ii), which states, “These structures and components include, but are not limited to, pump casings, valve bodies….”

Pumps and valves were just an example here, meant to focus the AMR process on the passive function of an SSC. That passive function is not limited to the pressure boundary of the Reactor Coolant System. The exclusion of an SSC due to its active nature only applies to that portion of the SSC with an active function and not to those portions of the SSC with a passive function. Therefore, at Palisades, pump casings, valve bodies, fan housings and damper housings are considered to be within scope and subject to an AMR.

2.1.1.3.7 **Scoping Guidance for Fire Protection (FP) Systems, Structures and Components (Draft ISG-7)**

The Palisades Fire Protection System scoping takes into consideration the interim staff guidance (ISG-7) that was developed to clarify the requirements of 10 CFR 54.4(a)(3) as it pertains to 10 CFR 50.48. This includes General Design Criterion 3, Appendix R and associated license conditions, and the proposed revisions to NUREG-1800, “Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants” concerning incorporation of this guidance into the improved license renewal guidance documents when this ISG is finalized.

The Fire Protection Program (FPP) was developed to maintain compliance with 10 CFR 50.48, Appendix R to 10 CFR 50, and Appendix A of Branch Technical Position APCSB 9.5-1 by meeting the following objectives in fire areas important to safety (the effects of fires on SSCs “important to safety” are addressed in 10 CFR 50.48 and provide a general level of protection that is afforded to all systems, not only those required for safe shutdown):

- Reduce the likelihood of fires.
- Promptly detect and extinguish fires that do occur.
- Maintain safe-shutdown capability if a fire does occur.
- Prevent release of a significant amount of radioactive material if a fire does occur.

The Palisades Fire Protection Program also includes the fire protection equipment required for insurance purposes for facilities such as the Service, Support, Security, and Dry Fuel Support Buildings, temporary trailers, etc. The SSCs that are not important to safety have been excluded from the scope of license renewal.

A review has been performed to identify the specific SSCs that fall within the scope of license renewal for the Fire Protection Program, including the SSCs relied upon in the Fire Hazards Analysis. As a result of this review, the following features and equipment have been included within the scope of license renewal for fire protection:

- Fire detection and suppression equipment
- Fire barriers (includes doors, walls, gap covers, penetration seals, etc.)
- Safe shutdown equipment (Appendix R), if not included in other systems
- Reactor coolant pump oil collection sub-system
- Fire fighting support (communications, lighting, bottled air/SCBA, fire hose, extinguishers)
- Panels
- Post-fire repair equipment (required for cold shutdown)
- Radiant energy shields

The screening methodology applied to the Appendix R post-fire repair equipment that is maintained in storage was also considered and is discussed below.

In response to the NRC letter from Chris Grimes to Doug Walters (NEI) dated February 11, 1999, Subject: Screening of Equipment Kept in Storage, a review has been performed to identify equipment that (1) is maintained in storage, (2) is reserved for installation in the plant in response to a design basis event (DBE), and (3) requires an AMR. In addition to passive components, the review has also considered stored active components that are not routinely inspected, tested, and maintained. The Appendix R stored equipment is used to restore power to pre-selected plant components and to provide cooling to certain areas after a fire in order to attain cold shutdown.

This scoping methodology used at Palisades is in accordance with the guidelines of the draft ISG-7.

2.1.1.3.8 **Updating the Approved Guidance Documents ISG Process (ISG-8)**

This discusses an administrative process to update guidance documents. No response is required.

2.1.1.3.9 **Scoping Criteria 54.4(a)(2) (Draft ISG-9)**

By letters dated December 3, 2001, and March 15, 2002, the NRC issued a staff position to the Nuclear Energy Institute (NEI) which described areas to be considered and options it expects licensees to use to determine what systems, structures, or components (SSCs) meet the 10 CFR 54.4(a)(2) criterion.

Palisades considered the guidance contained in this ISG during the enhanced license renewal (a)(2) scoping process.

2.1.1.3.10 “Class of 03” Standard License Renewal Application Format (ISG-10)

This ISG endorsed the NEI-proposed application format later embodied in NEI 95-10 Revision 4. This application has been prepared consistent with this guidance.

2.1.1.3.11 Aging Management of Environmental Fatigue for Carbon/Low-Alloy Steel (Draft ISG-11)

This draft ISG was closed by the NRC in 2003. No response is required.

2.1.1.3.12 Addition of GALL Aging Management Program XI.M35, “One-Time Inspection of Small-Bore Piping” (Draft ISG-12)

ISG-12 clarifies the circumstances under which the NRC expects a one-time inspection for small bore Class 1 piping to provide additional assurance that small-bore piping is not aging, or that the effects of aging will be insignificant during the extended period of operation. See Appendix B to the LRA for additional information.

2.1.1.3.13 Management of Loss of Preload on Reactor Vessel Internals Bolting Using the Loose Parts Monitoring System (Draft ISG-13)

This ISG is under NRC development and has not been issued for use. No response is required.

2.1.1.3.14 Operating Experience with Cracking in Bolting (Draft ISG-14)

This ISG is under NRC development and has not been issued for use. No response is required.

2.1.1.3.15 Revision to Generic Aging Lessons Learned (GALL) Aging Management Program (AMP) XI.E2 (Draft ISG-15)

This ISG provides clarification to the NUREG-1801 Aging Management Program X1.E2 to allow specific calibrations or surveillance in lieu of Technical Specification surveillance.

Consistent with the ISG, the Palisades program for aging management of electrical cables and connections used in instrumentation circuits that are sensitive to reduction in conductor IR requires periodic testing of cables in applicable circuits. See the Non-EQ Electrical Commodities Condition Monitoring Program discussion in Appendix B for additional information.

2.1.1.3.16 Time Limited Aging Analyses Supporting Information for license renewal applications (Draft ISG-16)

Draft ISG-16 provides guidance on information that should be provided about plant specific time-limited aging analyses (TLAAs). The guidance in draft ISG-16 was considered in the development of the information included in section 4.0, "Time-Limited Aging Analyses."

2.1.1.3.17 Periodic Inspections of Bus ducts and Develop GALL AMP XI.E4 for bus ducts (Draft ISG-17)

This ISG provides direction to review bus insulation due to water intrusion in bus ducts and bus bar connections due to thermal cycles, and develops a NUREG-1801 Aging Management Program X1.E4 for bus ducts.

Consistent with Draft ISG-17, the Palisades program for non-segregated phase bus and connections requires periodic inspections. See the Non-EQ Electrical Commodities Condition Monitoring Program discussion in Appendix B for additional information.

2.1.1.3.18 Revision to GALL AMP XI.E3 for inaccessible cable (Draft ISG-18)

This ISG provides an enhancement to NUREG-1801 Aging Management Program X1.E3 to prevent moisture collection in manholes where medium-voltage cables are located.

Consistent with Draft ISG-18, the Palisades program for Inaccessible medium-voltage cables and connections requires periodic inspection of manholes that contain in scope medium-voltage cables. See the Non-EQ Electrical Commodities Condition Monitoring Program discussion in Appendix B for additional information.

2.1.1.3.19 Revision of Aging Management Program XI.M11 (Draft ISG-19)

This ISG is under NRC development and has not been issued for use. No response is required.

2.1.1.3.20 Revision of Aging Management Program XI.M19 (Draft ISG-20)

This ISG is under NRC development and has not been issued for use. No response is required.

2.1.1.4 Consideration of Power Uprating in License Renewal Evaluation Process

Palisades has implemented a 1.4% power uprate associated with margin uncertainty recapture. This small power uprate was considered in the evaluations supporting License Renewal. No further power uprates are currently anticipated.

2.1.1.5 Industry Guidance

In addition to NRC-issued guidance, a number of EPRI reports, vendor reports, and other nuclear industry documents were used as input to the IPA process. A primary reference for the design of license renewal work processes and the format of this application was NEI 95-10, Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - the License Renewal Rule.

2.1.2 Scoping Methodology

10 CFR 54 provides specific criteria for determining which systems, structures, and components should be reviewed and evaluated for inclusion in the scope of License Renewal. Specifically, 10 CFR 54.4 of the rule states that:

(a) Plant systems, structures, and components within the scope of this part are:

(1) Safety related systems, structures, and components which are those relied upon to remain functional during and following design basis events (as defined in 10 CFR 50.49(b)(1)) to ensure the following functions:

(i) The integrity of the reactor coolant pressure boundary;

(ii) The capability to shut down the reactor and maintain it in a safe shutdown condition; or

(iii) The capability to prevent or mitigate the consequences of accidents that could result in potential off-site exposures comparable to those referred to in 10 CFR 50.34(a)(1), 10 CFR 50.67(b)(2), or 10 CFR 100.11, as applicable.

(2) All non-safety related systems, structures, and components whose failure could prevent satisfactory accomplishment of any of the functions identified in paragraphs (a)(1)(I), (ii), or (iii) above.

(3) All systems, structures, and components relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the NRC's regulations for fire protection (10 CFR 50.48), environmental qualification (10 CFR 50.49), pressurized thermal shock (10 CFR 50.61), anticipated transients without scram (10 CFR 50.62), and station blackout (10 CFR 50.63).

(b) The intended functions that these systems, structures, and components must be shown to fulfill in 10 CFR 54.21 are those functions that are the bases for including them within the scope of License Renewal as specified in paragraphs (a)(1) - (3) of this section.

The Scoping Process defined the entire plant in terms of major license renewal systems, structures, and commodity groups. System, structure, and commodity group system-level functions were identified and evaluated against criteria provided in 10 CFR 54.4(a)(1), (2), and (3), to determine whether the system, structure, or commodity should be considered in-scope for license renewal. Even if only a portion of a system, structure, or commodity fulfilled a scoping criterion, the system/structure/commodity was identified as in-scope for license renewal and received further screening. System boundaries were adjusted as needed to permit more efficient treatment of in-scope SSC.

The scoping methodology utilized by Palisades is consistent with the guidance provided in NEI 95-10. Existing plant documentation was used for this review, including the Palisades Current Licensing Basis documents, controlled drawings, and the equipment database. Once the systems and structures were identified for inclusion in the scope, they were further evaluated with the next step in the IPA process - screening.

2.1.2.1 Application of License Renewal Scoping Criteria

2.1.2.1.1 Scoping Criterion 1 - Safety Related Systems and Structures

The first scoping category in 10 CFR 54.4 involves safety related systems, structures, and components. The license renewal criteria for safety related SSCs are consistent with Palisades CLB, Q-List, and Maintenance Rule classification criteria. Therefore, SSCs that support a safety related function as defined in the CLB are considered in-scope of license renewal. In addition, a review was made of SSCs that have a safety related designation within the Q-List or the Maintenance Rule.

The Palisades CLB definition of Safety Related is not identical to the definition provided within the license renewal rule. 10 CFR 54 cites 10 CFR 50 Section 50.67(b)(2) and Section 50.34(a)(1), in addition to Section 100.11, as causing SSC to fall within the definition of safety related, and the Palisades CLB cites only Section 100.11. The differences related to §50.34(a)(1) and §50.67(b)(2) were investigated, and this did not result in any components being considered safety related in addition to those identified using the CLB.

2.1.2.1.2 Scoping Criterion 2 - Non-Safety Related Systems and Structures Affecting Safety Related Functions

10 CFR 54.4(a)(2) states that SSCs within the scope of License Renewal include non-safety related SSCs whose failure could prevent satisfactory accomplishment of any of the functions identified for safety related SSCs.

Components meeting the scoping criterion of 10 CFR 54.4(a)(2) will generally fall into the three categories listed below:

(1) Current Licensing Basis (CLB). The Palisades CLB includes a number of specific issues that identify non-safety related SSCs that meet the intent of 10 CFR 54.4(a)(2).

(2) Non-safety related SSCs directly connected to safety related SSCs (typically piping systems).

(3) Non-safety related SSCs that are not directly connected to safety related SSCs.

Categories 2 and 3 apply to non-safety related systems and components that may not be specifically identified in the CLB. These two categories were part of an expanded scoping effort conducted by Palisades relatively late in the application development process.

In many cases components were brought into scope in more than one category. Each of the three categories is described in the following text.

(1) Non-Safety Related SSCs Identified in the CLB

a. High Energy Line Break (HELB)

Non-safety related whip restraints, jet impingement shields, blowout panels, etc., that are designed and installed to protect safety related equipment from the effects of a HELB, are part of the CLB and should be included within the scope of license renewal per 54.4(a)(2) as mitigative features. These protective features are typically associated with the structure, and addressed in the Civil/Structural area review.

The CLB design is based on HELB analysis that typically assumes failure only occurs at specific locations. If the mitigative option was chosen, then it was necessary to demonstrate that the choice of break locations remain valid for all potential aging mechanisms, or the mitigative features are adequate to address all potential failure locations that could result from aging. Otherwise, it was necessary to include all or part of the piping system within the scope of license renewal per 54.4(a)(2).

b. Internal/External Flooding

If level instrumentation and alarms are utilized to warn the operators of flood conditions, and operator action is necessary to mitigate the flood, then these instruments and alarms are within the scope of license renewal per 10 CFR 54.4(a)(2). If non-safety related sump pumps, piping and valves, are necessary to mitigate the effects of a flood which threatens safety related SSCs intended functions, then these components are also within the scope of license renewal per 10 CFR 54.4(a)(2).

Non-safety related walls, curbs, dikes, doors, etc., that provide flood barriers to protect safety related SSCs, are within the scope of license renewal per 10 CFR 54.4(a)(2). These are typically included as part of the building structure, and evaluated in the civil/structural area review.

c. Internal/External Missiles

Missiles can be generated from internal or external events such as failure of rotating equipment or tornados. Inherent non-safety related features that protect safety related equipment from missiles are within the scope of license renewal per 10 CFR 54.4(a)(2). These protection features (missile barriers) are typically included as part of the building structure, and evaluated in the civil/structural area review.

d. Heavy Load Lifting Equipment

The overhead-handling systems from which a load drop could result in damage to any system that could prevent the accomplishment of a safety related function, are considered to meet the criteria of 10 CFR 54.4(a)(2) and are within the scope of license renewal. These are included as part of the building structure, and evaluated in the civil/structural area review.

e. Operator Habitability and Access

Civil/Structural components/commodities that support operator habitability and access to operate safety related equipment in the Auxiliary Building and Turbine Building are included in-scope of license renewal. These items are evaluated by the license renewal civil/structural discipline.

(2) Non-Safety Related SSCs Directly Connected to Safety Related SSCs

For non-safety related SSCs directly connected to safety related SSCs, there are two primary areas of concern to address. First was the possibility that the pressure boundary related spatial interactions (pipe whip, jet impingement, spray, flood, etc.) of the non-safety piping could damage adjacent safety related components. This is addressed the same as the non-safety related SSCs not directly connected to safety related SSCs discussed in category (3) below.

The second issue was to ensure the attached non-safety related piping and supports maintain their structural integrity to ensure the structural integrity of the attached safety related piping. This was accomplished by ensuring the pipe supports up to and including the first seismic or equivalent seismic anchor beyond the safety/non-safety interface, were within the scope of license renewal per 54.4(a)(2).

a) Initial (a)(2) Scoping Effort for NSR Directly Connected to SR

The initial (a)(2) scoping effort for NSR directly connected to SR was based on industry experience and the positions take by early applicants for license renewal. The philosophy was that if pipe supports of the connected NSR piping are age managed using a spaces approach, and the connected piping itself was subject to the same aging management (e.g., chemistry control) as the safety-related piping, then there was no need to explicitly identify the locations of anchors or scoping boundaries. The connected NSR piping was determined to be in scope for license renewal, but a generic definition of the piping and supports, similar to a commodity, was considered to be sufficient. This position was later revised and the scoping effort was expanded to more explicitly define the scoping boundaries.

b) Expanded (a)(2) Scoping Effort for NSR Directly Connected to SR

For non-safety SSCs directly connected to safety-related SSCs (typically piping systems), the non-safety piping and supports, up to and including the first seismic or equivalent seismic anchor beyond the safety/non-safety interface, are within the scope of license renewal per 54.4(a)(2). For this purpose, Palisades defined the “first seismic or equivalent seismic anchor” such that the failure in the non-safety related pipe run beyond the first seismic or equivalent seismic anchor would not render the safety-related portion of the piping unable to perform its intended function under CLB design conditions. An actual seismic anchor is taken to be a physical 6-way support (restraint against forces and moments in each of three orthogonal directions) that can be either a fabricated restraint or a base mounted component (e.g., pump, heat exchanger, tank, etc.). If a base mounted component is credited as an anchor, it is included in scope. An equivalent seismic anchor is typically defined as at least two rigid supports in each of the three orthogonal directions. Components that are part of the non-safety related piping segment up to and including the first seismic or equivalent seismic anchor were included in the scope of license renewal, and a more detailed description of the scoping boundary was provided for the affected components.

3) Non-Safety Related SSCs Not Directly Connected to Safety Related SSCs

For non-safety related SSCs that are not directly connected to safety related SSCs, or are connected downstream of the first seismic or equivalent seismic anchor, the non-safety related SSCs may be in-scope if their failure could prevent the performance of the safety function of a safety related SSC. The failures to be considered are two-fold: consequences due to the loss of pressure boundary of the non-safety related piping on adjacent SSCs (e.g., pipe whip, jet impingement, spray, flooding, etc.); and loss of support of the non-safety related piping resulting in it potentially falling on adjacent SSCs (seismic II/I).

a) Pressure Boundary Spatial Interactions

To determine which non-safety related SSCs may be in-scope for 54.4(a)(2) due to pressure boundary spatial interaction, including HELB, two options exist: a mitigative option or a preventive option. These options are discussed in the following text:

Mitigative Option

Palisades defines “mitigative” consistent with NEI-95-10. It means that the effects of failures of a non-safety related SSC on safety related SSCs are mitigated by some means. This mitigation is such that the failure of the non-safety related SSC will not prevent the performance of a safety related SSC intended function identified in 54.4(a)(1). If the mitigative option is used, then the mitigative features (whip restraints, spray shields, supports, barriers, etc.) need to be included within the scope of license renewal per 54.4(a)(2), and the non-safety system can be excluded from the scope of license renewal provided the mitigative features are adequate to address all potential failure locations that could result from aging.

This option typically includes non-safety related SSCs identified in the CLB as discussed in Category 1 above.

Preventive Option

If mitigative features are not installed, or cannot be shown to adequately protect safety related SSCs, then the preventive option is used. For the preventive option, vulnerable safety related components in proximity to the non-safety related systems are identified. This is performed via review of plant documentation and/or via walkdowns to identify non-safety systems, or portions of systems, that have spatial interaction potential (e.g., pipe whip, jet impingement, spray, flooding, etc.) with vulnerable safety related equipment assuming a failure anywhere along the length of the non-safety system.

Palisades initially conducted scoping under this preventive option using industry precedents and judgment to exclude certain components from scope, but subsequently expanded the license renewal scope under this category by removing those exclusions guidelines.

- Original (a)(2) Scoping Effort for Preventive Option

  Originally, Palisades used specific limitations that were based upon engineering judgment and earlier applications. For example, the spatial interactions evaluations initially relied upon were based upon consequence analysis work that had already been completed for another project. Limitations in that work included use of the analysts' judgment on spray distances as well as only considering active components as being susceptible targets.

- Expanded (a)(2) Scoping Effort for Preventive Option

  Based upon recent NRC questions, current NRC expectations require a comprehensive strategy whereby there is no limitation to the duration of exposure to spray/leakage. This necessitates consideration of passive as well as active components to potential functional failure due to spray/leakage.

  Additionally, current expectations are that there be no distance or other subjective criteria allowed in assessing whether a safety related component may be excluded from consideration if it exists in the same general area as the non-safety related piping system. General area is defined as being on the same floor of a building with no barrier walls between the fluid source and the safety related component. The original criteria did permit judgment to be exercised in determining susceptibility to spray or leakage.

  Thus, the augmented methodology required that all pressurized liquid/steam systems in the general area of safety related components, passive or active, be considered in scope for license renewal. This expanded approach was applied to a review of SSCs not already in scope for license renewal. SSCs already in-scope were not reevaluated. A significant amount of additional non-safety related piping and components were brought into scope for license renewal because of this expanded scoping effort.

At Palisades, both the preventive and mitigative options were utilized. All mitigative SSCs were identified via review of the CLB and included in-scope as discussed in Category 1 above. In addition, a review of potential consequences due to pressure boundary failure was performed for all non-safety related piping in the plant. Consequences that affect the ability of safety related components to perform their intended functions were identified and those systems, or portions of systems, were included in-scope. This was performed by use of piping failure consequence information developed from detailed plant documentation reviews (CLB, Design Basis Documents, piping area drawings, equipment layout drawings, flooding analysis, HELB analysis, etc.) augmented by walkdown inspections, assuming failure anywhere along the pipe.

b) Seismic II/I

Potential spatial interaction of non-safety piping systems that may fall on or otherwise physically impact safety related SSCs is considered as seismic II/I. Palisades was not originally designed to seismic II/I criteria. To address seismic II/I considerations at Palisades, guidance from NEI 95-10 was utilized.

In accordance with NEI 95-10, NUREG CR-6239 (section 4.2.6) reviewed earthquake experience data, including aged pipe, and concluded the following:

- No experience data exists of welded steel pipe segments falling due to a strong motion earthquake.
- Falling of piping segments is extremely rare and only occurs when there is a failure or unzipping of the supports.
- These observations hold for new and aged pipe.

Piping supports for non-seismic piping need to be intact in order to prevent physical impacts on safety related equipment during a seismic event. Consistent with leak-before-break philosophy, it can also be assumed that piping which has retained its functional integrity will remain supported as long as its supports do not fail. If aged non-safety related piping does not fall during a seismic event, it will also not fall due to the aging process of the pipe, as long as its supports are intact.

Therefore, as long as the supports for these piping systems are managed, falling of piping sections is not credible, and the piping section itself would not be in-scope for 54.4(a)(2) due to physical impact hazard (although the leakage/spray/flooding hazard may still apply).

All supports for non-seismic piping systems with a potential for spatial interaction with safety-related SSCs are included within the scope of license renewal. These supports are addressed in a commodity fashion within the civil / structural area utilizing a “spaces” approach wherein all pipe supports that exist in areas that contain safety related components are included in-scope.

4) Non-Safety Related Structures That Can Affect Safety Related Structures or Components

With regards to failure of non-safety related structures affecting the function of safety related structures and components, only the turbine building contains safety related components whose functions could be affected by failure of the non-safety related turbine building. The turbine building and discharge structure (warm water recirculating pump P-5 pumphouse, specifically) are both non-safety related structures that contain non-safety related components that could affect safety related components. There are no non-safety related structures that have been identified whose failure could affect the ability of a safety related structure from performing its safety related function

2.1.2.1.3 Scoping Criterion 3 - Systems and Structures Required by Other Regulations Identified in 10 CFR 54

The third scoping category in 10 CFR 54.4 involves SSCs relied upon by five regulated events identified in the Rule. Specifically, 10 CFR 54.4(a)(3) defines in-scope SSCs as those relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with one or more of the regulated events:

1) Fire Protection (10 CFR 50.48)
2) Environmental Qualification (10 CFR 50.49)
3) Pressurized Thermal Shock (10 CFR 50.61)

4) Anticipated Transients Without Scram (10 CFR 50.62)

5) Station Blackout (10 CFR 50.63)

Any system, structure, or component that is required to meet one or more of these regulations is identified as a Criterion 3 component. All criterion 3 systems/components are considered to be in-scope of License Renewal.

A separate section is provided below for each of these regulations that describes the methodology used to determine any system, structure, or component that is required to support the regulations.

1) Fire Protection (FP)

The design of the Palisades Fire Protection program is based upon the defense in depth concept. Multiple levels of protection are provided so that should a fire occur, it will not prevent safe plant shutdown and the risk of a radioactive release to the environment will be minimized. These levels of protection include Fire Prevention, Fire Detection and Mitigation, and the Capability to Achieve and Maintain Safe Shutdown should a fire occur. This protection is provided through commitments made to Branch Technical Position APCSB 9.5 1, Appendix A, 10 CFR 50.48, and 10 CFR 50 Appendix R. Fire protection features and commitments are described in detail in the Fire Protection Program Report (FPPR). The SSCs at Palisades that support these multiple levels of protection are considered to be within the scope of license renewal. Note that Palisades' scoping and screening process is consistent with the NRC Staff's guidance on consumables provided in NUREG-1800, Table 2.1 3. For fire protection, this would include such items as fire extinguishers, fire hoses, portable lighting, and air packs.

Information sources used in performing this portion of the scoping effort were the Post Fire Safe Shutdown Analysis Report (PSSA), the Safe Shutdown Equipment drawings, and the Fire Hazards Analysis Report (FHA). These are further discussed below.

a. Safe Shutdown Analysis Report

Section III.G.1 of Appendix R to 10 CFR 50 requires that fire protection features be provided for systems, structures and components important to safe shutdown. In order to meet these requirements, all equipment required for safe shutdown, including the associated power and control cables, and any equipment which could adversely affect safe shutdown if spuriously actuated by fire-induced faults, have been identified for every fire area in the plant in order to assess the fire protection required. Safe shutdown for

Palisades is defined as hot standby conditions as a minimum, with the capability to proceed to cold shutdown should conditions warrant. Using this information, a Safe Shutdown Analysis was performed to determine the impact of a postulated fire on the safe shutdown equipment and circuitry within each fire area. Where a safe shutdown function was prevented, corrective actions (e.g., cable rerouting, cable protection, procedure changes, etc.) have been implemented to resolve the concern, or operator manual actions have been specified. In some cases credit is taken for equipment (other than the redundant counterpart) that provides a redundant function to the equipment affected by a postulated fire.

b. Safe Shutdown Equipment

The first step of the safe shutdown analysis process was to establish the safe shutdown functions required to be performed. This was followed by selection of the systems, specific system equipment, and electrical/control circuits required to accomplish these functions. The selection of systems and how the systems fulfill the Appendix R performance goals were then depicted on system-level logic diagrams. Process flow paths for each of the required systems were then traced on plant flow diagrams. Based on these system flow paths, the minimum equipment necessary to bring the plant to cold shutdown was compiled. This compilation contains all power generation and distribution equipment (e.g., diesel generators, batteries, switchgear, motor control centers, power panels, etc.) that are required for the operation of the listed equipment. In addition, it includes equipment that, although not required for safe shutdown, could adversely affect safe shutdown if spuriously actuated by a fire-induced electrical fault. Safe shutdown equipment is shown on the color coded Appendix R safe shutdown equipment drawings.

c. Fire Hazards Analysis (FHA)

A systematic approach was used for the review of the fire hazards and their exposure to safety related equipment and components necessary for safe shutdown within the area. The type and quantity of combustible materials, type of fire hazards these materials present in the area, and the fire protection features (passive, active and manual) for the area were reviewed. The effects of postulated fires on the performance of safe shutdown functions and the minimization of radioactive releases to the environment were evaluated for each fire area. These evaluations identify those portions of the plant Fire Protection System that are relied on to support the safe shutdown function of Appendix R.

Using the described information sources, the components in the Palisades equipment database were reviewed to determine those required to support the safe shutdown function for Fire Protection. Components supporting safe shutdown but not included in the equipment database were added to the license renewal database as necessary. Specifically, the FHA was the primary source of information to identify the classical fire protection related components to be included in-scope of license renewal. Components supporting fire detection, barriers (i.e., dampers), suppression, and fire fighting were identified primarily by the FHA and fire implementing procedures. Components and commodities of systems other than the Fire Protection System that support Appendix R safe shutdown requirements were identified primarily by use of the Appendix R color coded drawings. Structural components included in-scope of license renewal were identified by review of the Fire Protection area drawings that depict Appendix R fire barriers and boundaries, as well as by review of the mechanical systems, and their locations, credited for Appendix R.

2) Environmental Qualification (EQ)

The criteria for determining which equipment requires environmental qualification is defined by 10 CFR 50.49.

Electric equipment covered in 10 CFR 50.49 is characterized as follows:

a) Safety related electric equipment that is relied upon to remain functional during and following design basis events to ensure -

(i) The integrity of the reactor coolant boundary,

(ii) The capability to shut down the reactor and maintain it in a safe shutdown condition, or

(iii) The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guidelines in §50.34(a)(1), §50.67(b)(2), or §100.11 of Title 10 CFR.

Design Basis Events are defined as conditions of normal operation, including anticipated operational occurrences, design basis accidents, external events, and natural phenomena for which the plant must be designed to ensure functions (i) through (iii) of this paragraph.

Requirements for (1) dynamic and seismic qualification of electric equipment important to safety, (2) protection of electric equipment important to safety against other natural phenomena and external events, and (3) environmental qualification of electric equipment important to safety located in a mild environment are not included within the scope of the EEQ Program. A mild environment is an environment that would at no time be significantly more severe than the environment that would occur during normal plant operation or during anticipated operational occurrences.

b) Non-safety related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions.

c) Certain post-accident monitoring equipment (Refer to Regulatory Guide 1.97, Revision 3, “Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident”).

The components, that meet these criteria for Palisades, are identified on the “Environmental Qualification Master Equipment List (MEL).”

Electrical equipment is qualified to withstand the normal and accident environments. In addition, consideration is given to all significant types of age-related degradation that can have an effect on the functional capability of the equipment.

The following documents describe the Palisades approach to environmental qualification of electrical equipment, give instructions for determining which equipment is qualified and for which environmental parameters, and identify the electrical equipment that has currently been qualified:

Environmental Qualification Master Equipment List (MEL)

The Palisade MEL defines for Palisades all equipment determined to be within the scope of 10 CFR 50.49. The equipment is listed by equipment tag number or by commodity group. Any newly installed 10 CFR 50.49 equipment is added to this list.

Harsh Environments List (HEL)

The Palisade HEL contains the accident temperature and pressure profiles for the containment building, and the normal and accident environments for all harsh environment areas of the plant.

For scoping and screening at Palisades, the MEL was used as the basis for determining if a SSC is in-scope of license renewal. Specifically, components and their associated systems were identified in the MEL. These systems were placed in-scope of license renewal for EQ and the components from the MEL defined the boundary. Similarly, the structure that the component is located was identified from the MEL and the structure was included in-scope of license renewal since it supports and protects the components. Applicable commodities, such as cables, Raychem, conduit seals, etc., were also included in the license renewal database.

3) Pressurized Thermal Shock (PTS)

This event deals with operations and inspections of equipment to minimize and monitor pressurized thermal transients to pressurized vessels. The only SSC relied on for pressurized thermal shock is the Reactor Vessel. Note there are no structures that are relied on for this scoping criterion.

4) Anticipated Transients Without a Scram (ATWS)

The Palisades design for accommodating ATWS events is described in Section 7.4 of the Palisades FSAR. The ATWS provides circuitry independent of the reactor protective system (RPS) to shut down the reactor, trip the turbine, and start the turbine driven auxiliary feedwater pump following a transient in which the Reactor Protective System (RPS) does not trip in the normal fashion.

5) Station Blackout (SBO)

Methodologies acceptable to the NRC to achieve compliance with the SBO Rule are provided in NUMARC 87-00 and NRC Regulatory Guide 1.155. Palisades provided responses to the SBO rule in four submittals from 1989 to 1991. The Palisades design satisfies the SBO Rule by providing adequate power to maintain Hot Standby conditions.

In support of Palisades responses to the SBO Rule, a minimum acceptable station blackout (SBO) duration of four hours was determined. Palisades selected coping as the means of maintaining the plant in a safe shutdown condition (hot standby) independent of AC power for the required SBO duration. Selected systems and design characteristics provide the availability, adequacy, and capability to achieve and maintain safe shutdown and to recover from an SBO for a four hour coping duration. Relevant analyses, in conjunction with plant Emergency Operation Procedures and the SBO SER, were used as the basis to determine the systems and system functions required to support the SBO Rule. System boundaries and the components required to support those system functions were identified and included in-scope of license renewal due to SBO.

Systems and components were identified that are required to restore offsite power consistent with the direction given by the NRC in ISG-2. An April 1, 2002 letter from the NRC to NEI (Interim Staff Guidance ISG-2) states the NRC's position “…that the plant system portion of the offsite power system that is used to connect the plant to the offsite power source should be included within the scope of the rule.”

The preferred source of offsite power is via the motor operated disconnect 24F1, fed from the front bus of the switchyard, which has multiple sources of supply from the transmission system to the Safeguards transformer. The scoping boundary for the preferred source of offsite power is via the motor operated disconnect 24F1, Safeguards transformer 1-1, direct buried cables, Safeguards bus, non-segregated bus, medium voltage cables, and isolation breakers. The alternate source of offsite power is via the motor operated disconnect 24R2, which is fed from the rear bus of the 345 KV switchyard, which has multiple sources of supply from the 345 KV transmission system. The scoping boundary for the alternate source of offsite power is via the motor operated disconnect 24R2, high voltage overhead lines and towers, startup transformers, load isolation breakers, and medium voltage cables.

2.1.2.2 Mechanical Discipline-Specific Scoping Methodology

The mechanical discipline was responsible for scoping evaluations for the plant mechanical piping systems. License renewal systems were initially based on the equipment database and Palisades Maintenance Rule system designations and boundaries. System level functions were also initially obtained from the Maintenance Rule database. The Maintenance Rule and License Renewal Rule have slightly differing criteria for system or structure inclusion. Therefore, system functions in-scope for the Maintenance Rule would not necessarily be in-scope for license renewal. In addition, the Maintenance Rule system functions were augmented to include functions to address the license renewal criterion as necessary.

To determine whether a system was in-scope for license renewal, all system level functions were evaluated against the criteria specified in 10 CFR 54.4 (a)(1), (2), and (3). Those that met the criteria were considered in-scope of License Renewal. There are unique clarifications and exceptions to this:

- When all in-scope components are “Active,” the system was considered in-scope of license renewal but had no components requiring AMR.
- Similarly, when the only in-scope portion of the system was comprised of components that were subject to a commodity group evaluation (e.g. fire barriers, equipment supports, cables, etc.), that system or structure could be identified as not being within the scope of license renewal. These instances are clearly documented and the commodity group evaluation was referenced.

Some components within a system were removed from the mechanical system scoping if they were addressed generically in commodity groups. For example, system pipe supports were removed from their respective mechanical systems since they are handled as component support commodities by the civil/structural discipline.

Electrical components contained in the mechanical system were usually left in their respective systems since the majority are active and were screened out later. Some electrical components were also kept in their respective mechanical systems where they served a passive pressure boundary function.

Within most systems, new assets or subcomponents were created in the license renewal database to ensure that all necessary assets/components were accurately described and addressed in the license renewal process. Some of these decisions were based on the knowledge of what would subsequently be needed for screening or aging management. The following examples were used in the mechanical area:

- Instrument manifolds, isolation, test, and vent valves, are not typically shown on P&IDs nor are they identified with individual component identifiers beyond the root valves. For this reason, a tubing and valve sub component was created for in-scope instruments. This represents all tubing, valves, and manifolds that would be associated with that instrument.
- A new asset of 'fastener' was created for each system that utilizes pressure boundary bolting.
- Heat exchangers were divided into subcomponents as necessary to identify all applicable material/environment/intended function combinations (i.e., HX ID tubes, HX ID - tube sheet, HX ID shell, etc.).

- Piping assets were created to identify all applicable materials.

Evaluation Boundaries

Application of all three10CRF54.4 criteria generated a listing of SSC that are considered in-scope for license renewal. Not every component of a system supports the system intended functions. Therefore, some components within an in-scope system are not in-scope for license renewal. A description of the in-scope boundaries was provided along with a listing of components within the boundaries. Where convenient, these boundaries were depicted on drawings with color overlays. The colored portions of each drawing are in-scope, and the black portions are not in-scope. (Note that not all colored portions of the drawings are subject to AMR; i.e., they may be in boundary but screen out as active, periodically replaced, etc.).

In some instances, components were reviewed as part of another interfacing system in order to more accurately portray system functional boundaries, or to streamline the overall license renewal process. In a few cases, all in-scope components for a single system could be reviewed as part of another interfacing system. When components were reassigned to commodity groups or other systems, the system function associated with those components was also reassigned to the new commodity/system, and that function was deleted (if no other components used that function) from the original system. In this way, some systems were de-populated of in-scope functions, and therefore shown to be out of scope. However, most Palisades systems were ultimately determined to be in scope. Table 2.2-1 provides a list of Palisades license renewal systems and indicates whether they are in or out of scope for license renewal.

2.1.2.3 Civil/Structural Discipline-Specific Scoping Methodology

Civil/Structural scoping was performed in a manner to ensure that all plant buildings and yard structures, and their constituent parts, were considered for License Renewal. The scoping was performed using a structures and commodity approach to ensure all C/S components and commodities that serve a license renewal intended function were identified and evaluated for aging management. Grouping of C/S structures, components and commodities varied over the course of the project. They were grouped and evaluated in a comprehensive manner for initial scoping, but were then regrouped to facilitate aging management review and final License Renewal Application reporting to best serve the requirements of each stage of the LR process.

Information sources used in C/S scoping included the FSAR, Q-List, Design Basis Documents (DBDs), Maintenance Rule Documentation, AMMS equipment database, Civil and other drawings, specifications, codes/standards, plant procedures, interviews with plant personnel, walkdowns of plant buildings and yard, as well as NEI 95-10 rev 4 (i.e., Appendix B “Typical Structure, Component and Commodity Groupings and Active/Passive Determinations for the Integrated Plant Assessment”).

The first step in C/S scoping involved identification of all site structures. The Maintenance Rule list of structures was used as a starting point. That list was augmented by review of the information sources discussed above and walkdowns to develop a final list of all plant structures, sub-structures, or groups of structures. The resulting structures were then loaded into the license renewal database for scoping evaluation. Structure functions were then developed and a determination of whether a structure was in-scope of license renewal was made based on: 1) whether the structure was safety related per the CLB; 2) whether its failure could prevent any safety related structure, system or component from performing its intended function; or 3) whether it performed a protection or support intended function for any in-scope system or component.

Then, generic commodity components (i.e., assets) were created based on design function, structure (or yard), material, and environment to aid in the AMR evaluation process. For example, structural steel framing components consisting of columns, brackets, beams, girders, stiffeners, trusses, etc. would be represented by an asset entitled “Building Framing Commodity- Auxiliary Bldg, Carbon Steel, Protected.” The specific structural components to be included in each generic commodity asset were defined. Assets were only created for in-scope intended functions of the structures and commodity groups.

For aging management review, the structural commodity assets were regrouped into five (5) AMR groups to more effectively perform AMR evaluations and NUREG 1801 Volume 2 (GALL) alignments. The five groups developed for AMR evaluations were: Containment Pressure Boundary (GALL Section II), Component Supports (GALL Section IIIB), Structural Concrete Commodities (GALL Section IIIA), Structural Steel Commodities (GALL Section IIIA), and Miscellaneous and Bulk Commodities (GALL Section VII and non-GALL).

Finally, in order to align with the standard license renewal application format, C/S scoping and AMR results were regrouped and are presented by individual structures, the component support commodity group, and the miscellaneous and bulk commodity group.

Evaluation Boundaries

Building evaluation boundaries were established at the building column lines and/or physical barriers. Flexibility exists in determining the exact location of a boundary between adjacent structures and is discussed in detail in the system/boundary descriptions.

2.1.2.4 Electrical and I&C Discipline-Specific Scoping Methodology

The Electrical discipline was responsible for performing scoping evaluations on plant electrical and instrumentation and control (I&C) systems for their applicability to license renewal rule requirements. License renewal system boundaries were initially based on the associated Palisades Equipment Database and Palisades Maintenance Rule system boundaries.

During the Scoping Process, evaluation boundaries were established for each system or commodity group based on the system functions. In some instances, components were reviewed as part of another interfacing system in order to more accurately portray system functional boundaries or to streamline the overall license renewal process. In a few cases, all in-scope components for a single system could be reviewed as part of another interfacing system. Junction boxes, panels, cabinets, and electrical penetration assemblies are identified in the Palisades equipment database, but the individual cables are not. The cables are all listed in a separate data base.

A cable asset was added to each Mechanical and Electrical system in the license renewal database, and then transferred to a cable commodity. The cable commodity was then broken down into commodities to evaluate the Non-EQ electrical cables and connections, inaccessible medium voltage cables and connections, Non-EQ connections in instrument circuits that are sensitive to reduction in conductor insulation resistance, and the Non-EQ electrical and I&C penetration assemblies. Asset commodities were also created for fuse holders, non-segregated phase bus and connectors, high-voltage transmission conductors, high-voltage switchyard bus and connections, and high-voltage insulators.

2.1.3 Screening Methodology

The Screening Process identified the components from the in-scope systems, and commodity groups that would be subject to an aging management review. These components were those that performed a component level intended function (IF) without moving parts or change in configuration or properties, and were not subject to replacement based on a qualified life or specified time period.

The screening process consisted of the following distinctive steps:

- Identification of the components that were subject to aging management review (long lived and passive) for each in-scope system, structure, or commodity.
- Identification of the component level IF(s) for all components subject to aging management review. A component level IF is one that supports the system / structure / commodity level IFs.
- Identification of the applicable references used during the screening process to make the determinations.

Components that are subject to an aging management review are those that perform component level IF(s) without moving parts or without a change in configuration or properties and are not subject to replacement based on a qualified life or specified time period.

2.1.3.1 Active/Passive Determination

As part of the screening process, components that were within the license renewal evaluation boundary, that functioned with moving parts or with a change in configuration or properties (i.e. active components), were identified. Appendix B to NEI 95-10 provided guidance regarding component types that are generally considered to be passive or active.

Most electrical and I&C system components are active per NEI 95-10, Appendix B. Using the license renewal database, active component types were identified and screened out since they do not require aging management review per 10 CFR 54. For “passive, long-lived” electrical components, a commodity group was established for Non-EQ electrical cables and connections, inaccessible medium voltage cables and connections, Non-EQ connections in instrument circuits that are sensitive to reduction in conductor insulation resistance, Non-EQ electrical and I&C penetration assemblies, fuse holders, non-segregated phase bus and connectors, high-voltage transmission conductors, high-voltage switchyard bus and connections, and high-voltage insulators.

Some components were screened uniquely:

- Temperature elements (TE) are typically active components per NEI 95-10. However, since Palisades does not typically identify thermowells as unique components in the Palisades equipment database, a review of Palisades documentation and interviews with system engineers was performed to determine those that are pressure boundary related. The TE components identified as pressure boundary related, or which could not be determined, were identified as passive in order to represent the pressure boundary function (either the thermowell or other fittings if it was a direct immersion TE).

- Solenoid valves (SVs) are also typically active components. However, in some cases, the solenoid valve body will actually need to perform a pressure boundary function. All SVs were reviewed against this criteria, and those that were needed to maintain a pressure boundary were identified as passive.
- All instruments are considered active unless they form an integral part of the pressure-retaining boundary, such as level glasses, flow glasses, and in-line flow switches.
- Fans and dampers are normally considered active components, but, in accordance with ISG-6, fan and damper housings were considered passive and subject to an AMR.

Since Civil / Structural components and commodities (assets) were only created for in-scope structures and commodity groups, and all (with the exception of some crane components) are long-lived and passive, there was very little screening required. With the exception of some crane related components, all civil/structural components that were included in scope of License Renewal received aging management review. Therefore, in general, “in-scope” and “requires aging management review” are synonymous for civil structural components.

2.1.3.2 Identification of Short-lived Components and Consumables

The screening process identified those components that can be treated as short-lived. If a work control document was found to provide for the periodic replacement of the component, or the component was found to have an established qualified life (e.g., for EQ purposes), the component was identified as short-lived and an aging management review is not required for that component.

Consumables are a special class of short-lived items that can include packing, gaskets, component seals, O-rings, oil, grease, component filters, system filters, fire extinguishers, fire hoses, and air packs. Many types of consumables are part of a component such as a valve or a pump and, therefore, have not been identified uniquely during scoping or screening. Items potentially treatable as consumables have been evaluated consistent with the information presented in Table 2.1-3 of NUREG-1800. The results of that evaluation are presented below.

Packing, Gaskets, Component Seals, and O-Rings

Packing, gaskets, component mechanical seals, and O-rings are typically used to provide a leak-proof seal when components are mechanically joined together. These items are commonly found in components such as valves, pumps, heat exchangers, ventilation units/ducts, and piping segments. These types of consumables are considered subcomponents of the identified components and, therefore, are not subject to their own condition or performance monitoring. Therefore, the AMR for the component included an evaluation of the sealing materials in those instances where it could not be demonstrated that one of the following conditions exist:

1) The sealing materials are short-lived because they are replaced on a fixed frequency or have a qualified life established (e.g., for EQ purposes), or
2) The sealing materials are not relied on in the CLB to maintain any of the following:
    - Leakage below established limits
    - System pressure high enough to deliver specified flow rates
    - A pressure envelope for a space

Oil, Grease, and Filters

Oil, grease, and filters (both system and component filters) have been treated as consumables because either:

1) A program for periodic replacement exists, or
2) A monitoring program (e.g., predictive analysis activities, condition monitoring) exists that replaces these consumables, based on established performance criteria, when their condition begins to degrade but before there is a loss of intended function.

Fire Extinguishers, Fire Hoses, and Air Packs

Components such as fire hoses, fire extinguishers, self-contained breathing apparatus (SCBA), and SCBA cylinders are considered to be consumables and are routinely tested or inspected. The Fire Protection Program complies with the applicable NFPA safety standards, which specify performance and condition monitoring programs for these specific components. Fire hoses and fire extinguishers are inspected and hydrostatically tested periodically, and must be replaced if they do not pass the test or inspection. SCBA and SCBA cylinders are inspected and periodically tested and must be replaced if they do not pass the test or inspection. The Fire Protection Program determines the replacement criterion of these components that are routinely checked by tests or inspections to assure operability. Therefore, while these consumables are in the scope of license renewal, they do not require an AMR.

2.1.3.3 **Screening of Stored Equipment**

In response to the NRC letter from Chris Grimes to Doug Walters (NEI) dated February 11, 1999, Subject: Screening of Equipment Kept in Storage, a review has been performed to identify equipment that (1) is maintained in storage, (2) is reserved for installation in the plant in response to a design basis event (DBE), and (3) requires an AMR. In addition to passive components, the review has also considered stored active components that are not routinely inspected, tested, and maintained.

The Appendix R stored equipment is used to restore power to pre-selected plant components and to provide cooling to certain areas after a fire in order to attain cold shutdown. Related components were identified and included in the Fire Protection System for consistency, and given their Appendix R intended functions. Spare motors and portable fan units retained in storage were treated as active components, however, since they are tested and maintained in accordance with the Preventive Maintenance Program. The stored Appendix R cables are included in the Non-EQ electrical cables and connections commodity.

2.1.3.4 **Screening of Thermal Insulation**

Palisades has reviewed the current licensing basis and determined that only a small portion of the thermal insulation is in scope for license renewal. Through AMR, Palisades further determined that insulation is not subject to any aging effects requiring management. The Palisades methods and conclusions are consistent with those precedents established by and for the majority of past applicants.

Palisades operating experience has shown no exposure to aggressive chemicals, such as sulfur dioxide in industrial areas and salt air in marine seashore areas.

There are 2 plant-specific situations where thermal insulation is within the scope of license renewal. They are the insulation at (1) the main feedwater and main steam penetrations that has the environmental control intended function to limit the adjacent concrete temperatures to less than 179F; and (2) the stainless steel mirror insulation on the reactor vessel supplied as part of the seismically qualified NSSS system for their spatial impact on safety related components.

There are no plant locations within Palisades where insulation on piping and components is credited to reduce heat transfer for individual room heat load calculations in support of accident analyses or safe shutdown for regulated events. In particular, (1) the HVAC calculations for the diesel generator rooms did not credit actual heat losses from the diesel generator exhaust, and (2) there are no operating experience of AFW pump tests reporting localized failure of insulation leading to overheating failure of the AFW pump.

Individual insulation systems were designed, selected, and installed for the specific service and conditions. Passive degradation of insulation systems in indoor and outdoor environments is not expected. Gross failure of insulation materials is not credible. Therefore, degradation of insulation systems is not an aging concern, and will not result in degradation/failure of the underlying metal.

2.1.3.5 Identification of Component Intended Functions

Component level IFs were identified and loaded into the license renewal database. A component level IF is a function that is required for the system to perform its system level IF(s). If a component did not have at least one component level IF, the component was not subject to aging management review. A list of component level IFs considered for all disciplines is provided in Table 2.1-1.

Table 2.1-1 Intended Function Definitions

INTENDED FUNCTION	DEFINITION
Component Structural Support	Provide structural support to safety-related components
Containment Isolation	Provide containment isolation
Direct Flow	Provide spray shield or curbs for directing flow (e.g. safety injection flow to containment sump)
Electrical Connections	Provide electrical connections to specified sections of an electrical circuit to deliver system voltage and current
Expansion/Separation	Provide for thermal expansion and / or seismic separation
Filtration	Provide filtration
Fire Barrier	Provide rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant
Flood Protection	Provide flood protection barrier (internal and external flooding event)
Flow Measurement	Orifice for purpose of fluid flow measurement
Flow Restriction	Provide flow restriction (throttle)
Fluid Pressure Boundary	Provide fluid pressure boundary
Gaseous Discharge Path	Provide path for release of filtered and unfiltered gaseous discharge
Heat Sink	Provide heat sink during SBO or design basis accidents.
Heat Transfer	Provide heat transfer
HELB Shielding	Provide shielding against high energy line breaks

Table 2.1-1 Intended Function Definitions

INTENDED FUNCTION	DEFINITION
IA Pressure Boundary	Part of IA Pressure Boundary for AOV
IA Supply Pressure Boundary	Part of IA Supply Pressure Boundary to AOV
Missile Barrier	Provide missile barrier (internally or externally generated)
Pipe Whip Restraint	Provide pipe whip restraint
Pressure Boundary	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered
Pressure Boundary/Fission Product Retention	Provide pressure boundary or fission product retention barrier to protect public health and safety in the event of any postulated design basis events.
Prevent Core Displacement	Prevent excessive core displacement if the core barrel fails
Radiation Shielding	Provide shielding against radiation
Reduce Flow Inequalities	Reduce core inlet flow inequalities
Shelter/Protection	Provide shelter/protection to safety-related components
Shutdown Cooling Water	Provide source of cooling water for plant shutdown.
Spray Pattern	Provide spray pattern at discharge nozzle
Structural Support for Non-Safety Related	Provide structural support to non-safety related components whose failure could prevent satisfactory accomplishment of any of the required safety-related functions
Structural Support for Safety Related	Provide structural and / or functional support to safety-related components

Table 2.1-1 Intended Function Definitions

INTENDED FUNCTION	DEFINITION
Structural Support for Regulated Events	Provide structural and/or functional support to Regulated Events components
Structure Functional Support	Provide structural and / or functional support to safety-related equipment
Water Suppression Support	Provide support of automatic water suppression for the protection of safe shutdown systems

Section 2.1 References

1. Letter of March 10, 2003 from David B. Matthews of the NRC to Alan Nelson of NEI, Subject: Interim Staff Guidance (ISG) - 5 on the Identification and Treatment of Electrical Fuse Holders for License Renewal.
2. Letter of February 11, 1999 from Christopher I. Grimes of the NRC to Doug Walters of NEI, Subject: Request for Additional Information Regarding Generic License Renewal Issue No. 98-0102, Screening of Equipment that is Kept in Storage.
3. Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, NUREG-1800, U.S. Nuclear Regulatory Commission, July 2001.
4. Letter of December 3, 2001 from Christopher I. Grimes of the NRC to Alan Nelson of NEI, Subject: License Renewal Issue: Scoping of Seismic II/I Piping Systems.
5. Letter of April 1, 2002 from David B. Matthews of the NRC to Alan Nelson of NEI, Subject: Staff Guidance on Scoping of Equipment Relied on to Meet the Requirements of the Station Blackout (SBO) Rule (10 CFR 50.63) for License Renewal (10 CFR 54.4(a)(3)).

2.2 Plant Level Scoping Results

The systems, structures, and commodities at Palisades were evaluated to determine whether they were within the scope of license renewal, using the methodology described in Section 2.1. The results are shown in Table 2.2-1.

Table 2.2-1 Plant Level Scoping Results

Description	Within Scope of License Renewal?	Comments
SRP Evaluation Group: Reactor Vessel, Internals, and Reactor Coolant System		
Primary Coolant System (Section 2.3.1.1)	Yes	This system includes Class 1 piping, pressurizer, and coolant pumps
Reactor Vessel (Section 2.3.1.2)	Yes	Includes Control Rod Drive Mechanisms
Reactor Vessel Internals (Section 2.3.1.3)	Yes	Includes passive components not directly attached to the reactor vessel
Replacement Steam Generators (Section 2.3.1.4)	Yes	Includes primary, secondary and internal structural components
SRP Evaluation Group: Engineered Safety Features		
Engineered Safeguards System (Section 2.3.2.1)	Yes	This system includes Containment Spray, Engineered Safety Features Actuation, High Pressure Safety Injection, Low Pressure Safety Injection, Normal Shutdown and Design Basis Accident Sequencers, Safety Injection Tanks, Shutdown Cooling, Safety Injection and Refueling Water Tank, and Containment Sump Suction.
SRP Evaluation Group: Auxiliary Systems		
Chemical Addition System (Section 2.3.3.17)	Yes	Portions of this system are in License Renewal scope for Criterion 2 only.
Chemical and Volume Control System (Section 2.3.3.1)	Yes	This system includes Chemical and Volume Control and Concentrated Boric Acid.

Table 2.2-1 Plant Level Scoping Results

Description	Within Scope of License Renewal?	Comments
Circulating Water System (Section 2.3.3.2)	Yes	Portions of this system are in License Renewal scope for Criterion 2 only.
Component Cooling Water System (Section 2.3.3.3)	Yes	
Compressed Air System (Section 2.3.3.4)	Yes	This system includes Feedwater Purity Air, High Pressure Air, and Instrument and Service Air.
Containment Air Recirculation and Cooling System (Section 2.3.3.5)	Yes	
Domestic Water System (Section 2.3.3.16)	Yes	Portions of this system are in License Renewal scope for Criterion 2 only.
Dry Fuel Storage System	No	Dry Fuel Storage Systems are managed under 10 CFR 72 and do not fall under the requirements of 10 CFR 54.
Emergency Power System (Section 2.3.3.6)	Yes	This system includes Emergency Diesel Generators, Fire Pump Diesels, 125-Volt Vital DC Power, 120-Volt Preferred AC Power, and Emergency Lighting. Passive electrical components for 125-Volt Vital DC Power, 120-Volt Preferred AC Power, and Emergency Lighting are included in the Cables and Terminations Commodity Group.
Fire Protection System (Section 2.3.3.7)	Yes	This system includes the Primary Coolant Pump oil collection equipment. Fire rated assemblies are included in the Miscellaneous and Bulk Structural Commodities Group.
Fuel Handling System	No	System is not in scope for license renewal. Selected components are addressed in civil/structural commodity groups.
Fuel Oil System (Section 2.3.3.8)	Yes	

Table 2.2-1 Plant Level Scoping Results

Description	Within Scope of License Renewal?	Comments
Heating, Ventilation, and Air Conditioning System (Section 2.3.3.9)	Yes	This system includes Containment Purge; Control Room HVAC; Electrical Equipment, Switchgear, and Cable Spreading Room HVAC; Emergency Diesel Generator Room Fans; Engineered Safeguards Room HVAC; Fuel Handling Area Ventilation; and Penetration and Fan Room HVAC.
Miscellaneous Gas System (Section 2.3.3.10)	Yes	This system includes Hydrogen Monitoring and Miscellaneous Gas (e.g., nitrogen).
OSHA Safety Items	No	This “system” exists primarily to categorize maintenance man-hours spent on improving Safety. The only plant equipment within this system is emergency showers and eye wash stations. The system contains no assets that are required to support a 10 CFR 54.4 intended function.
Post Accident Sampling	No	The Post Accident Sampling System was originally installed as a post-TMI item. The requirement for system has subsequently been deleted from Technical Specifications and alternate methods are used to assess post-accident fuel damage. Since all systems that interface with Post Accident Sampling have their own license renewal boundaries, or are not in-scope, there are no system functions that meet 10 CFR 54.4 criteria, and this system is not in license renewal scope.
Radwaste (Section 2.3.3.11)	Yes	This system includes Liquid Radwaste and Solid Radwaste.

Table 2.2-1 Plant Level Scoping Results

Description	Within Scope of License Renewal?	Comments
Security	No	The Security System consists of equipment used by the security forces to fulfill their responsibilities. Major components include security cameras, E-Field detectors, fence, crash barriers, locked doors and hatches, door and hatch alarms, hand scanners, card readers, and the access control computer. The system contains no assets that are required to support a 10 CFR 54.4 intended function.
Service Water (Section 2.3.3.12)	Yes	This system includes Critical Service Water, Non-Critical Service Water, and Ultimate Heat Sink.
Shield Cooling (Section 2.3.3.13)	Yes	Evaluated under Closed Cooling Water
Spent Fuel Pool Cooling (Section 2.3.3.14)	Yes	Evaluated under Closed Cooling Water
Waste Gas (Section 2.3.3.15)	Yes	This system includes Waste Gas and Hydrogen Recombiners.
SRP Evaluation Group: Steam and Power Conversion System		
Condensate and Condenser (Section 2.3.4.1)	Yes	
Demineralized Makeup Water (Section 2.3.4.2)	Yes	
Feedwater (Section 2.3.4.3)	Yes	This system includes Auxiliary Feedwater and Feedwater.
Heater Extraction and Drain (Section 2.3.4.4)	Yes	Some components are in-scope due to criterion a(2)
Main Air Ejection and Gland Seal (Section 2.3.4.5)	Yes	Some components are in-scope due to criterion a(2)

Table 2.2-1 Plant Level Scoping Results

Description	Within Scope of License Renewal?	Comments
Main Steam (Section 2.3.4.6)	Yes	This system includes Main Steam and Steam Generator Blowdown.
Turbine Generator (Section 2.3.4.7)	Yes	This system includes the main turbines, main electrical generator and various supporting subsystems.
SRP Evaluation Group: Containments, Structures and Component Supports		
Auxiliary Building (Section 2.4.1)	Yes	
Component Supports (Section 2.4.2)	Yes	Includes supports for mechanical and electrical components
Containment (Section 2.4.3)	Yes	
Containment Interior Structures (Section 2.4.4)	Yes	
Cooling Towers	No	The Cooling Towers, including the basins, are non-safety related structures. The basins are concrete structures and the cooling towers are primarily made of wood materials. The basins are not used as a backup source of water for safety functions.
Cooling Tower Pump House	No	The Cooling Tower Pump House contains the two pumps which circulate the tube side condenser cooling water up to the cooling tower inlet near the tower top. the The failure of the structure would not result in the release of radioactivity and would not prevent reactor shutdown but may interrupt power generation.
Discharge Structure (Section 2.4.5)	Yes	
Feedwater Purity Building (Section 2.4.6)	Yes	

Table 2.2-1 Plant Level Scoping Results

Description	Within Scope of License Renewal?	Comments
Intake Structure (Section 2.4.7)	Yes	
Miscellaneous and Bulk Structural Commodities (Section 2.4.8)	Yes	Includes elastomers, fire barriers, cranes, and other miscellaneous commodities
Minor Buildings Inside and Outside Protected Area	No	Includes Construction Building, Construction Fabrication shop, Ground Water Remediation Building, Building and Grounds Shop Building, Dry Fuel Storage Building, Training (Outage) Building, Warehouse.
Radwaste Buildings	No	Consists of five radwaste storage buildings separated from the main plant, including South Radwaste Storage Building, Radwaste Storage Pole Building, North Radwaste Storage Building, East Radwaste Storage Building, and Old Steam Generator Storage Building.
Service, Support and Security Buildings	No	Consists of four buildings, including Service Building, Service Building Expansion, Support Building, and Security Building.
Small Electrical, Mechanical, Fire Equipment Shelters and Storage Buildings	No	Consists of various small buildings, sheds, and shelters outside the Protected Area
Switchyard and Yard Structures (Section 2.4.9)	Yes	
Turbine Building (Section 2.4.10)	Yes	
SRP Evaluation Group: Electrical and Instrumentation and Controls		
Cables and Terminations (Section 2.5.2)	Yes	

Table 2.2-1 Plant Level Scoping Results

Description	Within Scope of License Renewal?	Comments
Containment Isolation and Penetration System (Section 2.5.3)	Yes	Electrical system only. No Assets Subject to AMR. Containment isolation valves and piping are included in their respective mechanical systems. Containment structural penetrations are included in Containment structure.
Control Rod Drive System (Section 2.5.4)	Yes	Electrical system only. No Assets Subject to AMR. Pressure boundary components are included with the Reactor Vessel.
Neutron Monitoring (Section 2.5.5)	Yes	Electrical system only. No assets subject to AMR.
Portable Measuring and Test Equipment	No	“System” contains equipment used for test or calibrate activities. This equipment is not associated with permanently installed plant equipment.
Radiation Monitoring (Section 2.5.6)	Yes	Electrical system only. Pressure boundary components are included in their respective mechanical systems.
Reactor Protection (Section 2.5.7)	Yes	All components are active. No assets subject to AMR.
Station Power (Section 2.5.8)	Yes	
Switchyard (Section 2.5.9)	Yes	

2.3 Scoping and Screening Results: Mechanical Systems

2.3.1 Reactor Vessel, Internals, and Reactor Coolant System

The following systems are addressed in this section:

- Primary Coolant System (Section 2.3.1.1)
- Reactor Vessel (Section 2.3.1.2)
- Reactor Vessel Internals (Section 2.3.1.3)
- Replacement Steam Generators (Section 2.3.1.4)

2.3.1.1 Primary Coolant System

System Description

The Primary Coolant System (PCS) is designed to remove heat from the reactor core and internals and transfer it to the secondary (steam generating) system by the controlled circulation of pressurized borated water which serves both as a coolant and a neutron moderator. The PCS serves as a barrier to the release of radioactive material to the Containment building, and is equipped with controls and safety features that assure safe conditions within the system.

The system contains two steam generators to transfer the heat generated in the Primary Coolant System to the secondary system. The steam generators are vertical U-tube heat exchangers. The steam generators operate with the primary coolant in the tube side and the secondary fluid on the shell side.

All mechanical components of the PCS are located within the Containment building. The system includes two identical heat transfer loops connected in parallel to the reactor vessel. Each loop contains one steam generator, two circulating pumps, flow and temperature instrumentation, and connecting piping. A pressurizer is connected to one of the reactor vessel outlet pipes by means of a surge line. The pressurizer is located with its base at a higher elevation than the reactor vessel piping. This eliminates the need for a separate drain on the pressurizer and ensures that it is drained before maintenance operations.

System pressure is maintained by regulation of the water temperature in the pressurizer where steam and water are held in thermal equilibrium. Steam is either formed by the pressurizer heaters or condensed by the pressurizer spray to limit the pressure variations caused by contraction or expansion of the primary coolant.

Overpressure protection is provided by spring-loaded safety valves connected to the top of the pressurizer. Steam discharged from the valves is cooled and condensed by water in a quench tank.

A subsystem of the PCS is the pressurizer pressure and level control system. The Pressurizer Pressure and Level Control is a redundant control system designed to maintain pressurizer level and pressure during heatup, cooldown and load changes. Level is accomplished by controlling the rate of water added to or removed from the PCS. Pressure is controlled by energizing electrical heaters or activating steam space spray water to the pressurizer.

The description above results in some SSCs in this system being considered in-scope in accordance with 10 CFR 54.4(a)(1). Since some SSCs in the Primary Coolant System are non-safety related and their failure could affect the capability of safety related SSCs to perform their safety function, they are considered to be in-scope in accordance with 10 CFR 54.4(a)(2). In addition, some SSCs are considered in-scope due to ATWS, Environmental Qualification, Fire Protection, and Station Blackout in accordance with 10 CFR 54.4(a)(3).

The boundaries of the in-scope portions of the Primary Coolant System are described as follows: 1) The PCS discharge from the reactor vessel safe ends to the steam generators, from the steam generators to the primary coolant pumps, and return to the safe ends of the reactor vessel inlet; 2) the PCS surge line to the pressurizer and through the RVs and PORVs to the quench tank; 3) The primary coolant pump seal flow paths; 4) the PCS lubricating oil system components (seismic only); and 5) the PCS piping that penetrates Containment penetration 40. The PCS also contains non-safety related components spatially oriented that could affect safety related components.

The original system scoping boundaries for license renewal included, where applicable, non-safety related components that could affect safety related components. The boundaries have been expanded in accordance with current NRC guidance for implementation of 10 CFR 54.4(a)(2). The expanded scoping boundary includes:

PCS non-safety related piping and components that are directly attached to safety related piping and components out to the first seismic anchor or equivalent anchor. Where base mounted components are credited as an anchor, they are also included in scope. Pipe supports for the newly added piping and components are also in scope. The new piping components are located in the Auxiliary Building and Containment

PCS non-safety related components containing steam and/or fluids that are located in an area that also contains a safety related component. The components types include piping, fittings, fasteners (to maintain the pressure boundary), and valves. The components are located in the Auxiliary Building and Containment.

The portions of the Primary Coolant System containing components subject to an AMR include the PCS Class 1 piping, valves, and associated fittings; steam generators; pressurizer; and primary coolant pump casings. The NSAS components brought into scope by the 10 CFR 54.4(a)(2) boundary expansion are subject to AMR.

System Function Listing

A comprehensive listing of functions associated with the Class 1 portions of the Primary Coolant System (including Class 1 piping, steam generators, and pressurizer) is provided in the summary below.

System Function: PCS-01	Cri 1	Cri 2	Cri 3				
Circulate subcooled water through the reactor core to remove core heat and reject it to the secondary side via the steam generators.			FP	EQ	PTS	AT	SB
	X						

Comment: The PCS cools the reactor fuel under normal and accident conditions and thus maintains the fuel within the specified acceptable design limits and transfer heat from the reactor core to the steam generators where steam is produced.

System Function: PCS-02	Cri 1	Cri 2	Cri 3				
Serve as a fission product boundary between the reactor core and the containment environment or interfacing systems.			FP	EQ	PTS	AT	SB
	X						

Comment: The PCS serves as a barrier to the release of fission products from the reactor core to the environment.

System Function: PCS-03	Cri 1	Cri 2	Cri 3				
Automatically isolate the PCS sample lines and quench tank spray line containment isolation valves on CHP or CHR.			FP	EQ	PTS	AT	SB
	X						

Comment: This system function automatically isolates the Primary Make-up Water to the Quench Tank and is associated with the valves and pipe required to ensure containment isolation in the PCS system. Components associated with the processing of electrical signals to initiate automatic isolation upon containment high pressure (CHP) and containment high radiation (CHR) are dispositioned under the Containment Isolation and Penetration System. The penetrations are dispositioned within the Containment structure evaluation.

System Function: PCS-04 Maintain PCS pump pressure boundary while minimizing and recycling (via CVC and RWS) PCS leakage.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This system function maintains PCS inventory control, but is not a license renewal IF. PCS-02 is the applicable license renewal IF.

System Function: PCS-05 Provide for the release or removal of steam or gas bubbles in the reactor vessel head or pressurizer.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: This system function is designed to permit the operator to vent the reactor vessel head or the pressurizer steam space from the control room under a variety of post-accident conditions.

System Function: PCS-06 Provide PCS parameter indications and alarms to operators and provide signal inputs to other systems.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: This is a normal operation system function. It provides operation parameters and signal inputs to various trip and actuation functions.

System Function: PCS-07 Provide high pressure lift oil (with or without AC power) to primary coolant pump motors.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This is a normal operation system function that protects the PCP and maintains the PCS heat removal system function.

System Function: PCS-08 Provide anti-reverse rotation function on primary coolant pump motors with backstop oil system on Allis Chalmers motors and locking pawls on Westinghouse motor.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This is a normal operation system function that prevents reverse flow in the PCS loops.

System Function: PCS-09 Provide capture and quenching of rejected inventory from the pressurizer, SDC system, SITs and CVC relief valves and transfers excessive inventory to RWS.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This system function provides collection of excess coolant, and is not a license renewal IF.

System Function: PCS-AT	Cri 1	Cri 2	Cri 3				
The system contains structures and/or components required by the current licensing basis for Anticipated Transients Without Scram.			FP	EQ	PTS	AT	SB
						X	

Comment: The PCS system contains components that are required for Anticipated Transients Without Scram.

System Function: PCS-EQ	Cri 1	Cri 2	Cri 3				
The system contains components required by the current licensing basis for Environmental Qualification.			FP	EQ	PTS	AT	SB
				X			

Comment: The PCS system contains components that are required per 10 CFR 50.49.

System Function: PCS-FP	Cri 1	Cri 2	Cri 3				
The system contains structures and/or components required by the current licensing basis for Fire Protection.			FP	EQ	PTS	AT	SB
			X				

Comment: The PCS system contains components that are required for “Appendix R” safe shutdown. The PCP oil leakage collection subsystem is evaluated for license renewal in the Fire Protection System.

System Function: PCS-NSAS	Cri 1	Cri 2	Cri 3				
This system has components in scope of license renewal in accordance with 10CFR54.4(a)(2) requirements for non-safety affecting safety (NSAS).			FP	EQ	PTS	AT	SB
		X					

Comment: PCS has components in scope for Criterion 2, potential seismic and spatial interaction.

System Function: PCS-SB	Cri 1	Cri 2	Cri 3				
The system contains structures and/or components required by the current licensing basis for Station Blackout (Loss of all AC power).			FP	EQ	PTS	AT	SB
							X

Comment: The PCS system contains components that are required for SBO.

System Function: PZR-01	Cri 1	Cri 2	Cri 3				
Provide automatic and manual level control of the Primary Coolant System.			FP	EQ	PTS	AT	SB

Comment: This system function maintains PCS inventory.

System Function: PZR-02	Cri 1	Cri 2	Cri 3				
Provide automatic, manual and passive pressure control of the Primary Coolant System.			FP	EQ	PTS	AT	SB

Comment: This system function maintains PCS pressure control.

System Function: PZR-03 Provide PCS heat removal via the PORVs on once through cooling.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: The PORVs depressurize the PCS and provide a flow path.

System Function: PZR-04 Provide pressurizer level and pressure indications.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: This system function provides safety related indication per Reg. Guide 1.97.

System Function: PZR-05 Provide pressure boundary between primary coolant system and containment environment.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: This system function provides a barrier to the release of fission products from the PCS to the environment.

FSAR Reference

Additional Primary Coolant System details are provided in Section 4.1 and Section 4.3 of the FSAR.

Scoping Boundary Drawings

The scoping boundary drawings for the Primary Coolant System are listed below:

LR-M-201, Sheet 1
LR-M-201, Sheet 2
LR-M-202, Sheet 1
LR-M-202, Sheet 1B
LR-M-203, Sheet 1
LR-M-203, Sheet 2
LR-M-204, Sheet 1
LR-M-209, Sheet 1
LR-M-210, Sheet 2
LR-M-214, Sheet 4
LR-M-214, Sheet 5
LR-M-219, Sheet 1B
LR-M-219, Sheet 2
LR-M-212, Sheet 4

Components Subject to an AMR

The component groups for this system that require aging management review are addressed in Table 2.3.1-1 along with each Component Group’s intended function(s). The Reactor Vessel and Reactor Vessel Internals are addressed separately and therefore are not identified in Table 2.3.1-1. Only the main loop piping, Steam Generators, Pressurizer, PCPs, and interfacing systems’ piping are addressed in this table.

Table 2.3.1-1 Primary Coolant System

Component Group	Intended Function
Alloy 600 Cladding	Shelter / Protection
Alloy 600 Safe Ends	Fluid Pressure Boundary
Alloy 600 Thermal Sleeves	Shelter / Protection
Bolting and Fasteners	Fluid Pressure Boundary
Carbon Steel Nozzles	Fluid Pressure Boundary
Carbon Steel Pipe (30” and 42”)	Fluid Pressure Boundary
Flow Element (PCP Controlled Bleed)	Fluid Pressure Boundary
Non-CASS Valves in PCS and Connected Systems	Fluid Pressure Boundary
PCS Spray and Drain Nozzles	Fluid Pressure Boundary
PORV Isolation, Quench Tank Spray Manual Valves	Fluid Pressure Boundary
PORV Isolation Valves	Fluid Pressure Boundary
Pressurizer Alloy 600 Instrument Penetrations	Fluid Pressure Boundary
Pressurizer Heater Sleeves	Fluid Pressure Boundary
Pressurizer Heaters	Fluid Pressure Boundary
Pressurizer Integral Support Weld	Fluid Pressure Boundary
Pressurizer Manway and Flange Bolting	Fluid Pressure Boundary
Pressurizer Manway and Flanges	Fluid Pressure Boundary
Pressurizer Quench Tank	Fluid Pressure Boundary

Table 2.3.1-1 Primary Coolant System

Component Group	Intended Function
Pressurizer Quench Tank Shell and Heads	Fluid Pressure Boundary
Pressurizer Spray Head	Spray Pattern
Primary Coolant Pump Casing	Fluid Pressure Boundary
Primary Coolant Sample Heat Exchanger Shell	Fluid Pressure Boundary
Reactor Head Vent	Fluid Pressure Boundary
Reactor Head Vent Orifice	Fluid Pressure Boundary
Sample Point (Quench Tank Liquid, Loop 2 Hot Leg, Pressurizer Surge Line)	Fluid Pressure Boundary
Small Bore Stainless Steel Pipe (PCS and Connected Systems)	Fluid Pressure Boundary
SS Cladding	Shelter/ Protection
Stainless Steel Pipe (PCS and Connected Systems)	Fluid Pressure Boundary
Stainless Steel Safe Ends (Pressurizer and Connected Systems)	Fluid Pressure Boundary
Stainless Steel Thermal Sleeves	Shelter / Protection
Stainless Steel Tubing	Fluid Pressure Boundary
Vessels, Pressure (Pressurizer)	Fluid Pressure Boundary

2.3.1.2 Reactor Vessel

Component Description

The Reactor Vessel (RVG) is the pressure vessel used to contain the core, core supports, and nuclear fuel in a Pressurized Water Reactor. The reactor vessel is an integral part of the PCS, which provides the means of removing heat from the fuel and generating steam in the steam generator. The reactor vessel is operated at a pressure high enough to ensure that the bulk primary coolant remains in a liquid phase. The reactor vessel and top head are designed in accordance with ASME B&PV Code, Section III, Class A, 1965, W65a.

The description above results in some SSCs in this system being considered in-scope in accordance with 10 CFR 54.4(a)(1). In addition, some SSCs are considered in-scope due to Pressurized Thermal Shock in accordance with 10 CFR 54.4(a)(3).

The boundaries of the in-scope portions of the Reactor vessel include the following: shells and lower head; removable head and closure bolts; control rod tube lateral supports; vessel primary coolant nozzles and safe ends; control rod drive nozzles, flanges, pressure housings and fasteners; in-core instrumentation nozzles, tubes, flanges, and fasteners; head vent line; vessel supports (covered in structural area); and attachments (core stabilizer and stop lugs, surveillance capsule holder, flow skirt, and lifting lugs).

The portions of the Reactor Vessel containing components subject to an AMR include nozzles, o-rings, flow skirt, upper shell, intermediate shell, lower shell, safe ends, seal ledge ring, stop lugs, studs, nuts, washers, capsule holders, upper shell flange, and Control Rod Drive Mechanism supports.

System Function Listing

A comprehensive listing of functions associated with the Reactor Vessel is provided in the summary below.

System Function: RVG-01	Cri 1	Cri 2	Cri 3				
Serve as a fission product boundary between the reactor core and the containment building.			FP	EQ	PTS	AT	SB
	X						

Comment: This RVG System Function also includes CRD-04, Maintain PCS Pressure Boundary, for the CRDM housings and seals, and the Incore Instrument Assembly Pressure Boundary. The cladding that protects the carbon steel components from the primary borated water is also included in this system function.

System Function: RVG-02 Provide structural support of the reactor core support barrel.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: The core support barrel which is part of the reactor vessel internals, carries all the weight of the reactor internals and is suspended from the core barrel support ledge. Other supports included in this system function are the reactor head, vessel flange, and primary coolant nozzles and safe ends for CRDM, control rods, incore instrumentation, core stabilizing and stop lugs, under-head tube supports, reactor closure head lifting lugs, and surveillance capsule holders.

System Function: RVG-PTS The system contains structures and/or components required by the current licensing basis for Pressurized Thermal Shock.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
					X		

Comment: 10CFR50.61, Pressurized Thermal Shock, provides design requirements for RVG fracture toughness and the protection against Pressurized Thermal Shock events.

FSAR Reference

Additional Reactor Vessel details are provided in Section 3.3 and Section 4.3 of the FSAR.

Scoping Boundary Drawings

The scoping boundary drawings for the Reactor Vessel are listed below:

LR-M-201, Sheet 1

Subcomponents Subject to an Aging Management Review

The subcomponents of the Reactor Vessel that require aging management review are addressed in Table 2.3.1-2 along with each subcomponent’s intended function(s).

Table 2.3.1-2 Reactor Vessel

Subcomponent	Intended Function
CRDM Seal Pressure Housing	Pressure Boundary/Fission Product Retention
CRDM Upper Pressure Housing & Flange	Pressure Boundary/Fission Product Retention

Table 2.3.1-2 Reactor Vessel

Subcomponent	Intended Function
CRDM/Incore Instrument Bolting	Pressure Boundary/Fission Product Retention
Incore Instrument Closure Flanges	Pressure Boundary/Fission Product Retention
Internal SS Cladding	Pressure Boundary/Fission Product Retention
Reactor Vessel Column Support	Structure Functional Support
Reactor Vessel Bottom Head	Pressure Boundary/Fission Product Retention
Reactor Vessel Closure Head	Component Structural Support Pressure Boundary/Fission Product Retention
Reactor Vessel Closure Head Lifting Lugs	Component Structural Support
Reactor Vessel Core Stabilizer Lugs	Structure Functional Support
Reactor Vessel CRDM Nozzles	Structure Functional Support Pressure Boundary/Fission Product Retention
Reactor Vessel Flow Skirt	Reduce Flow Inequalities
Reactor Vessel Head O-ring Leakage Monitoring	Pressure Boundary/Fission Product Retention
Reactor Vessel Head Vent Nozzle	Pressure Boundary/Fission Product Retention
Reactor Vessel Incore Instrument Nozzles	Pressure Boundary/Fission Product Retention
Reactor Vessel Intermediate Shell	Pressure Boundary/Fission Product Retention

Table 2.3.1-2 Reactor Vessel

Subcomponent	Intended Function
Reactor Vessel Lower Shell	Pressure Boundary/Fission Product Retention
Reactor Vessel Nozzle Safe Ends	Pressure Boundary/Fission Product Retention
Reactor Vessel Primary Coolant Nozzles	Pressure Boundary/Fission Product Retention Structure Functional Support
Reactor Vessel Seal Ledge Ring	Pressure Boundary/Fission Product Retention
Reactor Vessel Stop Lugs	Prevent Core Displacement
Reactor Vessel Studs, Nuts, Washers	Pressure Boundary/Fission Product Retention
Reactor Vessel Surv. Capsule Holder	Structure Functional Support
Reactor Vessel Upper Shell	Pressure Boundary/Fission Product Retention
Reactor Vessel Upper Shell Flange	Pressure Boundary/Fission Product Retention
Under Head CRDM Support	Structure Functional Support

2.3.1.3 Reactor Vessel Internals

Component Description

The Reactor Vessel Internals (RVI) support and orient the reactor core fuel bundles and control rods, absorb the control rod dynamic loads and transmit these and other loads to the reactor vessel flange, provide a passageway for the reactor coolant, and support incore instrumentation.

The reactor vessel internals safely perform their functions during all steady-state conditions and during normal operating transients. The internals safely withstand the forces due to deadweight, handling, system pressure, flow impingement, temperature differential, shock and vibration. All reactor components are considered Class 1 for seismic design.

The description above results in some SSCs in this system being considered in-scope in accordance with 10 CFR 54.4(a)(1).

The boundaries of the in-scope portions of the Reactor Internals include the following: core support barrel (including the lower core support structure and core shroud); the upper guide structure (including the control rod shrouds, and the incore instrumentation guide tubes); and the flow skirt (flow skirt is addressed as part of the reactor vessel).

The portions of the Reactor Vessel Internals containing components subject to an AMR include control rod shroud assemblies, core shroud assembly, core support barrel assembly, incore instrument guide tubes, lower internals assembly, and upper guide structure.

System Function Listing

A comprehensive listing of functions associated with the Reactor Vessel Internals is provided in the summary below.

System Function: RVI-01	Cri 1	Cri 2	Cri 3				
Provide a nuclear chain reaction to generate heat for power production.			FP	EQ	PTS	AT	SB

Comment: The reactor core, together with its control systems and the reactor protection system, is designed to function over it lifetime without exceeding fuel damage limits of excessive fuel temperature, cladding strain and cladding stress during normal operating conditions and anticipated transients. The fuel bundles are designed to maintain their structural integrity under steady-state and transient operating conditions, as well as for normal handling, shipping and refueling loads.

Providing a nuclear chain reaction to generate heat for power production is a normal operating system function that is not in boundary of License Renewal.

System Function: RVI-02 Provide controlled distribution of PCS flow through the core while limiting the amount of core bypass flow.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: The flow skirt is used to reduce inequalities in core inlet flow distributions and to prevent formation of large vortices in the lower plenum. The skirt provides a nearly equalized pressure distribution across the bottom of the core support barrel. The core shroud follows the perimeter of the core and limits the coolant bypass flow. The core barrel directs the reactor coolant flow to the core and after leaving the core directs the flow to the outlet nozzles.

System Function: RVI-03 Provide alignment of the fuel bundles and control rods, structural support of the core, in-core nuclear instrumentation and other internals; and prevent lift out of the fuel bundles during accident conditions.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: The Upper Guide Structure Assembly consists of a flanged grid structure, 45 control rod shrouds, a fuel bundle alignment plate and a ring shim. The upper guide structure aligns and supports the upper end of the fuel bundles, maintains the control rod channel spacing, prevents fuel bundles from being lifted out of position during a severe accident condition and protects the control rods from the effect of coolant cross flow in the upper plenum. It also supports the incore instrumentation guide tubing. The upper guide structure is handled as one unit during installation and refueling.

System Function: RVI-04 Provide a fission product boundary via fuel cladding.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Fuel rods are clad with Zircaloy-4 tubing and sealed by welding end caps to each end. The atmosphere within the rods is pressurized helium. This pressure will assure that the fuel rod cladding will be free-standing under all anticipated reactor operating conditions.

System Function: RVI-05 Provide neutrons to ensure neutron monitors can monitor for criticality at low power levels.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This function is associated with neutron sources. Performance criteria is startup visibility of neutron population, which is a normal operation system function and is not in boundary of License Renewal.

System Function: RVI-06 Provide Reactor Vessel Level Monitoring (RVLMS).	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: The components in boundary of License Renewal that monitor PCS inventory are being evaluated in Primary Coolant System (function PCS-06).

FSAR Reference

Additional Reactor Vessel Internals details are provided in Section 3.2, Section 3.3, Figure 3-1, Figure 3-10, Figure 3-11 and Figure 3-12 of the FSAR.

Scoping Boundary Drawings

The boundaries of Reactor Vessel Internals components in scope of license renewal are shown in FSAR Figure 3-1, Figure 3-10, Figure 3-11, and Figure 3-12.

Subcomponents Subject to an Aging Management Review

The subcomponents of the Reactor Vessel Internals that require aging management review are addressed in Table 2.3.1-3 along with each subcomponent's intended function(s).

Table 2.3.1-3 Reactor Vessel Internals

Subcomponent	Intended Function
IV.B3.2 - CEA Shroud Assemblies: Control Rod Shroud, Shroud Support Lug, Fuel Guide Pin, Fuel Guide Pin Nuts, Shroud Top Support, Control Rod Support Lug, Fuel Plate Cap Screw	Structure Functional Support
IV.B3.4 - Core Shroud Assembly: Anchor Block, Centering Plate, Core Shroud Plate, Anchor Screw & Pin, Centering Screw & Pin, Positioning Screw, Shroud Bolt & Pin,	Structure Functional Support
IV.B3.3 - Core Support Barrel Assembly Core Support Barrel, Core Support Barrel Integral Upper Flange	Structure Functional Support

Table 2.3.1-3 Reactor Vessel Internals

Subcomponent	Intended Function
Incore Instrument Guide Tube (Not Addressed in GALL IV.B3) Instrument Guide Tube, Guide Tube Bracket, Guide Tube Plugs, Guide Tube Plug Screw, Guide Tube Support	Structure Functional Support
IV.B3.5 - Lower Internal Assembly Core Support Plate, Core Support Column, Core Support Barrel Snubber Lug, Core Support Barrel Cap Screws, Fuel Alignment Pins, Core Support Column Support Beams and Tie Rods	Structure Functional Support
Upper Guide Structure - Not in GALL Spacer Shim, Instrument Sleeve	Structure Functional Support
IV.B3.1 - Upper Internal Assembly Fuel Alignment Plate, Fuel Plate Align Lug, Fuel Plate Cap Screw, Fuel Plate Guide Pin, Holddown Ring Plunger, Holddown Ring Strap, Holddown Ring, Brace Grid Beam, Cross Brace Screw, Shroud Grid Ring	Structure Functional Support

2.3.1.4 Replacement Steam Generators

Component Description

The two (2) replacement steam generators are of the vertical pressurized water type wherein the heating surface and steam separation equipment are in the same vessel. The steam generators have been designed and fabricated as Class 1 vessels as defined in ASME Boiler and Pressure Vessel Code, Section

III, 1977 Edition. Materials of construction are as specified in ASME Code, Section III.

Each steam generator is a vertical U-tube heat exchanger which operates with the primary coolant on the tube side and secondary coolant on the shell side. A vertical divider plate separates the inlet and outlet plenums of the primary head. Secondary system feedwater enters the steam generator through the feed ring, mixes with the recirculating water from the moisture separators, and flows down the annulus between the tube bundle wrapper plate and the steam generator shell. Here it turns, flows inward around the base of the tube bundle and then upward around the tubes. As it flows up past the tubes, it is heated to saturation temperature and boils. The wet steam passes through sets of moisture separators and steam driers in the upper portion of the steam generator and leaves as dry steam through the outlet flow restricting nozzle at the top. Manways and handholes are provided for access to the steam generator internals for inspection or repairs.

The primary side steam generator boundaries are: the lower hemispherical head of the vessel, the tubesheet, and the 8,219 tubes. The primary head divider plates divide the head into two (2) separate plenums. Reactor coolant enters the inlet plenum through one (1) inlet nozzle, flows through the U-tubes to the outlet plenum, and exits through two (2) outlet nozzles.

The secondary side steam generator boundaries are: the lower shell, conical transition section, upper shell, top head, outside the tubes, and the secondary side of the tubesheet. Connections are provided in the shells and head for feedwater, steam outlet, emergency feedwater, pressure tap, bottom blowdown, sampling, recirculation, and water level instrumentation connections. Nozzle safe-ends are provided on the emergency feedwater and feedwater nozzles. Thermal sleeves are provided in the feedwater nozzles and the recirculating nozzle. The upper shell is provided with two (2) 18-inch nominal I.D. gasketed manways. The manway cover seats externally and is held in place by sixteen (16) studs and nuts. The lower shell is provided with four (4) 6-inch nominal I.D. gasketed handholes for inspection of the tube bundle. The handhole cover seats externally and is held in place by ten (10) studs and nuts.

System Function Listing

A comprehensive listing of functions associated with the Replacement Steam Generators is provided in the summary below.

System Function: RSG-01 Provide a Primary Coolant System pressure boundary	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components within the Class 1 Piping/Components System perform this primary design system function by containing the coolant for heat transfer and serving as a closed pressure boundary that limits leakage to the Containment Building Structure and interconnecting systems.The replacement steam generators serve as a fission product boundary between the reactor core and the containment environment and interfacing systems.

System Function: RSG-02 Provide a Main Steam System and Feedwater System pressure boundary	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: The replacement steam generators serve as a fission product boundary between the reactor core and the containment environment and interfacing systems.

System Function: RSG-03 Transfer heat from the PCS to the secondary systems	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Heat is transferred from the reactor core to the steam generators where steam is produced.

System Function: RSG-04 Provide a closed system in-side containment pressure boundary, or fission product retention barrier to protect public health and safety in the event of a postulated accident	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: The RSG serves as a barrier to the release of fission products from the reactor core to the environment.

System Function: RSG-05 Provide a throttling function to limit steam flow in the event of a postulated steam line break	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Steam outlet nozzles incorporate an integral flow restriction feature which will serve to limit the steam generator blowdown during a main steam line break accident.

Closure of MSIVs isolates the affected steam generator from the intact steam generator and minimizes radiological releases.

FSAR Reference

Additional Replacement Steam Generators details are provided in Section 4.3 and Section 4.4 of the FSAR.

Scoping Boundary Drawings

The scoping boundary drawings for the Replacement Steam Generators are listed below:

LR-M-201, Sheet 1	LR-M-207, Sheet 2
LR-M-207, Sheet 1	LR-M-218, Sheet 2

Subcomponents Subject to an Aging Management Review

The subcomponents of the Replacement Steam Generators that require aging management review are addressed in Table 2.3.1-4 along with each subcomponent's intended function(s).

Table 2.3.1-4 Replacement Steam Generators

Subcomponent	Intended Function
Alloy 690 Tube Plugs	Fluid Pressure Boundary
Handhole Cover	Fluid Pressure Boundary
Tube Bundle Support Assembly	Structural Support for Safety Related
Tube Bundle Wrapper	Direct Flow
Lower Head	Fluid Pressure Boundary
Primary Manway Cover	Fluid Pressure Boundary
Manway Cover Diaphragm	Fluid Pressure Boundary
Nozzle Safe Ends	Fluid Pressure Boundary
Primary Divider Plate	Fluid Pressure Boundary
Primary Inlet and Outlet Nozzles	Fluid Pressure Boundary
Feedwater Inlet Nozzles and Thermal Sleeves	Fluid Pressure Boundary

Table 2.3.1-4 Replacement Steam Generators

Subcomponent	Intended Function
Steam Outlet Nozzle and Flow Limiter, Blowdown Nozzle	Fluid Pressure Boundary
Secondary Side Inspection Port Cover	Fluid Pressure Boundary
Wide and Narrow Range Water Level Nozzles, Sampling and Instrument Nozzles	Fluid Pressure Boundary
Shells (Lower, Upper, Transition)	Fluid Pressure Boundary
Tubesheet	Fluid Pressure Boundary
Upper Head	Fluid Pressure Boundary
U-Tubes	Fluid Pressure Boundary’ Heat Transfer
Fasteners	Fluid Pressure Boundary

2.3.2 Engineered Safety Features

The following systems are addressed in this section:

- Engineered Safeguards System (Section 2.3.2.1)

2.3.2.1 Engineered Safeguards System

System Description

The Engineered Safeguards System (ESS) is a two train independent and diverse system designed to identify inadequate core cooling, and then start the necessary pumps and open the needed valves to establish adequate core cooling conditions. The system is subdivided functionally into seven mechanical subsystems: 1) High Pressure Safety Injection (HPSI), 2) Low Pressure Safety Injection (LPSI), 3) Containment Spray (CSS), 4) Safety Injection Tanks (SIT), 5) Safety Injection and Refueling Water (SIRW) Tank and Containment Sump Suction, 6) Shutdown Cooling (SDC), 7) Reactor Cavity Flood (RCF), and two electrical subsystems: 1) Engineered Safety Features (ESF) Actuation, and 2) Normal shutdown (NSD) and Design Basis Accident (DBA) Sequencers. Except for Reactor Cavity Flood, the mechanical subsystems use most of the same system components for the various subsystem functions, and those components are hydraulically interconnected. Therefore, ESS is most accurately presented as a single system, and the groups of components that provide each major function are characterized as subsystems for license renewal purposes.

The ESF Actuation subsystem (electrical) consists of two independent and isolated circuits that initiate operation of redundant engineered safeguards equipment. These control circuits monitor whether offsite and/or emergency power is available and select load groups in accordance with the available power supply. The sensors are arranged in a two out of four matrix, which, in turn, actuates two safety injection control circuits. Within each control circuit, relays are provided to initiate redundant devices so that individual relay failure will not cause a complete circuit failure. Each circuit is supplied by a separate AC source.

The Normal Shutdown and Design Basis Accident Sequencer (electrical) is designed to sequentially load the safe shutdown equipment on to the emergency buses and diesel generators. Sequencing of loads ensures that the equipment is energized when needed while preventing excessive step loads from being imposed on the diesel generator which could result in loss of the diesel generator. This subsystem is evaluated as part of the electrical Containment Isolation System and is listed here only for completeness.

The High Pressure Safety Injection subsystem is designed to supply high pressure cooling water to the Primary Coolant System (PCS) under accident conditions. The major components of this system are two High Pressure Safety Injection pumps and eight high-pressure injection valves.

The Low Pressure Safety Injection subsystem is designed to supply low pressure cooling water to the PCS under accident conditions. The major components of this system are two Low Pressure Safety Injection pumps, four Safety Injection Tanks, the Safety Injection and Refueling Water (SIRW) Tank, and multiple control valves.

The Containment Spray subsystem is designed to limit post-accident containment pressure and remove heat from the containment atmosphere under accident conditions. The major components of this system are three Containment Spray pumps, two sets of full capacity (100%) spray headers and nozzles, and two full capacity (100%) pump suction lines from both the SIRW Tank and the containment sump.

The four Safety Injection Tanks are designed to flood the core with borated water following a depressurization of the PCS.

The SIRW And Containment Sump Suction subsystem delivers water to the Engineered Safeguards System pumps through one of four paths. It provides the suction source for High Pressure Injection, Low Pressure Injection, Containment Spray subsystems, and Chemical and Volume Control System.

The Shutdown Cooling subsystem is designed to transfer heat from the PCS to the Component Cooling System after reactor shutdown and after an accident, and to maintain a suitable temperature for refueling and maintenance during plant shutdowns.

The Reactor Cavity Flood subsystem consists of a network of floor drain piping designed to 1) collect normal expected floor drains and transport these drains to the containment sump for subsequent disposal outside the containment, and 2) collect a portion of the containment spray water during accident conditions and transport this water into the cavity (annulus) between the biological shield and reactor vessel for flooding and cooling the outside of the reactor vessel bottom head. Neither function is safety related or credited in the FSAR accident analyses.

The description above results in some SSCs in this system being considered in-scope in accordance with 10 CFR 54.4(a)(1). Since some SSCs in the Engineering Safeguards System are non-safety related and their failure could affect the capability of safety related SSCs to perform their safety function, they are considered to be in-scope in accordance with 10 CFR 54.4(a)(2). In addition, some SSCs are considered in-scope due to Environmental Qualification and Fire Protection in accordance with 10 CFR 54.4(a)(3).

The boundaries of the in-scope portions of the Engineering Safeguards System are described as follows: 1) the piping and valves from the SIRW Tank through the SIRW Tank Recirculation pump, the SIRW heat exchanger (tube side only) and return to the SIRW, 2) the piping from the SIRW to manual valve MV-ES3263, 3) the piping, valves and components from the SIRW Tank through Containment Spray pumps, the LPSI pumps, the HPSI pumps, through the shutdown heat exchangers, and to the LPSI flow path to the PCS loop nozzles, the Containment spray headers, and the recirculation/spray recirculation headers back to the SIRW Tank, 4) the piping, valves and components from the safety injection tanks, including the tanks to the LPSI injection headers, 5) the nitrogen supply piping, valves and components from the SIRW Tank to the Containment sump, 6) the piping, valves and components from the Containment sump to the ESS pumps, 7) the piping and valves from the ESS pumps and SIRW Tank to the process sampling system, 8) the Spent Fuel Pool cooling pumps recirculation piping located downstream of MV-SFP126 to the SIRW and 10) the Reactor Cavity Flood filter Canisters.

The original system scoping boundaries for license renewal included, where applicable, non-safety related components that could affect safety related components. The boundaries have been expanded in accordance with current NRC guidance for implementation of 10 CFR 54.4(a)(2). The expanded scoping boundary includes:

ESS non-safety related piping and components that are directly attached to safety related piping and components out to the first seismic anchor or equivalent anchor. Where base mounted components are credited as an anchor, they are also included in scope. Pipe supports for the newly added piping and components are also in scope. The new piping components are located in the Auxiliary Building.

ESS non-safety related components containing steam and/or fluids that are located in an area that also contains a safety related component. The components types include fasteners (to maintain the pressure boundary), piping, fittings and valves. The components are located in the Auxiliary Building.

The portions of the Engineered Safeguards System containing components subject to an AMR include accumulators, heat exchangers, fasteners, baskets, pipe and fittings, pumps, and valves. The NSAS components brought into scope by the 10 CFR 54.4(a)(2) boundary expansion are subject to AMR.

System Function Listing

A comprehensive listing of functions associated with the Engineered Safeguards System is provided in the summary below.

System Function: CSS-01	Cri 1	Cri 2	Cri 3				
Limit post-accident containment pressure by spray-cooling the containment atmosphere following a LOCA or MSLB.			FP	EQ	PTS	AT	SB
	X						

Comment: Credit is taken for containment spray in the LOCA and MSLB accident analyses.

System Function: CSS-02	Cri 1	Cri 2	Cri 3				
Provide cooled suction water supply to the HPSI pumps following a Recirculation Actuation Signal.			FP	EQ	PTS	AT	SB
	X						

Comment: Credited in accident analyses as part of emergency core cooling.

System Function: CSS-03	Cri 1	Cri 2	Cri 3				
Isolate the containment spray system containment isolation valves to ensure containment integrity.			FP	EQ	PTS	AT	SB
	X						

Comment: Containment spray containment isolation valves are closed during normal plant operation and are automatically opened on containment high pressure (CHP). The isolation maintains containment integrity during normal plant operation.

This function is associated with containment spray valves and piping required to ensure containment isolation. Components associated with the processing of electrical signals to initiate automatic isolation upon CHP and Containment high radiation (CHR) are included in the Containment Isolation System (electrical). The Containment penetrations are included in the Containment Structure evaluation.

System Function: CSS-04	Cri 1	Cri 2	Cri 3				
Provide an alternate source of shutdown cooling.			FP	EQ	PTS	AT	SB

Comment: This function is not in-scope of license renewal because it does not meet any of the three license renewal Criteria. This function provides the operator with the flexibility of using the containment spray pumps instead of the LPSI pumps for shutdown cooling, if desired. For this function to be required, multiple equipment failures would have to occur, which is beyond the required single active failure coincident with a loss of offsite power.

System Function: ESF-01	Cri 1	Cri 2	Cri 3				
Initiate SIS on pressurizer low-low pressure or CHP and automatically initiate the necessary engineered safeguards equipment with or without offsite power available.			FP	EQ	PTS	AT	SB
	X						

Comment: Credit is taken for ESF actuation in the LOCA accident analyses.

System Function: ESF-02	Cri 1	Cri 2	Cri 3				
Allow blocking of Safety Injection during normal shutdown conditions to prevent spurious actuation.			FP	EQ	PTS	AT	SB

Comment: This function is not in-scope of license renewal because it does not meet any of the three license renewal Criteria. Blocking of Safety Injection occurs during normal plant shutdowns. It is not a post-accident function.

System Function: ESF-03	Cri 1	Cri 2	Cri 3				
Initiate CHP signal on high containment pressure which will automatically initiate Containment spray, Containment isolation, Safety Injection, Main Steam Isolation, and Control Room HVAC emergency mode.			FP	EQ	PTS	AT	SB
	X						

Comment: Credit is taken for ESF actuation in the LOCA accident analyses.

System Function: ESF-04	Cri 1	Cri 2	Cri 3				
Initiate CHR on containment area alarms or refueling monitor alarm which will automatically initiate Containment isolation, prevent auto start of ESS rooms sump pumps, and initiate Control Room HVAC emergency mode.			FP	EQ	PTS	AT	SB
	X						

Comment: These actions mitigate the consequences of the accident.

System Function: ESF-05	Cri 1	Cri 2	Cri 3				
Initiate Recirculation Actuation Signal (RAS) on low SIRW tank level which automatically transfers suction of HPSI and containment spray pumps from SIRW tank to containment sump, stops LPSI pumps and provides Component Cooling and maximum Service Water to SDC Heat Exchanger.			FP	EQ	PTS	AT	SB
	X						

Comment: Credit is taken for RAS in the LOCA accident analyses.

System Function: ESF-06	Cri 1	Cri 2	Cri 3				
Initiate a low steam generator pressure signal with initiation logic to initiate Main Steam Isolation Signal (MSIS).			FP	EQ	PTS	AT	SB
	X						

Comment: Credited in main steam line break (MSLB) accident analyses. MSIS will isolate the steam generators in the event of a steam generator tube failure following a MSLB. It will also prevent release to the containment of the contents of the secondary sides of both steam generators in the event of a MSLB inside containment.

System Function: ESS-EQ	Cri 1	Cri 2	Cri 3				
The system contains components required by the current licensing basis for Environmental Qualification.			FP	EQ	PTS	AT	SB
				X			

Comment: The ESS system contains components that are required per 10 CFR 50.49

System Function: ESS-FP The system contains structures and/or components required by the current licensing basis for Fire Protection.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: The ESS system contains components that are required for “Appendix R” safe shutdown.

System Function: ESS-NSAS This system has components in scope of license renewal in accordance with 10CFR54.4(a)(2) requirements for non-safety affecting safety (NSAS).	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: In-scope of License Renewal to protect safety related components from spray and flooding.

System Function: HPI-01 Automatically provide borated water at high pressure to the PCS following a Safety Injection actuation.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Credited in accident analyses as part of emergency core cooling.

System Function: HPI-02 Isolate the HPSI system to containment to ensure containment integrity.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: HPI containment isolation valves are closed during normal plant operation and are automatically opened on Safety Injection actuation. The isolation maintains reactor coolant pressure boundary integrity as well as containment integrity during normal plant operation.

This function is associated with high pressure safety injection valves and piping required to ensure containment isolation. Components associated with the processing of electrical signals to initiate automatic isolation upon CHP and CHR are included in the electrical Containment Isolation System. The penetrations are included in the Containment Structure evaluation.

System Function: HPI-03 Provide simultaneous hot and cold leg injection to the PCS following a LOCA to ensure core flushing flow is maintained and excessive boron precipitation and core flow blockage is prevented.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Credited in accident analyses as part of emergency core cooling.

System Function: HPI-04 Provide an alternate borated water flow path during shutdown conditions.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This function is not in-scope of license renewal because it does not meet any of the three license renewal Criteria. The function provides for the use of HPSI pumps as emergency makeup for shutdown cooling if both LPSI and CSS pumps are unavailable. For this to occur, multiple equipment failures would have to occur, which is beyond the required single active failure coincident with a loss of offsite power. It is noted that the HPSI system is normally made inoperable during shutdown conditions due to Low Temperature Overpressure Protection concerns.

System Function: LPI-01 Automatically provide large volumes of borated water at low pressures to the PCS following a Safety Injection actuation.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Credited in accident analyses as part of emergency core cooling.

System Function: SDC-01 Provide core decay heat removal and circulation of the PCS during cooldown and shutdown conditions, and provide heat removal capability during the recirculation mode of a LOCA, via heat rejection to Component Cooling.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: This function applies to the use of the SDC Heat Exchangers with ESF pumps. The SDC Heat Exchangers are used during plant shutdowns in conjunction with the LPSI pumps, piping and valves to remove core decay heat. For the design basis LOCA, the SDC Heat Exchangers are used in conjunction with the containment spray pumps, piping and valves to remove heat during the recirculation phase of the accident.

System Function: SDC-02 Provide two independent means of reactor vessel level and temperature indication while in mid loop operation.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This function is not in-scope of license renewal because it does not meet any of the three license renewal Criteria. It applies to normal plant shutdowns (PCS level lowered to mid-loops). It is not a post-accident function.

System Function: SDC-03 Provide alternate cooling to the spent fuel pool.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This function is not in-scope of license renewal because it does not meet any of the three license renewal Criteria. It provides operators with the flexibility of using the SDC Heat Exchangers to cool the Spent Fuel Pool, if required.

<table>
<tr><td rowspan="3">System Function: SDC-04
Isolate the shutdown cooling system lines from containment to ensure containment isolation.</td><td rowspan="2">Cri 1</td><td rowspan="2">Cri 2</td><td colspan="5">Cri 3</td></tr>
<tr><td>FP</td><td>EQ</td><td>PTS</td><td>AT</td><td>SB</td></tr>
<tr><td>X</td><td></td><td></td><td></td><td></td><td></td><td></td></tr>
</table>

Comment: SDC/LPSI containment isolation valves are normally closed during normal plant operation and are automatically opened on Safety Injection actuation, or manually opened during plant shutdowns. The isolation maintains reactor coolant pressure boundary integrity as well as containment integrity.

This function is associated with SDC/LPSI valves and piping required to ensure containment isolation. Components associated with the processing of electrical signals to initiate automatic isolation upon CHP and CHR are included in the electrical CIS System. The penetrations are included in the Containment Structure evaluation.

<table>
<tr><td rowspan="3">System Function: SIT-01
Passively discharge borated water into the PCS at low pressures during accident conditions.</td><td rowspan="2">Cri 1</td><td rowspan="2">Cri 2</td><td colspan="5">Cri 3</td></tr>
<tr><td>FP</td><td>EQ</td><td>PTS</td><td>AT</td><td>SB</td></tr>
<tr><td>X</td><td></td><td></td><td></td><td></td><td></td><td></td></tr>
</table>

Comment: Credited in accident analyses as part of emergency core cooling.

<table>
<tr><td rowspan="3">System Function: SIT-02
Isolate the safety injection tank drain lines from containment to ensure containment isolation.</td><td rowspan="2">Cri 1</td><td rowspan="2">Cri 2</td><td colspan="5">Cri 3</td></tr>
<tr><td>FP</td><td>EQ</td><td>PTS</td><td>AT</td><td>SB</td></tr>
<tr><td>X</td><td></td><td></td><td></td><td></td><td></td><td></td></tr>
</table>

Comment: SIT drain line containment isolation valves are locked closed during normal plant operation. Isolation maintains containment integrity during normal plant operation.

This function is associated with safety injection tank valves and piping required to ensure containment isolation. Components associated with the processing of electrical signals to initiate automatic isolation upon CHP and CHR are included in the electrical CIS System. The penetrations are included in the Containment Structure.

<table>
<tr><td rowspan="3">System Function: SIT-03
Isolate or vent the safety injection tanks to prevent nitrogen cover gas from being discharged into the PCS.</td><td rowspan="2">Cri 1</td><td rowspan="2">Cri 2</td><td colspan="5">Cri 3</td></tr>
<tr><td>FP</td><td>EQ</td><td>PTS</td><td>AT</td><td>SB</td></tr>
<tr><td></td><td></td><td></td><td></td><td></td><td></td><td></td></tr>
</table>

Comment: This function is not in-scope of license renewal because it does not meet any of the three license renewal Criteria. The SIT MOVs are closed during normal plant shutdowns. During normal operation, the SIT MOVs are open and power is removed from motor operators to prevent inadvertent closure. SIT vent lines are normally isolated. SIT level alarms will alert the operator to fill the tank with borated water if water level drops. This is not a post-accident function.

System Function: SSS-01 Maintain suction source of water to all engineered safeguards pumps via the SIRW tank or the containment sump.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Credited in accident analyses as part of emergency core cooling.

System Function: SSS-02 Control pH in the containment sump to retain iodine in solution via passive addition of trisodium phosphate.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: This is a safety related function. The trisodium phosphate (TSP) baskets in containment are credited in the accident analysis. The LOCA radiological consequences analysis takes credit for iodine retention in the sump solution.

System Function: SSS-03 Provide SIRW tank water to the suction of the charging pumps.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This function is not in-scope of license renewal because it does not meet any of the three license renewal Criteria. The charging pumps are used to test certain HPSI valves during cold shutdown.

System Function: SSS-04 Maintain minimum water temperature in SIRW tank.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: The lower limit of 40 degrees F on SIRW tank water temperature is assumed in the accident analysis and temperatures below this limit could have an adverse effect on the accident analysis. During normal operation, the SIRW tank temperature is maintained, per Technical specification 3.5.4, greater than 40 degrees F.

The tank is equipped with a heating steam heat exchanger and pump to maintain water temperature above this limit during normal operation. Since the heating steam is not needed to mitigate a DBE, only the tube side of the SIRW tank heat exchanger is in boundary of license renewal. Therefore, this system function is not in boundary of license renewal.

System Function: RCF-01 Collect a portion of containment spray water and transport the water into the cavity between the biological shield and reactor vessel for flooding and cooling the outside of the reactor vessel bottom head following an accident	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: RCF benefits post-LOCA recovery but is not credited in accident analyses. This function is not in scope of license renewal because it does not meet any of the three criteria. The RCF subsystem is not safety-related per FSAR 6.8. Failure of RCF components would not prevent the satisfactory accomplishment of Criterion 1 functions.

System Function: RCF-02 Collect normal expected floor drainage and transport this drainage to the containment sump for disposal outside the containment.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This function is not in-scope of license renewal because it does not meet any of the three license renewal Criteria. The RCF subsystem is non-safety-related per FSAR 6.8. Failure of RCF components would not prevent the satisfactory accomplishment of Criterion 1 functions.

System Function: RCF-03 The RCF system contains components that are safety related for seismic considerations.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment:

System Function: RCF-NSAS Filter canisters prevent core debris from entering the sump and ESF system.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This function is not in scope of license renewal because the filter canisters slow the release of core debris to the Containment sump after a core meltdown. This is not a safety function.

FSAR Reference

Additional Engineering Safeguards System details are provided in Section 6.1, Section 6.2, Section 6.4, Section 6.8, Section 7.3, Section 8.4, Section 10.2, Section 14.14, Section 14.17, Section14.18, and Section 14.22, of the FSAR.

Scoping Boundary Drawings

The scoping boundary drawings for the Engineering Safeguards System are listed below:

LR-M-201, Sheet 1
LR-M-203, Sheet 1
LR-M-203, Sheet 2
LR-M-204, Sheet 1
LR-M-204, Sheet 1A
LR-M-204, Sheet 1B
LR-M-209 Sheet 1
LR-M-209 Sheet 2
LR-M-212 Sheet 4
LR-M-218 Sheet 2
LR-M-219, Sheet 1B
LR-M-221, Sheet 2
LR-M-225 Sheet 1
LR-M-225 Sheet 1A

Components Subject to an AMR

The component groups for this system that require aging management review are addressed in Table 2.3.2-1 along with each Component Group's intended function(s).

Table 2.3.2-1 Engineered Safeguards System

Component Group	Intended Function
SIRW Tank	Fluid Pressure Boundary
Safety Injection Tank	Fluid Pressure Boundary
SDC HX Shell	Fluid Pressure Boundary
SIRWT HX Shell	Fluid Pressure Boundary
SDC, SIRWT HX Shell	Fluid Pressure Boundary
SIRWT HX Shell	Fluid Pressure Boundary
SDC HX Shell	Fluid Pressure Boundary
SDC HX Channel Head	Fluid Pressure Boundary
SDC HX Tube Sheet shell side	Fluid Pressure Boundary
SDC HX Channel Head shell side	Fluid Pressure Boundary
Cont. Spray Pump HX shell, LPSI Pump HX shell	Fluid Pressure Boundary
SIRWT HX Tubes	Fluid Pressure Boundary
Cont. Spray, LPSI Pump coils	Fluid Pressure Boundary
PCP Seal Cooler Coils, Cont. Spray Pump coils, LPSI pump Coils, SDC HX Tubes	Fluid Pressure Boundary Heat Transfer
SDC HX Tubes	Fluid Pressure Boundary Heat Transfer
PCP Seal Cooler Coils, SIRWT HX Tubes	Fluid Pressure Boundary Heat Transfer
PCP Seal Cooler Coils	Fluid Pressure Boundary

Table 2.3.2-1 Engineered Safeguards System

Component Group	Intended Function
SIRWT HX Tubes	Fluid Pressure Boundary Heat Transfer
Cont. Spray Pump Coils, SDC HX Tubes	Fluid Pressure Boundary Heat Transfer
SIRWT HX Tubes, Cont. Spray Pump coils, LPSI Pump coils, SDC HX Tubes	Fluid Pressure Boundary
Fasteners	Fluid Pressure Boundary
Containment Spray System Fasteners	Fluid Pressure Boundary
Containment Spray Pump Bolting	Fluid Pressure Boundary
Containment Spray System Valves Bolting	Fluid Pressure Boundary
Containment Spray System Valves Header Bolting	Fluid Pressure Boundary
HPSI, LPSI Pumps Bolting	Fluid Pressure Boundary
SIRWT HX Bolting	Fluid Pressure Boundary
SIRWT Bolting	Fluid Pressure Boundary
HPSI Check Valves, SCDC from PCS MOVs	Fluid Pressure Boundary
HPSI Check Valves	Fluid Pressure Boundary
Hot Leg Injection Check Valves	Fluid Pressure Boundary
HPSI Check Valves Loops 1A, 2A	Fluid Pressure Boundary

2.3.3 Auxiliary Systems

The following systems are addressed in this section:

- Chemical Addition System (Section 2.3.3.17)
- Chemical Volume and Control System (Section 2.3.3.1)
- Circulating Water System (Section 2.3.3.2)
- Component Cooling Water System (Section 2.3.3.3)
- Compressed Air System (Section 2.3.3.4)
- Containment Air Recirculation and Cooling System (Section 2.3.3.5)
- Domestic Water System (Section 2.3.3.16)
- Emergency Power System (Section 2.3.3.6)
- Fire Protection System (Section 2.3.3.7)
- Fuel Oil System (Section 2.3.3.8)
- Heating, Ventilation, and Air Conditioning System (Section 2.3.3.9)
- Miscellaneous Gas System (Section 2.3.3.10)
- Radwaste System (Section 2.3.3.11)
- Service Water System (Section 2.3.3.12)
- Shield Cooling System (Section 2.3.3.13)
- Spent Fuel Pool Cooling System (Section 2.3.3.14)
- Waste Gas System (Section 2.3.3.15)

2.3.3.1 Chemical and Volume Control System

System Description

The Chemical and Volume Control (CVC) System design basis is to: maintain the required volume of water in the Primary Coolant System (PCS) over the range of full to zero reactor power, maintain the chemistry and purity of the primary coolant, maintain the desired boric acid concentration in the PCS, and pressure test the PCS.

Letdown coolant from the cold leg of the PCS passes through the tube side of the regenerative heat exchanger and is partially cooled. The cooled fluid is then partially depressurized as it passes through the letdown stop valves and orifices. The temperature and pressure of the letdown coolant are finally reduced to the operating requirements of the purification system by the letdown heat exchanger and back pressure valve, respectively. The coolant then passes through an ion exchanger and a filter and is sprayed into the volume control tank and pumped back to the PCS by way of the shell side of the regenerative heat exchanger. The regenerative heat exchanger transfers heat from the letdown coolant to the charging coolant before the charging coolant is returned to the PCS.

The boric acid concentration and chemistry of the primary coolant are maintained by the CVC system. A concentrated boric acid solution is prepared in a batching tank and is stored in two concentrated boric acid storage tanks. Two pumps are provided to transfer concentrated boric acid to a blender where the boric acid is mixed with primary makeup water in a predetermined ratio. The solution is added to the PCS by the charging pumps.

Chemicals are introduced to the PCS by means of a metering pump which pumps the chemical solution from a chemical addition tank and introduces it to the charging pumps suction header.

Depleted zinc ions are introduced to the PCS via the zinc addition system for reduction of dose to personnel through the removal of radioactive cobalt ions from the inner wall of PCS piping.

The PCS may be pressure tested for leaks by means of the variable speed charging pump. The system is also provided with connections for installing a hydrostatic test pump.

The CVC system is not credited in the Chapter 14 accident analyses with any mitigating actions. However, the system responds to safety injection actuation signal and the charging pumps inject concentrated boric acid into the primary coolant system.

The description above results in some SSCs in this system being considered in-scope in accordance with 10 CFR 54.4(a)(1). Since some SSCs in the Chemical and Volume Control System are non-safety related and their failure could affect the capability of safety related SSCs to perform their safety function, they are considered to be in-scope in accordance with 10 CFR 54.4(a)(2). In addition, some SSCs are considered in-scope due to Environmental Qualification, Fire Protection, and Station Blackout in accordance with 10 CFR 54.4(a)(3).

The boundaries of the in-scope portions of the Chemical and Volume Control System are described as follows: 1) The PCS letdown through the regenerative heat exchanger (tube side), the letdown orifices, the letdown heat exchanger, Containment penetration #36, to the purification ion exchangers, 2) From the purification ion exchangers to the purification filters, 3) From the purification filters to the volume control tank (VCT), 4) From the primary coolant pumps controlled bleedoff, through Containment penetration #44 to the Containment isolation valves, 5) From the VCT to the charging pumps, Containment penetration #45, the regenerative heat exchanger (shell side), and to the PCS, 6) From the concentrated boric acid tanks to the concentrated boric acid pumps, the boric acid filters and to the suction of the charging pumps, and 7) From the safety injection and refueling water tank to the suction of the charging pumps.

The original system scoping boundaries for license renewal included, where applicable, non-safety related components that could affect safety-related components. The boundaries have been expanded in accordance with current NRC guidance for implementation of 10 CFR 54.4(a)(2). The expanded scoping boundary includes:

CVC non-safety related piping and components that are directly attached to safety related piping and components out to the first seismic anchor or equivalent anchor. Where base mounted components are credited as an anchor, they are also included in scope. Pipe supports for the newly added piping and components are also in scope. The new piping components are located in the Auxiliary Building and Containment.

CVC non-safety related components containing steam and/or fluids that are located in an area that also contains a safety related component. The components are located in the Auxiliary Building and Containment.

The portions of the Chemical and Volume Control system containing components subject to an AMR include accumulators, coolers, filters, heat exchangers, fasteners, level glasses, pipe and fittings, pumps, flow elements, and valves. The NSAS components added into scope by the 10 CFR 54.4 scoping boundary expansion are subject to AMR.

System Function Listing

A comprehensive listing of functions associated with the Chemical and Volume Control System, or specific components contained in the system, is provided in the summary below.

System Function: CBA-01 Provide concentrated boric acid from the SIRW Tank to the suction of the charging pumps.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Provides boration during a design basis accident.

System Function: CBA-02 Maintain temperature of the boration flow paths.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Not in scope of License Renewal due to the concentrated boric acid flow path has been down-graded from Safety-Related to Non-Safety Related in the Palisades Technical Specifications and the Inservice Inspection Program.

System Function: CVC-01 Provide borated makeup water to the primary coolant system from the volume control tank, Concentrated Boric Acid storage tank, SIRW tank.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This is a normal operating system function that does not meet 10 CFR 54.4 Criteria 1, 2, or 3 and is not in boundary of License Renewal. Safety functions of CVC System have been eliminated from Chapter 14 analyses with the issuance of FSAR revision 21. The letdown portion of the system is isolation on Containment Isolation and performs no safety functions.

System Function: CVC-02 Provide backup PCS pressure control via pressurizer sprays from CVC.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Auxiliary spray capability is provided from the CVCS charging pumps when no PCP is operating.

During shutdown, the charging subsystem provides auxiliary spray flow to the pressurizer to cooldown the pressurizer if normal pressurizer spray via the primary coolant pumps is not available.

This system function is a normal operating system function and does not meet 10 CFR 54.4 Criteria 1 intended functions. However system components are required by NSAS (Criterion 2) and Appendix R (Criterion 3), are addressed in System Functions CVC-NSAS and CVC-FP, respectively, and are in the boundary of License Renewal.

System Function: CVC-03 Manually or Automatically close PCP bleedoff and letdown containment isolation valves on CHP and CHR. Also ensure manual isolation valves can perform containment isolation.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: This function is associated with the valves and pipe required to ensure containment isolation in the CVC system. Components associated with the processing of electrical signals to initiate automatic isolation upon CHP and CHR are dispositioned under the electrical CIS system. The penetrations are dispositioned within the Containment Structure evaluation.

System Function: CVC-04 Maintain PCS chemistry within specification.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: The chemical control and purification functions ensure that during normal operation an average operating activity of 1.0 uc/cc, excluding gases, is maintained in the primary coolant.

This system function is not in boundary of License Renewal. The system components that are required by NSAS (Criterion 2) and Appendix R (Criterion 3) requirements are addressed in System Function CVC-NSAS and CVC-FP, respectively, and are in the boundary of License Renewal.

System Function: CVC-05 Provide an alternate charging PCS injection path through the high pressure injection piping.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Generic Letter 88-17, recommended expeditious action No 6, required that there be at least two available or operable means of adding inventory to the PCS that are in addition to pumps that are a part of the normal decay heat removal systems.

System Function: CVC-06 Provide process radiation monitoring of the PCS for indication of failed fuel.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This function is not in the boundary of License Renewal.

Currently, the installed monitor RIA-0202A is not being used and is slated for removal. PCS radioactivity level is being measured by batch sampling.

System Function: CVC-EQ The system contains -components required by the current licensing basis for Environmental Qualification.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
				X			

Comment: The CVC system contain components that are required for 10 CFR 50.49 requirements.

System Function: CVC-FP The system contains structures and/or components required by the current licensing basis for Fire Protection.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: The CVC system contains components that are required for Appendix R safe shutdown.

System Function: CVC-NSAS	Cri 1	Cri 2	Cri 3				
This system has components in scope of license renewal in accordance with 10CFR54.4(a)(2) requirements for non-safety affecting safety (NSAS).			FP	EQ	PTS	AT	SB
		X					

Comment: The CVC system contains non-safety related components that are attached to safety related components and non-safety related components containing liquids located in an area that also contains safety related components.

System Function: CVC-SBO	Cri 1	Cri 2	Cri 3				
The system contains structures and/or components required by the current licensing basis for Station Blackout (Loss of all AC power).			FP	EQ	PTS	AT	SB
							X

Comment: The CVC system contains components required to operate (valves) during an SBO to reduce the primary coolant system's inventory losses.

FSAR Reference

Additional Chemical and Volume Control System details are provided in Section 9.10 of the FSAR.

Scoping Boundary Drawings

The scoping boundary drawings for the Chemical and Volume Control System are listed below:

LR-M-201, Sheet 1
LR-M-201, Sheet 2
LR-M-202, Sheet 1
LR-M-202, Sheet 1A
LR-M-202, Sheet 1B
LR-M-204, Sheet 1A
LR-M-204, Sheet 1B
LR-M-209, Sheet 1
LR-M-209, Sheet 3
LR-M-212 Sheet 3
LR-M-222, Sheet 1A
LR-M-650, Sheet 1A

Components Subject to an AMR

The component groups for this system that require aging management review are addressed in Table 2.3.3-1 along with each Component Group’s intended function(s).

Table 2.3.3-1 Chemical and Volume Control System

Component Group	Intended Function
Boric Acid Storage Tanks	Fluid Pressure Boundary

Table 2.3.3-1 Chemical and Volume Control System

Component Group	Intended Function
Oil Cooler Shell	Fluid Pressure Boundary
Oil Cooler Tubes	Fluid Pressure Boundary
Letdown Heat Exchanger Shell	Fluid Pressure Boundary
Letdown Heat Exchanger Channel Head, Tubes, Tube Sheet	Fluid Pressure Boundary
Letdown Heat Exchanger Tubes, Tube Sheet	Fluid Pressure Boundary
Letdown Heat Exchanger Tubes	Fluid Pressure Boundary
Letdown Heat Exchanger Channel Head, Tubes, Tube Sheet	Fluid Pressure Boundary
Fasteners	Fluid Pressure Boundary
Pipe - CVC Cooler	Fluid Pressure Boundary
Nozzle - CVC Spray	Fluid Pressure Boundary
Tubing - CVC Oil	Fluid Pressure Boundary
Pipe and Pressure Test Fittings	Fluid Pressure Boundary
Flow Elements	Fluid Pressure Boundary
Regenerative Heat Exchanger Tubes, Tube Sheet	Fluid Pressure Boundary
Regenerative Heat Exchanger Channel Head, Shell, Tubes, Tube Sheet	Fluid Pressure Boundary
Control Valves	Fluid Pressure Boundary
Letdown Stop Valve CV-2001	Fluid Pressure Boundary
Check, Control, Manual & Relief valves; Instrument Assemblies	Fluid Pressure Boundary

2.3.3.2 Circulating Water System

System Description

The Circulating Water System (CWS) is a closed cycle system using two mechanical draft cooling towers. Each loop supplies one-half of the main condenser with cooling water by gravity flow from the two 18-cell, induced draft cross-flow cooling towers. The cooling towers are erected to the south of the Plant over concrete basins. Two half-capacity vertical wet pit cooling tower pumps receive heated circulating water from the condenser pump suction makeup basin. The cooling tower pumps return the circulating water to the cooling tower distribution headers through two 96-inch pipes. Improved cooling efficiency and reduced system scaling are obtained by injection of dilution water pump discharge into the condenser inlet.

The description above results in some non-safety related SSCs in this system, whose failure could affect the capability of a safety related SSC to perform its safety function, as being considered in-scope in accordance with 10 CFR 54.4(a)(2).

The boundaries of the in-scope portions of the Circulating Water System are described as follows: 1) the dilution pumps and discharge piping located in the intake structure pump house, 2) the portion of the piping from A Cooling Tower to MO-5301 (West Water Box Inlet Valve) with splits to MO-5326A (Basin 'A' Cooling Tower Blowdown Line Isolation located outside and south of intake structure pump house) and MO-5315 (P-40A/B Discharge To E-30A Makeup/Fill located outside and northeast of intake structure pump house), and 3) the portion of the piping from B Cooling Tower to MO-5302 (East Water Box Inlet Valve) with splits to MO-5326B (Basin 'B' Cooling Tower Blowdown Line Isolation located outside and south of intake structure pump house) and MO-5316 (P-40A/B Discharge To E-30B Makeup/Fill located outside and southeast of intake structure pump house).

The original system scoping boundaries for license renewal included, where applicable, non-safety related components that could affect safety-related components. The in-scope boundaries have been expanded in accordance with current NRC guidance for implementation of 10 CFR 54.4(a)(2). The expanded scoping boundary includes:

CWS non-safety related components containing steam and/or fluids that are located in an area that also contains a safety related component. The components types include piping, fittings, and valves. The components are located in the Turbine Building and Screen House.

The portions of the Circulating Water System containing components subject to an AMR include fasteners, pipe and fittings, pumps and valves. The NSAS components added into scope by the 10 CFR 54.4(a)(2) scope expansion are subject to AMR.

System Function Listing

A comprehensive listing of functions associated with the Circulating Water System, or specific components contained in the system, is provided in the summary below.

System Function: CWS-01	Cri 1	Cri 2	Cri 3				
Circulate cooling water through the condenser tubes and cooling towers to remove heat from the secondary system and to provide condenser vacuum.			FP	EQ	PTS	AT	SB

Comment: The removal of heat from the main condenser is the main function of the Circulating Water System. The Circulation Water System is not safety related.

System Function: CWS-02	Cri 1	Cri 2	Cri 3				
Circulate cooling air through the cooling tower to remove circulating water heat via evaporation.			FP	EQ	PTS	AT	SB

Comment: This function relates to the fans that are used to circulate air through the cooling towers.

System Function: CWS-NSAS	Cri 1	Cri 2	Cri 3				
The system contains structures and/or components whose failure could cause failure of safety related components due to flooding.			FP	EQ	PTS	AT	SB
		X					

Comment: In-scope of License Renewal to protect safety related components from flooding.

A failure of the segment of CSW system pipe from E-30A to MO-5301 and E-30B to MO-5302 that runs through the screen house as well as the 36 inch piping from P-40A and P-40B to the edge of the screen house could cause the loss of safety related equipment.

FSAR Reference

Additional Circulating Water System details are provided in Section 10.2 of the FSAR.

Scoping Boundary Drawings

The scoping boundary drawings for the Circulating Water System are listed below:

LR-M-653, Sheet 1

LR-M-653, Sheet 3

Components Subject to an AMR

The component groups for this system that require aging management review are addressed in Table 2.3.3-2 along with each Component Group's intended function(s).

Table 2.3.3-2 Circulating Water System

Component Group	Intended Function
Fasteners	Fluid Pressure Boundary
Pipe & Fittings	Fluid Pressure Boundary
Pumps	Fluid Pressure Boundary
Valves & Dampers	Fluid Pressure Boundary

2.3.3.3 Component Cooling Water System

System Description

The Component Cooling Water System (CCS) is designed to cool components carrying radioactive and potentially radioactive fluids. It provides a monitored intermediate barrier between these fluids and the service water system which transfers the heat to the outside environment. Thus, the probability of leakage of contaminated fluid into the lake is greatly reduced.

System components are rated for the maximum heat removal requirements that occur during normal, shutdown or accident operation as applicable. The parts of the system located inside containment are isolated in the event of a containment high-pressure signal (CHP). The component cooling water to the radwaste evaporators and spent fuel cooling system are isolated on safety injection actuation signal (SIAS).

The system is a closed loop consisting of three motor-driven circulating pumps, two heat exchangers, a surge tank, associated valves, piping, instrumentation and controls. The system is continuously monitored by a process monitor which detects radioactivity which may have leaked into the system from the fluids being cooled.

The Component Cooling Water System uses demineralized water to which an inhibitor is added for corrosion control. Makeup to the system is automatically supplied from the primary system makeup storage tank.

Heat is transferred from the system to plant service water by means of two component cooling heat exchangers. Service water from the critical service water header is provided to the tube side of the heat exchangers and the rejected heat from the system is discharged by service water into the cooling tower pump makeup basin.

Four main supply lines are provided to the various areas of the Plant as follows:

1. To Shutdown Cooling Heat Exchangers
2. To Engineered Safeguards Pumps
3. To Spent Fuel Pool Heat Exchangers and Radwaste Equipment
4. To Services Inside the Containment

Two full capacity valves installed in parallel are provided in the line supplying component cooling water to the Shutdown Cooling heat exchangers.

The description above results in some SSCs in this system being considered in-scope in accordance with 10 CFR 54.4(a)(1). Since some SSCs in the Component Cooling Water System are non-safety related and their failure could affect the capability of safety related SSCs to perform their safety function, they are considered to be in-scope in accordance with 10 CFR 54.4(a)(2). In addition, some SSCs are considered in-scope due to Environmental Qualification and Fire Protection in accordance with 10 CFR 54.4(a)(3).

The boundaries of the in-scope portions of the Component Cooling Water System are described as follows: 1) The pressure boundary from the Primary Makeup supply valve and bypass to the Component Cooling Expansion tank and the suction header to the CCS Pumps. 2) From the CCS pumps suction header, through the CCS pumps, the CCS discharge header, through the CCS heat exchangers to the CCS main supply header, 3) From the CCS main supply header to: a. Shutdown Cooling Heat Exchangers, b. the Engineered Safeguards Pumps, c. the Spent fuel Pool Heat Exchangers, d. through Containment penetration #14 to the Letdown Heat Exchanger, shield cooling heat exchanger and the Primary Coolant Pumps internal heat exchangers located inside Containment, and 4) the CCS return headers to the CCS pumps suction header and surge tank.

The original system scoping boundaries for license renewal included, where applicable, non-safety related components that could affect safety-related components. The boundaries have been expanded in accordance with current NRC guidance for implementation of 10 CFR 54.4(a)(2). The expanded scoping boundary includes:

CCS non-safety related piping and components that are directly attached to safety related piping and components out to the first seismic anchor or equivalent anchor. Where base mounted components are credited as an anchor, they are also included in scope. Pipe supports for the newly added piping and components are also in scope. The new piping components are located in the Auxiliary Building and Containment Building.

CCS non-safety related components containing steam and/or fluids that are located in an area that also contains a safety related component. The components types include an accumulator, fasteners (to maintain the pressure boundary), pipe, fittings and valves. The components are located in the Auxiliary Building and Containment Building.

The portions of the Component Cooling System containing components subject to an AMR include accumulators, flow switches, heat exchangers, coolers, fasteners, pipe and fittings, pumps and valves. The NSAS components added into scope by the 10 CFR 54.4(a)(2) scoping boundary expansion are subject to AMR.

System Function Listing

A comprehensive listing of functions associated with the Component Cooling Water System, or specific components contained in the system, is provided in the summary below.

System Function: CCS-01	Cri 1	Cri 2	Cri 3				
Circulate cooling water through essential loads and heat exchangers during normal and shutdown operations.			FP	EQ	PTS	AT	SB
	X						

Comment: The CCS system provides sufficient heat removal for each of the components supported by the system in order to maintain operability and function of each component serviced during normal and emergency modes of operation. This function includes the Component Cooling Surge tank, Pumps, Heat Exchangers and piping to and from the cooled components.

System Function: CCS-02	Cri 1	Cri 2	Cri 3				
Automatically isolate the CCS supply and return lines from containment on CHP for containment isolation.			FP	EQ	PTS	AT	SB
	X						

Comment: The automatic activation by the Containment High Pressure (CHP) signal is required by Containment Pressure And Temperature Analysis, FSAR 14.18.

This function is associated with the valves and pipe required to ensure containment isolation in the CCS system. Components associated with the processing of electrical signals to initiate automatic isolation upon CHP and CHR are dispositioned under the electrical Containment Isolation system. The penetrations are dispositioned within the Containment Structure evaluation.

System Function: CCS-03	Cri 1	Cri 2	Cri 3				
Start idle CCS pumps and reposition valves to circulate water through essential loads/heat exchangers and isolate non-essential loads/heat exchangers upon receipt of an ESF actuation signal.			FP	EQ	PTS	AT	SB
	X						

Comment: The CCW system is automatically activated by the Safety Injection Actuation Signal (SIAS), Recirculation Actuation Signal (RAS), and Containment High Pressure (CHP) signal under certain accident conditions. The CHP signal is covered in function CCS-02.

This function includes all CCS SSCs required to start the pumps, reposition the valves and isolate the system. However the sequencers and the air system components to the solenoid valves are not included.

The FSAR Design Basis Accidents that affect CCS are: Increase In Steam Flow (Excess Load), FSAR 14.10, Loss Of Normal Feedwater, FSAR 14.13, Steam Line Rupture Incident, FSAR 14.14, Steam Generator Tube Rupture With A Loss Of Offsite Power, FSAR 14.15, Large Break LOCA (LBLOCA), FSAR 14.17, Containment Pressure And Temperature Analysis, FSAR 14.18.

System Function: CCS-04	Cri 1	Cri 2	Cri 3				
Provide cooling to non-essential loads for normal operations.			FP	EQ	PTS	AT	SB

Comment: This function does not meet Criterion 1, 2, or 3 and is not in boundary.

System Function: CCS-EQ	Cri 1	Cri 2	Cri 3				
The system contains structures and/or components required by the current licensing basis for Environmental Qualification.			FP	EQ	PTS	AT	SB
				X			

Comment: This function is developed to ensure the equipment subject to a harsh environment during an accident, will be able to perform its function.

System Function: CCS-FP	Cri 1	Cri 2	Cri 3				
The system contains structures and/or components required by the current licensing basis for Fire Protection.			FP	EQ	PTS	AT	SB
			X				

Comment: This function includes the equipment required to ensure the requirements of the 10CFR50.48 fire protection program are functional.

This function also includes the 10CFR50 Appendix R equipment required for a post fire safe shutdown.

System Function: CCS-NSAS	Cri 1	Cri 2	Cri 3				
This system has components in scope of license renewal in accordance with 10 CFR 54.4(a)(2) requirements for non-safety affecting safety (NSAS).			FP	EQ	PTS	AT	SB
		X					

Comment: Selected components were determined to be in boundary for NSAS.

FSAR Reference

Additional Component Cooling Water System details are provided in Section 9.3, Section 14.10, Section 14.13, Section 14.14, Section 14.15, Section 14.17, and Section 14.18 of the FSAR.

Scoping Boundary Drawings

The scoping boundary drawings for the Component Cooling Water System are listed below:

LR-M-202, Sheet 1B
LR-M-204, Sheet 1
LR-M-208, Sheet 1A
LR-M-209, Sheet 1
LR-M-209, Sheet 2
LR-M-209, Sheet 3
LR-M-212, Sheet 3
LR-M-212, Sheet 4
LR-M-214, Sheet 4
LR-M-214, Sheet 5
LR-M-219, Sheet 1B
LR-M-221, Sheet 1
LR-M-221, Sheet 2
LR-M-223, Sheet 1B
LR-M-224, Sheet 1
LR-M-655, Sheet 1

Components Subject to an AMR

The component groups for this system that require aging management review are addressed in Table 2.3.3-3 along with each Component Group's intended function(s).

Table 2.3.3-3 Component Cooling Water System

Component Group	Intended Function
Accumulators	Fluid Pressure Boundary

Table 2.3.3-3 Component Cooling Water System

Component Group	Intended Function
Bistable/Switch (In-line Flow Indicator)	Fluid Pressure Boundary
Component Cooling Heat Exchanger	Fluid Pressure Boundary
Cooler	Fluid Pressure Boundary
Heat Exchanger	Fluid Pressure Boundary Heat Transfer
Fasteners	Fluid Pressure Boundary
Primary Coolant Pump Motor Oil Cooler	Fluid Pressure Boundary
Pipe & Fittings	Fluid Pressure Boundary
Pumps	Fluid Pressure Boundary
Valves & Dampers	Fluid Pressure Boundary
Waste Gas Compressor Cooler	Fluid Pressure Boundary

2.3.3.4 Compressed Air System

System Description

The Compressed Air Systems (CAS) consist of the Instrument Air System, the High Pressure Air System, various backup systems, and the Feedwater Purity Air system.

The Instrument Air System is a non-safety related system that is required for normal plant operation. The system is designed to provide a reliable supply of dry, oil-free air for instruments and controls, and for service air requirements. The design of the system is based on an estimated instrument air consumption rate of 80 scfm for the Nuclear Steam Supply System and 115 scfm for the remainder of the Plant. Portions of the systems are called out in the Fire Protection Plan for use in mitigating the results of fires in the plant.

Three air cooled, oil free compressors are provided for the system, each with an in-line air receiver tank. Two have after-coolers served by critical service water, with the third compressor after-cooler served by non-critical service water. The air receivers are connected to a common discharge air header. The common air header branches into two separate air headers, one to the instrument air dryer and filter assembly, and one to the Service Air System. The instrument air headers are divided into branch lines supplying the Turbine Building, Containment Building, Intake Structure, and Auxiliary Building.

The High Pressure Air System consists of three oil-lubricated air compressors, each with its own dryer and air receiver. One of these High Pressure Air Compressors resides in the Turbine Building and is non-safety related, while the other two are located in the East and West Safeguards Rooms. The compressors in the East and West Safeguards Rooms supply air to safety-related air receivers which supply air to valves located in the two Safeguards Rooms. The non-safety related Turbine Building system can supply either the East or West Safeguards Systems in any plant mode. Moisture is removed from the high pressure air by dryers that are in series with the compressors' air-cooled aftercoolers. Any remaining moisture is removed by periodic blowdown of the air receivers and the low point drains. The safety related portion of the High Pressure Air System in the East and West Safeguards Rooms extends from the air receivers to the control valves serviced and is isolated from the non-safety related portion by check valves.

Backup systems consist of bottled nitrogen stations, bottled air station, bulk nitrogen, local accumulators, and manual valve actuators.

The Feedwater Purity system is not a safety-related system. When manually aligned the system is capable of supplying air to the Instrument and Service air systems.

The description above results in some SSCs in this system being considered in-scope in accordance with 10 CFR 54.4(a)(1). Since some SSCs in the Compressed Air System are non-safety related and their failure could affect the capability of safety related SSCs to perform their safety function, they are considered to be in-scope in accordance with 10 CFR 54.4(a)(2). In addition, some SSCs are considered in-scope due to Environmental Qualification and Fire Protection in accordance with 10 CFR 54.4(a)(3).

The boundaries of the in-scope portions of the Compressed Air System are described as: 1) the piping and valves from instrument air compressors, through aftercoolers, through air receivers, through air filters, to various control valves located in the turbine building, auxiliary building and Containment; and 2) the piping and valves from the high pressure air compressors through their aftercoolers, through air dryer, through air receivers, to various high pressure air operated valves, and to the instrument air header.

The original system scoping boundaries for license renewal included, where applicable, non-safety related components that could affect safety-related components. The scoping boundaries have been expanded in accordance with current NRC guidance for implementation of 10 CFR 54.4(a)(2). The expanded scoping boundary includes:

CAS non-safety related piping and components that are directly attached to safety related piping and components out to the first seismic anchor or equivalent anchor. Where base mounted components are credited as an anchor, they are also included in scope. Pipe supports for the newly added piping and components are also in scope. The new piping components are located in the Auxiliary Building, Containment Building and Turbine Building.

CAS non-safety related components contain compressed air. They are located in an area that also contains a safety related component. The components types include fasteners (to maintain the pressure boundary), piping, fittings and valves. The components are located in the Auxiliary Building, Containment Building and Turbine Building.

The portions of the Compressed Air System containing components subject to an AMR include accumulators, dryers, filters, heat exchangers, fasteners, pipe and fittings, steam traps, and valves. The NSAS components added into scope by the 10 CFR 54.4(a)(2) scoping boundary expansion are subject to AMR.

System Function Listing

A comprehensive listing of functions associated with the Compressed Air System, or specific components contained in the system, is provided in the summary below.

System Function: CAS-EQ	Cri 1	Cri 2	Cri 3				
The system contains components required by the current licensing basis for Environmental Qualification.			FP	EQ	PTS	AT	SB
				X			

Comment: EQ components are located in the Compressed Air System.

System Function: CAS-FP	Cri 1	Cri 2	Cri 3				
The system contains structures and/or components required by the current licensing basis for Fire Protection.			FP	EQ	PTS	AT	SB
			X				

Comment: Appendix R components are supplied air from both the Instrument Air subsystem or the High Pressure Air subsystem.

System Function: CAS-NSAS	Cri 1	Cri 2	Cri 3				
This system has components in scope of license renewal in accordance with 10CFR54.4(a)(2) requirements for non-safety affecting safety (NSAS).			FP	EQ	PTS	AT	SB
		X					

Comment: Compressed air components are non-safety related. They are attached to and located above safety related components.

System Function: FPA-01	Cri 1	Cri 2	Cri 3				
The Feedwater Purity Air subsystem provides a backup source of instrument/service air.			FP	EQ	PTS	AT	SB

Comment: Feedwater Purity air is not a safety related system. When manually aligned, the system is capable of supplying air to the Instrument and Service Air System. Feedwater Purity air is not required for normal operation and is, therefore, not in-scope for license renewal.

System Function: HPA-01	Cri 1	Cri 2	Cri 3				
The High Pressure Air subsystem provides high pressure air to assure operability of cylinder operated Engineered Safeguards, Main Feedwater, and Condensate valves necessary for accident conditions. Provides air to the Main Steam Isolation Valves.			FP	EQ	PTS	AT	SB
	X						

Comment: The HPA subsystem is safety related and, therefore, in-scope for CAS license renewal.

System Function: IAS-01	Cri 1	Cri 2	Cri 3				
The Instrument and Service Air subsystem provides instrument air to components to avoid plant trips and to mitigate consequences of an accident.			FP	EQ	PTS	AT	SB

Comment: Except for IAS-03, below, the Instrument and Service air subsystem is not safety related. It does supply air to Appendix R components as identified in function CAS-FP above. The balance of the system is not in-scope for license renewal.

System Function: IAS-02 The Instrument and Service Air subsystem provides service air to the cooling tower pump basin level transmitter.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This system function does not meet any license renewal criterion.

System Function: IAS-03 Manually isolate the Instrument Air subsystem to containment to ensure containment isolation.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Components accomplishing this system function are safety related. Additionally, valve MV-CA10190, located in the Control Room, isolates Instrument air to the Control Room and is safety related for seismic considerations. Therefore, this valve is in-scope for License Renewal.

System Function: IAS-04 Automatically isolate the Service Air subsystem in the event of low air header pressure.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Components accomplishing this system function do not meet a license renewal criterion.

FSAR Reference

Additional Compressed Air System details are provided in Section 9.5 of the FSAR.

Scoping Boundary Drawings

The scoping boundary drawings for the Compressed Air System are listed below:

LR-M-201, Sheet 1
LR-M-201, Sheet 2
LR-M-202, Sheet 1
LR-M-202, Sheet 1A
LR-M-202, Sheet 1B
LR-M-203, Sheet 1
LR-M-203, Sheet 2
LR-M-204, Sheet 1
LR-M-204, Sheet 1A
LR-M-204, Sheet 1B
LR-M-205, Sheet 1
LR-M-205, Sheet 2
LR-M-207, Sheet 1
LR-M-207, Sheet 1A
LR-M-207, Sheet 2
LR-M-208, Sheet 1A
LR-M-208, Sheet 1B
LR-M-209, Sheet 1
LR-M-209, Sheet 2
LR-M-209, Sheet 3
LR-M-210, Sheet 1A
LR-M-210, Sheet 1B
LR-M-210, Sheet 2
LR-M-211, Sheet 1
LR-M-211, Sheet 2
LR-M-211, Sheet 3
LR-M-212, Sheet 1
LR-M-212, Sheet 1A
LR-M-212, Sheet 2
LR-M-212, Sheet 3
LR-M-212, Sheet 4
LR-M-212, Sheet 5
LR-M-213
LR-M-215, Sheet 1
LR-M-216, Sheet 2
LR-M-218, Sheet 2
LR-M-218, Sheet 5
LR-M-218, Sheet 6
LR-M-218, Sheet 6A
LR-M-219, Sheet 1B
LR-M-220, Sheet 1
LR-M-221, Sheet 1
LR-M-222, Sheet 2
LR-M-225, Sheet 1
LR-M-225, Sheet 1A
LR-M-225, Sheet 2
LR-M-226, Sheet 1
LR-M-650, Sheet 1B
LR-M-651, Sheet 1B
LR-M-657, Sheet 1

Components Subject to an AMR

The component groups for this system that require aging management review are addressed in Table 2.3.3-4 along with each Component Group's intended function(s).

Table 2.3.3-4 Compressed Air System

Component Group	Intended Function
Accumulators	Fluid Pressure Boundary
Air Dryers	Fluid Pressure Boundary
Blowers Fans Compressors Vacuum	Fluid Pressure Boundary
Filters/Strainers	Fluid Pressure Boundary Filtration
Heat Exchangers	Fluid Pressure Boundary
Fasteners	Fluid Pressure Boundary
Pipe & Fittings	Fluid Pressure Boundary
Pumps	Fluid Pressure Boundary
Traps (Steam)	Fluid Pressure Boundary
Valves & Dampers	Fluid Pressure Boundary

2.3.3.5 Containment Air Recirculation and Cooling System

System Description

The Containment Air Recirculation and Cooling System (CRS) includes four air handling and cooling units located entirely within the Containment Building. Plant service water from the critical service water is circulated through the air cooling units.

Air is drawn through the coils by two matched vaneaxial fans with direct connected motors. One fan motor is rated for normal operating conditions and the second fan motor is rated for post-DBE conditions. The fan motors rated for the post-DBE condition are fed from the emergency power buses.

Four units are normally in operation with two fans in each unit operating. The coolers are automatically changed to the emergency mode by a safety injection actuation signal. This signal will trip the normal rated fan motor in each unit.

The description above results in some SSCs in this system being considered in-scope in accordance with 10 CFR 54.4(a)(1). In addition, some SSCs are considered in-scope due to Environmental Qualification and Fire Protection in accordance with 10 CFR 54.4(a)(3).

The boundaries of the in-scope portions of the Containment Air Recirculation and Cooling System (CRS) are described as follows: ducting from the Containment atmosphere, through filters, through the Containment Air Cooler (CAC) units (including drip pans), and returning to the Containment atmosphere. The boundary of the CRS also includes the cooling water side (critical service water) of the CAC cooling coils.

The portions of the Containment Air Recirculation and Cooling System containing components subject to an AMR include fans, filters, heat exchanger, drip pans, pipe and fittings, flow elements, valves, dampers and fasteners.

System Function Listing

A comprehensive listing of functions associated with the Containment Air Recirculation and Cooling System, or specific components contained in the system, is provided in the summary below.

System Function: CRS-01	Cri 1	Cri 2	Cri 3				
Remove heat and vapor from the containment atmosphere during normal plant operation.			FP	EQ	PTS	AT	SB

Comment: This is a normal plant operation function and does not meet the criteria of 10 CFR 54.4. Therefore, this function is not in boundary of license renewal.

System Function: CRS-02	Cri 1	Cri 2	Cri 3				
In the event of a DBE, limit the containment building pressure rise and reduce the leakage of airborne radioactivity by providing a means of cooling the containment atmosphere.			FP	EQ	PTS	AT	SB
	X						

Comment: A LOCA is initiated by the rupture of the primary coolant system piping. The primary coolant will flash to steam and escape through the break. As the steam is released to the Containment building, the pressure and temperature of the Containment atmosphere quickly increases. The structures in Containment will absorb energy and condense steam, counteracting the initial pressure and temperature increase. The Containment Air Coolers and Containment Spray System, which are activated by the pressure rise, then act to reduce the pressure and temperature and remove energy released from decay heat.

System Function: CRS-EQ	Cri 1	Cri 2	Cri 3				
The system contains structures and/or components required by the current licensing basis for Environmental Qualification.			FP	EQ	PTS	AT	SB
				X			

Comment: The CRS system contains components that are required per 10 CFR 50.49.

System Function: CRS-FP	Cri 1	Cri 2	Cri 3				
The system contains structures and/or components required by the current licensing basis for Fire Protection.			FP	EQ	PTS	AT	SB
			X				

Comment: The Containment Air Coolers contain components that are required for “Appendix R” safe shutdown.

FSAR Reference

Additional Containment Air Recirculation and Cooling System details are provided in Section 6.3 and Section 14.18 of the FSAR.

Scoping Boundary Drawings

The scoping boundary drawings for the Containment Air Recirculation and Cooling System are listed below:

LR-M-208, Sheet 1B
LR-M-218, Sheet 2

Components Subject to an AMR

The component groups for this system that require aging management review are addressed in Table 2.3.3-5 along with each Component Group’s intended function(s).

Table 2.3.3-5 Containment Air Recirculation and Cooling System

Component Group	Intended Function
Containment Air Cooler Coils	Fluid Pressure Boundary Heat Transfer
Containment Air Cooler Filter	Filtration Fluid Pressure Boundary
Containment Air Cooler Flow Element	Flow Measurement Fluid Pressure Boundary
Containment Air Cooler Housing	Fluid Pressure Boundary
Containment Air Cooler Recirculation Fans	Fluid Pressure Boundary
Dampers	Fluid Pressure Boundary
Drip Pans	Fluid Pressure Boundary
Duct	Pressure Boundary
Fasteners	Structure Functional Support
Manual and Instrumentation Valves	Fluid Pressure Boundary

2.3.3.6 Emergency Power System

System Description

The Emergency Power System (EPS) has four major subsystems: 1) Emergency Diesel Generator, 2) 125 Volt Vital DC, 3) 120 Volt Preferred AC System and 4) Emergency Lighting. The emergency power sources are designed to furnish onsite power to reliably shut down the Plant and maintain it in a safe shutdown condition under all conditions, including DBE, upon loss of normal and standby power. The emergency power sources are part of the engineered safeguards electrical system and are identified as Class 1E systems.

The Diesel Fire Pump engines have also been included in the EPS system for convenience to permit fire pump diesels to be evaluated in conjunction with the diesel generator diesel engines.

The Emergency Diesel Generator subsystem consists of two independent, physically separate, diesel-engine driven generators of equal size. Support systems associated with each diesel generator include a fuel oil system, air starting system, lube oil system, jacket water system, crankcase exhauster, two starting circuits and a load sequencer. Supply of electric power for these support subsystems is obtained from the generator they are supporting. Each unit is installed in a separate room in the seismic Class 1 Auxiliary Building with the exception of the Fuel Oil Storage Tank, Fuel Oil Transfer Pumps, and the engine combustion air intake and exhaust. The load sequencers are located separate from one another in the main control room.

The 125-Volt Vital DC Power subsystem consists of two independent and redundant safety related Class 1E DC power sources. Each DC source consists of one 125 V battery, one battery charger in-service and one battery charger in-standby, and the associated control equipment, distribution panels, switchgear, instrumentation, fuse panels and interconnecting cabling. On loss of normal and standby ac power, the batteries will supply power to required preferred ac and dc loads until one of the diesel generators has started and can supply power for the chargers. Assuming that neither diesel emergency generator is available, the batteries have ample capacity to supply required dc loads and preferred ac loads during a complete loss of ac power for at least four hours.

The 120-Volt Preferred AC subsystem is part of the Engineered Safeguards Electrical System, and provides a Class 1E service. The inverters are the normal source of power for the four Preferred AC buses. The function of each inverter is to convert the 125-Volt Vital DC power from the batteries and provide continuous uninterruptible Preferred AC power for the instrumentation required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBE. The Preferred AC System buses supply power to the four Reactor Protection System channels and other Engineered Safety Features (ESF) controls and instrumentation.

The Emergency Lighting subsystem is composed of three subsystems: 1) Emergency AC Lighting (The portion of the normal lighting system which is supplied by the Alternating Current portion of the Class 1E Engineered Safeguards Electrical System), 2) Emergency DC Lighting (Supplied by the 125-VOLT Vital DC battery), and 3) Emergency Lighting Units (Supplied by internal battery packs which are continuously maintained in the charged condition when normal lighting power is available). These fixed battery pack lights are provided for access/egress and at the location of all manual actions required to achieve and maintain hot shutdown per 10 CFR 50, Appendix R. Emergency lighting inside containment is provided for personnel safety and to assist in safe handling of fuel during refueling outages.

The description above results in some SSCs in this system being considered in-scope in accordance with 10 CFR 54.4(a)(1). Since some SSCs in the Emergency Power System are non-safety related and their failure could affect the capability of safety related SSCs to perform their safety function, they are considered to be in-scope in accordance with 10 CFR 54.4(a)(2). In addition, some SSCs are considered in-scope due to Fire Protection and Station Blackout in accordance with 10 CFR 54.4(a)(3).

The boundaries of the in-scope portions of the Emergency Power System are described as follows: 1) Emergency Diesel Generators and components associated with the various support systems, including starting air, lube oil, fuel oil, intake and exhaust, and the jacket water system; 2) diesel fire pump engines, including auxiliary subsystems of cooling water, fuel oil, lube oil, intake air and exhaust air; 3) 125 Volt Vital DC subsystem, including two station batteries (including racks and output terminals), two redundant divisions consisting of battery chargers, bus, and distribution panels; 4) 120 Volt Preferred AC subsystem, including two divisions consisting of inverters, input breakers, instrument AC bypass regulator, distribution panels, and feeder breakers connected to instrument and control loads. 5) Emergency AC lighting, including lighting loads, branch circuit breakers and interconnecting cabling; a portion of emergency DC lighting, including one supply breaker; and self-contained emergency lighting units.

The original system scoping boundaries for license renewal included, where applicable, non-safety related components that could affect safety related components. The boundaries have been expanded in accordance with current NRC guidance for implementation of 10 CFR 54.4(a)(2). The expanded scoping boundary includes:

EPS non-safety related components containing steam and/or fluids that are located in an area that also contains a safety related component. The component types include piping and fittings. The components are located in the Auxiliary Building.

The portions of the Emergency Power System containing components subject to an AMR include accumulators, fans, coolers, filters, heat exchangers, instrument valve assemblies, heaters, fasteners, mufflers, pipe and fittings, pumps, traps, tubing, valves and oil pans. The NSAS components brought into scope by the 10 CFR 54.4(a)(2) boundary expansion are subject to AMR.

System Function Listing

A comprehensive listing of functions associated with the Emergency Power System, or specific components contained in the system, is provided in the summary below.

System Function: EDC-01 The 125-Volt Vital DC Power subsystem supplies DC power to safety related equipment and controls.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: The 125-Volt Vital DC Power subsystem consists of two independent and redundant safety related Class 1E DC power sources. On loss of normal and standby ac power, the batteries will supply power to preferred ac and dc loads until one of the diesel generators has started and can supply power for the chargers. Power to these loads is required for safe shutdown of the reactor. This EPS system function is within the boundary of License Renewal.

System Function: EDC-02 The 125-Volt Vital DC Power subsystem provides DC power to inverters supplying preferred AC busses/loads.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: The 125-Volt Vital DC Power subsystem supplies power to the inverters which are the normal source of power for the Preferred AC buses. The function of the inverter is to provide continuous AC electrical power to the Preferred AC buses, even in the event of an interruption to the normal AC power distribution system. The inverters are required to be operable to ensure that redundant sources of Preferred AC power for instrumentation and control are available to support engineered safeguards equipment in the event of an accident or transient and for power operation, plant heatups and cooldowns, and shutdown operation. Power to these loads is required for safe shutdown of the reactor. This EPS system function is within the boundary of License Renewal.

System Function: EDG-01 The Emergency Diesel Generators are able to be locally started and controlled to provide minimum safeguards equipment loads.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: The Emergency Diesel Generators' control circuits, in addition to the "automatic" functions, are arranged for manual start-stop at the diesel and in the control room. The governor and voltage regulator also have local controls in the diesel generator room. Procedures direct the use of this system feature which provides an alternative start and control of the EDG. This EPS system function increases the availability of the EDG which is used to place the plant in a safe shutdown condition. This EPS system function is within the boundary of License Renewal.

System Function: EDG-02	Cri 1	Cri 2	Cri 3				
The Emergency Diesel Generators are able to auto start and provide minimum safeguards equipment electric loads.			FP	EQ	PTS	AT	SB
	X						

Comment: The emergency generators are designed to provide a dependable onsite power source capable of starting and supplying the essential loads to safely shut down the Plant and maintain it in a safe shutdown condition under all conditions. The reliability of this onsite power is provided by its duplication wherein each emergency generator supplies redundant loads and each is capable of providing power to the minimum necessary safeguards equipment.

The two DGs each supply a separate 2400 V bus. They provide backup power in the event of loss of off-site power, or loss of power to the associated 2400 V bus. If either 2400 V bus, 1C or 1D, experiences a sustained undervoltage, the associated DG is started, the affected bus is separated from its offsite power sources, loads are stripped from that bus and its supported buses, the DGs are connected to the bus, and ECCS or shutdown loads are started by an automatic load sequencer. The EDG is required for safe shutdown of the plant. This EPS system function is within the boundary of License Renewal.

The following system support auto start of the diesels:

The Lube Oil System provides lubrication for engine bearings, vibration damper, turbocharger, and internal crankcase components during engine operation. A prelube pump circulates the lube oil when the diesel is in the standby condition. The Lube Oil System maintains the engine at an elevated “Standby” temperature of approximately 120 deg. F for fast starting when shutdown. The lube oil components for this diesel generator support system are safety-related and are required to maintain the diesel in standby readiness and operable. This EPS system function is within the boundary of License Renewal.

Each diesel engine has its own self-contained jacket cooling system for use during operation and preheat heating system when in standby. The jacket water system removes heat from the engine cylinders, block, heads, turbocharger, and aftercooler during operation. The jacket water components for this diesel generator support system are safety-related and are required to maintain operability. This EPS system function is within the boundary of License Renewal.

System Function: EDG-03	Cri 1	Cri 2	Cri 3				
Supply compressed air to the diesel engines for cranking.			FP	EQ	PTS	AT	SB
	X						

Comment: The starting air system for each diesel engine consists of an air compressor with diverse drives, two air storage tanks, two air start motors, and associated piping and valves. This auxiliary system is required to start the diesel and support operation of the EDG which is needed for safe shutdown of the plant. This EPS system function is within the boundary of License Renewal.

System Function: EDG-04	Cri 1	Cri 2	Cri 3				
Supply fuel oil from the diesel generator day tanks/belly tanks to the diesel engine.			FP	EQ	PTS	AT	SB
	X						

Comment: The Fuel Oil System provides storage of an adequate fuel supply, transfers fuel to the diesel engine, and delivers the correct amount of fuel to the engine cylinder. This auxiliary system is required to operate the EDG which is needed for safe shutdown of the plant. This EPS system function is within the boundary of License Renewal.

System Function: EDG-05	Cri 1	Cri 2	Cri 3				
Provide indication/alarms of critical diesel generator operating parameters.			FP	EQ	PTS	AT	SB
	X						

Comment: Each emergency generating system is equipped with a local alarm station and local instrumentation to allow supervision and monitoring of the system during different modes of operation and testing. The system provides sufficient alarms and indications to alert or inform the operator of system conditions, including those which are abnormal, adverse or potentially adverse to proper system operation. The alarms are safety-related and needed to support operation of the diesel generator. This EPS system function is within the boundary of License Renewal.

System Function: ELU-01	Cri 1	Cri 2	Cri 3				
The Emergency Lighting subsystem provides emergency AC lighting during normal, off normal and emergency conditions.			FP	EQ	PTS	AT	SB

Comment: During normal operation, the plant is served by the normal AC lighting system and the emergency AC lighting system (which is part of the normal AC lighting system). The Emergency AC Lighting subsystem is supplied by the Alternating Current portion of the Class 1E Engineered Safeguards Electrical System. The AC lighting system is non-Class 1E and can be isolated from Class 1E components by breaker. This EPS system function is not within the boundary of License Renewal.

System Function: ELU-02	Cri 1	Cri 2	Cri 3				
The Emergency Lighting subsystem provides emergency DC lighting during off normal and emergency conditions.			FP	EQ	PTS	AT	SB

Comment: The Emergency DC Lighting subsystem is supplied by the Direct Current portion of the Class 1E Engineered Safeguards Electrical System. The DC lighting system is non-Class 1E and can be isolated from Class 1E components by breaker. equipment and actions necessary for safe shutdown of the reactor. This EPS system function is not within the boundary of License Renewal.

System Function: ELU-03	Cri 1	Cri 2	Cri 3				
Emergency Lighting subsystem provides emergency fixed battery pack lighting (ELUs) during off normal and emergency conditions.			FP	EQ	PTS	AT	SB

Comment: See EPS-FP and EPS-SB for the functions of the ELUs in scope of license renewal.

This function addresses general emergency lighting functions for those ELUs that are not in the boundary of License Renewal.

System Function: EPS-FP	Cri 1	Cri 2	Cri 3				
The system contains structures and/or components required by the current licensing basis for Fire Protection.			FP	EQ	PTS	AT	SB
			X				

Comment: Emergency Diesels, Emergency Lighting, Preferred AC and 125-Volt DC are all systems required for safe-shutdown by 10 CFR 50, Appendix R. Also, a portable air operated pump is used to transfer fuel oil from tank T-926 to the diesel generator day tanks to support Appendix R safe shutdown. The air is supplied from the starting air receivers and the receivers are replenished using the backup gasoline engine to drive the starting air compressors.

The diesel fire pump engines and auxiliaries are also addressed within the EPS system.

System Function: EPS-NSAS	Cri 1	Cri 2	Cri 3				
This system has components in scope of license renewal in accordance with 10 CFR 54.4(a)(2) requirements for non-safety affecting safety (NSAS).			FP	EQ	PTS	AT	SB
		X					

Comment: The EPS contains non-safety related components containing liquids located in an area that also contains safety related components.

System Function: EPS-SB	Cri 1	Cri 2	Cri 3				
The system contains structures and/or components required by the current licensing basis for Station Blackout (Loss of all AC power).			FP	EQ	PTS	AT	SB
							X

Comment: The Emergency Lighting Units at Palisades are considered equipment for coping with a Station Blackout.

System Function: PAC-01 120-Volt Preferred AC Power subsystem provides AC power to safety related instrumentation and controls during normal and emergency conditions.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: The 120 Volt Preferred AC buses are normally powered from the inverters. The function of the inverter is to convert 125 Volt DC to AC and provide continuous AC electrical power to the Preferred AC buses, even in the event of an interruption to the normal AC power distribution system. The inverters are required to be operable to ensure that redundant sources of Preferred AC power for instrumentation and control are available to support engineered safeguards equipment in the event of an accident or transient and for power operation, plant heatups and cooldowns, and shutdown operation. Power to these loads is required for safe shutdown of the plant. This EPS system function is within the boundary of License Renewal.

FSAR Reference

Additional Emergency Power System details are provided in Section 8.3.5, Section 8.3, Section 8.4, Section 7.2.8, Section 8.4.1 and Section 8.9 of the FSAR.

Scoping Boundary Drawings

The scoping boundary drawings for the Emergency Power System are listed below:

LR-E-13, Sheet 1
LR-E-13, Sheet 2
LR-M-208, Sheet 1A
LR-M-214, Sheet 1
LR-M-216, Sheet 1
LR-WD950, Sheet 1
LR-WD950, Sheet 4
LR-WD950, Sheet 21
LR-WD950, Sheet 21A

Components Subject to an AMR

The component groups for this system that require aging management review are addressed in Table 2.3.3-6 along with each Component Group's intended function(s).

Table 2.3.3-6 Emergency Power System

Component Group	Intended Function
Accumulators	Fluid Pressure Boundary

Table 2.3.3-6 Emergency Power System

Component Group	Intended Function
Blowers Fans Compressor Vacuum	Fluid Pressure Boundary
Cooler	Fluid Pressure Boundary Heat Transfer
Fasteners	Fluid Pressure Boundary
Filters/Strainers	Filtration Fluid Pressure Boundary
Heat Exchangers	Fluid Pressure Boundary Heat Transfer
Heaters, Electric	Fluid Pressure Boundary
Misc Mechanical (Mufflers, oil pans)	Fluid Pressure Boundary
Motors	Fluid Pressure Boundary
Pipe & Fittings	Fluid Pressure Boundary
Pumps	Fluid Pressure Boundary
Traps (Steam)	Fluid Pressure Boundary
Valves & Dampers	Fluid Pressure Boundary

2.3.3.7 Fire Protection System

System Description

The Fire Protection System (FPS) includes the diverse design and operational features intended to prevent and mitigate the effects of fires. Building structures have been designed and arranged to prevent the spread of fire and to ensure integrity of redundant safe shutdown systems and areas. Fire suppression is provided by fixed water spray systems, such as sprinkler systems and deluge systems, fire hose reels and cabinets, portable fire extinguishers, fire barriers and fire detection systems. These fire suppression provisions are found throughout the Plant site.

Fire hoses from fire hydrants and a standpipe system provide protection in accordance with the guidance of NFPA 14, 20 and 24. The fire hydrant piping system is designed, installed and tested in accordance with the guidance of NFPA 24-1965, Outside Protection. The pumping supply system and fire pumps are designed and installed in accordance with NFPA 20-1959, Installation of Centrifugal Fire Pumps. The standpipe system is designed, installed and tested as a Class II system in accordance with the guidance of NFPA 14-1963, Installation of Standpipe and Hose Systems.

Fixed water spray systems, such as wet pipe fusible link sprinkler systems, dry pipe fusible link sprinkler systems, and fixed fog deluge spray systems are designed, installed and tested in accordance with the guidance of NFPA13-1968, Installation of Sprinkler Systems, and NFPA 15-1966, Water Spray Fixed Systems for Fire Protection. Indication of individual systems in various areas is provided by an annunciator panel in the main control room.

Fixed fog deluge systems protect the main, start-up and station auxiliary transformers. Each of these deluge systems are automatically actuated and annunciated by a general alarm in the main control room. A manual operated fixed fog deluge system protects the charcoal filters used to maintain control room habitability.

Fire detection is provided in the form of smoke and ultraviolet detectors. These detectors were located and installed in accordance with the guidance of NFPA72E-1974. The fire detectors are located in selected plant areas. Alarms from any of these detector zones will be indicated on the annunciator panel located in the main control room and in Switchgear Room1D.

Other equipment required by Appendix R is included in the scope of License Renewal, and is addressed in the various systems where the equipment resides. The Appendix R equipment included in the Fire Protection System includes the Alternate Shutdown Panels EC-150, -150A (panels and power supply - individual instruments belong to their respective systems). Appendix R FPS equipment also includes components associated with providing backup supply to auxiliary feedwater and critical service water. The primary coolant pump oil collection system function, which is required by 10CFR50 Appendix R, is included in the Fire Protection System for evaluation.

Portable fire extinguishers are provided at convenient and accessible locations. The extinguishing media are pressurized water, CO2 or dry chemical as appropriate for the service requirements of the area.

Water for the fire suppression system is supplied by one of three full-capacity fire pumps. Each fire pump is capable of providing water to the largest system demand plus fire hose streams in the area of demand. One fire pump is electrically driven; the other two are diesel engine-driven. Any fire pump will start automatically and can be manually started from the pump control panel. The diesel engine-driven fire pumps can also be manually started from the control room.

A jockey pump with local controls is provided to maintain the fire suppression system full and pressurized. The building structure has been designed and arranged to prevent the spread of fire and to ensure integrity of redundant safe shutdown systems and areas.

The description above results in some SSCs in this system being considered in-scope in accordance with 10 CFR 54.4(a)(1). Since some SSCs in the System are non-safety related and their failure could affect the capability of safety related SSCs to perform their safety function, they are considered to be in-scope in accordance with 10 CFR 54.4(a)(2). In addition, some SSCs are considered in-scope due to Fire Protection in accordance with 10 CFR 54.4(a)(3).

The boundaries of the in-scope portions of the Fire Protection System are described as follows: 1)Two diesel driven fire pumps, one motor driven fire pump, and all supporting components attached to the pumps and drivers. (The diesel driven fire pumps diesel engines are evaluated in the Emergency Power System), 2) Jockey pump and motor, 3) Discharge from three fire pumps and jockey pump to the fire main loop, 4) Fire main loop encircling the plant that includes fire hydrants FH-1 through FH-7, 5) Five legs that tap off the fire main loop to supply various sprinkler systems, deluge systems and hose stations within the auxiliary building, turbine building, intake structure, and hydrant stations, 6) Transformer deluge systems, 7) Backup supplies from the FPS to Auxiliary Feedwater, and Critical Service Water, 8) Fire dampers, 9) Primary Coolant Pump lubricating oil collection tanks, enclosures, drip pans and piping.

The original system scoping boundaries for license renewal included, where applicable, non-safety related components that could affect safety related components. The boundaries have been expanded in accordance with current NRC guidance for implementation of 10 CFR 54.4(a)(2). The expanded scoping boundary includes:

FPS non-safety related piping and components that are directly attached to safety related piping and components out to the first seismic anchor or equivalent anchor. Where base mounted components are credited as an anchor, they are also included in scope. Pipe supports for the newly added piping and components are also in scope. The new piping components are located in the Auxiliary Building and Turbine Building.

FPS non-safety related components containing steam and/or fluids that are located in an area that also contains a safety related component. The components are located in the Auxiliary Building and Turbine Building.

The portions of the Fire Protection System containing components subject to an AMR include accumulators, filter, strainers, sprinkler heads, fasteners, drip pans, oil enclosures, pipe and fittings, pumps and valves. The NSAS components brought into scope by the 10 CFR 54.4(a)(2) boundary expansion are subject to AMR.

System Function Listing

A comprehensive listing of functions associated with the Fire Protection System, or specific components contained in the system, is provided in the summary below.

System Function: ASP-01	Cri 1	Cri 2	Cri 3				
The Alternate Shutdown Panel provides alternate controls and indication during an Appendix R fire or when Control Room habitability is a concern.			FP	EQ	PTS	AT	SB
	X		X				

Comment: The alternate hot shutdown panels include EC-150 and EC-150A (panels and power supplies only - individual instruments belong to their respective systems)

The most critical individual functions provided are S/G pressure/level indication and auxiliary feedwater pump controls.

System Function: FPS-01	Cri 1	Cri 2	Cri 3				
Provide an alternate source of feedwater inventory for the auxiliary feedwater pumps.			FP	EQ	PTS	AT	SB
			X				

Comment: This function for this system does NOT include the portion of piping tying into the AFW pump suction that is ASME III class 3

System Function: FPS-02	Cri 1	Cri 2	Cri 3				
Provide an alternate source of critical service water inventory. (Can only supplement total supply required.)			FP	EQ	PTS	AT	SB
			X				

Comment: FPS can provide partial backup supply to Service Water as noted in FSAR Section 9.1.2.1

System Function: FPS-03 Fire suppression and Fire Barriers	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: This function includes fire suppression equipment, including fixed water sprinkler and deluge systems, portable fire extinguishers, hoses/reels/cabinets, fire barriers/penetration seals.

System Function: FPS-04 Provide alternate source of inventory to the spent fuel	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This is an auxiliary function of the FPS. Fuel pool normal makeup water is supplied from the Safety Injection and Refueling Water (SIRW) tank. A secondary backup supply of water is available from the fire system. This could be utilized to replenish the fuel pool inventory in the event of considerable loss of pool water. Therefore, this backup function is not considered within the scope of license renewal.

System Function: FPS-05 Fire detection, control and annunciation	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: Includes detectors cabling, and fire detector control panels.

System Function: FPS-FP This system contains components required by the Current Licensing Basis for fire protection	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: The FP system contains components required for appendix R safe shutdown. This function also includes the primary coolant pump oil collection system.

System Function: FPS-NSAS This system has components in scope of license renewal in accordance with 10CFR54.4(a)(2) requirements for non-safety affecting safety (NSAS).	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: In-scope of License Renewal to protect safety related components from flooding, seismic II/I failures, spray impingement.

FSAR Reference

Additional Fire Protection System details are provided in Section 9.6 and Table 9-10 of the FSAR.

Scoping Boundary Drawings

The scoping boundary drawings for the Fire Protection System are listed below:

LR-M-207, Sheet 2
LR-M-213
LR-M-214, Sheet 4
LR-M-214, Sheet 5
LR-M-216, Sheet 1
LR-M-216, Sheet 2
LR-M-216, Sheet 3
LR-M-218, Sheet 6
LR-M-221, Sheet 2

Components Subject to an AMR

The component groups for this system that require aging management review are addressed in Table 2.3.3-7 along with each Component Group's intended function(s).

Table 2.3.3-7 Fire Protection System

Component Group	Intended Function
Accumulators	Fluid Pressure Boundary
Fasteners	Fluid Pressure Boundary
Filters/Strainers	Fluid Pressure Boundary
Pipe & Fittings	Fluid Pressure Boundary
Pumps	Fluid Pressure Boundary
Sprinkler Heads	Fluid Pressure Boundary Spray Pattern
Valves & Dampers	Fluid Pressure Boundary

2.3.3.8 Fuel Oil System

System Description

The Fuel Oil System (FOS) is designed to provide storage of an adequate volume of fuel oil for accident conditions, to transfer fuel to the diesel engines or boilers at an adequate rate, and to stop delivery to the unit storage tank when filled. The primary loads on the fuel oil system are the two Emergency Diesel engines and the two Fire Pump engines.

The description above results in some SSCs in this system being considered in-scope in accordance with 10 CFR 54.4(a)(1). Since some of the SSCs in the system are non-safety related and their failure could affect the capability of safety related SSCs to perform their safety function, they are considered to be in scope in accordance with 10 CFR 54.4(a)(2). In addition, some SSCs are considered in-scope due to Fire Protection in accordance with 10 CFR 54.4(a)(3).

The boundaries of the in-scope portions of the Fuel Oil System are described as follows: 1) The tank, piping, valves and components from the fuel oil storage tank (T-10A), through the diesel oil transfer pumps, and to the emergency diesel generator day tank isolation valve, 2) The piping and valves from the fuel oil storage tank (T-10A), through the diesel oil transfer pumps, through the diesel fire pump tanks, and to the fire pump diesel drivers, and 3) The tank, piping, valves and components from fuel oil tank (T-926), through the fuel oil transfer pump to manual valve MV-FO119.

The original system scoping boundaries for license renewal included, where applicable, non-safety related components that could affect safety-related components. The boundaries have been expanded in accordance with current NRC guidance for implementation of 10 CFR 54.4(a)(2). The expanded scoping boundary includes:

FOS non-safety related piping and components that are directly attached to safety related piping and components out to the first seismic anchor or equivalent anchor. Where base mounted components are credited as an anchor, they are also included in scope. Pipe supports for the newly added piping and components are also in scope. The new piping components are located in the turbine building and feedwater purity building.

FOS non-safety related components containing steam and/or fluids that are located in an area that also contains a safety related component. The components types include accumulator, fasteners (to maintain the pressure boundary), piping and fittings. The components are located in the turbine building and feedwater purity building.

The portions of the Fire Protection System containing components subject to an AMR include accumulators, filters, level glasses, fasteners, pipe and fittings, pumps and valves. The NSAS components added into scope by the 10 CFR 54.4(a)(2) scope expansion are subject to AMR.

System Function Listing

A comprehensive listing of functions associated with the Fuel Oil System is provided in the summary below.

System Function: FOS-01	Cri 1	Cri 2	Cri 3				
Store a sufficient supply of fuel oil in storage tank T-10A, as defined in the Tech Specs and provide it to the Diesel Generator day tanks (T-25A and T-25B).			FP	EQ	PTS	AT	SB
	X						

Comment: The fuel oil system supplies fuel oil to the emergency diesel generators.

System Function: FOS-02	Cri 1	Cri 2	Cri 3				
Provide fuel oil to the Fire Protection System diesel pump day tanks (T-24 and T-40).			FP	EQ	PTS	AT	SB

Comment: The FOS provides a fuel oil makeup source to the diesel fire pump day tanks. This makeup capability is not a license renewal intended function. The equipment to supply fuel from the diesel fire pump day tanks to the diesel fire pump engines has a Fire Protection intended function which is addressed in System Function FOS-FP.

System Function: FOS-03	Cri 1	Cri 2	Cri 3				
Provide fuel oil to the Service Building Heating Boiler (M-950), Feedwater Purity Heating Boiler (M-901), Plant Heating Boiler (M-8), and the Radwaste Evaporator Heating Boiler (M-61).			FP	EQ	PTS	AT	SB

Comment: This intended function of the Fuel Oil System does not perform an license renewal intended function per 10 CFR 54.4 and is not in the boundary of license renewal.

System Function: FOS-04	Cri 1	Cri 2	Cri 3				
The system contains an alternate pumping system to supply fuel oil to the day tanks (T-25A/B) from T-926 for Appendix R safe shutdown.			FP	EQ	PTS	AT	SB

Comment: The system consists of an portable air driven diaphragm pump and rubber hose installed between T-926 and tanks T-25A/B per SOP-22, Attachment 5. Refer to System Function FOS-FP.

System Function: FOS-FP	Cri 1	Cri 2	Cri 3				
The system contains structures and/or components required by the current licensing basis for Fire Protection.			FP	EQ	PTS	AT	SB
			X				

Comment: The fuel oil system contains components that are required for “Appendix R” safe shutdown. This includes components required to supply fuel to the diesel fire pump engines from the day tanks.

System Function: FOS-NSAS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
This system has components in scope of license renewal in accordance with 10CFR54.4(a)(2) requirements for non-safety affecting safety (NSAS).		X					

Comment: The FOS system contains non-safety related components that are attached to safety related components and non-safety related components containing liquids located in an area that also contains safety related components

FSAR Reference

Additional System details are provided in Section 8.4 of the FSAR.

Scoping Boundary Drawings

The scoping boundary drawings for the Fuel Oil System are listed below:

LR-M-214, Sheet 1
LR-M-215, Sheet 1
LR-M-216, Sheet 1
LR-M-655, Sheet 1
LR-M-907

Components Subject to an AMR

The component groups for this system that require aging management review are addressed in Table 2.3.3-8 along with each Component Group's intended function(s).

Table 2.3.3-8 Fuel Oil System

Component Group	Intended Function
Accumulators	Fluid Pressure Boundary
Filters/Strainers	Fluid Pressure Boundary
Indicators/Recorders (level glasses)	Fluid Pressure Boundary
Fasteners	Fluid Pressure Boundary

Table 2.3.3-8 Fuel Oil System

Component Group	Intended Function
Pipe & Fittings	Fire Barrier Flow Restriction Fluid Pressure Boundary
Pumps	Fluid Pressure Boundary
Valves & Dampers	Fluid Pressure Boundary

2.3.3.9 Heating, Ventilation, and Air Conditioning System

System Description

The Heating, Ventilation, and Air Conditioning System (VAS) provides air flow to spaces in the Plant. Various supply and exhaust fan combinations provide ventilation air for breathing, heated air to prevent equipment freezing and for personnel comfort in cold weather, and cooled air to selected locations to remove heat from lights, equipment, etc. This system is a collection of independent ventilation subsystems, with the major ones being: 1) Control Room Heating, Ventilation and Air Conditioning (HVAC), 2) Containment Purge, 3) Engineered Safeguards Room HVAC, 4) Emergency Diesel Generator Room Fans, 5) Electrical Equipment Room HVAC, including Battery Room, 6) Fuel Handling Area Ventilation.

The Control Room HVAC subsystem provides conditioned air to the Control Room, the Technical Support Center (TSC), the Viewing Gallery and the Mechanical Equipment Room (MER). The subsystem has separate and redundant air handling units, air filtering units, condensing units, steam humidifiers, and continuous air monitors (CAMs). There are two normal outside air intakes, one associated with each of the air handling units. A single common remote emergency outside air intake serves both of the air filtering units. The Control Room HVAC subsystem also includes a smoke purge exhaust fan with duct and a toilet exhaust fan with duct.

The Containment Purge subsystem supplies air to the Air Room area of the Containment Building and provides an exhaust line that connects the Containment Building to the Main Exhaust Fans. The Containment Purge subsystem is mostly located in the Auxiliary Building, with only the air outlet and return suction connection are located in the Containment Building.

The Engineered Safeguard Rooms HVAC subsystem normally supplies ventilation air to the East and West Engineered Safeguard Rooms via the Radwaste Area HVAC subsystem. The ductwork to these rooms is automatically isolated if airborne radiation in the exhaust ductwork exceeds preset levels. Each room has one cooling coil (cooled by critical service water) with two fans that recirculate room air to remove the heating load in these rooms.

The Emergency Diesel Generator Room Fans provide ventilation air to remove heat generated by the diesel generators. Each room is supplied with air from the common plenum by separate intake fans. The exhaust air from each room exits through louvered discharge ducts in the diesel generator muffler enclosure. The fans are powered by the particular emergency diesel generator for which they provide cooling. Each room also includes a unit heater, which uses plant heating steam.

The Electrical Equipment, Switchgear, & Cable Spreading Room HVAC subsystem (includes Battery Rooms) draws outside air through a filter and heating coil and distributes the air to these spaces, which contain mainly electrical equipment. Air is circulated back to the supply fan suction or exhausted to atmosphere from these areas by a recirculation fan, depending on supply air temperature. The electrical equipment room part of the subsystem includes a cooling coil whose heat sink is non-critical service water. The North Electrical Penetration Room receives minimal ventilation from the 1D Switchgear room via the cable way to the penetration room, it has no forced ventilation. The South Electrical Penetration Room has a roof-mounted ventilator controlled by a wall-mounted thermostat. There is no significant heat source in either Electrical Penetration Room.

The Fuel Handling Area Ventilation subsystem provides the capability of filtering potential airborne radioactive particulates from the area of the spent fuel pool following a fuel handling accident or a fuel cask drop accident. During plant evolutions when the possibility for a fuel handling accident or fuel cask drop accident exist, the Fuel Handling Area Ventilation subsystem is configured such that all fans are stopped except one exhaust fan that is aligned to the emergency filter bank.

The Penetration and Fan Room HVAC subsystem was installed in conjunction with modifications to protect essential structures, systems and components from the effects of high energy line breaks. The system includes a supply filter and fan, and an exhaust filter and fan. It also ran new ductwork for the supply and return systems down to the Feedwater Penetration Room, and provided branch connections for the Main Steam Penetration Room and Fan Room elevations.

Steam for the various heating coils that are part of the above-described systems is provided by extraction steam from the low pressure turbine that is reduced to nominally 15 psig. When the main turbine is not operating, heating steam is supplied by two auxiliary heating boilers. The piping system associated with the steam source from the low pressure turbine is evaluated in HED system. The piping system associated with the heating boilers and the steam distribution system is evaluated in this system.

The description above results in some SSCs in this system being considered in-scope in accordance with 10 CFR 54.4(a)(1). Since some SSCs in the Heating, Ventilation, and Air Conditioning System are non-safety related and their failure could affect the capability of safety related SSCs to perform their safety function, they are considered to be in-scope in accordance with 10 CFR 54.4(a)(2). In addition, some SSCs are considered in-scope due to Environmental Qualification, Fire Protection, and Station Blackout in accordance with 10 CFR 54.4(a)(3).

The boundaries of the in-scope portions of the HVAC systems include the components and ducting of each subsystem as follows: 1) Control Room HVAC system, including the flow paths of Emergency mode operation and part of the Normal mode operation, 2) Fuel Handling Area HVAC System, including the original Fuel Pool HVAC System and the Fuel Pool HVAC System addition, 3) Engineered Safeguards Room (East and West) HVAC System, 4) Electrical Equipment, Switchgear, Cable Spreading & Battery Rooms HVAC System, 5) Emergency Diesel Generator Room HVAC System, 6) Containment Air Room Purge System, 7) Control Rod Drive Mechanism Ventilation system (Seismic), and 8) a portion of the Plant Heating Steam piping.

The original system scoping boundaries for license renewal included, where applicable, non-safety related components that could affect safety-related components. The boundaries have been expanded to incorporate additional non-safety related components in accordance with current NRC guidance for implementation of 10 CFR 54.4(a)(2). The expanded scoping boundary includes:

VAS non-safety related piping and components that are directly attached to safety related piping and components out to the first seismic anchor or equivalent anchor. Where base mounted components are credited as an anchor, they are also included in scope. Pipe supports for the newly added piping and components are also in scope. The new piping components are located in the Auxiliary Building and Containment.

VAS non-safety related components containing steam and/or fluids that are located in an area that also contains a safety related component. The components types include blower, ductwork, pipe, fittings, fasteners (to maintain the pressure boundary), filters, heaters, steam traps, and valves. The components are located in the Auxiliary Building and Turbine Building.

The portions of the Heating, Ventilation and Air Conditioning System containing components subject to an AMR include fans, duct, filters, heat exchangers, flexible connections, mufflers, fasteners, seals, pipe and fittings, traps, valves, and dampers. The NSAS components brought into scope by the boundary expansion for 10 CFR 54(a)(2) are subject to AMR.

System Function Listing

A comprehensive listing of functions associated with the Heating, Ventilation, and Air Conditioning System, or specific components contained in the system, is provided in the summary below.

System Function: CPG-01	Cri 1	Cri 2	Cri 3				
CONTAINMENT PURGE: Automatically isolate the containment isolation valves for the Containment Purge subsystem on CHP or CHR.			FP	EQ	PTS	AT	SB
	X						

Comment: Maintaining the containment Operable limits the leakage of fission product radioactivity from the containment to the environment. This function is associated with the valves and pipe required to ensure containment isolation in the Containment Purge subsystem. Components associated with the processing of electrical signals to initiate automatic isolation upon CHP and CHR are dispositioned under the Containment Isolation and Penetration (CIS) system. The penetrations are dispositioned within the Containment Structure evaluation.

System Function: CPG-02	Cri 1	Cri 2	Cri 3				
CONTAINMENT PURGE: Circulate air through charcoal filters to remove iodine from the containment atmosphere prior to containment entry for outages.			FP	EQ	PTS	AT	SB

Comment: As indicated in FSAR 9.8.2.4.15, this function is in service only during cold shutdown condition. The function of containment purge is not safety-related. The associated piping and components are involved in the safety-related function of Containment Isolation, which is covered in System Function CPG-01 above.

System Function: CRV-01 CONTROL ROOM HVAC: Maintain ambient temperature in the control room at acceptable levels.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: The Control Room HVAC subsystem provides temperature control for the control room during normal and emergency conditions. This subsystem is capable of removing sensible and latent heat loads from the control room, considering equipment heat loads and personnel occupancy requirements, to ensure equipment Operability. The normal operating mode is also required to be operable to support Appendix R Safe Shutdown.

System Function: CRV-02 CONTROL ROOM HVAC: Maintain acceptable air quality in the control room.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: The emergency mode operation with charcoal filtration and positive pressurization are credited in the radiological analysis in FSAR Section 14.24. The filtration provides a protected environment from which operators can control the plant following an uncontrolled release of radioactivity. Pressurization of the Control Room, including Technical Support Center, Viewing Gallery, and Mechanical Equipment Room (MER), prevents infiltration of unfiltered air from the surrounding areas.

System Function: CRV-03 CONTROL ROOM HVAC: Provide actuation and control signals for control room HVAC.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: The control room ventilation emergency mode of operation is actuated either by a containment high radiation signal or a containment high pressure signal, or manually from the control room. During emergency mode operation, the air handling units and the charcoal filter units of both trains are actuated automatically.

System Function: DGV-01 EMERGENCY DIESEL GENERATOR ROOM FANS: Maintain temperatures in the Emergency Diesel Generator Rooms.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: As indicated in FSAR Section 9.8.5.2.7, the reliable operation of these ventilation systems is considered essential to Plant safety. Analysis has demonstrated that room temperatures will exceed the design temperature in a matter of minutes without cooling. Note the unit heaters and the associated equipment are not required for this safety-related function since they are not essential for diesel generator operability.

System Function: ESV-01 ENGINEERED SAFEGUARDS ROOM HVAC: Maintain temperature in the engineered safeguards equipment rooms.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: FSAR 9.8.2.4.13 states that the room coolers and associated fans are required to be operable to support operability of the required equipment to support the accident analysis.

System Function: SCV-01 ELECTRICAL EQUIPMENT, SWITCHGEAR, & CABLE SPREADING ROOM HVAC: Maintain ambient temperatures in the Cable Spreading, Switchgear and Battery Rooms at acceptable levels to ensure operability of essential equipment within the rooms.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: FSAR 9.8.2.4.21 and 9.8.5.2.6 indicate that the cable spreading room, switchgear rooms and battery rooms are considered essential because they house the safe shutdown equipment which are considered important to safety. However, the ventilation system that service these areas are not safety grade. Ventilation testing showed that upon loss of normal ventilation, the operator has up to six hours to take corrective measures. Based on Appendix R safe shutdown analysis, this ventilation system is considered safe shutdown equipment.

System Function: SCV-02 ELECTRICAL EQUIPMENT, SWITCHGEAR, & CABLE SPREADING ROOM HVAC: Remove hydrogen gas from Battery rooms to prevent explosive atmosphere.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: Battery Room exhaust fans V-15A, V-15B and the exhaust ducting are not safety-related. The amount of time it takes for the battery room atmosphere to build up to 2% hydrogen, when the batteries are being equalized and ventilation is lost, is 1.8 hours. Based on Appendix R safe shutdown Logic Diagram, the Battery Room exhaust fans V-15A and V-15B are safe shutdown equipment.

System Function: VAS-01 FUEL HANDLING AREA HVAC: Circulate air through HEPA and charcoal filters to remove contamination and iodine from the fuel handling area atmosphere during fuel handling.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: As indicated in Tech Spec Bases 3.7.12, filtration of the fuel handling area atmosphere following a fuel cask drop on irradiated fuel assemblies with <90 days decay is required to maintain the offsite doses within the guidelines of 10 CFR 100.

System Function: VAS-02 RADWASTE AREA VENTILATION SYSTEM: Maintain ambient temperatures in the Auxiliary Building at acceptable levels.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This function of temperature control in the Radwaste Area Ventilation System is to support normal plant operation only. Also, in the transient analyses of radiological events in the Auxiliary Building as described in FSAR 14.20, 14.21, and 14.23, it is assumed the radionuclides are uncontrolled and released directly into the environment.

System Function: VAS-03 AFW PUMP ROOM HVAC: Maintain ambient temperatures in the AFW pump room at acceptable levels.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This function is applicable to AFW pumps P-8A and P-8B only, and is not a license renewal intended function. AFW pump P-8C is located in the Engineered Safeguards Room, which is covered by System Function ESV-01. As described in FSAR 9.8.5.2.4, tests were performed to verify the room can safely withstand a loss of ventilation units for a long period of time, at least 24 hours. The simple measure of opening a door would provide adequate ventilation for an indefinite period.

System Function: VAS-04 CONTAINMENT HIGH PRESSURE MONITORING: Provide containment pressure inputs to the Reactor Protective System for containment high pressure and to Engineered Safety Features Actuation for safety injection initiation.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: The Containment High Pressure trip provides a reactor trip in the event of a Loss of Coolant Accident (LOCA) or Main Steam Line Break (MSLB). The Containment High Pressure trip shares sensors with the Containment High Pressure sensing logic for Safety Injection (SIS), Containment Isolation, and Containment Spray. As indicated in Tech Spec B3.3.3, the SIS ensures acceptable consequences during Loss of Coolant Accident (LOCA) events, including steam generator tube rupture, and Main Steam Line Breaks (MSLBs) or Feedwater Line Breaks (FWLBs) (inside containment).

System Function: VAS-05 CONTAINMENT TEMPERATURE AND PRESSURE MONITORING: Provide containment temperature and pressure indication.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: The containment average air temperature and containment pressure are limited during normal operation to preserve the initial conditions assumed in the accident analyses for a Loss of Coolant Accident (LOCA) or Main Steam Line Break (MSLB). The containment temperature and pressure indications are required for proper monitoring of these parameters to ensure that initial conditions assumed in the analysis of containment response to a DBE are not violated.

System Function: VAS-06 INTAKE STRUCTURE VENTILATION SYSTEM: Maintain ambient temperatures in the intake structure at acceptable levels.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: Although service water pumps are located in the intake structure and considered important to safety, as described in FSAR 9.8.5.2.8, the room is not airtight allowing some limited convection to take place, and doors opening to outside are available providing sufficient air flow even with multiple fan failures. Screen House Roof fans are Seismic Class 2, which meets license renewal scoping Criterion 2.

System Function: VAS-07 CRDM VENTILATION: Provide forced air cooling to Control Rod Drive Mechanisms.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This function is to support normal operation only. It is not required following a reactor trip since the drive motors and clutches are de-energized.

System Function: VAS-08 PENETRATION AND FAN ROOM HVAC: Maintain ambient temperatures in the Penetration (Main Steam and Feedwater) and Fan Rooms.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: As discussed in FSAR 9.8.5.2.9, this function is not considered essential because the essential equipment in this area is qualified to survive a main steam line break within this area.

System Function: VAS-09 PENETRATION SLEEVE VENTILATION: Provides forced-air cooling in the air gap between the insulated pipe and the penetration sleeve of Main Steam and Feedwater penetrations.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Based on the analysis cited in FSAR 5.8.6.4, it was determined that no cooling is necessary for these high temperature penetrations with insulation. A temperature switch is mounted in each penetration to provide alarm indication “Steam/Feedwater Penetration High Temp” in the Control Room if any of these penetrations reaches alarming point.

System Function: VAS-10	Cri 1	Cri 2	Cri 3				
CONTAINMENT ISOLATION: Maintain containment integrity at penetrations associated with the Heating, Ventilating and Air Conditioning System, including containment purge lines, various instrument lines, and heating steam supply/return lines.			FP	EQ	PTS	AT	SB
	X						

Comment: Maintaining the containment integrity limits the leakage of fission product radioactivity from the containment to the environment. The function is associated with the valves and pipe required to ensure containment isolation. Components associated with the processing of electrical signals to initiate automatic isolation upon CHP and CHR are dispositioned under the Containment Isolation and Penetration (CIS) system. The penetrations are dispositioned within the Containment Structure evaluation.

System Function: VAS-EQ	Cri 1	Cri 2	Cri 3				
The system contains components required by the current licensing basis for Environmental Qualification.			FP	EQ	PTS	AT	SB
				X			

Comment: Includes components associated with Engineered Safeguards Room Coolers, containment pressure switches, containment purge isolation controls, etc.

System Function: VAS-FP	Cri 1	Cri 2	Cri 3				
The system contains structures and/or components required by the current licensing basis for Fire Protection.			FP	EQ	PTS	AT	SB
			X				

Comment: HVAC systems associated with Control Room, Engineered Safeguards Rooms, Cable Spreading Room, Switchgear Room, Battery Room, and EDG Rooms are required to support the design function of Safe Shutdown equipment.

System Function: VAS-NSAS	Cri 1	Cri 2	Cri 3				
This system has components in scope of license renewal in accordance with 10CFR54.4(a)(2) requirements for non-safety affecting safety (NSAS).			FP	EQ	PTS	AT	SB
		X					

Comment: The VAS system contains non-safety related components that are attached to safety related components and non-safety related components containing liquids located in an area that also contains safety related components.

System Function: VAS-SB	Cri 1	Cri 2	Cri 3				
The system contains structures and/or components required by the current licensing basis for Station Blackout (Loss of all AC power).			FP	EQ	PTS	AT	SB
							X

Comment: The components involved in SBO 4-hour coping are Containment pressure and temperature indicators and associated components.

HVAC would not be required for Control Room, Containment, AFW, cable spreading room, battery and switchgear rooms during the SBO coping period.

FSAR Reference

Additional Heating, Ventilation, and Air Conditioning System details are provided in Section 5.8, Section 6.5, Section 7.2, Section 7.4, Section 9.6, Section 9.8, Section 14.11, Section 14.19, Section 14.20, Section 14.21, Section 14.23, and Section 14.24 of the FSAR.

Scoping Boundary Drawings

The scoping boundary drawings for the Heating, Ventilation, and Air Conditioning System are listed below:

LR-M-208, Sheet 1
LR-M-208, Sheet 1A
LR-M-208, Sheet 1B
LR-M-212, Sheet 3
LR-M212, Sheet 5
LR-M-215, Sheet 1
LR-M-215, Sheet 1A
LR-M-218, Sheet 1
LR-M-218, Sheet 2
LR-M-218, Sheet 4
LR-M-218, Sheet 5
LR-M-218, Sheet 6
LR-M-218, Sheet 6A
LR-M-218, Sheet 7
LR-M-223, Sheet 2
LR-M-650, Sheet 2
LR-M-658, Sheet 1

Components Subject to an AMR

The component groups for this system that require aging management review are addressed in Table 2.3.3-9 along with each Component Group’s intended function(s).

Table 2.3.3-9 Heating, Ventilation, and Air Conditioning System

Component Group	Intended Function
Blowers Fans Compressor Vacuum	Fluid Pressure Boundary

Table 2.3.3-9 Heating, Ventilation, and Air Conditioning System

Component Group	Intended Function
Ductwork	Filtration Fluid Pressure Boundary
Filters/Strainers	Filtration Fluid Pressure Boundary
Heat Exchangers	Fluid Pressure Boundary Heat Transfer
Heaters, Electric	Fluid Pressure Boundary
Muffler (CRHVAC refrig. condensing units)	Fluid Pressure Boundary
Elastomers in Flex. Connections and Seals inside/outside of Containment	Fluid Pressure Boundary
CRHVAC Duct Silencer	Fluid Pressure Boundary
Fasteners	Fluid Pressure Boundary
Dampers	Fluid Pressure Boundary
Pipe & Fittings	Fluid Pressure Boundary Containment Isolation
Traps (Steam)	Fluid Pressure Boundary
Valves & Dampers	Fluid Pressure Boundary

2.3.3.10 Miscellaneous Gas System

System Description

The Miscellaneous Gas System (MGS) is a collection of all the compressed bottles and liquid storage of gases used in various plant process and equipment. The stored gases are nitrogen (liquid and gaseous), hydrogen, carbon dioxide (liquid), propane, helium, argon, acetylene, and air bottles (Air bottles are included in the Compressed Air System scoping/screening.). Each system has headers to attach the gaseous source, pressure regulators, monitoring gauges, valving and piping. The Appendix R evaluations identified the need for a backup to the air supply to air operated valves. This evaluation resulted in numerous nitrogen supply stations being strategically located throughout the plant.

The description above results in some SSCs in this system being considered in-scope in accordance with 10 CFR 54.4(a)(1). Since some SSCs in the Miscellaneous Gas System are non-safety related and their failure could affect the capability of safety related SSCs to perform their safety function, they are considered to be in-scope in accordance with 10 CFR 54.4(a)(2). In addition, some SSCs are considered in-scope due to Environmental Qualification, Fire Protection, and Station Blackout in accordance with 10 CFR 54.4(a)(3).

The boundaries of the in-scope portions of the Miscellaneous Gas System are described as follows: 1) The bulk nitrogen storage and the nitrogen gas supply bottles and stations that supply backup nitrogen to air-operated valves required to operate in order to support license renewal criteria, including Containment Isolation, 2) The backup high pressure air supply, 3) The Containment hydrogen monitoring system, and 4) The piping and valves from the compressed nitrogen bottles to the spent fuel pool gate, to various tank isolation valves, and to various control valves, including the nitrogen outboard Containment isolation valve CV-1358.

The original system scoping boundaries for license renewal included, where applicable, non-safety related components that could affect safety related components. The boundaries have been expanded in accordance with current NRC guidance for implementation of 10 CFR 54.4(a)(2). The expanded scoping boundary includes:

MGS non-safety related piping and components that are directly attached to safety related piping and components out to the first seismic anchor or equivalent anchor. Where base mounted components are credited as an anchor, they are also included in scope. Pipe supports for the newly added piping and components are also in scope. The new piping components are located in the Turbine Building, Auxiliary Building and Containment.

MGS non-safety related components containing steam and/or fluids that are located in an area that also contains a safety related component. The components types include piping, fittings, fasteners (to maintain the pressure boundary), and valves. The components are located in the Turbine Building, Auxiliary Building and Containment.

The portions of the Miscellaneous Gas System containing components subject to an AMR include accumulators, controllers, filters, heat exchangers, fasteners, manifold, monitor, pipe & fittings, pumps, valves, tanks and sample point. The NSAS components brought into scope by the 10 CFR 54.4(a)(2) boundary expansion are subject to AMR.

System Function Listing

A comprehensive listing of functions associated with the Miscellaneous Gas System is provided in the summary below.

System Function: HYM-01	Cri 1	Cri 2	Cri 3				
HYDROGEN MONITORING: Provide continuous hydrogen monitoring of the containment atmosphere during post accident conditions.			FP	EQ	PTS	AT	SB
	X						

Comment: Safety related panels EC-11A, EC-161, EC-162, EC-163, and EC-164 contain MGS components.

System Function: HYM-02	Cri 1	Cri 2	Cri 3				
HYDROGEN MONITORING: Automatically isolate the MGS (H2 monitor) containment isolation valves on Containment High Pressure and Containment High Radiation, and ensure the manual containment isolation valves can perform containment isolation.			FP	EQ	PTS	AT	SB
	X						

Comment: Containment isolation solenoid valves are safety related.

System Function: MGS-01	Cri 1	Cri 2	Cri 3				
MISCELLANEOUS GAS SYSTEM: Provide nitrogen cover gas to the Safety Injection Tanks (SITs) for driving head to ensure SIT injection on low Primary Coolant System pressure.			FP	EQ	PTS	AT	SB
	X						

Comment: Required for a safety related function.

System Function: MGS-02	Cri 1	Cri 2	Cri 3				
MISCELLANEOUS GAS SYSTEM: Provide nitrogen backup to the AFW pumps discharge valves.			FP	EQ	PTS	AT	SB
	X						

Comment: Required for a safety related function.

System Function: MGS-03 MISCELLANEOUS GAS SYSTEM: Provide nitrogen backup to the steam driven AFW pump steam inlet valves.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Required for a safety related function.

System Function: MGS-04 MISCELLANEOUS GAS SYSTEM: Provide nitrogen backup to the SWS supply and return valves to and from containment.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Required for a safety related function.

System Function: MGS-05 MISCELLANEOUS GAS SYSTEM: Provide nitrogen backup to the HPSI subcooling valves from CSS.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Required for a safety related function.

System Function: MGS-06 MISCELLANEOUS GAS SYSTEM: Provide nitrogen backup to the instrument air containment isolation valves.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Required for a safety related function.

System Function: MGS-07 MISCELLANEOUS GAS SYSTEM: Provide nitrogen backup to the containment spray valves.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Required for a safety related function.

System Function: MGS-08 MISCELLANEOUS GAS SYSTEM: Provide nitrogen backup to the SIRW tank recirculation valves.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Required for a safety related function.

System Function: MGS-09 MISCELLANEOUS GAS SYSTEM: Provide nitrogen backup to the atmospheric steam dump valves.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: MGS also supplies motive pressure for selected AFW valves and steam to AFW. Also, see MGS-FP below.

System Function: MGS-10 MISCELLANEOUS GAS SYSTEM: Provide carbon dioxide purge of main generator.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This system function does not meet any license renewal criterion.

System Function: MGS-11	Cri 1	Cri 2	Cri 3				
MISCELLANEOUS GAS SYSTEM: Provide air backup to HPSI pump P-66B discharge valve to Train 2.			FP	EQ	PTS	AT	SB
	X						

Comment: Normally isolated, seismically safety related valve.

System Function: MGS-12	Cri 1	Cri 2	Cri 3				
MISCELLANEOUS GAS SYSTEM: Provide containment isolation boundary for North and South containment electrical penetrations.			FP	EQ	PTS	AT	SB

Comment: Provides an inert atmosphere to minimize the potential for corrosion by providing a dry nitrogen blanket to prevent condensed moisture buildup in the penetration canisters. Does not support a license renewal criterion.

System Function: MGS-FP	Cri 1	Cri 2	Cri 3				
The system contains structures and/or components required by the current licensing basis for Fire Protection.			FP	EQ	PTS	AT	SB
			X				

Comment: Provides backup pressure source for valve operations should an Appendix R fire occur. Also, see MGS-09 above.

System Function: MGS-SBO	Cri 1	Cri 2	Cri 3				
The system contains structures and/or components required by the current licensing basis for Station Blackout (Loss of all AC power).			FP	EQ	PTS	AT	SB
							X

Comment: Backup nitrogen is provided to the atmospheric steam dump valves and Bottled Nitrogen Gas Backup Stations 1 and 2 provide motive pressure for AFW valve operation to support recovery from an SBO event.

System Function: MSG-EQ	Cri 1	Cri 2	Cri 3				
The system contains structures and/or components required by the current licensing basis for Environmental Qualification.			FP	EQ	PTS	AT	SB
				X			

Comment: EQ components are present in the hydrogen analyzer portion of the MGS system.

System Function: MSG-NSAS	Cri 1	Cri 2	Cri 3				
This system has components in scope of license renewal in accordance with 10CFR54.4(a)(2) requirements for non-safety affecting safety (NSAS).			FP	EQ	PTS	AT	SB
		X					

Comment: The MGS contains non-safety related components that are attached to safety related components and are located over safety related equipment.

FSAR Reference

Additional Miscellaneous Gas System details are provided in Section 5.8, Section 6.1, Section 6.7, Section 7.4, Section 8.5, and Section 9.9 of the FSAR.

Scoping Boundary Drawings

The scoping boundary drawings for the Miscellaneous Gas System are listed below:

LR-M-201, Sheet 2
LR-M-202, Sheet 1A
LR-M-203, Sheet 1
LR-M-205, Sheet 2
LR-M-206, Sheet 1
LR-M-206, Sheet 1A
LR-M-206, Sheet 1B
LR-M-208, Sheet 1B
LR-M-210, Sheet 1
LR-M-210, Sheet 1A
LR-M-210, Sheet 2
LR-M-211, Sheet 1
LR-M-211, Sheet 2
LR-M-211, Sheet 3
LR-M-212, Sheet 3
LR-M-219, Sheet 2
LR-M-222, Sheet 1
LR-M-222, Sheet 2
LR-M-222, Sheet 3
LR-M-224, Sheet 1
LR-M-224, Sheet 2
LR-M-650, Sheet 1

Components Subject to an AMR

The component groups for this system that require aging management review are addressed in Table 2.3.3-10 along with each Component Group's intended function(s).

Table 2.3.3-10 Miscellaneous Gas System

Component Group	Intended Function
Accumulators	Fluid Pressure Boundary
Filters/Strainers	Fluid Pressure Boundary
Misc Mechanical (fasteners, manifold, monitor)	Fluid Pressure Boundary
Pipe & Fittings	Fluid Pressure Boundary
Valves & Dampers	Fluid Pressure Boundary
Tanks	Fluid Pressure Boundary

2.3.3.11 Radwaste System

System Description

The Radwaste System (RWS) contains two major subsystems: 1) Liquid Radwaste and 2) Solid Radwaste. The systems are designed and operated to achieve a near-zero discharge to the environment.

The Liquid Radioactive Waste System is divided into three sections: (a) the clean waste section which processes high-activity, high-purity (low solids) liquid waste, (b) the dirty waste section which processes low-activity, low-purity (high solids) liquid waste and (c) the laundry waste.

The Solid Waste Management System is designed to collect, process, package and store for future offsite disposal low-level liquid and solid wastes, evaporator concentrates, spent ion-exchange resins and assorted solid wastes. Solid wastes are, as applicable, stored on site, shipped to contractors for incineration, immobilized by the addition of additives, and (resins) dewatered and compacted.   There are two onsite storage/processing facilities.

The description above results in some SSCs in this system being considered in-scope in accordance with 10 CFR 54.4(a)(1). Since some SSCs in the Radwaste System are non-safety related and their failure could affect the capability of safety related SSCs to perform their safety function, they are considered to be in-scope in accordance with 10 CFR 54.4(a)(2). In addition, some SSCs are considered in-scope due to Environmental Qualification in accordance with 10 CFR 54.4(a)(3).

The boundaries of the in-scope portions of the Radwaste System are described as follows: 1) The piping, valve and strainer located downstream of manual valve MV-CRW175 to the Safety Injection and Refueling Water tank, 2 Manual valve MV-CRW819, 3) Containment isolation components to and from Clean Waste Receiver Tanks for Containment penetrations, 4) Containment isolation components to and from Primary System Drain Tank for Containment penetrations, 5) Containment isolation components associated with containment sump level instrumentation for Containment penetrations, 6) Flood control for the engineered safeguards rooms, emergency diesel generator rooms, auxiliary feedwater room, including floor/equipment drains and pipes, sump pits, sump pumps, and discharge piping to route flood water out of the rooms, and 7) The utility water storage tank.

The original system scoping boundaries for license renewal included, where applicable, non-safety related components that could affect safety-related components. The boundaries have been expanded in accordance with current

NRC guidance for implementation of 10 CFR 54.4(a)(2). The expanded scoping boundary includes:

RWS non-safety related piping and components that are directly attached to safety related piping and components out to the first seismic anchor or equivalent anchor. Where base mounted components are credited as an anchor, they are also included in scope. Pipe supports for the newly added piping and components are also in scope. The new piping components are located in the Turbine Building, Auxiliary Building and Containment Building.

RWS non-safety related components containing steam and/or fluids that are located in an area that also contains a safety related component. The components are located in the Turbine Building, Auxiliary Building and Containment Building.

The portions of the Radwaste System containing components subject to an AMR include accumulators, filters, strainers, heat exchangers, fasteners, pipe and fittings, pumps, and valves. The NSAS components brought into scope by the 10 CFR 54.4(a)(2) scoping boundary expansion are subject to AMR.

System Function Listing

A comprehensive listing of functions associated with the Radwaste System, or specific components contained in the system, is provided in the summary below.

System Function: LRW-01 LIQUID RADIOACTIVE WASTE: Provide for collection, monitoring, filtering, processing, storage, reuse, and controlled release of liquid radioactive waste (LRW).	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: The radioactive waste treatment system was designed so that discharge of radioactivity to the environment is in accordance with the requirements of 10CFR20 and Appendix I to 10CFR50. However, release of liquid radwaste is not required for safe plant shutdown, does not support any safety related equipment, and is not related to any of the five regulated events.

The following tasks of the LRW system are not required for safe plant shutdown, do not support any safety related equipment, and are not related to any of the five regulated events: storage of water discharged from Steam Generators following a Steam Generator Tube Rupture event; collection, monitoring and processing of dirty liquid radwaste into concentrate for disposal by the solid radwaste system and into distillate utility water for reuse; collection, filtering and monitoring of laundry waste; collection, monitoring and processing of clean LRW.

Therefore, this function is not within the scope of license renewal.

System Function: LRW-02	Cri 1	Cri 2	Cri 3				
LIQUID RADIOACTIVE WASTE: Automatically isolate the LRW containment isolation valves on CHP and CHR. Also ensure manual containment isolation valves can perform containment isolation.			FP	EQ	PTS	AT	SB
	X						

Comment: This function is associated with the valves and pipe required to ensure containment isolation in the LRW system. Components associated with the processing of electrical signals to initiate automatic isolation upon CHP and CHR are dispositioned under the Containment Isolation and Penetration (CIS) system. The penetrations are dispositioned within the Containment Structure evaluation.

System Function: LRW-03	Cri 1	Cri 2	Cri 3				
LIQUID RADIOACTIVE WASTE: Override automatic operation of engineered safeguards sump pumps on a high sump level on receipt of a CHR signal.			FP	EQ	PTS	AT	SB

Comment: See ESF-04 in ESS System. This is not an intended function of Liquid Radwaste.

System Function: LRW-04	Cri 1	Cri 2	Cri 3				
Store water discharged from the Steam Generators, when responding to a Steam Generator Tube Rupture, in the miscellaneous waste holdup tanks.			FP	EQ	PTS	AT	SB

Comment: Storage of water discharged from the Steam Generators following a Steam Generator Tube Rupture event is not required for safe plant shutdown, does not support any safety related equipment, and is not related to any of the five regulated events. Therefore, this function is not within the scope of license renewal.

System Function: LRW-EQ	Cri 1	Cri 2	Cri 3				
The system contains components required by the current design basis for Environmental Qualification.			FP	EQ	PTS	AT	SB
				X			

Comment: The RWS system contains components that are required per 10 CFR 50.49.

System Function: LRW-NSAS	Cri 1	Cri 2	Cri 3				
This system has components in scope of license renewal in accordance with 10CFR54.4(a)(2) requirements for non-safety affecting safety (NSAS).			FP	EQ	PTS	AT	SB
		X					

Comment: In-scope of License Renewal to protect safety related components from spray, flooding and seismic II/I considerations. Specifically, this includes a function to route flood-water out of the Engineered Safeguards rooms, Emergency Diesel Generator rooms, and Auxiliary Feedwater pump room to the sump via floor and equipment drains, and/or sump pumps.

System Function: SRW-01 SOLID RADIOACTIVE WASTE: Collect, process, package and store dirty radwaste evaporator concentrates, spent ion-exchange resins and assorted solid wastes.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Collection, processing, packaging and storage of dirty liquid radwaste concentrates, spent ion-exchanger resins and assorted solid wastes is not required for safe plant shutdown, does not support any safety related equipment, and is not related to any of the five regulated events. Therefore, this function is not within the scope of license renewal.

FSAR Reference

Additional Radwaste System details are provided in Section 1.4.8, Section 6.7, Section 7.3, Section 11.2, and Section 11.4 of the FSAR.

Scoping Boundary Drawings

The scoping boundary drawings for the Radwaste System are listed below:

LR-M-201, Sheet 1
LR-M-201, Sheet 2
LR-M-202, Sheet 1
LR-M-202, Sheet 1A
LR-M-202, Sheet 1B
LR-M-203, Sheet 1
LR-M-204, Sheet 1
LR-M-204, Sheet 1A
LR-M-204, Sheet 1B
LR-M-209, Sheet 3
LR-M-210, Sheet 1
LR-M-210, Sheet 1A
LR-M-210, Sheet 1B
LR-M-210, Sheet 1C
LR-M-210, Sheet 2
LR-M-211, Sheet 1
LR-M-211, Sheet 2
LR-M-211, Sheet 3
LR-M-218, Sheet 6A
LR-M-219, Sheet 1A
LR-M-219, Sheet 2
LR-M-221, Sheet 2
LR-M-223, Sheet 1
LR-M-223, Sheet 1B
LR-M-224, Sheet 1
LR-M-227, Sheet 1
LR-M-650, Sheet 1
LR-M-650, Sheet 1A
LR-M-650, Sheet 1B
LR-M-651, Sheet 1A
LR-M-651, Sheet 1B
LR-M-652, Sheet 1
LR-M-654
LR-M-655, Sheet 2
LR-M-657, Sheet 1

Components Subject to an AMR

The component groups for this system that require aging management review are addressed in Table 2.3.3-11 along with each Component Group’s intended function(s).

Table 2.3.3-11 Radwaste System

Component Group	Intended Function
Accumulators	Flood Protection Fluid Pressure Boundary
Demineralizer	Fluid Pressure Boundary
Fasteners	Fluid Pressure Boundary
Filters/Strainers	Fluid Pressure Boundary
Heat Exchangers	Fluid Pressure Boundary
Pipe & Fittings	Flood Protection Fluid Pressure Boundary
Pumps	Flood Protection Fluid Pressure Boundary
Valves & Dampers	Flood Protection Fluid Pressure Boundary

2.3.3.12 Service Water System

System Description

The Service Water System (SWS) supplies Lake Michigan water as the cooling medium (Ultimate Heat Sink) for removal of waste heat from the nuclear and steam plant auxiliary systems during normal, shutdown, or emergency conditions.

Three half-capacity electric motor-driven pumps draw screened and intermittently chlorinated Lake Michigan water from the Intake Structure. Each service water pump discharges through a simplex strainer into a common header. The common header has a full-capacity takeoff at each end, which supplies critical plant systems. A third takeoff at one end of the common header supplies the noncritical auxiliary systems.

The two critical service water (CSW) lines run underground by different paths from the Intake Structure to the Auxiliary Building. Each line supplies cooling water to one set of the redundant components including emergency diesel generator lube oil and jacket water coolers, a control room air-conditioning unit, an air compressor after-cooler and an engineered safeguards room cooler. In addition, Train A supplies cooling water to the component cooling water heat exchangers while Train B supplies cooling water to the containment air coolers.

The noncritical service water (NSW) header is isolated on a Safety Injection Signal, thus ensuring that all available service water is routed to the critical systems.

The Service Water System, also includes the Ultimate Heat Sink (UHS) subsystem. Basically, UHS subsystem of SWS consists of the components that take water from Lake Michigan to the suction of the SWS system (intake crib, intake structure, and connecting pipe), take the water from the SWS system and discharges it to the lake (discharge structure), and transfer water from the discharge structure to the intake structure (P-5 and associated components) for the purpose of deicing or supplying alternate SWS supply water should the intake crib collapse.

The description above results in some SSCs in this system being considered in-scope in accordance with 10 CFR 54.4(a)(1). Since some SSCs in the Service Water System are non-safety related and their failure could affect the capability of safety related SSCs to perform their safety function, they are considered to be in-scope in accordance with 10 CFR 54.4(a)(2). In addition, some SSCs are considered in-scope due to Environmental Qualification and Fire Protection in accordance with 10 CFR 54.4(a)(3).

The boundaries of the in-scope portions of the Service Water System are described as follows: (1) The piping systems from the intake structure, including the nominal 3000-feet long piping from the intake crib to the intake structure, through the service water pumps, through critical headers A & B, and return to the discharge structure, (2) the non-safety related warm water recirculation pump and its discharge piping and valves, (3) the non-safety related screen wash pump and its discharge piping and valves, including the spray nozzles and screens, (4) a section of the non-critical header located downstream of the non-critical service water isolation valve for flood protection, (5) piping and components used to provide makeup water to the spent fuel pool, (6) piping and components used to provide alternate water to the Auxiliary Feedwater System, (7) piping and components used to provide alternate water to the Engineered Safeguards pumps seal cooling, and (8) the piping and components that provide cooling to the Compressed Air System aftercoolers credited for Appendix R. It is noted that the de-icing function of the warm water recirculating pump and the trash removal function of the traveling screens and screen wash pump are conservatively included in scope to ensure the safety-related SWS functions are maintained.

The original system scoping boundaries for license renewal included, where applicable, non-safety related components that could affect safety related components. The boundaries have been expanded in accordance with current NRC guidance for implementation of 10 CFR 54.4(a)(2). The expanded scoping boundary includes:

SWS non-safety related piping and components that are directly attached to safety related piping and components out to the first seismic anchor or equivalent anchor. Where base mounted components are credited as an anchor, they are also included in scope. Pipe supports for the newly added piping and components are also in scope. The new piping components are located in the screen house, turbine building, and Containment

SWS non-safety related components containing steam and/or fluids that are located in an area that also contains a safety related component. The components are located in the turbine building and auxiliary building.

The portions of the Service Water System containing components subject to an AMR include accumulator, heat exchanger, filters, strainers, spray nozzles, flex connections, instrument valve assemblies, fasteners, pipe and fittings, pumps, and valves. The NSAS components brought into scope by the 10 CFR 54.4(a)(2) boundary expansion are subject to AMR.

System Function Listing

A comprehensive listing of functions associated with the Service Water System, or specific components contained in the system, is provided in the summary below.

System Function: CSW-01	Cri 1	Cri 2	Cri 3				
Circulate cooling water through safety related loads and heat exchangers.			FP	EQ	PTS	AT	SB
	X						

Comment: In boundary of License Renewal to supply cooling water to safety related components.

System Function: CSW-02 Upon receipt of ESF actuation signals, start idle SWS pumps and reposition valves to circulate water through essential loads/heat exchangers and isolate non-essential loads/heat exchangers.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: In boundary of License Renewal due to being a response to an Safety Injection actuation signal.

System Function: CSW-03 Supply alternate water supply to the Auxiliary Feedwater System.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: In boundary of License Renewal due to supply a backup water source to Auxiliary Feedwater Pump P-8C.

System Function: CSW-04 Provide alternate cooling to the Engineered Safeguards System pumps.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: In boundary of License Renewal due to providing a alternate seal cooling water source to Engineered Safeguards pumps.

System Function: CSW-05 Isolate the service water system to and from containment.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: This function is associated with the valves and pipe required to ensure containment isolation in the Critical Service Water system. Components associated with the processing of electrical signals to initiate automatic isolation upon CHP and CHR are dispositioned under the Containment Isolation and Penetration (CIS) system. The penetrations are dispositioned within the Containment Structure evaluation.

System Function: CSW-06 Provide isolation of non-critical service water from critical service water.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: In boundary of License Renewal due to isolating the Non-critical Service Water header.

System Function: NSW-01 Circulate cooling water through non-essential loads and heat exchangers during normal operations.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This system function is not in boundary of License Renewal due to not performing any Criterion 1, 2 or 3 functions per 10 CFR 54.

System Function: NSW-02	Cri 1	Cri 2	Cri 3				
Circulate cooling water through non-essential loads and heat exchangers during emergency operating conditions following restoration of Noncritical Service Water following an Safety Injection Actuation Signal.			FP	EQ	PTS	AT	SB

Comment: This system function is not required for safe shutdown of the plant. The function can be used at the discretion of plant operators. Therefore, this system function is not in the boundary of License Renewal.

System Function: SWS-EQ	Cri 1	Cri 2	Cri 3				
The system contains components required by the current licensing basis for Environmental Qualification.			FP	EQ	PTS	AT	SB
				X			

Comment: The SWS system contains components that are required for 10 CFR 50.49 requirements.

System Function: SWS-FP	Cri 1	Cri 2	Cri 3				
The system contains structures and/or components required by the current licensing basis for Fire Protection.			FP	EQ	PTS	AT	SB
			X				

Comment: The SWS system contains components that are required for Appendix R safe shutdown.

System Function: SWS-NSAS	Cri 1	Cri 2	Cri 3				
This system has components in scope of license renewal in accordance with 10CFR54.4(a)(2) requirements for non-safety affecting safety (NSAS).			FP	EQ	PTS	AT	SB
		X					

Comment: The SWS system contains non-safety related components that are attached to safety related components and non-safety related components containing liquids located in an area that also contains safety related components.

System Function: UHS-01	Cri 1	Cri 2	Cri 3				
Maintain adequate level in the intake structure.			FP	EQ	PTS	AT	SB
	X						

Comment: The UHS performs two principal safety functions: 1) dissipation of decay heat after reactor shutdown and during cool down and 2) dissipation of decay heat after an accident. Additionally, the UHS provides cooling water for critical equipment needed to cool down the plant and maintain it in a shutdown mode. Also, the screen wash system is in-scope of license renewal to provide trash removal from intake water.

System Function: UHS-02 Provide an emergency backup water supply to the intake structure via the warm water recirculation pump P-5.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Per FSAR 9.1.2.1, in the event that water is not available to the intake structure, makeup water may be supplied via the warm water recirculation pump P-5, if it is available. The warm water recirculation pump's capability to provide water to the intake structure is an original design feature installed to mitigate circulation water system icing (see function UHS-03), and is not intended or required to provide a safety-related capability. Thus the SWS backup supply water function is not in-scope of license renewal.

System Function: UHS-03 Provides warm water flow (P-5) to prevent ice build-up on the trash racks and screens.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: The warm water recirculation pump (P-5) was designed to provide an adequate flow to the intake structure to prevent ice build-up.

FSAR Reference

Additional Service Water System details are provided in Section 9.1 of the FSAR.

Scoping Boundary Drawings

The scoping boundary drawings for the Service Water System are listed below:

LR-M-207, Sheet 1A
LR-M-207, Sheet 2
LR-M-208, Sheet 1
LR-M-208, Sheet 1A
LR-M-208, Sheet 1B
LR-M-209, Sheet 2
LR-M-209, Sheet 3
LR-M-212, Sheet 1
LR-M-213
LR-M-214, Sheet 1
LR-M-218, Sheet 1
LR-M-218, Sheet 4
LR-M-226, Sheet 1A
LR-M-653, Sheet 3
LR-M-658, Sheet 1
LR-C-24
LR-C-25

Components Subject to an AMR

The component groups for this system that require aging management review are addressed in Table 2.3.3-12 along with each Component Group’s intended function(s).

Table 2.3.3-12 Service Water System

Component Group	Intended Function
Accumulator	Fluid Pressure Boundary
Fasteners in Containment	Fluid Pressure Boundary
Fasteners Not in Containment	Fluid Pressure Boundary
Filters/Strainers	Filtration Fluid Pressure Boundary
Heat Exchanger	Fluid Pressure Boundary
Pipe & Fittings	Flow Restriction Fluid Pressure Boundary
Pumps	Fluid Pressure Boundary
Traveling Screen Spray Nozzles	Spray Pattern
Valves & Dampers	Fluid Pressure Boundary

2.3.3.13 Shield Cooling System

System Description

The reactor Shield Cooling System (SCS) is designed to remove heat from the biological shield surrounding the reactor vessel thereby limiting the thermal stresses in the structural concrete. It is not a safety-related system.

The system is designed to maintain structural concrete temperature below 165 degrees F. The system assures the concrete in the reactor cavity does not overheat and develop excessive thermal stress. The Shield Cooling System is a closed loop system consisting of two full-capacity sets of cooling coils, two full-capacity pumps, a heat exchanger, a surge tank, associated piping, valves, instrumentation and controls.

Each set of shield cooling coils is composed of individual cooling coils embedded in the concrete shield. The distance between the inner surface of the concrete and the centerline of the cooling coils is three inches. The closed loop system transfers heat to the Component Cooling Water System by means of the shield cooling heat exchanger. Demineralized water with a corrosion inhibitor is used in the shield cooling loop.

The description above results in some SSCs in this system being considered in-scope in accordance with 10 CFR 54.4(a)(1). Since some SSCs in the Shield Cooling System are non-safety related and their failure could affect the capability of safety related SSCs to perform their safety function, they are considered to be in-scope in accordance with 10 CFR 54.4(a)(2). In addition, some SSCs are considered in-scope due to Environmental Qualification in accordance with 10 CFR 54.4(a)(3).

The boundaries of the in-scope portions of the Shield Cooling System are described as follows: (1) The makeup water piping and valves from Containment isolation check valve CK-CD401, through control valve CV-0939 and to the Containment penetration, and 2) components that are safety related due to being seismic.

The original system scoping boundaries for license renewal included, where applicable, non-safety related components that could affect safety related components. The boundaries have been expanded in accordance with current NRC guidance for implementation of 10 CFR 54.4(a)(2). The expanded scoping boundary includes:

SCS non-safety related piping and components that are directly attached to safety related piping and components out to the first seismic anchor or equivalent anchor. Where base mounted components are credited as an anchor, they are also included in scope. Pipe supports for the newly added piping and components are also in scope. The new piping components are located in the Containment Building and Auxiliary Building.

SCS non-safety related components containing steam and/or fluids that are located in an area that also contains a safety related component. The components types include an accumulator, filters, fasteners (to maintain the pressure boundary), piping, fittings and valves. The components are located in the Containment Building and Auxiliary Building.

The portions of the Shield Cooling System containing components subject to an AMR include accumulators, filters, fasteners, pipe and fittings, and valves. The NSAS components brought into scope by the 10 CFR 54.4(a)(2) boundary expansion are subject to AMR.

System Function Listing

A comprehensive listing of functions associated with the Shield Cooling System is provided in the summary below.

System Function	Cri 1	Cri 2	Cri 3 FP	Cri 3 EQ	Cri 3 PTS	Cri 3 AT	Cri 3 SB
System Function: SCS-01 SHIELD COOLING SYSTEM: Remove heat from the biological shield surrounding the reactor vessel.							

Comment: This system function does not support a license renewal criterion.

System Function	Cri 1	Cri 2	Cri 3 FP	Cri 3 EQ	Cri 3 PTS	Cri 3 AT	Cri 3 SB
System Function: SCS-02 SHIELD COOLING SYSTEM: Automatically isolate the containment isolation valve for the shield cooling surge tank fill line on CHP or CHR.	X						

Comment: The only portion of SCS that performs a safety related function is the containment isolation function for the surge tank fill line.

System Function	Cri 1	Cri 2	Cri 3 FP	Cri 3 EQ	Cri 3 PTS	Cri 3 AT	Cri 3 SB
System Function: SCS-EQ The system contains components required by the current licensing basis for Environmental Qualification.				X			

Comment: The SCS contains equipment in compliance with the EQ requirements of 10CFR50.49.

System Function	Cri 1	Cri 2	Cri 3 FP	Cri 3 EQ	Cri 3 PTS	Cri 3 AT	Cri 3 SB
System Function: SCS-NSAS This system has components in scope of license renewal in accordance with 10CFR54.4(a)(2) requirements for non-safety affecting safety (NSAS).		X					

Comment: In-scope of License Renewal to protect safety related components from spray, flooding and seismic II/I considerations.

FSAR Reference

Additional Shield Cooling System details are provided in Section 9.2 of the FSAR.

Scoping Boundary Drawings

The scoping boundary drawings for the Shield Cooling System are listed below:

LR-M-221, Sheet 1

Components Subject to an AMR

The component groups for this system that require aging management review are addressed in Table 2.3.3-13 along with each Component Group's intended function(s).

Table 2.3.3-13 Shield Cooling System

Component Group	Intended Function
Accumulators	Fluid Pressure Boundary
Fasteners	Fluid Pressure Boundary
Filters/Strainers	Fluid Pressure Boundary
Pipe & Fittings	Fluid Pressure Boundary
Valves & Dampers	Fluid Pressure Boundary

2.3.3.14 Spent Fuel Pool Cooling System

System Description

The Spent Fuel Pool Cooling (SFP) System removes decay heat from spent fuel stored in the spent fuel pool. The System is a closed loop system consisting of two pumps, a heat exchange unit consisting of two heat exchangers in series, a bypass filter, a bypass demineralizer, a booster pump, piping, valves and instrumentation.

The description above results in some SSCs in this system being considered in-scope in accordance with 10 CFR 54.4(a)(1). Since some SSCs in the Spent Fuel Pool Cooling System are non-safety related and their failure could affect the capability of safety related SSCs to perform their safety function, they are considered to be in-scope in accordance with 10 CFR 54.4(a)(2).

The boundaries of the in-scope portions of the Spent Fuel Pool Cooling system are described as follows: 1) The SFP cooling loop from the spent fuel pool, through the fuel pool cooling pumps, the spent fuel pool heat exchangers (tube side), the fuel pool recirculation booster pump, the fuel pool demineralizer, and return to the spent fuel pool; 2) the fuel transfer tube isolation valve; and 3) the temporary Fire Protection System connection for makeup water to the spent fuel pool.

The original system scoping boundaries for license renewal included, where applicable, non-safety related components that could affect safety-related components. The scoping boundaries have been expanded in accordance with current NRC guidance for implementation of 10 CFR 54.4(a)(2). The expanded scoping boundary includes:

SFP non-safety related piping and components that are directly attached to safety related piping and components out to the first seismic anchor or equivalent anchor. Where base mounted components are credited as an anchor, they are also included in scope. Pipe supports for the newly added piping and components are also in scope. The new piping components are located in the Auxiliary Building and Containment Building.

SFP non-safety related components containing steam and/or fluids that are located in an area that also contains a safety related component. The components types include accumulators, filters, fasteners (to maintain the pressure boundary), pipe, fittings and valves. The components are located in the Auxiliary Building and Containment Building.

The portions of the Spent Fuel Pool Cooling System containing components subject to an AMR include accumulators, demineralizers, filters, fasteners, pipe & fittings, pumps, heat exchangers and valves. The NSAS components added into scope by the 10 CFR 54.4(a)(2) scope expansion are subject to AMR.

System Function Listing

A comprehensive listing of functions associated with the Spent Fuel Pool Cooling System, or specific components contained in the system, is provided in the summary below.

System Function: SFP-01	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
SPENT FUEL POOL COOLING: Remove decay heat from fuel stored in the spent fuel pool and cool the reactor cavity water during spent fuel transfer.	X						

Comment: The spent fuel pool cooling system is in boundary of License Renewal for removing decay heat from the fuel stored in the spent fuel pool (normal operation), thereby preventing boiling of the spent fuel pool and uncovering the spent fuel bundles, which could result in exceeding the 10 CFR 100 limits.

Cooling the reactor cavity during refueling operation does not meet the requirements of 10 CFR 54.4.

System Function: SFP-02 SPENT FUEL POOL COOLING: Provide alternate shutdown cooling.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This function provides an alternate means of cooling the reactor core with SFP cross-tied to the refueling cavity if Shutdown Cooling is lost. This option is described in an off normal procedure to provide an alternate method to mitigate the event of Loss of Shutdown Cooling. However, this function is not described in FSAR and is not credited in design basis transients.

System Function: SFP-03 SPENT FUEL POOL COOLING: Maintain SFP boron concentration at or greater than required concentrations.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: This system function provides reactivity control. The components associated with the addition of boron to the SFP are not in-scope of license renewal.

System Function: SFP-04 SPENT FUEL POOL COOLING: Able to isolate the SFP system to containment to ensure containment isolation.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: This function is associated with the valves and pipe required to ensure containment isolation in the SFP system. Components associated with the processing of electrical signals to initiate automatic isolation upon CHP and CHR are dispositioned under the Containment Isolation and Penetration (CIS) system. The penetrations are dispositioned within the Containment Structure evaluation.

System Function: SFP-05 SPENT FUEL POOL COOLING: Maintain clarity and purity of the water in the SFP.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This system function does not meet the requirements of 10 CFR 54.4. This is a normal operation function.

System Function: SFP-06 SPENT FUEL POOL COOLING: Maintain clarity and purity of the water in the SIRW tank.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This system function is not in the boundary of License Renewal due to not meeting the requirements of 10 CFR 54.4. This is a refueling operation function.

System Function: SFP-07	Cri 1	Cri 2	Cri 3				
SPENT FUEL POOL COOLING: Fill and drain the reactor cavity for refueling using Safety Injection and Refueling Water tank water transported via the SFP system.			FP	EQ	PTS	AT	SB

Comment: This system function is not in the boundary of License Renewal due to not meeting the requirements of 10 CFR 54.4. This is a refueling operation function.

System Function: SFP-NSAS	Cri 1	Cri 2	Cri 3				
This system has components in scope of license renewal in accordance with 10CFR54.4(a)(2) requirements for non-safety affecting safety (NSAS).			FP	EQ	PTS	AT	SB
		X					

Comment: In-scope of License Renewal to protect safety related components from spraying by the postulated failure of the SFP system.

FSAR Reference

Additional Spent Fuel Pool Cooling System details are provided in Section 9.4 of the FSAR.

Scoping Boundary Drawings

The scoping boundary drawings for the Spent Fuel Pool Cooling System are listed below:

LR-M-218, Sheet 2 LR-M-221, Sheet 2

Components Subject to an AMR

The component groups for this system that require aging management review are addressed in Table 2.3.3-14 along with each Component Group's intended function(s).

Table 2.3.3-14 Spent Fuel Pool Cooling System

Component Group	Intended Function
Accumulator	Fluid Pressure Boundary
Demineralizers	Fluid Pressure Boundary
Fasteners	Fluid Pressure Boundary

Table 2.3.3-14 Spent Fuel Pool Cooling System

Component Group	Intended Function
Filters/Strainers	Fluid Pressure Boundary
Pipe & Fittings	Fluid Pressure Boundary
Pumps	Fluid Pressure Boundary
SFP Heat Exchanger Shell and Channel Head	Fluid Pressure Boundary
SFP Heat Exchanger Tube and Tubesheet	Fluid Pressure Boundary Heat Transfer
Valves & Dampers	Fluid Pressure Boundary

2.3.3.15 **Waste Gas System**

System Description

The Waste Gas System (WGS) stores gaseous isotopes for a time period, which will permit sufficient radioactive decay, prior to their discharge to the environment. The Waste Gas System is divided into two sections: (a) the gas collection header which collects low-activity gases from liquids which have been previously degassed and/or vented in other waste handling steps, and (b) the gas processing section which collects gases from potentially high-activity sources. The Waste Gas system also contains components to provide Containment isolation.

The Hydrogen Recombiners are a subsystem to the WGS. The recombiner units are natural convection, thermal reactor type, hydrogen/oxygen recombiners. The subsystem consists of two 100% recombiner units, each containing an electric heater bank, power supply panel, and control panel. The recombiner units are located inside the containment building and the power supplies and control panels are in the Cable Spreading Room. These units rely on the Containment Air Coolers to mix the air for optimal performance.

The description above results in some SSCs in this system being considered in-scope in accordance with 10 CFR 54.4(a)(1). Since some SSCs in the Waste Gas System are non-safety related and their failure could affect the capability of safety related SSCs to perform their safety function, they are considered to be in-scope in accordance with 10 CFR 54.4(a)(2). In addition, some SSCs are considered in-scope due to Environmental Qualification in accordance with 10 CFR 54.4(a)(3).

The boundaries of the in-scope portions of the Waste Gas System are described as follows: 1) The piping and valves for Containment isolation located between Containment penetration #46 to control valve CV-1102, 2) the piping and valves for Containment isolation located between Containment penetration #52 to CV-1104, 2) the hydrogen recombiner units, and 3) numerous seismic components.

The original system scoping boundaries for license renewal included, where applicable, non-safety related components that could affect safety related components. The boundaries have been expanded in accordance with current NRC guidance for implementation of 10 CFR 54.4(a)(2). The expanded scoping boundary includes:

WGS non-safety related piping and components that are directly attached to safety related piping and components out to the first seismic anchor or equivalent anchor. Where base mounted components are credited as an anchor, they are also included in scope. Pipe supports for the newly added piping and components are also in scope. The new piping components are located in the Auxiliary Building and Containment Building.

WGS non-safety related components contain condensate and/or fluids that are located in an area that also contains a safety related component. The components are located in the Auxiliary Building and Containment Building.

The portions of the Waste Gas System containing components subject to an AMR include accumulators, fasteners, filters, pipe & fittings, and valves. The NSAS components brought into scope by the 10 CFR 54.4(a)(2) boundary expansion are subject to AMR.

System Function Listing

A comprehensive listing of functions associated with the Waste Gas System is provided in the summary below.

System Function: HYR-01 HYDROGEN RECOMBINERS: Provide hydrogen recombiners to control hydrogen concentration in the containment atmosphere following an event affecting the reactor core with releases into containment.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: The hydrogen recombiners are safety related and EQ.

System Function: WGS-01 WASTE GAS SYSTEM: Collect and store high activity gases and allow a controlled discharge to the atmosphere.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: A release of the stored high activity gases would not result in exceeding 10 CFR 100 limits.Therefore, this system function does not support a license renewal criterion.

System Function: WGS-02 WASTE GAS SYSTEM: Collect and filter gases with low activity levels and dilute and vent them to the atmosphere via a monitored vent path.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This system function does not support a license renewal criterion.

System Function: WGS-03 WASTE GAS SYSTEM: Automatically isolate the WGS containment isolation valves on CHP and CHR. Also ensure manual containment isolation valves can perform containment isolation.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Containment isolation valves and piping are safety related.

System Function: WGS-EQ The system contains components required by the current licensing basis for Environmental Qualification.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
				X			

Comment: The hydrogen recombiners are managed by the Environmental Qualification Program.

System Function: WGS-NSAS	Cri 1	Cri 2	Cri 3				
This system has components in scope of license renewal in accordance with 10CFR54.4(a)(2) requirements for non-safety affecting safety (NSAS).			FP	EQ	PTS	AT	SB
		X					

Comment: The WGS System contains non-safety related components that are attached to safety related components and is located above safety related components.

FSAR Reference

Additional Waste Gas System details are provided in Section 6.6 and Section 11.3 of the FSAR.

Scoping Boundary Drawings

The scoping boundary drawings for the Waste Gas System are listed below:

LR-M-201, Sheet 2
LR-M-202, Sheet 1A
LR-M-202, Sheet 1B
LR-M-203, Sheet 1
LR-M-209, Sheet 3
LR-M-210, Sheet 1
LR-M-210, Sheet 1A
LR-M-210, Sheet 1C
LR-M-210, Sheet 2
LR-M-211, Sheet 1
LR-M-211, Sheet 2
LR-M-211, Sheet 3
LR-M-218, Sheet 2
LR-M-218, Sheet 4
LR-M-219, Sheet 2A
LR-M-223, Sheet 1A
LR-M-650, Sheet 1
LR-M-650, Sheet 1A
LR-M-650, Sheet 1B
LR-M-651, Sheet 1
LR-M-651, Sheet 1A
LR-M-651, Sheet 1B
LR-M-655, Sheet 2
LR-M-657, Sheet 1

Components Subject to an AMR

The component groups for this system that require aging management review are addressed in Table 2.3.3-15 along with each Component Group's intended function(s).

Table 2.3.3-15 Waste Gas System

Component Group	Intended Function
Accumulators	Fluid Pressure Boundary
Fasteners	Fluid Pressure Boundary

Table 2.3.3-15 Waste Gas System

Component Group	Intended Function
Filters/Strainers	Fluid Pressure Boundary
Pipe & Fittings	Fluid Pressure Boundary
Valves & Dampers	Fluid Pressure Boundary

2.3.3.16 **Domestic Water System**

System Description

The Domestic Water System is a source of general purpose filtered well and/or city water to buildings of the plant. The major components are: a storage tank, level control system, accumulator, water heaters, pumps, piping, valves, and instrumentation. The accumulator is pressurized from the plant instrument air system. The system is non-safety related.

The Domestic Water System (DWS) has components in scope of license renewal in accordance with 10CFR54.4(a)(2) due to spatial considerations.

The boundaries of in-scope portions of the Domestic Water System are described as follows: 1) The well/city water line as it enters the Turbine building and exists the Turbine building going to the Domestic Water Storage Tank, 2) the DWS supply and return lines to and from the permanganate filter (T-36), including the filter, 3) the DWS supply lines to the potassium permanganate and sodium hydroxide tanks through the metering pumps, including the tanks and pumps, and return to the DWS supply line to T-36, 4) the DWS supply line to the permanganate filter (T-36), 5) the DWS supply line from the Domestic Water Storage Tank as it enters the Turbine building, through the Domestic Water Pumps to the Hydro Pneumatic Accumulator, 6) the DWS supply through the Service Building Domestic Water Pumps to the discharge line as it exist the turbine building, 7) the DWS supply to the Domestic Water Tank Heat Exchanger, including the heat exchanger, to the Domestic Water Storage Tank upstream of MV-DW106A, 8) the DWS lines located in Area 4, elevation 625', 8) the DWS lines located Area 2, elevation 611' going to and from the emergency shower, and 10) the DWS lines located in Area 9, elevation 590' going to and from the decontamination sink.

The original system scoping boundaries for license renewal included, where applicable, non-safety related components that could affect safety related components. The boundaries have been expanded in accordance with current NRC guidance for implementation of 10 CFR 54.4(a)(2). The expanded scoping boundary includes:

DWS non-safety related components containing steam and/or fluids that are located in an area that also contains a safety related component. The components types include accumulators, pumps, fasteners (to maintain pressure boundary), heat exchanger, piping, fittings, and valves. The components are located in the turbine building and auxiliary building.

The portions of the Domestic Water System containing components subject to an AMR include accumulators, heat exchanger, pump casings, fasteners, piping and fittings, and valves. The NSAS components brought into scope by the 10 CFR 54.4(a)(2) boundary expansion are subject to AMR.

System Function Listing

A comprehensive listing of functions associated with the Domestic Water System is provided in the summary below.

System Function: DWS-01 Pump, chemically treat, store and distribute potable water to plant facilities.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This function is not in scope of LR because it does not meet any of the three LR Criteria.

System Function: DWS-NSAS This system has components in scope of license renewal in accordance with 10CFR54.4(a)(2) requirements for non-safety affecting safety (NSAS).	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: The DWS System is a non-safety related system with pressurized raw water and is in areas that also contain safety related components.

FSAR Reference

None.

Scoping Boundary Drawings

The scoping boundary drawings for the Domestic Water System are listed below:

LR-M-215, Sheet 1A

LR-M-220, Sheet 2

Components Subject to an AMR

The component groups for this system that require aging management review are addressed in Table 2.3.3-16 along with each Component Group's intended function(s).

Table 2.3.3-16 Domestic Water System

Component Group	Intended Function
Accumulators	Fluid Pressure Boundary
Fasteners	Fluid Pressure Boundary
Heat Exchangers	Fluid Pressure Boundary
Pipe and Fittings	Fluid Pressure Boundary
Pumps	Fluid Pressure Boundary
Valves	Fluid Pressure Boundary

2.3.3.17 Chemical Addition System

System Description

Provide chemical addition to; Primary Coolant, Main Feedwater, Condensate, Circulating Water, and Service Water Systems. These additions are to reduce radiation levels, preserve piping integrity, maintain heat transfer, and prevent biological growths.

The Chemical Addition System (CHM) has components in scope of license renewal in accordance with 10CFR54.4(a)(2) due to spatial considerations.

The boundaries of in-scope portions of the Chemical Addition System are described as follows: 1) Chemical Addition Tanks T-15, T-16, T-19A, T-19B, and T-19C, 2) the piping and valves from T-15, T-16, and T-19A through the Chemical Addition Pumps, P-15A/G, to the exhaust hood spray header, the feedwater headers, the condensate pump discharge header, the condensate effluent header, the feedwater pump suction header, and the steam generator blowdown header. 3) Chemical Addition Tank, T-35, and the piping and valves to MV-FW238/737, 4) the Anti-foam Injection Tank and the piping and valves downstream of the Anti-foam Injection Pump (P-101), including the pump, to the radwaste evaporators M-59A/B, to the miscellaneous waste tanks T-92A/B/C, and to the clean waste holdup tank T-85, 5) the Caustic Injection Tank and the piping and valves downstream of the Caustic Injection Pumps (P-100A/B), including the pumps, to the radwaste evaporators M-59A/B, to the miscellaneous waste tanks T-92A/B/C, to the clean waste holdup tank T-85, and to the evaporator concentrator tanks T-94/95, and 6) the piping, pumps (P-47A/B) and valves located between basket strainers BS-5393/4 and valves MV-CHM115/116, including the valves.

The original system scoping boundaries for license renewal included, where applicable, non-safety related components that could affect safety related components. The boundaries have been expanded in accordance with current NRC guidance for implementation of 10 CFR 54.4(a)(2). The expanded scoping boundary includes:

CHM non-safety related piping and components that are directly attached to safety related piping and components out to the first seismic anchor or equivalent anchor. Where base mounted components are credited as an anchor, they are also included in scope. Pipe supports for the newly added piping and components are also in scope. The new piping components are located in the Turbine Building and Auxiliary Building.

CHM non-safety related components containing steam and/or fluids that are located in an area that also contains a safety related component. The components are located in the Turbine Building and Auxiliary Building.

The portions of the Chemical Addition System containing components subject to an AMR include accumulators, pump, fasteners, piping and fittings, and valves. The NSAS components brought into scope by the 10 CFR 54.4(a)(2) boundary expansion are subject to AMR.

System Function Listing

A comprehensive listing of functions associated with the Chemical Addition System is provided in the summary below.

System Function: CHM-01	Cri 1	Cri 2	Cri 3				
Provide a method to control chemistry in the Primary Coolant System via the Chemical and Volume Control System.			FP	EQ	PTS	AT	SB

Comment: This function is not in scope of LR because it does not meet any of the three LR Criteria.

System Function: CHM-02 Provide a method to control Main Feedwater and Condensate Systems chemistry for steam generator reliability and heat transfer performance.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This function is not in scope of LR because it does not meet any of the three LR Criteria.

System Function: CHM-03 Provide a method to control Circulating Water System chemistry for condenser tube heat transfer and integrity.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This function is not in scope of LR because it does not meet any of the three LR Criteria.

System Function: CHM-04 Provide a method to control Service Water System chemistry to minimize biological growth to ensure piping and valve integrity.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This function is not in scope of LR because it does not meet any of the three LR Criteria.

System Function: CHM-NSAS This system has components in scope of license renewal in accordance with 10CFR54.4(a)(2) requirements for non-safety affecting safety (NSAS).	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: The CHM system contains non-safety related components that are attached to safety related components and non-safety related components containing liquids located in an area that also contains safety related components.

FSAR Reference

None.

Scoping Boundary Drawings

The scoping boundary drawings for the Chemical Addition System are listed below:

LR-M-205, Sheet 1B
LR-M-207, Sheet 1A
LR-M-207, Sheet 1B
LR-M-220, Sheet 1
LR-M-220, Sheet 2
LR-M-226, Sheet 1
LR-M-227, Sheet 1
LR-M-650, Sheet 1
LR-M-650, Sheet 1A
LR-M-651, Sheet 1
LR-M-651, Sheet 1A
LR-M-653, Sheet 1
LR-M-655, Sheet 2

Components Subject to an AMR

The component groups for this system that require aging management review are addressed in Table 2.3.3-17 along with each Component Group's intended function(s).

Table 2.3.3-17 Chemical Addition System

Component Group	Intended Function
Accumulator	Fluid Pressure Boundary
Fasteners	Fluid Pressure Boundary
Pipe and Fittings	Fluid Pressure Boundary
Pump	Fluid Pressure Boundary
Valves and Dampers	Fluid Pressure Boundary

2.3.4 Steam and Power Conversion Systems

The following systems are addressed in this section:

- Condensate and Condenser System (Section 2.3.4.1)
- Demineralized Makeup Water System (Section 2.3.4.2)
- Feedwater System (Section 2.3.4.3)
- Heater Extraction and Drain System (Section 2.3.4.4)
- Main Air Ejection and Gland Seal System (Section 2.3.4.5)
- Main Steam System (Section 2.3.4.6)
- Turbine Generator System (Section 2.3.4.7)

2.3.4.1 Condensate and Condenser System

System Description

Within the Condensate and Condenser System (CDS), the main condenser condenses the steam from the main turbine and main feed pump driver exhausts, as well as the flashed steam from the feedwater heater drains, and provides the design vacuum which establishes the steam-water cycle efficiency. Noncondensible gases are removed from the main condenser by air ejectors or vacuum pump which are evaluated in Air Ejector System (AES). The condensate system provides the means for transferring the deaerated condensate from the condenser hotwell through the heat transfer surfaces of the air ejector condensers, the gland steam condenser, and the low pressure feedwater heaters to the suction of the main feed water pumps.

Also included in this system are the condensate storage tank (CST) and the associated piping system for makeup to and reject of the condenser hot well. The CST and the primary makeup (PMW) tank, which is covered in the demineralized make-up water system, provide a passive flow of water, by gravity, to the Auxiliary Feedwater (AFW) System. Piping from the CST to the AFW pump suctions is included in the Feedwater system. The combined CST and PMW tank contain a sufficient amount of cooling water to remove decay heat following a reactor trip to allow for cool down of the Primary Coolant System to Shutdown Cooling entry conditions.

The description above results in some SSCs in this system being considered in-scope in accordance with 10 CFR 54.4(a)(1). Since some SSCs in the Condensate and Condenser System are non-safety related and their failure could affect the capability of SR SSCs to perform their safety function, they are considered to be in-scope in accordance with 10 CFR 54.4(a)(2). In addition, some SSCs are considered in-scope due to Fire Protection and Station Blackout in accordance with 10 CFR 54.4(a)(3).

The boundaries of the in-scope portions of the Condensate and Condenser System are described as follows: 1) The piping and valves from the condensate storage tank (CST), including the tank, through the condensate transfer pump, through the CST heat exchanger and return to the CST. 2) The condenser makeup line piping and valves from the CST to makeup and reject valves, including portions of the auxiliary feedwater pump recirculation lines, 3) some segments of CDS piping from feedwater heaters E-2A/B, through the downstream feedwater heaters (E-3A/B, E-4A/B, and E-5A/B) to the suction of the main feedwater pumps. In addition, the non-safety related condenser makeup and reject piping in the Auxiliary Feedwater pump room are in scope.

The original system scoping boundaries for license renewal included, where applicable, non-safety related components that could affect safety related components. The boundaries have been expanded in accordance with current NRC guidance for implementation of 10 CFR 54.4(a)(2). The expanded scoping boundary includes:

CDS non-safety related piping and components that are directly attached to safety related piping and components out to the first seismic anchor or equivalent anchor. Where base mounted components are credited as an anchor, they are also included in scope. Pipe supports for the newly added piping and components are also in scope. The new piping components are located in the Turbine Building.

CDS non-safety related components containing steam and/or fluids that are located in an area that also contains a safety related component. The components are located in the Turbine Building.

The portions of the Condensate and Condenser System containing components subject to an AMR include accumulators, heaters, strainers, pipe and fittings, fasteners, heat exchangers, pumps, and valves. The NSAS components brought into scope by the 10 CFR 54.4(a)(2) boundary expansion are subject to AMR.

System Function Listing

A comprehensive listing of functions associated with the Condensate and Condenser System is provided in the summary below.

System Function: CDS-01	Cri 1	Cri 2	Cri 3				
Provide a continuous supply of water from the condenser hotwell to the suction of the main feedwater pumps.			FP	EQ	PTS	AT	SB

Comment: Condensate supply to the main feedwater pumps is to support normal operation only. However, failure of some segments of condensate supply pipes in the pathway may cause damage to some safety related components, such as AFW pumps, controls of AFW turbine steam supply valve, and controls of FW regulator valves. The effect of the failure on the adjacent safety related components is addressed in Function CDS-NSAS.

System Function: CDS-02	Cri 1	Cri 2	Cri 3				
Ensure adequate condensate inventory is available for the Auxiliary Feedwater System.			FP	EQ	PTS	AT	SB
	X						

Comment: The CDS provides a safety grade source of water to the steam generators to make up for steaming required for removing decay heat and cooling down the plant following all events discussed in the FSAR, Chapters 5 and 14. To satisfy accident analysis assumptions, the condensate inventory must contain sufficient cooling water to allow for cool down of the Primary Coolant System to Shutdown Cooling entry conditions.

System Function: CDS-03	Cri 1	Cri 2	Cri 3				
Ensure ability to manually isolate the hotwell makeup valves to conserve condensate inventory for Auxiliary Feedwater during an accident.			FP	EQ	PTS	AT	SB
	X						

Comment: The Auxiliary Feedwater pumps operate with a continuous recirculation to the CST. The minimum flow recirculation is returned to the CST through the header of condenser hotwell reject and makeup. The associated manual makeup and reject isolation valves are relied on to minimize the losses and ensure the CST will maintain adequate inventory for its safety function of CDS-02.

System Function: CDS-04	Cri 1	Cri 2	Cri 3				
Provide ability to feed the steam generators from the condenser hotwell via the condensate pumps during a Loss of Feedwater accident.			FP	EQ	PTS	AT	SB

Comment: As described in FSAR 9.7.4, the condensate pumps may be used to pump water through the normal feedwater train to the steam generators in the event of a failure of the auxiliary feedwater piping system. The failure of the Auxiliary Feedwater system is not postulated in the analysis of a Loss of Feedwater accident, and is outside the design basis of this plant.

System Function: CDS-05 Steam discharged from the low-pressure turbine is condensed and deaerated in the main condenser.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This function of the main condenser provides the design vacuum to establish the steam-water cycle efficiency, which is to support normal operation only.

System Function: CDS-FP The system contains structures and/or components required by the current licensing basis for Fire Protection.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: The components of CDS involved in Appendix R Safe Shutdown Analysis include CST (normal source of AFW supply), control switches and breakers (breakers are addressed in Station Power System) of Condensate Pumps (secured for FW isolation without closing block valves), etc.

System Function: CDS-NSAS This system has components in scope of license renewal in accordance with 10CFR54.4(a)(2) requirements for non-safety affecting safety (NSAS).	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: The CDS system contains non-safety related components that are attached to safety related components and non-safety related components containing liquids located in an area that also contains safety related components.

System Function: CDS-SB The system contains structures and/or components required by the current licensing basis for Station Blackout (Loss of all AC power).	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
							X

Comment: As described in FSAR 10.2.3.1, the minimum required condensate storage tank (CST) inventory will ensure that sufficient inventory is available to cope with a Station Blackout for 4 hours.

FSAR Reference

Additional Condensate and Condenser System details are provided in Section 9.7 and Section10.2 of the FSAR.

Scoping Boundary Drawings

The scoping boundary drawings for the Condensate and Condenser System are listed below:

LR-M-206, Sheet 1
LR-M-206, Sheet 1B
LR-M-207, Sheet 1A
LR-M-207, Sheet 1B
LR-M-207, Sheet 1C
LR-M-207, Sheet 1D
LR-M-207, Sheet 2
LR-M-212, Sheet 2
LR-M-215, Sheet 1A
LR-M-220, Sheet 1
LR-M-225, Sheet 2

Components Subject to an AMR

The component groups for this system that require aging management review are addressed in Table 2.3.4-1 along with each Component Group’s intended function(s).

Table 2.3.4-1 Condensate and Condenser System

Component Group	Intended Function
Accumulators	Fluid Pressure Boundary
CST Heater Shell	Fluid Pressure Boundary
CST Heater Tubes	Fluid Pressure Boundary
Filters/Strainers	Fluid Pressure Boundary
FW Heater Shell and Channel Head	Fluid Pressure Boundary
Fasteners	Fluid Pressure Boundary
Heat Exchangers	Fluid Pressure Boundary
Pipe & Fittings	Fluid Pressure Boundary
Pumps	Fluid Pressure Boundary
Valves & Dampers	Fluid Pressure Boundary

2.3.4.2 Demineralized Makeup Water System

System Description

The Demineralized Makeup Water (DMW) System is the source of high quality filtered and demineralized water throughout the plant. The major components of this system are pumps, storage tanks, a reverse osmosis system, filters, tank heating heat exchangers, extensive piping, valves, and instrumentation.

The description above results in some SSCs in this system being considered in-scope in accordance with 10 CFR 54.4(a)(1). Since some SSCs in the Demineralized Makeup Water System are non-safety related and their failure could affect the capability of SR SSCs to perform their safety function, they are considered to be in-scope in accordance with 10 CFR 54.4(a)(2). In addition, some SSCs are considered in-scope due to Fire Protection and Station Blackout in accordance with 10 CFR 54.4(a)(3).

The boundaries of the in-scope portions of the Demineralized Makeup Water System are described as follows:1) Primary System Makeup Storage Tank T-81, 2) Primary System Makeup Storage Tank T-90, 3) piping and components comprising the pressure boundary between tank T-81 to the Condensate Storage Tank, 4) piping located downstream of T-81 to valves MV-PMU100 and MV-PMU109, 5) piping header HC-11-3” located between headers HCD-92-1” and HCD-7-2”, and 6) piping header HB-26-1” located in Diesel Generator 1-1 and 1-2 rooms.

The original system scoping boundaries for license renewal included, where applicable, non-safety related components that could affect safety-related components. The boundaries have been expanded in accordance with current NRC guidance for implementation of 10 CFR 54.4(a)(2). The expanded scoping boundary includes:

DMW non-safety related piping and components that are directly attached to safety related piping and components out to the first seismic anchor or equivalent anchor. Where base mounted components are credited as an anchor, they are also included in scope. Pipe supports for the newly added piping and components are also in scope. The new piping components are located in the Turbine Building and Auxiliary Building.

DMW non-safety related components containing steam and/or fluids that are located in an area that also contains a safety related component. The components are located in the Turbine Building and Auxiliary Building.

The portions of the Demineralized Makeup Water System containing components subject to an AMR include accumulators, heat exchanger, filters, pipe & fittings, fasteners, pumps, valves, and conductivity element. The NSAS components added into scope by the 10 CFR 54.4(a)2 scoping boundary expansion are subject to AMR.

System Function Listing

A comprehensive listing of functions associated with the Demineralized Makeup Water System is provided in the summary below.

System Function: DMW-01 Provide condensate/feedwater supply from tank T-81 via pump feed or gravity feed.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Gravity feed from Primary Makeup Water (PMW) Tank to Condensate Storage Tank is in-scope of License Renewal, but supply via pump feed is not in-scope of License Renewal because the feed pumps are not safety related. Technical Specification 3.7.6 requires that a minimum of 100,000 gallons be maintained in the Condensate Storage Tank and PMW Tank, combined.

System Function: DMW-02 Provide alternate supply of condensate/feedwater supply from tank T-939 via pump feed.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Tank and feed pump are not safety related. This function is not in-scope of License Renewal.

System Function: DMW-03 Provide primary makeup water to the emergency diesel generator jacket water surge tanks.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: The emergency diesel generator jacket water surge tanks have sufficient capacity for duration of accident without addition of makeup water. This function is not in scope of license renewal.

System Function: DMW-04 Provide primary makeup water to the component cooling water system surge tank.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: The component cooling water surge tank has sufficient capacity for duration of accident without addition of makeup water. This function is not in scope of license renewal.

System Function: DMW-05 Provide demineralized water to the shield cooling system surge tank.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: The shield cooling system, including surge tank, is not safety related and is not required for design basis accidents, per FSAR 9.2.

System Function: DMW-06	Cri 1	Cri 2	Cri 3				
Provide demineralized water for miscellaneous plant uses.			FP	EQ	PTS	AT	SB

Comment: Not required for design basis accidents.

System Function: DMW-FP	Cri 1	Cri 2	Cri 3				
The system contains structures and/or components required by the current licensing basis for Fire Protection.			FP	EQ	PTS	AT	SB
			X				

Comment: In scope of license renewal for Appendix R.

System Function: DMW-NSAS	Cri 1	Cri 2	Cri 3				
This system has components in scope of license renewal in accordance with 10CFR54.4(a)(2) requirements for non-safety affecting safety (NSAS).			FP	EQ	PTS	AT	SB
		X					

Comment: The DMW system contains non-safety related components that are attached to safety related components and non-safety related components containing liquids located in an area that also contains safety related components.

System Function: DMW-SB	Cri 1	Cri 2	Cri 3				
The system contains structures and/or components required by the current licensing basis for Station Blackout (Loss of all AC power).			FP	EQ	PTS	AT	SB
							X

Comment: in-scope of License Renewal based on TS 3.7.6, which requires that a minimum of 100,000 gallons be maintained in the Condensate Storage Tank and Primary Makeup Water Tank, combined.

FSAR Reference

Additional Demineralized Makeup Water System details are provided in Section 9.2 and Section 9.7 of the FSAR.

Scoping Boundary Drawings

The scoping boundary drawings for the Demineralized Makeup Water System are listed below:

LR-M-201, Sheet 2
LR-M-202, Sheet 1A
LR-M-209, Sheet 3
LR-M-210, Sheet 1B
LR-M-210, Sheet 2
LR-M-212, Sheet 2
LR-M-214, Sheet 1
LR-M-215, Sheet 1
LR-M-215, Sheet 1A
LR-M-219, Sheet 1A
LR-M-219, Sheet 1B
LR-M-219, Sheet 2
LR-M-220, Sheet 1
LR-M-220, Sheet 2
LR-M-221, Sheet 1
LR-M-227, Sheet 1
LR-M-650, Sheet 1
LR-M-650, Sheet 1B
LR-M-651, Sheet 1B
LR-M-652, Sheet 1
LR-M-653, Sheet 4
LR-M-654
LR-M-655, Sheet 1
LR-M-657, Sheet 1

Components Subject to an AMR

The component groups for this system that require aging management review are addressed in Table 2.3.4-2 along with each Component Group's intended function(s).

Table 2.3.4-2 Demineralized Makeup Water System

Component Group	Intended Function
Accumulators	Fluid Pressure Boundary
Fasteners	Fluid Pressure Boundary
Filters	Fluid Pressure Boundary
Heat Exchangers	Fluid Pressure Boundary
Pipe and Fittings	Fluid Pressure boundary
Pumps	Fluid Pressure Boundary
Transmitter/Element	Fluid Pressure Boundary
Valves & Dampers	Fluid Pressure Boundary

2.3.4.3 Feedwater System

System Description

The Feedwater System (FWS) consists of Main Feedwater (MFW) and Auxiliary Feedwater (AFW) Systems. The MFW System provides a reliable source of water to the Steam Generators during normal operations. This water is preheated in the feedwater heaters to increase overall thermal efficiency. Flow rate is manually and/or automatically controlled to maintain Steam Generator level. Condensate from two parallel streams of low pressure Feedwater Heaters and from the discharge of two Heater Drain Pumps is supplied to the suction of the two variable speed, turbine driven Feedwater Pumps. These pumps deliver the feedwater through parallel high pressure feedwater heaters to the Steam Generators.

The AFW System supplies water to the secondary side of the steam generators for reactor decay heat removal when normal feedwater sources are unavailable during accident conditions. AFW is also designed to provide a supply of feedwater to the steam generators during start-up operations and to remove primary system sensible and decay heat during initial stages of shutdown operations.

AFW can provide feedwater to any combination of steam generators from any one or combination of three pumps; two are motor-driven, and the third is steam-driven. Steam can be supplied to the steam-driven pump from the steam generator E-50A. The pumps can take suction from the Condensate Storage Tank, which is the normal source, or from the Service Water System (P-8C) or the Fire Protection System (P-8A/B) if the Condensate Storage Tank is not available. The steam driven pump provides an independent and diversely powered means of providing feedwater to the steam generators. In the event of automatic initiation, the AFW system is designed to automatically start the AFW pumps and direct flow to the steam generators via the flow path to the AFW nozzles.

The description above results in some SSCs in this system being considered in-scope in accordance with 10 CFR 54.4(a)(1). Since some SSCs in the Feedwater System are non-safety related and their failure could affect the capability of SR SSCs to perform their safety function, they are considered to be in-scope in accordance with 10 CFR 54.4(a)(2). In addition, some SSCs are considered in-scope due to ATWS, Environmental Qualification, Fire Protection, and Station Blackout in accordance with 10 CFR 54.4(a)(3).

The boundaries of the in-scope portions of the Feedwater System are described as follows: 1) The piping, valves, and components from the S/G Feed

Pumps, including the pumps, through feedwater heaters to the Steam Generators, 2) The piping, valves, and components from the Condensate Storage Tank (T-2) through the AFW pumps to the Steam Generators, and 3) the steam-driven pump turbine and the piping, valves and components for its steam supply from S/G E-50A.

The original system scoping boundaries for license renewal included, where applicable, non-safety related components that could affect safety-related components. The boundaries have been expanded in accordance with current NRC guidance for implementation of 10 CFR 54.4(a)(2). The expanded scoping boundary includes:

FWS non-safety related piping and components that are directly attached to safety related piping and components out to the first seismic anchor or equivalent anchor. Where base mounted components are credited as an anchor, they are also included in scope. Pipe supports for the newly added piping and components are also in scope. The new piping components are located in the Turbine Building, Auxiliary Building and Yard.

FWS non-safety related components containing steam and/or fluids that are located in an area that also contains a safety related component. The components are located in the Turbine Building and Auxiliary Building.

The portions of the Feedwater System containing components subject to an AMR include accumulator, strainers, heat exchangers, flow indicators, pipe & fittings, fasteners, filters, pumps, steam traps, turbines and valves. The NSAS components added into scope by the 10 CFR 54.4(a)(2) scoping boundary expansion are subject to AMR.

System Function Listing

A comprehensive listing of functions associated with the Feedwater System, or specific components contained in the system, is provided in the summary below.

System Function: AFW-01	Cri 1	Cri 2	Cri 3				
Provide water to the steam generator (S/G) secondary side.			FP	EQ	PTS	AT	SB
	X						

Comment: Components of AFW are Safety-Related; FSAR 9.7 includes the detail description of the design basis function of AFW. Also, Safety Analyses in FSAR 14.13, 14.14, 14.15, and 14.18 credit the AFW system for the mitigation of the transients.

System Function: AFW-02	Cri 1	Cri 2	Cri 3				
Provide Auxiliary Feedwater Actuation Signals (AFAS) and flow control signals to AFW.			FP	EQ	PTS	AT	SB
	X						

Comment: AFW Control components are safety-related; FSAR 7.4.3 details the design basis functions of AFW Controls.

System Function: AFW-03	Cri 1	Cri 2	Cri 3				
Able to manually isolate AFW to containment to ensure containment integrity.			FP	EQ	PTS	AT	SB
	X						

Comment: FSAR 7.4.3 and 9.7.2.3 indicate that in the event of a main steam line break inside containment, the AFW flow toward the affected steam generator must be terminated. The Emergency Operating Procedures direct operators to isolate the affected steam generator using the flow control valves.

System Function: AFW-04	Cri 1	Cri 2	Cri 3				
Provide level instrumentation for steam generator level inputs to Auxiliary Feedwater Actuation and the Reactor Protective System for low steam generator level trips.			FP	EQ	PTS	AT	SB
	X						

Comment: FSAR Table 5.2-5, I&C Component Classification, FSAR 7.4.3, AFW Controls, and FSAR 7.2.3.7, Reactor Protective System (Low Steam Generator Water Level) indicate the safety function of the level instrumentation for steam generators.

System Function: AFW-05	Cri 1	Cri 2	Cri 3				
Provide decay heat removal via steaming the steam generators through steam-driven AFW pump P-8B.			FP	EQ	PTS	AT	SB

Comment: Steaming of the Steam Generators via steam-driven AFW pump is an ancillary function that pump can perform that is a backup to the normal steaming paths via the turbine bypass valve, the Atmospheric Dump Valves, and the PCS code safeties. This function is included in the Emergency Operating Procedures to provide the operators a means to minimize challenges to the code safeties. Therefore, this is neither a safety related function nor a regulated event function.

System Function: AFW-06	Cri 1	Cri 2	Cri 3				
Isolate steam supply to AFW pump turbine.			FP	EQ	PTS	AT	SB
	X						

Comment: FSAR 14.15, Steam Generator Tube Rupture, states that one of the operator actions assumed in the analysis is to isolate the affected steam generator when the hot leg is 525 deg F or less.

System Function: AFW-07	Cri 1	Cri 2	Cri 3				
Provides AFW pump trips on low suction pressure, overcurrent, or load shed.			FP	EQ	PTS	AT	SB

Comment: This function is for equipment protection.

System Function: AFW-08	Cri 1	Cri 2	Cri 3				
Provide means of containment isolation for Class C-1 penetrations.			FP	EQ	PTS	AT	SB
	X						

Comment: Because the steam supply lines to the steam driven AFW pump turbine branch off the main steam lines upstream of the main steam isolation valves, control valve CV-0522B is considered a containment isolation valve.

The AFW piping to the steam generators is routed through Containment Penetrations. In each situation, a single check valve is the means of containment isolation, similar to the main feedwater isolation. FSAR 5.1.6.9 provides justification of using single check valve for containment isolation for MFW and AFW systems.

This function is associated with the valves and pipe required to ensure containment isolation in the AFW system. Components and commodities (e.g., Cables and supports) associated with the processing of electrical signals to initiate automatic isolation upon CHP and CHR are dispositioned under the Containment Isolation and Penetration (CIS) system. The penetrations are dispositioned within the Containment Structure evaluation.

System Function: FWS-AT	Cri 1	Cri 2	Cri 3				
The system contains structures and/or components required by the current licensing basis for Anticipated Transients Without Scram.			FP	EQ	PTS	AT	SB
						X	

Comment: FSAR 7.4.3 indicates an auto start of the turbine driven AFW pump on loss of DC control power was added as an independent actuation signal to meet the ATWS requirements.

System Function: FWS-EQ	Cri 1	Cri 2	Cri 3				
The system contains components required by the current licensing basis for Environmental Qualification.			FP	EQ	PTS	AT	SB
				X			

Comment:

System Function: FWS-FP	Cri 1	Cri 2	Cri 3				
The system contains structures and/or components required by the current licensing basis for Fire Protection.			FP	EQ	PTS	AT	SB
			X				

Comment: FSAR 9.6.8 summarizes the plant specific safe shutdown analysis. FSAR 7.7.4 describes that the Auxiliary Hot Shutdown Panels have been provided as a centralized location for controlling safe shutdown of the plant for the alternate shutdown accomplished from outside the control room.(3) Technical Specification Table 3.3.8-1, Alternate Shutdown System Instrumentation and Controls, includes AFW control and indication of turbine driven pump.

System Function: FWS-NSAS	Cri 1	Cri 2	Cri 3				
This system has components in scope of license renewal in accordance with 10CFR54.4(a)(2) requirements for non-safety affecting safety (NSAS).			FP	EQ	PTS	AT	SB
		X					

Comment: The FWS system contains non-safety related components that are attached to safety related components and non-safety related components containing liquids located in an area that also contains safety related components.

System Function: FWS-SB	Cri 1	Cri 2	Cri 3				
The system contains structures and/or components required by the current licensing basis for Station Blackout (Loss of all AC power).			FP	EQ	PTS	AT	SB
							X

Comment: AFW is involved in the mitigation of Station Blackout events

System Function: MFW-01	Cri 1	Cri 2	Cri 3				
Provide water to the steam generators to maintain water level above trip setpoint.			FP	EQ	PTS	AT	SB

Comment: This system function is to support Normal Operation, not credited in mitigation of design basis transients. However, piping failure of some segments will affect safety related components. The system function associated with Criterion 2 is covered in FWS-NSAS.

System Function: MFW-02	Cri 1	Cri 2	Cri 3				
Automatically isolate the MFW regulation and MFW regulation bypass valves on CHP or the respective Steam Generator low pressure signal.			FP	EQ	PTS	AT	SB
	X						

Comment: FSAR Table 14.14-4 (Steam Line Break Sequence Of Events) and FSAR 7.5.1.3 indicate that in the event of low steam generator pressure < 500 psia or containment high pressure (CHP), the main feedwater regulating and regulating bypass valves are closed to prevent excessive flow into the steam generators. This ensures containment pressure is not exceeded during a main steam line break inside containment. FSAR 10.2.3.3 indicates this feature also prevent the possibility of a condensate pump supplying water to a depressurized steam generator causing overcooling of the Primary Coolant System.

System Function: MFW-03	Cri 1	Cri 2	Cri 3				
Provide hot start capability of the MFW pumps during a loss of all feedwater event.			FP	EQ	PTS	AT	SB

Comment: Emergency Operating Procedures provide the direction to implement this function to recover the MFW pumps. But the recovery of MFW pumps following a transient is not credited in any safety analysis.

System Function: MFW-04	Cri 1	Cri 2	Cri 3				
Provide pressure instrumentation for steam generator pressure inputs for Main Steam Isolation and to the Reactor Protective System for low Steam Generator pressure trips.			FP	EQ	PTS	AT	SB
	X						

Comment: Bases of Technical Specification 3.3.3 indicates this signal ensures acceptable consequence in the event of Main Steam Line Break by isolating the affected steam generator. The input to Reactor Protective System is described in the Bases of Technical Specification 3.3.1.

System Function: MFW-05	Cri 1	Cri 2	Cri 3				
Isolate Steam Generators on high level override at 85% level.			FP	EQ	PTS	AT	SB

Comment: Provides equipment protection only and no credit is taken in the safety analysis per FSAR 7.5.2.3. The associated level transmitters are in scope as safe shutdown equipment shown on Fire Protection drawings, but only for pressure boundary in support of functions MFW-01 and AFW-01.

System Function: MFW-06	Cri 1	Cri 2	Cri 3				
Manually isolate the Main Feed Water System with the main feed water block valves.			FP	EQ	PTS	AT	SB

Comment: This is an operator action directed by Emergency Operating Procedures and no credit is taken in the accident analysis. This is redundant to closure of the MFW regulating valves and the MFW regulating bypass valves.

System Function: MFW-07 Provide Steam Generator level and pressure indication.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: The Bases of Technical Specification 3.3.7 indicates that the Steam Generator level indication is used during a tube rupture event to determine which Steam Generator has the ruptured tube. It is also used to determine when to initiate once through cooling on low water level. The Bases also indicates Steam Generator pressure is a Type A, Category 1 variable used in accident identification, including Loss of Coolant, and Steam Line Break. Redundant monitoring capability is provided by two channels of instrumentation for each Steam Generator. FSAR Table 5.2-5 (I&C Component Classification) includes Wide Range Steam Generator Levels, which are Safety Class 1E.

System Function: MFW-08 Manually align the Main Feed Water System valves to support low pressure feed with the condensate pumps.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This is an operator action driven by Emergency Operating Procedures and no credit is taken in the accident analysis.

System Function: MFW-09 Provide means of containment isolation for Class C-1 penetrations.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: The MFW piping is routed through Containment Penetrations. In each situation, a single check valve is the means of containment isolation. FSAR 5.1.6.9 provides justification of using single check valve for containment isolation for MFW and AFW systems. This function is associated with the valves and pipe required to ensure containment isolation in the MFW system. Components associated with the processing of electrical signals to initiate automatic isolation upon CHP and CHR are dispositioned under the Containment Isolation and Penetration (CIS) system. The penetrations are dispositioned within the Containment Structure evaluation.

FSAR Reference

Additional Feedwater System details are provided in Section 14.13, Section 14.14, Section 14.15, Section 14.18, Section 5.2, Section 7.4, Section 9.7, Section 7.2, Section 5.1, Section 7.7, Section 9.6, Section 10.2, and Section 7.5 of the FSAR.

Scoping Boundary Drawings

The scoping boundary drawings for the Feedwater System are listed below:

LR-M-201, Sheet 2
LR-M-205, Sheet 2
LR-M-205, Sheet 2A
LR-M-206, Sheet 1
LR-M-207, Sheet 1
LR-M-207, Sheet 1A
LR-M-207, Sheet 1C
LR-M-207, Sheet 2
LR-M-212, Sheet 2
LR-M-212, Sheet 3
LR-M-214, Sheet 3
LR-M-220, Sheet 1
LR-M-220, Sheet 2
LR-M-225, Sheet 2

Components Subject to an AMR

The component groups for this system that require aging management review are addressed in Table 2.3.4-3 along with each Component Group's intended function(s).

Table 2.3.4-3 Feedwater System

Component Group	Intended Function
Accumulators	Fluid Pressure Boundary
Fasteners	Fluid Pressure Boundary
Filters/Strainers	Fluid Pressure Boundary
Heat Exchangers	Fluid Pressure Boundary
Indicators/Recorders	Fluid Pressure Boundary
Pipe & Fittings	Flow Restriction Fluid Pressure Boundary
Pumps	Fluid Pressure Boundary
Traps (Steam)	Fluid Pressure Boundary
Turbines	Fluid Pressure Boundary
Valves & Dampers	Fluid Pressure Boundary

2.3.4.4 Heater Extraction and Drain System

System Description

In the Heater Extraction and Drain (HED) System, extraction steam is supplied to the feedwater heaters from the high pressure (HP) and low pressure (LP) turbines as required by the plant thermal cycle. During normal operation, feedwater heating stages 1 through 4 are supplied by the LP turbines extraction steam, and heating stage 5 is supplied from the HP turbine exhaust crossunder to the moisture separator reheater. Feedwater heating stage 6, high pressure, is supplied by HP turbine extraction and the reheater drain tanks. All feedwater heaters with sufficient energy to overspeed the turbine have extraction nonreturn valves or turbine reheat stop and intercept valves to stop reverse steam flow following a turbine trip.

The HED System collects drains from various heaters and returns them to the secondary water cycle. Feedwater heater 6 drains cascade to heater 5 and then to the moisture separator drain tank which supplies the heater drain pumps. Feedwater heaters 4, 3, 2, and 1 are also arranged in a cascaded drain system, but ultimately discharge to the main condenser. The system design is to inject the higher temperature drains to the shell side of the feedwater heaters closer to the steam generators. This allows the feedwater to be progressively heated until it is just a few degrees below main steam temperature when the feedwater is injected into the steam generators. This also increases the thermal efficiency of the steam/water cycle.

Since the failure of some non-safety related SSCs in the Heater Extraction and Drain System could affect the capability of SR SSCs to perform their safety function, they are considered to be in-scope in accordance with 10 CFR 54.4(a)(2). In addition, some SSCs are considered in-scope due to Fire Protection in accordance with 10 CFR 54.4(a)(3).

The boundaries of the in-scope portions of the Heater Extraction and Drain System are described as follows: 1) The piping, valves and components from extraction steam supply lines downstream of the HP turbine Bleeder Trip Valves to feedwater heaters E-6A/B, 2) A portion of the HP turbine exhaust crossunder piping to feedwater heaters E-5A/B, 3) The piping, valves and components from the heater drains from heaters E-6A/B, cascade to heater E-5A/B, and then to the moisture separator drain tank (T-5), 4) The piping, valves and components from T-5 through the heater drain pumps, and the discharge headers to flow element FE-0786 and 5) Part of the extraction steam lines to the plant heating system located upstream of the plant heating steam trap ST-0672 and pipe header HB-27-8” through CV-0668 to pipe header JB-11-2”.

The original system scoping boundaries for license renewal included, where applicable, non-safety related components that could affect safety-related components. The boundaries have been expanded in accordance with current NRC guidance for implementation of 10 CFR 54.4(a)(2). The expanded scoping boundary includes:

HED non-safety related piping and components that are directly attached to safety related piping and components out to the first seismic anchor or equivalent anchor. Where base mounted components are credited as an anchor, they are also included in scope. Pipe supports for the newly added piping and components are also in scope. The new piping components are located in the Turbine Building.

HED non-safety related components containing steam and/or fluids that are located in an area that also contains a safety related component. The components are located in the Turbine Building.

The portions of the Heater Extraction and Drain System containing components subject to an AMR include accumulator, heat exchangers, pipe & fittings, fasteners, pumps, flow elements, steam traps, and valves. The NSAS components added into scope by the 10 CFR 54.4(a)(2) scoping boundary expansion are subject to AMR.

System Function Listing

A comprehensive listing of functions associated with the Heater Extraction and Drain System is provided in the summary below.

System Function: HED-01	Cri 1	Cri 2	Cri 3				
Utilize extraction steam to preheat condensate and feedwater and provide a warm water suction supply to the MFW pumps via the heater drain pumps.			FP	EQ	PTS	AT	SB

Comment: This function is to support normal plant operation only. No safety functions are involved and this function is not in scope for license renewal. Some segments of the associated piping systems have potential to impact design function of the surrounding safety-related equipment upon postulated pipe failures. The evaluation of this failure is covered by System Function HED-NSAS.

System Function: HED-02 Close bleeder trip valves to prevent turbine overspeed via reverse steam flow from the heaters to the turbine following a turbine trip.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: As described in FSAR 7.5.3.6, the steam required to produce turbine overspeed has to come from either the main steam system or from water flashing in feedwater heaters and moisture separators after a turbine trip. All feedwater heaters with sufficient energy to overspeed the turbine have extraction nonreturn valves. However, this system function does not support a license renewal criterion.

System Function: HED-03 The HVAC system normally uses extraction steam, from the low-pressure turbine, as source of plant heating Steam.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This function is to support normal plant operation only. No safety functions are involved. Some segments of the associated piping systems have potential to impact design function of the surrounding safety-related equipment upon postulated pipe failures. The evaluation of this failure is covered by System Function HED-NSAS.

System Function: HED-FP The system contains structures and/or components required by the current licensing basis for Fire Protection.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: The main feedwater, condensate and heater drain pumps must be off to provide feedwater isolation to the steam generators. Ensuring that these pumps are off will accomplish the feedwater isolation function to the steam generators.

System Function: HED-NSAS This system has components in scope of license renewal in accordance with 10CFR54.4(a)(2) requirements for non-safety affecting safety (NSAS).	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: The HED system contains non-safety related components that are attached to safety related components and non-safety related components containing liquids located in an area that also contains safety related components.

FSAR Reference

Additional Heater Extraction and Drain System details are provided in Section 7.5 and Section 9.8 of the FSAR.

Scoping Boundary Drawings

The scoping boundary drawings for the Heater Extraction and Drain System are listed below:

LR-M-206, Sheet 1
LR-M-206, Sheet 1A
LR-M-206, Sheet 1B
LR-M-206, Sheet 2
LR-M-207, Sheet 1C
LR-M-215, Sheet 1A

Components Subject to an AMR

The component groups for this system that require aging management review are addressed in Table 2.3.4-4 along with each Component Group's intended function(s).

Table 2.3.4-4 Heater Extraction and Drain System

Component Group	Intended Function
Accumulators	Fluid Pressure Boundary
Fasteners	Fluid Pressure Boundary
Heat Exchangers	Fluid Pressure Boundary
Pipe & Fittings	Fluid Pressure Boundary
Pumps	Fluid Pressure Boundary
Transmitter/Element	Fluid Pressure Boundary
Traps (Steam)	Fluid Pressure Boundary
Valves & Dampers	Fluid Pressure Boundary

2.3.4.5 Main Air Ejection and Gland Seal System

System Description

In the Main Air Ejection and Gland Seal (AES) System, noncondensible gases are removed from the main condenser during operation by the steam jet air ejectors, and during start-up by the condenser vacuum pump, hogging air ejector and the steam jet air ejectors. The condenser vacuum pump is used to establish a partial condenser vacuum during start-up, and to allow testing of the main condenser for leakage while the plant is shut down.

The gland seal system provides steam seal on main turbine shaft to prevent air inleakage and maintain condenser vacuum. Steam supplies for sealing the main turbines pass through pressure reducing stations off the main steam supply line. The gland seal exhaust is condensed through the gland seal condenser and returned to steam-condensate cycle at the condenser hotwell.

Since the failure of some non-safety related SSCs in the Main Air Ejection and Gland Seal System could affect the capability of SR SSCs to perform their safety function, they are considered to be in-scope in accordance with 10 CFR 54.4(a)(2). In addition, some SSCs are considered in-scope due to Fire Protection in accordance with 10 CFR 54.4(a)(3).

The boundaries of the in-scope portions of the Main air Ejection and gland Seal System can be described as follows: 1) The piping, valves and components from Main Steam piping, through the Hogging Air Ejector, the muffler, and to outside the turbine building; and 2) The piping, valves and components from Main Steam piping, through the Steam Jet Air Ejectors, to the after-condenser.

The original system scoping boundaries for license renewal included, where applicable, non-safety related components that could affect safety-related components. The boundaries were expanded in accordance with current NRC guidance for implementation of 10 CFR 54.4(a)(2). The expanded scoping boundary includes:

AES non-safety related components containing steam and/or fluids that are located in an area that also contains a safety related component. The components are located in the Turbine Building.

The portions of the Main Air Ejection and Gland Seal System containing components subject to an AMR include fans, strainers, heat exchangers, filters, piping & fittings, fasteners, steam trap, and valves. The NSAS components added into scope by the 10 CFR 54.4(a)(2) scoping boundary expansion are subject to AMR.

System Function Listing

A comprehensive listing of functions associated with the Main Air Ejection and Gland Seal System is provided in the summary below.

System Function: AES-01	Cri 1	Cri 2	Cri 3				
Provide condenser vacuum via steam jet air ejectors.			FP	EQ	PTS	AT	SB

Comment: Maintaining condenser vacuum by removing noncondensible gases from the condenser supports plant start-up and normal operation. This function does not meet any of the LR criteria.

System Function: AES-02	Cri 1	Cri 2	Cri 3				
Provide decay heat removal via steaming of the steam generators through the hogging air ejector.			FP	EQ	PTS	AT	SB

Comment: Hot standby/hot shutdown decay heat removal can be accomplished by releasing steam to the atmosphere using the Atmospheric Dump Valves or the hogging air ejector. This is a normal operating function and is not in scope for license renewal. However, as described in FSAR 9.6.8.3, this same method and systems/components are used to accomplish pressure reduction and cooldown in Appendix R safe shutdown analysis. See AES-FP for the fire-protection-related function.

System Function: AES-03	Cri 1	Cri 2	Cri 3				
Provide steam seal on main turbine shaft to maintain condenser vacuum.			FP	EQ	PTS	AT	SB

Comment: This system function of maintaining condenser vacuum is to support plant start-up and normal operation only. No LR criteria are involved in this system function.

System Function: AES-FP	Cri 1	Cri 2	Cri 3				
The system contains structures and/or components required by the current licensing basis for fire protection.			FP	EQ	PTS	AT	SB
			X				

Comment: The AES system contains components that are required for Appendix R safe shutdown.

System Function: AES-NSAS	Cri 1	Cri 2	Cri 3				
This system has components in scope of license renewal in accordance with 10CFR54.4(a)(2) requirements for non-safety affecting safety (NSAS).			FP	EQ	PTS	AT	SB
		X					

Comment: The AES system contains non-safety related components containing liquids located in an area that also contains safety related components.

FSAR Reference

Additional Main Air Ejection and Gland Seal System details are provided in Section 10.2 of the FSAR.

Scoping Boundary Drawings

The scoping boundary drawings for the Main Air Ejection and Gland Seal System are listed below:

LR-M-205, Sheet 1
LR-M-206, Sheet 1B
LR-M-206, Sheet 1C
LR-M-207, Sheet 1B

Components Subject to an AMR

The component groups for this system that require aging management review are addressed in Table 2.3.4-5 along with each Component Group's intended function(s).

Table 2.3.4-5 Main Air Ejection and Gland Seal System

Component Group	Intended Function
Blowers Fans Compressor Vacuum	Fluid Pressure Boundary
Filters/Strainers	Fluid Pressure Boundary
Heat Exchangers	Fluid Pressure Boundary
Fasteners	Fluid Pressure Boundary
Pipe & Fittings	Fluid Pressure Boundary
Traps (Steam)	Fluid Pressure Boundary
Valves & Dampers	Fluid Pressure Boundary

2.3.4.6 Main Steam System

System Description

In the Main Steam System (MSS), steam generated in the steam generators passes through two 36-inch headers and main steam isolation valves (MSIVs) to the turbine stop valves. Each main steam header is provided with 12 spring-loaded safety valves (MSSVs) and 2 atmospheric dump valves (ADVs) upstream of the MSIVs. In addition, there is a steam bypass to condenser downstream of the MSIVs. The main steam line also supplies steam for the steam jet air ejectors, the heating steam for the reheaters, the steam supply to the steam generator feed pump turbine drivers, and steam supply to the turbine-driven auxiliary feed pump turbine driver which is connected upstream of an MSIV.

The MSIVs are provided to isolate the steam generators to (1) prevent the uncontrolled release of radioactivity in the unlikely event of a steam generator tube failure, (2) lessen a rapid uncontrolled cool down of the Primary Coolant System in the event of a main steam line break by limiting the blowdown from a single steam generator, and (3) limit the steam discharge to the containment from the affected steam generator in the event of the rupture of one main steam line inside containment. The design basis of the ADVs is to prevent lifting of the MSSVs following a turbine and reactor trip, and to provide the capability to cool the plant to Shutdown Cooling (SDC) System entry conditions when condenser vacuum is lost. The turbine bypass to the main condenser provides for removal of reactor decay heat following normal reactor shutdown. Overpressure protection for the shell side of the steam generators and the main steam line piping up to the inlet of the turbine stop valve is provided by the MSSVs. The MSSVs also provide protection against overpressurizing the Primary Coolant Pressure Boundary by providing a heat sink for the removal of energy from the Primary Coolant System if the preferred heat sink, provided by the condenser and Circulating Water System, is not available.

A subsystem of Main Steam is Steam Generator Blowdown. The steam generator blowdown subsystem is designed to process steam generator blowdown water. A minimum continuous blowdown is normally maintained for effective steam generator chemistry control. Other functions of the subsystem include the capability to clean up the condenser hotwell prior to start up by recirculating the water through the blowdown demineralizers, and the capability to recirculate steam generator secondary side water, for treatment purposes, during cold shutdown conditions.

The steam generator blowdown subsystem consists of flash tank, blowdown tank, two blowdown pumps, blowdown heat exchanger, blowdown filter, blowdown demineralizers, piping, valves and instrumentation. The subsystem is continuously monitored by a process monitor to detect radioactivity which may have leaked into the steam generator from the primary system.

The description above results in some SSCs in this system being considered in-scope in accordance with 10 CFR 54.4(a)(1). Since some SSCs in the Main Steam System are non-safety related and their failure could affect the capability of SR SSCs to perform their safety function, they are considered to be in-scope in accordance with 10 CFR 54.4(a)(2). In addition, some SSCs are considered in-scope due to ATWS, Environmental Qualification, Fire Protection, and Station Blackout in accordance with 10 CFR 54.4(a)(3).

The boundaries of the in-scope portions of the Main Steam System are described as follows: 1) The piping, valves and components from each S/G E-50A/B, through flow elements FE-0702/0704, through Containment penetrations 2 and 3 (including steam generator blowdown system to the flash tank, the main steam safety valves (MSSVs), and atmospheric dump valves (ADVs)), through the Main Steam Isolation Valves (MSIVs) to the main steam stop valves; 2) The piping, valves and components to the driving turbines of auxiliary feedwater pump and main feedwater pumps, and some segments of the main turbine crossunder piping and connections to the moisture separators and reheaters (MSRs); 3) The piping, valves, and components supplying steam through the steam jet air ejectors (SJAE) to the after-condenser; and 4) The piping, valves and components supplying steam through the hogging air ejector and silencer to outside the turbine building.

The original system scoping boundaries for license renewal included, where applicable, non-safety related components that could affect safety-related components. The boundaries have been expanded in accordance with current NRC guidance for implementation of 10 CFR 54.4(a)(2). The expanded scoping boundary includes:

MSS non-safety related piping and components that are directly attached to safety related piping and components out to the first seismic anchor or equivalent anchor. Where base mounted components are credited as an anchor, they are also included in scope. Pipe supports for the newly added piping and components are also in scope. The new piping components are located in the Containment Building, Auxiliary Building and the Turbine Building.

MSS non-safety related components containing steam and/or fluids that are located in an area that also contains a safety related component. The components are located in the Containment Building, Auxiliary Building and Turbine Building.

The portions of the Main Steam System containing components subject to an AMR include accumulators, strainers, filters, heat exchangers, pumps, level glasses, piping & fittings, fasteners, ejectors, steam traps and valves. The NSAS components added into scope by the 10 CFR 54.4(a)(2) scoping boundary expansion are subject to AMR.

System Function Listing

A comprehensive listing of functions associated with the Main Steam System, or specific components contained in the system, is provided in the summary below.

System Function: BLD-01	Cri 1	Cri 2	Cri 3				
Provide ability to sample each steam generator for activity and lithium as directed by the EOP.			FP	EQ	PTS	AT	SB

Comment: This function is not within the scope of license renewal. Steam Generator sampling for activity and lithium is one of several potential methods available for diagnosis of Steam Generator Tube Rupture (SGTR) to determine the most affected steam generator for isolation. This system function may be not available with loss of service water to the sample coolers. Other diagnosis information included in Emergency Operating Procedures includes Steam Generator level, Pressurizer pressure, Pressurizer level.

System Function: BLD-02	Cri 1	Cri 2	Cri 3				
Automatically close the containment isolation valves and blowdown system valves on Containment High Pressure or Containment High Radiation.			FP	EQ	PTS	AT	SB
	X						

Comment: This function is associated with the valves and pipe required to ensure containment isolation in the blowdown subsystem. Components associated with the processing of electrical signals to initiate automatic isolation upon CHP and CHR are dispositioned under the Containment Isolation and Penetration (CIS) system. The penetrations are dispositioned within the Containment Structure evaluation.

System Function: MSS-01	Cri 1	Cri 2	Cri 3				
Pathway for steam which provides motive force to the main turbine.			FP	EQ	PTS	AT	SB

Comment: Steam supply to the Main Turbine is to support normal operation only, and is outside the scope of license renewal. However, failure of some segments of steam pipes in the pathway may cause damage to some safety related components, such as Component Cooling Water pumps. The effects of failure of non-safety segments on the safety related components are addressed in System Function MSS-NSAS.

System Function: MSS-02	Cri 1	Cri 2	Cri 3				
Pathway for steam which provides motive force to the main feedwater pump turbines.			FP	EQ	PTS	AT	SB

Comment: Steam supply to the feedwater pump turbines is to support normal operation only, and is outside the scope of license renewal. However, failure of some segments of steam pipes in the pathway may cause damage to some safety related components. The effects of failure of non-safety segments on the safety related components are addressed in System Function MSS-NSAS.

System Function: MSS-03	Cri 1	Cri 2	Cri 3				
Pathway for steam which provides motive force to the auxiliary feedwater pump turbine.			FP	EQ	PTS	AT	SB
	X						

Comment:

System Function: MSS-04	Cri 1	Cri 2	Cri 3				
Provide a steaming path via the atmospheric steam dump valves during a plant transient or plant cooldown.			FP	EQ	PTS	AT	SB

Comment: Although ADVs are assumed to be used to cool down the plant to Shutdown Cooling entry condition, it is not a safety-related requirement to take the plant to cold shutdown. Palisades is licensed as a “Hot Shutdown” plant) following a transient. Thus this function of the ADVs does not meet Criterion #1. However, these ADVs are credited to meet the requirements of Station Blackout events, and are credited for Appendix R safe shutdown. The equivalent Criterion 3 functions are captured under function MSS-SB and MSS-FP.

System Function: MSS-05	Cri 1	Cri 2	Cri 3				
Provide a steaming path via the turbine bypass valve during a plant transient or plant cooldown.			FP	EQ	PTS	AT	SB

Comment: Although the turbine bypass valve can be used by the operator to cool down the plant to Shutdown Cooling entry conditions if the condenser, as the preferred heat sink, is available, this function is not credited for mitigation of plant transients.

System Function: MSS-06	Cri 1	Cri 2	Cri 3				
Close both MSIVs on receipt of a low Steam Generator pressure signal.			FP	EQ	PTS	AT	SB
	X						

Comment: Main steam isolation will occur on containment high pressure or steam generator low pressure (see FSAR Section 7.2.3.8) based on containment pressurization considerations for a steam line break. A low steam generator secondary pressure signal is provided to protect against excessively high steam flow caused by a steam line break.

System Function: MSS-07	Cri 1	Cri 2	Cri 3				
Close the MSIVs upon receipt of Containment High Pressure.			FP	EQ	PTS	AT	SB
	X						

Comment: Containment high-pressure (CHP) signal will initiate closure of the main steam isolation valves to reduce the inventory blowdown from the intact steam generator in the case of a main steam line break, reducing the peak containment pressure and temperature as required in the accident analysis.

System Function: MSS-08	Cri 1	Cri 2	Cri 3				
Prevent release of contents of the secondary side of the steam generator(s) in the event of a tube leak/rupture.			FP	EQ	PTS	AT	SB
	X						

Comment: In the event of a steam generator tube rupture, closure of the MSIVs isolates the affected steam generator from the intact steam generator and minimizes radiological releases.

System Function: MSS-09	Cri 1	Cri 2	Cri 3				
Provide passive overpressure protection of the Main Steam System.			FP	EQ	PTS	AT	SB
	X						

Comment: The Main Steam Safety Valves (MSSVs) provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the Primary Coolant pressure boundary by providing a heat sink for the removal of energy from the Primary Coolant System.

System Function: MSS-AT	Cri 1	Cri 2	Cri 3				
The system contains structures and/or components required by the current licensing basis for Anticipated Transients Without Scram.			FP	EQ	PTS	AT	SB
						X	

Comment: The design for compliance with the ATWS rule includes an independent and diverse turbine trip. This function only includes the closure of main turbine stop valves to support ATWS. The actuation of associated relays is covered in Turbine Generator System function TGS-AT.

System Function: MSS-EQ	Cri 1	Cri 2	Cri 3				
The system contains components required by the current licensing basis for Environmental Qualification.			FP	EQ	PTS	AT	SB
				X			

Comment:

System Function: MSS-FP	Cri 1	Cri 2	Cri 3				
The system contains structures and/or components required by the current licensing basis for Fire Protection.			FP	EQ	PTS	AT	SB
			X				

Comment: Some components of MSS are involved in the mitigation of this Regulated Event. Also, FSAR 9.6.8.3 indicates that for post fire safe shutdown, the systems and equipment which may be utilized for decay heat removal while maintaining the plant at hot standby or hot shutdown include the main steam system, the atmospheric steam dump valves, the hogging air ejector, and the main steam safety valves.

System Function: MSS-NSAS	Cri 1	Cri 2	Cri 3				
This system has components in scope of license renewal in accordance with 10CFR54.4(a)(2) requirements for non-safety affecting safety (NSAS).			FP	EQ	PTS	AT	SB
		X					

Comment: Some Main Steam components are non-safety related. They are attached to and located above safety related components. The MSS contains steam in an area that also contains safety related components.

System Function: MSS-SB	Cri 1	Cri 2	Cri 3				
The system contains structures and/or components required by the current licensing basis for Station Blackout (Loss of all AC power).			FP	EQ	PTS	AT	SB
							X

Comment: As described in FSAR 10.2.1, the atmospheric steam dump valves have a back up nitrogen supply to allow steam generator pressure control during station blackout.

FSAR Reference

Additional Main Steam System details are provided in Section 5.6, Section 7.3, Section 7.2, Section 7.5, and Section 9.6, Section 10.2, and Section 14.15 of the FSAR.

Scoping Boundary Drawings

The scoping boundary drawings for the Main Steam System are listed below:

LR-M-201, Sheet 1
LR-M-205, Sheet 1
LR-M-205, Sheet 1A
LR-M-205, Sheet 1B
LR-M-205, Sheet 2
LR-M-205, Sheet 2A
LR-M-206, Sheet 1
LR-M-206, Sheet 1B
LR-M-207, Sheet 1
LR-M-210, Sheet 1
LR-M-212, Sheet 3
LR-M-222, Sheet 1
LR-M-223, Sheet 1B
LR-M-226, Sheet 1
LR-M-226, Sheet 1A
LR-M-651, Sheet 1

Components Subject to an AMR

The component groups for this system that require aging management review are addressed in Table 2.3.4-6 along with each Component Group's intended function(s).

Table 2.3.4-6 Main Steam System

Component Group	Intended Function
Accumulators	Fluid Pressure Boundary
Ejectors	Fluid Pressure Boundary
Fasteners	Fluid Pressure Boundary
Filters/Strainers	Fluid Pressure Boundary
Heat Exchangers	Fluid Pressure Boundary
Indicators/Recorders (Level glasses)	Fluid Pressure Boundary
Pipe & Fittings	Fluid Pressure Boundary
Pumps	Fluid Pressure Boundary
Traps (Steam)	Fluid Pressure Boundary
Valves & Dampers	Fluid Pressure Boundary

2.3.4.7 Turbine Generator System

System Description

The Turbine Generator System (TGS) consists of the main turbines, main electrical generator and various supporting subsystems. The turbine is an 1,800 r/min tandem compound, 3 cylinder, quadruple flow, indoor unit. Saturated steam is supplied to the turbine throttle from the steam generators through four stop valves and four governing control valves. The steam flows through a double-flow, high-pressure turbine and four combination moisture separator-reheaters in parallel, and two double-flow, low-pressure turbines that exhaust to the main condenser.

Turbine control is accomplished with a rapid response electrohydraulic control system. In the event of turbine trip initiated from a solenoid trip, overspeed, low bearing oil pressure, low condenser vacuum, thrust bearing failure or a manual trip, a signal is supplied from the turbine auto-stop oil system to the Reactor Protective System to trip the reactor.

The turbine lubricating oil system supplies oil for lubricating the bearings. A bypass stream of turbine lubricating oil flows continuously through coalescing filters to remove water and other impurities.

The generator is a hydrogen inner cooled unit connected directly to the turbine. It is rated for 955 MVA and has the capability to accept the gross output of the turbine at rated steam conditions. The generator is made up of a housing, stator, exciter, rotor and shaft with sleeve bearings and ventilation blower. Generator operation is supported by a hydrogen gas system, a seal oil system and a signal system.

The turbine generator controls are the means by which the turbine generator is made to meet the electrical load demand placed upon it. The controls consist of the following five parts: 1) Operators Interface Panel, 2) Digital Controller & Engineers Console, 3) Steam valve servo actuator assemblies, 4) High-pressure fluid supply, and 5) Emergency trip. The electronic controller performs basic digital computations on reference signals and turbine feed-back signals and generates an output to the steam valve actuators. Manual inputs and manual functions, plus valve contingencies are also built-in capabilities.

Since the failure of some non-safety related SSCs in the Turbine Generator System could affect the capability of SR SSCs to perform their safety function, they are considered to be in-scope in accordance with 10 CFR 54.4(a)(2). In addition, some SSCs are considered in-scope due to ATWS in accordance with 10 CFR 54.4(a)(3).

The boundaries of in-scope portions of the Turbine Generator and Crane System are described as follows: 1) The HP turbine steam inlet drains to the condenser, 2) portions of the HP turbine glands seals and gland seals exhaust located below elevation 625', 3) portions of the LR turbine gland seals and gland seals exhaust located below elevation 625', 4) Hydrogen Side Seal Oil Cooler (E-16A), 5) Air Side Seal Oil Cooler (E16B), 6) portions of the bearing oil supply piping, the high pressure seal oil backup piping, the low pressure seal oil backup piping and the bearing/control oil return piping located below elevation 625', 7) the Electro-Hydraulic Control System piping, valves, and components located below elevation 625', and 8) the Main Generator Seal Oil System piping, valves and components located below elevation 625'.

The original system scoping boundaries for license renewal included, where applicable, non-safety related components that could affect safety-related components. The boundaries have been expanded in accordance with current NRC guidance for implementation of 10 CFR 54.4(a)(2). The expanded scoping boundary includes:

TGS non-safety related components containing steam and/or fluids that are located in an area that also contains a safety related component. The component types include accumulators, heat exchangers, fasteners (to maintain the pressure boundary), filters, piping, fittings, pumps, and valves. The components are located in the Turbine Building.

The NSAS components added into scope by the 10 CFR 54.4(a)(2) scoping boundary expansion are subject to AMR.

System Function Listing

A comprehensive listing of functions associated with the Turbine Generator System is provided in the summary below.

System Function: DEH-01	Cri 1	Cri 2	Cri 3				
Provide control and monitoring of the turbine governor and stop valves which controls turbine speed and generator load.			FP	EQ	PTS	AT	SB

Comment: This system function does not support a license renewal criterion.

System Function: DEH-02	Cri 1	Cri 2	Cri 3				
Provide turbine protective trips.			FP	EQ	PTS	AT	SB

Comment: This system function does not support a license renewal criterion.

System Function: EHC-01	Cri 1	Cri 2	Cri 3				
Provide high pressure fluid which acts as a motive force to the main turbine steam admission valves and repositions the valves in response to electronic commands.			FP	EQ	PTS	AT	SB

Comment: This system function does not support a license renewal criterion.

System Function: EHC-02 Provide direct turbine protective trips and provide a reactor trip signal to the RPS.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This system function provides an immediate reactor trip upon a turbine trip. The trip is not taken credit for in FSAR Section 14.12. Additionally, FSAR Section 7.2.3.6 states: “The loss of load reactor trip is an anticipatory trip which is not required to protect the reactor since the primary trip is high primary system pressure.” Therefore, this system function does not support a license renewal criterion.

System Function: TGS-01 Provide motive force to the main generator.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This system function does not support a license renewal criterion.

System Function: TGS-02 Provide control and monitoring of the main generator excitation.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This system function does not support a license renewal criterion.

System Function: TGS-03 Provide lubrication and seal oil between the main generator shaft and seal.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This system function does not support a license renewal criterion.

System Function: TGS-04 Provide isophase bus cooling.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This system function does not support a license renewal criterion.

System Function: TGS-05 Provide gland sealing steam to the main turbine.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This system function does not support a license renewal criterion.

System Function: TGS-06 Provide a 10 second time delay on tripping the main generator output breaker following a turbine trip occurring simultaneously with a loss of offsite power.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This system function adds additional main coolant pump coastdown time to facilitate core heat removal following a turbine trip. FSAR Section 4.4.1 states: “The most severe electrical failure analyzed, however, is the loss of offsite power and no turbine coastdown. The results are within core design limits and are presented in Section 14.7." Therefore, this system function does not support a license renewal criterion.

System Function: TGS-07 Close the turbine stop valves and the governor valves on demand.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This system function does not support a license renewal criterion. The turbine stop and governor valves are included in the Main Steam System (MSS).

System Function: TGS-08 Circulate cooled hydrogen through the main generator.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This system function does not support a license renewal criterion.

System Function: TGS-09 Provide electrical connection between main generator and main transformer.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This system function does not support a license renewal criterion.

System Function: TGS-10 Produce electrical power for transmission to the power grid.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This system function does not support a license renewal criterion.

System Function: TGS-11 Generate, transfer and regulate the field strength that creates and controls the Main Generator's power output.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This system function does not support a license renewal criterion.

System Function: TGS-AT The system contains structures and/or components required by the current licensing basis for Anticipated Transients Without Scram.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
						X	

Comment: Turbine trip signal is an input to ATWS. The signal circuitry is in TGS. The turbine stop valves are in the Main Steam System (MSS).

System Function: TGS-NSAS This system has components in scope of license renewal in accordance with 10CFR54.4(a)(2) requirements for non-safety affecting safety (NSAS).	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: The TGS has non-safety related components containing liquids located in an area that also contains safety related components.

FSAR Reference

Additional Turbine Generator System details are provided in Section 7.5, Section 7.2, Section 14.12, Section 14.7, Section 4.4, and Section 8.4 of the FSAR.

Scoping Boundary Drawings

The scoping boundary drawings for the Turbine Generator System are listed below:

LR-M-205, Sheet 1
LR-M-205, Sheet 1A
LR-M-205, Sheet 1B
LR-M-206, Sheet 1B
LR-M-208, Sheet 1
LR-M-214 Sheet 2A
LR-M-214, Sheet 2B

Components Subject to an AMR

The component groups for this system that require aging management review are addressed in Table 2.3.4-7 along with each Component Group's intended function(s).

Table 2.3.4-7 Turbine Generator System

Component Group	Intended Function
Accumulators	Fluid Pressure Boundary
Fasteners	Fluid Pressure Boundary
Filters/Strainers	Fluid Pressure Boundary
Heat Exchangers	Fluid Pressure Boundary
Pipe & Fittings	Fluid Pressure Boundary
Pumps	Fluid Pressure Boundary
Valves & Dampers	Fluid Pressure Boundary

2.4 Scoping and Screening Results: Containments, Structures, and Component Supports

The following structures and structural commodities are addressed in this section:

- Auxiliary Building (Section 2.4.1)
- Component Supports (Section 2.4.2)
- Containment (Section 2.4.3)
- Containment Interior Structures (Section 2.4.4)
- Discharge Structure (Section 2.4.5)
- Feedwater Purity Building (Section 2.4.6)
- Intake Structure (Section 2.4.7)
- Miscellaneous Structural and Bulk Commodities (Section 2.4.8)
- Switchyard and Yard Structures (Section 2.4.9)
- Turbine Building (Section 2.4.10)

2.4.1 Auxiliary Building

Description

The Auxiliary Building, with the exception of the administration area and the access control area, is a Class 1 structure. The following facilities, systems and equipment are among those located in the auxiliary building:

- Control room
- Emergency diesel generators and related auxiliaries
- New and spent fuel handling, storage and shipment facilities
- Radwaste, chemical and volume control equipment
- Safety Injection System (majority)
- Component Cooling System (majority)
- Containment Spray System (majority)

The reinforced concrete enclosure containing the engineered safeguards equipment is located below grade. It is partitioned into two rooms so that one room is operable in the event a pipe rupture floods the other room. The partition wall is designed to withstand hydrostatic loading over its full height. This building also houses the access control area, which controls access to and exit from the various radiation controlled zones.

The new and spent fuel pools are located adjacent to the three-story concrete enclosure, which houses the control room, switchgear and the emergency diesel generators. To ensure that the spent fuel pool and tilt pits have a high degree of leak tightness, the walls and floors of these cavities are lined with stainless steel plates. Monitoring trenches have been provided behind the liner plates to detect any leakage that might occur.

The main steam and main feedwater lines pass through the southwest corner of the auxiliary building. In this region, pipe whip restraints are provided on the turbine building side of the main steam and main feedwater isolation valves. These structural steel frames are anchored to various auxiliary building concrete walls and slabs. In addition, the jet forces resulting from a failure in either one of these lines were considered in the design of the walls and slabs that separate these lines from the switchgear and cable spreading rooms and from the containment ventilation isolation valves.

At elevation 590' there are electrical manholes in the 2.4kV Switchgear Room associated with safety related Bus 1-C, used for routing cables to various plant areas. Duct banks (reinforced concrete encasing rigid steel conduit) run from these manholes, connecting to another manhole just outside of the Auxiliary Building.

Auxiliary Building Radwaste Addition

During 1972 a 124’ x 38’ x 106’-high addition (approximate maximum dimensions) was added to the north side of the auxiliary building. This Class 1 structure houses additional gaseous, liquid and solid radwaste equipment. The building is constructed on a mat foundation. The reinforced concrete slabs and walls were designed as two-way slabs and bearing walls.

Auxiliary Building TSC/EER/HVAC Addition

During 1983 a reinforced concrete addition was appended to the north side of the auxiliary building for a technical support center (TSC), an electrical equipment room (EER) and a heating, ventilating and air conditioning (HVAC) area. The TSC was constructed pursuant to NUREG-0696, the HVAC area as a result of the control room habitability requirements of NUREG-0737, and the EER area as a result of loads placed on the electrical system by the addition of the TSC and HVAC areas. The L-shaped structure was added to the roofs of the existing waste gas decay tank/baler room (elevation 607’ - 6”) and the diesel generator exhaust muffler enclosure (elevation 629’ - 2”.

The portions of the Auxiliary Building that are in-boundary and contain components subject to an AMR include the original Auxiliary Building (except administrative areas), the Auxiliary Building Radwaste Addition and the Auxiliary Building TSC/EER/HVAC

Addition. These structures shelter or support ATWS, EQ, FP (Appendix R), SBO, Safety Related and/or NSAS components.

System Function Listing

A comprehensive listing of functions associated with the Auxiliary Building, or specific components contained in the structure, is provided in the summary below.

Code S0200-AT	Cri 1	Cri 2	Cri 3				
Provides shelter or support to ATWS related components.			FP	EQ	PTS	AT	SB
						X	

Comment: Most ATWS components are located in the Control Room and Cable Spreading Room.

Code S0200-EQ	Cri 1	Cri 2	Cri 3				
Provides shelter or support to EQ related components.			FP	EQ	PTS	AT	SB
				X			

Comment: Many EQ components are in the Class 1 portion of the Auxiliary Building.

Code S0200,B-FP	Cri 1	Cri 2	Cri 3				
Provides rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant.			FP	EQ	PTS	AT	SB
			X				

Comment: Auxiliary Building contains Appendix R fire barriers.

Code S0200-SB	Cri 1	Cri 2	Cri 3				
Provides shelter or support to SBO related components			FP	EQ	PTS	AT	SB
							X

Comment: Several systems required to perform during SBO are in the Class 1 portion of the Auxiliary Building.

Code S0200-SR	Cri 1	Cri 2	Cri 3				
Provides shelter/protection to safety related component.			FP	EQ	PTS	AT	SB
	X						

Comment: Most areas of the Auxiliary Building are Consumers Design Class 1 (Category 1).

Code S0200B-NSAS Provides storage/handling of spent fuel assemblies and provides structural support of safety related components	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: The Access Control area of the Auxiliary Building is Consumers Design Class 3. However, concrete walls below the roof slab of Access Control Area supports the Load Distribution System (LDS) that is used to transport spent fuel assemblies. They are in-scope of 10CFR 54.4. These walls are part of the Auxiliary Building framing system which supports safety related components.

Code Architectural (BLA)-FP2 Provides access to remote shutdown panels & components in the event of a fire	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: Provides access for the operator from the Control Room to Redundant Safety Injection Panel EC-33 in Auxiliary Building and Hot Shutdown Panels EC-150 & EC-150A in Turbine Building for alternate shutdown station. Other Auxiliary Building platforms, stairs, and similar commodities that are necessary to provide access to the operators that affects safety operations are in-scope.

FSAR Reference

Additional Auxiliary Building details are described in Section 5.9 of the FSAR.

Scoping Boundary Drawings

The scoping boundary drawings for the Auxiliary Building are listed below:

LR-C3

Components/Commodities Subject to an AMR

The component groups for the Auxiliary Building that require aging management review are addressed in Table 2.4.1-1 along with each component group's intended function(s).

Table 2.4.1-1 Auxiliary Building

Component Group	Intended Function
Building Framing - Carbon Steel, Protected (column, hanger, beam, truss, decking, floor grating or plate, catwalk, threaded fastener, concrete expansion bolt, column base plate, weld, etc.)	HELB Shielding Pipe Whip Restraint Shelter/ Protection Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related
Building Framing - Concrete, Below Grade (wall footing, foundation slab, grout, reinforcement, duct banks, cable pits, tunnels, etc.)	Flood Protection Structural Support for Safety Related
Building Framing - Concrete, Exposed (foundations, concrete & masonry wall, beam, roof slab, grout, reinforcements, concrete around expansion & grouted anchors)	Flood Protection Missile Barrier Radiation Shielding Shelter/ Protection Structural Support for Safety Related

Table 2.4.1-1 Auxiliary Building

Component Group	Intended Function
Building Framing - Concrete, Protected (foundations, concrete & masonry wall, column, pedestal, beam, floor slab, grout, reinforcements, concrete around expansion & grouted anchors, cable pits, tunnels, etc.)	Direct Flow Fire Barrier HELB Shielding Missile Barrier Pipe Whip Restraint Radiation Shielding Shelter/ Protection Structural Support for Safety Related
Flood Barrier - Carbon Steel, Exposed (water tight doors)	Flood Protection Structural Support for Safety Related
Flood Barrier - Carbon Steel, Protected (water tight doors and gate)	Flood Protection Structural Support for Safety Related
Fuel Related component - Carbon Steel, Protected (anchor bolts for SFP gates, liners, transfer tube appurtenances)	Structural Support for Safety Related
Fuel Related Component - Stainless, Protected (anchor bolts for SFP gates, liners, transfer tube appurtenances)	Structural Support for Safety Related

Table 2.4.1-1 Auxiliary Building

Component Group	Intended Function
Fuel Related Component - Stainless, Borated (liner plates, gates, transfer tube expansion bellows)	Expansion / Separation Fluid Pressure Boundary Structural Support for Safety Related
HELB/MELB Component - Carbon Steel, Protected (doors, scuttle, blowout panels, floor drains & screens, guard pipes, louvers, whip restraints, bellows, spray shields, etc.)	Direct Flow Flood Protection HELB Shielding Pipe Whip Restraint
HELB/MELB Component - Concrete, Protected (curbs & pipe whip restraint grout, concrete at locations of expansion & grouted anchors)	Direct Flow Flood Protection HELB Shielding Pipe Whip Restraint
HVAC Component - Stainless, Protected (control room vestibule doors)	Fluid Pressure boundary Structure Functional Support
HVAC Component - Carbon Steel, Protected (Control Room vestibule door)	Fluid Pressure Boundary Structural Support for Safety Related
HVAC Component - Concrete, Protected (Control Room vestibules, concrete & masonry walls, floors, ceilings)	Fluid Pressure Boundary Structural Support for Safety Related
HVAC Component - Galvanized, Protected (damper & louver frames)	Fire Barrier Structural Support for Safety Related

Table 2.4.1-1 Auxiliary Building

Component Group	Intended Function
Missile Shield - Carbon Steel, Protected (steel doors and structural steel missile barrier)	Missile Barrier Structural Support for Safety Related
Operator Access Component - Carbon Steel, Protected (stairs, floors, platforms, etc.)	Structural Support for Non-Safety Related Structural Support for Regulated Events
Operator Access Component - Concrete, Protected (stairs, floors, platforms, concrete at locations of expansion & grouted anchors)	Structural Support for Non-Safety Related Structural Support for Regulated Events
Operator Access Component - Galvanized, Protected (stairs, walkways, removable platform, welds, bolted connections)	Structural Support for Non-Safety Related Structural Support for Regulated Events

2.4.2 Component Supports

Description

This commodity group includes ASME and non-ASME pipe supports, ASME Class 1 equipment supports, as well as general mechanical and electrical component supports (e.g., HVAC, cable trays, conduits, etc.). These assets consist primarily of steel and concrete assets included under GALL Section IIIB for supports and GALL Section VII A1 and A2 for new fuel and spent fuel storage, respectively. They are grouped together based on their similar “support” function, material, environments, and expected aging management programs. This commodity group includes the following categories of support components:

ASME CLASS 1 PIPING & MECHANICAL COMPONENT SUPPORTS

This support commodity includes as applicable, but is not limited to, support/hanger/frame/rack/housing/skid composed of prefab or rolled steel shapes, rods, clamps, fasteners (screws, bolts, clips, concrete expansion bolts, inserts, anchor bolts) and welds. ASME Class 1 Components include Reactor Vessel, Steam Generators, Primary Coolant Pumps, Pressurizer, and Regenerative Heat Exchanger. Also included are stainless steel pipe support lug, rolled plates, shapes, and Incore Instrumentation (ICI) Support Stand Assemblies; Reactor Head Vent pipe supports; Lubrite bearing plates; building concrete at locations of expansion and grouted anchors; and grout pads for support base plates. (GALL IIIB1)

ASME CLASS 2 & 3 PIPING & MECHANICAL COMPONENT SUPPORTS

This support commodity includes as applicable, but is not limited to, support/hanger/frame/rack/housing/skid composed of prefab or rolled steel shapes, fasteners (screws, bolts, clips, concrete expansion bolts, inserts, anchor bolts) and welds. Also included are manholes, Lubrite or Fluorogold bearing plates, building concrete at locations of expansion and grouted anchors, and grout pads for support base plates. (GALL IIIB1)

NON-ASME PIPING & MECHANICAL COMPONENT SUPPORTS

This support commodity includes as applicable, but is not limited to: support/hanger/tray/frame/rack composed of prefab or rolled steel shapes, rods, clamps, fasteners (screws, bolts, clips, concrete expansion bolts, inserts, anchor bolts) and welds. Commodity includes support and anchorage of HVAC Ducts, TubeTrack, instrument tubing, manhole covers, building concrete at locations of expansion and grouted anchors, and grout pads for support base plates. This includes supports for Emergency Diesel Generator (EDG), HVAC System Components, and other miscellaneous mechanical equipment. (GALL IIIB2 & B4)

ELECTRICAL COMPONENT SUPPORTS

This support commodity includes, as applicable, electrical component supports; anchorage of cable trays, conduit, racks, panels, cabinets, and enclosures for electrical equipment and instrumentation; support/hanger/tray/frame/rack composed of prefab or rolled steel shapes, fasteners (screws, bolts, clips, concrete expansion bolts, inserts, anchor bolts) and welds; manholes, cabinets, panels, and boxes; building concrete at locations of expansion and grouted anchors; and grout pads for support base plates. (GALL IIIB2 & B3)

NEW FUEL STORAGE RACK

This commodity is the aluminum New Fuel Storage Rack in the Auxiliary Building in the new fuel storage pit. (GALL VIIA1)

SPENT FUEL STORAGE RACK-AUXILIARY BUILDING

This commodity is the stainless steel spent fuel racks. It also includes the boron carbide neutron absorber material credited in the spent fuel pool criticality analysis. Note that Boraflex is not credited in the criticality analysis. Supports/anchorage for the racks are included.(GALL VIIA2)

The portions of the Component Supports commodity containing components subject to an AMR include ASME and non-ASME pipe supports, ASME Class 1 equipment supports (including new fuel and spent fuel storage), as well as general mechanical and electrical component supports (e.g., HVAC, cable trays, conduits, etc.).

System Function Listing

A comprehensive listing of functions associated with Component Supports is provided in the summary below.

Code ELECTRICAL SUPPORT (BLE) -AT	Cri 1	Cri 2	Cri 3				
Provides Civil/Structural supports for Electrical ATWS related commodities			FP	EQ	PTS	AT	SB
						X	

Comment: ATWS components are located in the Control Room (Auxiliary Bldg.), Cable Spreading Room (Auxiliary Bldg.), and Containment Bldg.

Code ELECTRICAL SUPPORT (BLE) -EQ	Cri 1	Cri 2	Cri 3				
Provides Civil/Structural supports for Electrical EQ related commodities			FP	EQ	PTS	AT	SB
				X			

Comment: Electrical components that are within the scope of EQ Program are basically identified within different Mechanical & Electrical Plant Systems. Civil/Structural supports for these components are within the scope of 10CFR 54. Various Civil/Structural devices protect and ensure the environmental conditions for electrical components evaluated in the EQ Program.

Code ELECTRICAL SUPPORT (BLE) -FP	Cri 1	Cri 2	Cri 3				
Provides Civil/Structural supports for Electrical Fire Protection related commodities.			FP	EQ	PTS	AT	SB
			X				

Comment: Electrical components supporting Fire Protection System are identified in Mechanical system FPS, Fire Protection. Supports for these components are within the scope of 10CFR54. Civil/Structural supports for Fire Protection System electrical components include, but are not limited to alarms and detectors

Code ELECTRICAL SUPPORT (BLE) -NSAS Provides Civil/Structural supports for non-safety related Electrical commodities affecting safety related commodities.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: This Civil/Structural group of supports are non-safety; however, they include components designed seismically to assure that the non-safety related electrical components being supported will not fall on other safety related components and prevent them from functioning as intended. They also include supports for non-safety related equipment credited with maintaining the functions of safety related equipment and whose failure could affect safety related components; such as, the Warm Water Recirc Pump that mitigates Service Water System icing. Anchorages and supports of non-safety related ancillary items (dollies, gas bottles, etc.), located such that during an earthquake, they might be dislodged and impact and damage safety related equipment during an earthquake are within the scope the 10CFR 54.

Code ELECTRICAL SUPPORT (BLE) -SB Provides Civil/Structural supports for Electrical SBO related commodities.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
							X

Comment: Provides support to Station Blackout components.

Code ELECTRICAL SUPPORT (BLE) -SR Provides Civil/Structural supports for safety related Electrical commodities.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Supports for safety related Class 1E equipment are safety related Civil/Structural supports.

Code MECHANICAL SUPPORT (BLM) -EQ Provides Civil/Structural supports for Mechanical EQ related commodities.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
				X			

Comment: Mechanical components that are within the scope of EQ Program are identified within different Mechanical & Electrical Plant Systems. Civil/Structural supports for these components are within the scope of 10CFR 54. Examples of these commodities are: flooding curbs, pipe whip restraints, jet impingement and spray shields.

Code MECHANICAL SUPPORT (BLM) -FP Provides Civil/Structural supports for Mechanical Fire Protection System	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: Mechanical components for the Fire Protection System are identified in Mechanical system FPS, Fire Protection. Supports for these components are within the scope of 10CFR 54. Also includes components credited in the appendix R safe shutdown analysis.

Code MECHANICAL SUPPORT (BLM) -NSAS Provides Civil/Structural supports for non-safety related Mechanical commodities affecting safety related commodities.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: This Civil/Structural group of supports are non-safety; however, they are in-scope of 10CFR54.4 to assure that the non-safety related mechanical components being supported will not fall on other safety related components and prevent them from functioning as intended.

Code MECHANICAL SUPPORT (BLM) -SR Provides Civil/Structural supports for safety related Mechanical commodities and fuel storage.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Supports for safety related Mechanical commodities meet Criterion 1 of 10CFR54.4 and are therefore, in-scope for License Renewal as a safety related Civil/Structural support commodity.

FSAR Reference

None.

Scoping Boundary Drawings

None.

Components/Commodities Subject to an AMR

The component groups for Component Supports that require aging management review are addressed in Table 2.4.2-1 along with each component group's intended function(s).

Table 2.4.2-1 Component Supports

Component Group	Intended Function
ASME 1 Support - Containment - Carbon Steel, Protected	Structural Support for Safety Related
ASME 1 Support - Containment - Concrete, Protected	Structural Support for Safety Related
ASME 1 Support - Containment - Sliding Material, Cont Cavity	Expansion/Separation Structural Support for Safety Related

Table 2.4.2-1 Component Supports

Component Group	Intended Function
ASME 1 Support - Containment - Sliding Material, Protected	Expansion/Separation Structural Support for Safety Related
ASME 2 & 3 Support - Turbine (Water Treatment Area) - Carbon Steel, Protected	Structural Support for Regulated Events Structural Support for Safety Related
ASME 2 & 3 Support - Turbine (Water Treatment Area) - Concrete, Protected	Structural Support for Regulated Events Structural Support for Safety Related
ASME Class 1 Tubing Support - Auxiliary Bldg, Carbon Steel, Protected	Structural Support for Safety Related
ASME Class 2 & 3 Piping & Mechanical Component Support - Auxiliary Bldg, Carbon Steel, Protected	Structural Support for Regulated Events Structural Support for Safety Related
ASME Class 2 & 3 Piping & Mechanical Component Support - Auxiliary Bldg, Cast Iron, Protected	Structural Support for Safety Related
ASME Class 2 & 3 Piping & Mechanical Component Support - Auxiliary Bldg, Concrete, Protected	Structural Support for Regulated Events Structural Support for Safety Related
ASME Class 2 & 3 Piping & Mechanical Component Support - Auxiliary Bldg, Galvanized, Protected	Structural Support for Regulated Events Structural Support for Safety Related

Table 2.4.2-1 Component Supports

Component Group	Intended Function
ASME Class 2 & 3 Piping & Mechanical Component Support - Auxiliary Bldg, Sliding Material, Protected	Expansion/Separation Structural Support for Safety Related
ASME Class 2 & 3 Piping & Mechanical Component Support - Containment Bldg, Carbon Steel, Cont Cavity	Structural Support for Safety Related
ASME Class 2 & 3 Piping & Mechanical Component Support - Containment Bldg, Carbon Steel, Protected	Structural Support for Safety Related
ASME Class 2 & 3 Piping & Mechanical Component Support - Containment Bldg, Concrete, Protected	Structural Support for Safety Related
ASME Class 2 & 3 Piping & Mechanical Component Support - Containment Bldg, Galvanized, Cont Cavity	Structural Support for Safety Related
ASME Class 2 & 3 Piping & Mechanical Component Support - Containment Bldg, Galvanized, Protected	Structural Support for Safety Related
ASME Class 2 & 3 Piping & Mechanical Component Support - Containment Bldg, Sliding Material, Protected	Expansion/Separation Structural Support for Safety Related
ASME Class 2 & 3 Piping & Mechanical Component Support - Containment Bldg, Stainless, Borated	Structural Support for Safety Related
ASME Class 2 & 3 Piping & Mechanical Component Support - Intake Structure Bldg, Carbon Steel, Protected	Structural Support for Safety Related
ASME Class 2 & 3 Piping & Mechanical Component Support - Intake Structure Bldg, Concrete, Protected	Structural Support for Safety Related
ASME Class 2 & 3 Piping & Mechanical Component Support - Intake Structure Bldg, Galvanized, Protected	Structural Support for Safety Related

Table 2.4.2-1 Component Supports

Component Group	Intended Function
ASME Class 2 & 3 Piping & Mechanical Component Support - Turbine Bldg, Carbon Steel, Protected	Structural Support for Regulated Events Structural Support for Safety Related
ASME Class 2 & 3 Piping & Mechanical Component Support - Turbine Bldg, Concrete, Protected	Structural Support for Regulated Events Structural Support for Safety Related
ASME Class 2 & 3 Piping & Mechanical Component Support - Turbine Bldg, Galvanized, Protected	Structural Support for Regulated Events Structural Support for Safety Related
ASME Class 2 & 3 Piping & Mechanical Component Support - Turbine Bldg, Sliding, Protected	Expansion/Separation Structural Support for Safety Related
ASME Class 2 & 3 Piping & Mechanical Component Support - Yard, Aluminum, Exposed	Structural Support for Safety Related
ASME Class 2 & 3 Piping & Mechanical Component Support - Yard, Carbon Steel, Below Grade	Structural Support for Regulated Events Structural Support for Safety Related
ASME Class 2 & 3 Piping & Mechanical Component Support - Yard, Carbon Steel, Exposed	Structural Support for Regulated Events Structural Support for Safety Related
ASME Class 2 & 3 Piping & Mechanical Component Support - Yard, Concrete, Exposed	Structural Support for Regulated Events Structural Support for Safety Related

Table 2.4.2-1 Component Supports

Component Group	Intended Function
Elec Component Support - Auxiliary Bldg, Carbon Steel, Protected	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related
Elec Component Support - Auxiliary Bldg, Carbon Steel, Raw Water	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related
Elec Component Support - Auxiliary Bldg, Concrete, Protected	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related
Elec Component Support - Auxiliary Bldg, Concrete, Raw Water	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related
Elec Component Support - Auxiliary Bldg, Galvanized, Protected	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related

Table 2.4.2-1 Component Supports

Component Group	Intended Function
Elec Component Support - Auxiliary Bldg, Galvanized, Raw Water	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related
Elec Component Support - Containment Bldg, Carbon Steel, Containment Cavity	Structural Support for Safety Related
Elec Component Support - Containment Bldg, Carbon Steel, Protected	Structural Support for Safety Related
Elec Component Support - Containment Bldg, Concrete, Protected	Structural Support for Safety Related
Elec Component Support - Containment Bldg, Galvanized, Containment Cavity	Structural Support for Safety Related
Elec Component Support - Containment Bldg, Galvanized, Protected	Structural Support for Safety Related
Elec Component Support - Discharge Structure, Carbon Steel, Protected	Structural Support for Non-Safety Related
Elec Component Support - Discharge Structure, Galvanized, Protected	Structural Support for Non-Safety Related
Elec Component Support - Intake Structure Bldg, Carbon Steel, Protected	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related

Table 2.4.2-1 Component Supports

Component Group	Intended Function
Elec Component Support - Intake Structure Bldg, Concrete, Protected	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related
Elec Component Support - Intake Structure Bldg, Galvanized, Protected	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related
Elec Component Support - Switch Yard Relay House Group Bldg, Carbon Steel, Protected	Structural Support for Regulated Events
Elec Component Support - Switch Yard Relay House Group Bldg, Concrete, Protected	Structural Support for Regulated Events
Elec Component Support - Switch Yard Relay House Group Bldg, Galvanized, Protected	Structural Support for Regulated Events
Elec Component Support - Turbine Bldg, Carbon Steel, Protected	Structural Support for Regulated Events
Elec Component Support - Turbine Bldg, Concrete, Protected	Structural Support for Regulated Events
Elec Component Support - Turbine Bldg, Galvanized, Protected	Structural Support for Regulated Events
Elec Component Support - Yard, Carbon Steel, Exposed	Structural Support for Regulated Events
Elec Component Support - Yard, Concrete, Exposed	Structural Support for Regulated Events

Table 2.4.2-1 Component Supports

Component Group	Intended Function
Elec Component Support - Yard, Galvanized, Exposed	Structural Support for Regulated Events
Elec Component Support - Yard, Galvanized, Raw Water	Structural Support for Regulated Events
High Strength Bolting - Containment Building, Carbon Steel, Protected	Structural Support for Safety Related
Non-ASME Component Support - Auxiliary Bldg, Aluminum, Protected	Structural Support for Safety Related
Non-ASME Piping & Mechanical Component Support - Auxiliary Bldg, Carbon Steel, Protected	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related
Non-ASME Piping & Mechanical Component Support - Auxiliary Bldg, Concrete, Protected	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related
Non-ASME Piping & Mechanical Component Support - Auxiliary Bldg, Galvanized, Protected	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related
Non-ASME Piping & Mechanical Component Support - Auxiliary Bldg, Stainless, Borated	Structural Support for Safety Related
Non-ASME Piping & Mechanical Component Support - Boiler Building, Carbon Steel, Protected	Structural Support for Non-Safety Related

Table 2.4.2-1 Component Supports

Component Group	Intended Function
Non-ASME Piping & Mechanical Component Support - Boiler Building, Concrete, Protected	Structural Support for Non-Safety Related
Non-ASME Piping & Mechanical Component Support - Containment Bldg, Aluminum, Protected	Structural Support for Non-Safety Related
Non-ASME Piping & Mechanical Component Support - Containment Bldg, Carbon Steel, Containment Cavity	Structural Support for Non-Safety Related Structural Support for Safety Related
Non-ASME Piping & Mechanical Component Support - Containment Bldg, Carbon Steel, Protected	Structural Support for Non-Safety Related Structural Support for Safety Related
Non-ASME Piping & Mechanical Component Support - Containment Bldg, Concrete, Protected	Structural Support for Non-Safety Related Structural Support for Safety Related
Non-ASME Piping & Mechanical Component Support - Containment Bldg, Galvanized, Containment Cavity	Structural Support for Non-Safety Related Structural Support for Safety Related
Non-ASME Piping & Mechanical Component Support - Containment Bldg, Galvanized, Protected	Structural Support for Non-Safety Related Structural Support for Safety Related
Non-ASME Piping & Mechanical Component Support - Containment Bldg, Stainless, Borated	Structural Support for Non-Safety Related Structural Support for Safety Related
Non-ASME Piping & Mechanical Component Support - Discharge Structure, Carbon Steel, Protected	Structural Support for Non-Safety Related

Table 2.4.2-1 Component Supports

Component Group	Intended Function
Non-ASME Piping & Mechanical Component Support - Discharge Structure, Cast Iron, Protected	Structural Support for Non-Safety Related
Non-ASME Piping & Mechanical Component Support - Discharge Structure, Concrete, Protected	Structural Support for Non-Safety Related
Non-ASME Piping & Mechanical Component Support - Discharge Structure, Galvanized, Protected	Structural Support for Non-Safety Related
Non-ASME Piping & Mechanical Component Support - Feedwater Purity Bldg, Carbon Steel, Protected	Structural Support for Regulated Events
Non-ASME Piping & Mechanical Component Support - Feedwater Purity Bldg, Concrete, Protected	Structural Support for Regulated Events
Non-ASME Piping & Mechanical Component Support - Intake Structure Bldg, Carbon Steel, Protected	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related
Non-ASME Piping & Mechanical Component Support - Intake Structure Bldg, Concrete, Protected	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related
Non-ASME Piping & Mechanical Component Support - Intake Structure Bldg, Galvanized, Protected	Structural Support for Non-Safety Related Structural Support for Safety Related

Table 2.4.2-1 Component Supports

Component Group	Intended Function
Non-ASME Piping & Mechanical Component Support - Turbine Bldg, Carbon Steel, Protected	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related
Non-ASME Piping & Mechanical Component Support - Turbine Bldg, Concrete, Protected	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related
Non-ASME Piping & Mechanical Component Support - Turbine Bldg, Galvanized, Protected	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related
Non-ASME Piping & Mechanical Component Support - Water Treatment Bldg, Carbon Steel, Protected	Structural Support for Non-Safety Related Structural Support for Regulated Events
Non-ASME Piping & Mechanical Component Support - Water Treatment Bldg, Concrete, Protected	Structural Support for Non-Safety Related Structural Support for Regulated Events
Non-ASME Piping & Mechanical Component Support - Yard, Carbon Steel, Exposed	Structural Support for Regulated Events Structural Support for Safety Related

Table 2.4.2-1 Component Supports

Component Group	Intended Function
Non-ASME Piping & Mechanical Component Support - Yard, Concrete, Exposed	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related
Non-ASME Piping & Mechanical Component Support - Yard, Galvanized, Exposed	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related
Spent Fuel Storage Rack - Auxiliary Building, Boron Carbide, Borated Water	Radiation Shielding
Spent Fuel Storage Rack - Auxiliary Building, Stainless Steel, Borated Water	Structural Support for Safety Related

2.4.3 **Containment**

Description

The Containment structure consists of a post tensioned, reinforced concrete cylinder and dome connected to and supported by a massive, reinforced concrete foundation slab. The containment structure is a Class 1 structure. The majority of the interior surface of the containment pressure boundary structure is lined with 1/4” thick welded, steel plate to ensure a high degree of leak tightness. Numerous mechanical and electrical systems penetrate the containment wall through steel penetrations that are welded to the containment liner plate. None of the mechanical or electrical penetrations require expansion joints.

Personnel and equipment access to the structure is provided by a personnel air lock with two 3’ - 6” x 6’ - 8” doors, an escape air lock with two 30” diameter doors, and an equipment hatch with a single door that provides total access to the 12’ diameter passageway.

Principal dimensions are as follows:

Inside Diameter - 116’

Inside Height (Including Dome) - 189’

Vertical Wall Thickness - 3-1/2’

Dome Thickness - 3’

Foundation Slab Thickness - 8-1/2’ to 13-1/2’

Internal Free Volume - 1,640,000 ft^3

Upon completion of initial testing of the containment, the tendon anchorages at the ring girder and the buttresses were enclosed by corrugated aluminum siding, chosen to coordinate the architectural appearance of the containment with the balance of the Plant. This siding is removable to permit access to the tendons for the surveillance program. No siding was provided for tendons that terminate within adjacent buildings or within the access gallery under the containment wall.

Primary Containment Concrete

The Palisades concrete structures and concrete are designed in accordance with ACI 318-63 & 71, and constructed using ingredients conforming to ACI and ASTM standards. Palisades’ specifications require all concrete to contain an air-entraining agent in sufficient quantity to maintain specified percentages based on nominal maximum size aggregate. For severe weather exposures, the air content of Primary Containment concrete varies from 3 to 5 percent. Containment replacement concrete for steam generator replacement access had 3 to 7 percent air entrainment specified.

Primary Containment Liner

The original and steam generator replacement used 1/4” thick, ASTM A-442, Grade 55, liner plate, welded to a gridwork of structural steel angles embedded in the concrete. The anchoring system is designed to prevent significant distortion of the liner plate during accident conditions and to ensure that the liner maintains its leak tight integrity. The liner plate has been coated on the inside with 4 1/2 mils of inorganic zinc paint for corrosion protection. There is no paint on the side in contact with the concrete.

The personnel air lock, the emergency air lock, and the equipment hatch contain double seals that can be pressurized from outside the containment building. This feature allows periodic local leak rate testing to be conducted during both operation and shutdown. The electrical penetration canisters also can be pressurized from outside the containment. The piping and ventilation penetrations are of the rigid welded type and are solidly anchored to the containment wall, thus precluding any requirement for expansion bellows.

The deformations of the containment structure during normal Plant operation, or during accident conditions, are relatively minor due to the low strain behavior of the shell. The largest deformations occur during or shortly after the post-tensioning operation. This low strain behavior together with the inherent strength of the structure permits the shell to be used as an anchor point for all piping that passes through it. This behavior eliminates the need for expansion bellows and significantly reduces the likelihood of leaks developing at these containment penetrations.

The transfer tube and flange are also 304 SS, and the transfer tube flange gaskets are neoprene, testable, double “O”-ring (short lived, since they are replaced every refueling outage).

Primary Containment Prestressing System

The post tensioning system consists of:

- Three groups of 55 dome tendons oriented at 120 degrees to each other for a total of 165 tendons anchored at the vertical face of the dome ring girder.
- 178 vertical tendons anchored at the top surface of the ring girder and at the bottom of the base slab.
- Six groups of hoop tendons enclosing 120 degrees of arc for a total of 502 tendons anchored at the 6 vertical buttresses.

After fabrication, the tendon was shop dipped in a petrolatum corrosion protection material, bagged, and shipped. After installation, the tendon sheathing and caps were filled with corrosion preventative grease. (Reference FSAR 5.8.2)

The boundary of the in-scope portions of the Containment Building are described as: the entire containment building is in scope.

The portions of the Containment containing components subject to an AMR include the Primary Containment Concrete, the Primary Containment Liner, the Primary Containment Pre-stressing System and the Primary Containment Foundation.

System Function Listing

A comprehensive listing of functions associated with the Containment, or specific components contained in the structure, is provided in the summary below.

Code S0100-AT	Cri 1	Cri 2	Cri 3				
Provides shelter or support to ATWS related components.			FP	EQ	PTS	AT	SB
						X	

Comment: Containment building provides supporting function to ATWS related components of Electrical and Instrumentation & Control systems.

Code S0100-EQ	Cri 1	Cri 2	Cri 3				
Provides shelter or support to EQ related components.			FP	EQ	PTS	AT	SB
				X			

Comment: Containment building provides supporting function to safety related components of Electrical and Instrumentation & Control systems, Class 1E.

Code S0100-FP	Cri 1	Cri 2	Cri 3				
Provides rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant.			FP	EQ	PTS	AT	SB
			X				

Comment: Containment exterior wall is an Appendix R fire barrier.

Code S0100-NSAS	Cri 1	Cri 2	Cri 3				
Provides structural or functional support to non-safety components that support safety related components.			FP	EQ	PTS	AT	SB
		X					

Comment: Provides structural support to non-safety related components that are directly attached to or located in proximity to safety related components such that their support failure could cause failure of the SR component.

Code S0100-SB	Cri 1	Cri 2	Cri 3				
Provides shelter or support to SBO related components.			FP	EQ	PTS	AT	SB
							X

Comment: Containment building provides supporting function to SBO related components of Electrical and Instrumentation & Control systems.

Code S0100-SR1	Cri 1	Cri 2	Cri 3				
Provides essentially leak tight barrier to prevent uncontrolled release of radioactivity			FP	EQ	PTS	AT	SB
	X						

Comment: Containment Building is a Consumers Design Class 1 (seismic category 1) structure.

Code S0100-SR2 Provides shelter/protection and functional support to safety related component	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Containment Building is a Consumers Design Class 1 (seismic category 1) structure.

FSAR Reference

Additional Containment details are provided in Section 2.3.4, Section 5.8.2, Section 5.8.7 and Section 5.8.9 of the FSAR.

Scoping Boundary Drawings

The scoping boundary drawings for the Containment are listed below:

LR-C3

Components/Commodities Subject to an AMR

The component groups for the Containment that require aging management review are addressed in Table 2.4.3-1 along with each component group's intended function(s).

Table 2.4.3-1 Containment

Component Group	Intended Function
Containment Shell & Base Slab - Containment Bldg - Carbon Steel, Protected (air locks, equipment hatch, liner plate, penetrations)	Heat Sink Pressure Boundary/ Fission Product Retention Structural Support for Regulated Events Structural Support for Safety Related
Containment Shell & Base Slab - Containment Bldg - Concrete, Below Grade (base mat, foundation, wall, embedded steel, etc.)	Pressure Boundary/ Fission Product Retention Radiation Shielding Shelter/Protection Structural Support for Safety Related

Table 2.4.3-1 Containment

Component Group	Intended Function
Containment Shell & Base Slab - Containment Bldg - Concrete, Exposed (dome, wall, base mat, embedded steel, etc.)	Pressure Boundary/ Fission Product Retention Heat Sink Missile Barrier Radiation Shielding Shelter/ Protection Structural Support for Regulated Events Structural Support for Safety Related
Containment Shell & Base Slab - Containment Bldg - Elastomer, Protected (seals, gaskets, moisture barriers)	Pressure Boundary/ Fission Product Retention Shelter/ Protection
Containment Shell & Base Slab - Containment Bldg - Stainless Steel, Protected (fuel transfer tube, closure flange)	Heat Sink Pressure Boundary/ Fission Product Retention
Containment Shell Prestressing system - Containment Bldg - Carbon steel, Exposed (includes tendons and anchorage components)	Pressure Boundary/ Fission Product Retention Structural Support for Safety Related

2.4.4 Containment Interior Structures

Description

The containment interior concrete structures, which consist of all structural elements within the containment shell, are Class 1 structures. The principal interior concrete structures are:

- The primary shield wall, which forms the reactor cavity
- Two steam generator compartments
- A refueling pool which is located between the steam generator compartments and above the reactor cavity
- An enclosed sump under the reactor cavity
- Major equipment supports including the steam generator pedestals

The primary shield wall (bioshield) is essentially a circular cylinder, lined with ¼" steel plate, with concrete ranging in thickness from 7' to 8', with the inner 10" thickness acting as a sacrificial shield. The sacrificial shield is not reinforced, except for two horizontal "bands/hoops" of reinforcing steel and is considered non-structural, non-load bearing concrete. The primary shield wall cooling coils are located within the sacrificial concrete to provide temperature control of a maximum of 165ºF for the outer 6' to 7' thick, reinforced concrete. Openings in the shield wall for the primary coolant pipes are lined with ¼" steel plate, with the space between the opening and the piping filled with non-structural concrete block for shielding. The primary shield wall supports the roof slab of the sump.

The steam generator compartment walls form the secondary shield walls around the primary coolant loops. The primary functions of these walls are:

- To provide biological shielding,
- To provide missile barriers, and
- To provide barriers to resist the jet impingement loads, and associated compartment pressurization, resulting from the rupture of any one primary coolant pipe.

These compartment walls span horizontally between vertical beams, which, in turn, are restrained by the floor systems at elevations 607' - 0" and 649' - 0", and by structural steel tie struts at intermediate levels.

Pipe whip restraints are provided for the primary coolant pump suction, main steam, feedwater, and other high-pressure piping. Pipe whip barriers are also provided. For example, a reinforced concrete slab is provided above the top head of the pressurizer to protect it from a ruptured main steam line.

The boundaries of the in-scope portions of the Containment Interior Structures are described as follows: all containment interior structures are in scope.

The portions of the Containment Interior Structures containing components subject to an AMR include the primary shield wall, the steam generator compartments, the refueling pool, the Containment sump and major equipment supports including the steam generator pedestals.

System Function Listing

A comprehensive listing of functions associated with the Containment Interior Structures, or specific components contained in the structures, is provided in the summary below.

Code S0100-AT Provides shelter or support to ATWS related components.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
						X	

Comment: Containment building provides supporting function to ATWS related components of Electrical and Instrumentation & Control systems.

Code S0100-EQ Provides shelter or support to EQ related components.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
				X			

Comment: Containment building provides supporting function to safety related components of Electrical and Instrumentation & Control systems, Class 1E.

Code S0100-NSAS Provides structural or functional support to non-safety components that support safety related components.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: Provides structural support to non-safety related components that are directly attached to or located in proximity to safety related components such that their support failure could cause failure of the SR component.

Code S0100-SB Provides shelter or support to SBO related components.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
							X

Comment: Containment building provides supporting function to SBO related components of Electrical and Instrumentation & Control systems.

Code S0100-SR2 Provides shelter/protection and functional support to safety related component.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Containment Building is a Consumers Design Class 1 (seismic category 1) structure.

FSAR Reference

Additional Containment Interior Structures details are provided in Section 5.9.2, and Section 9.2 of the FSAR.

Scoping Boundary Drawings

None.

Components/Commodities Subject to an AMR

The component groups for the Containment Interior Structures that require aging management review are addressed in Table 2.4.4-1 along with each component group's intended function(s).

Table 2.4.4-1 Containment Interior Structures

Component Group	Intended Function
Building Framing - Carbon Steel, Protected (column, beam, truss, platform, floor grating or plate, catwalk, bracing, threaded fastener, concrete expansion bolt, column base plate, welds, etc.)	Heat Sink HELB Shielding Pipe Whip Restraint Structural Support for Regulated Events Structural Support for Safety Related Support for Non-Safety Related
Building Framing - Concrete, Containment Cavity (reactor shield walls including reinforcements, inserts, grouted anchors)	Heat Sink HELB Shielding Missile Barrier Radiation Shielding Structural Support for Safety Related

Table 2.4.4-1 Containment Interior Structures

Component Group	Intended Function
Building Framing - Concrete, Protected (concrete & masonry wall, column, pedestal, beam, slab, grout, reinforcements, concrete around expansion & grouted anchors, etc.)	Direct Flow Heat Sink HELB Shielding Pipe Whip Restraint Radiation Shielding Structural Support for Regulated Events Structural Support for Non-Safety Related Structural Support for Safety Related
Fuel Related Component - Stainless, Protected (refueling cavity liner, containment sump liner and screen, transfer tube)	Structural Support for Safety Related
HELB/MELB Component - Carbon Steel, Protected (steel curbs, pipe whip restraints, spray shields, etc.)	HELB Shielding Pipe Whip Restraint Structural Support for Safety Related
HELB/MELB Component - Concrete, Protected (curbs, sump, pipe whip restraint, grout, concrete at locations of expansion & grouted anchors, etc.)	Direct Flow Flood Protection Heat Sink HELB Shielding Pipe Whip Restraint
HVAC Component - Carbon Steel, Protected (damper & louver mounting frames)	Structural Support for Safety Related

Table 2.4.4-1 Containment Interior Structures

Component Group	Intended Function
HVAC Component - Galvanized, Protected (damper & louver mounting frames)	Structural Support for Safety Related
Missile Shield - Concrete, Containment Cavity (removable missile shield above reactor vessel)	Heat Sink Missile Barrier

2.4.5 **Discharge Structure**

Description

The Discharge Structure is a non-safety related, concrete structure with retaining walls that includes the Mixing Basin and Make-up Basin.

The safety related and non-safety related Service Water System (SWS) piping combine into a common 24" discharge header. This discharge header runs underground, splits into two 24" headers, and discharges into the Discharge Structure through the north and south concrete walls.

The Discharge Structure can also be used as a backup source of service water supply. In the event that water is not available to the intake structure due to a collapse of the intake crib or a similar type failure, approximately 17,000 gpm flow to the intake structure may be supplied from the mixing basin or makeup basin via the warm water recirculation pump P-5, if it is available. The warm water recirculation pump’s capability to provide water to the intake structure is an original design feature installed to mitigate circulating water system icing, and is not intended or required to provide a safety-related Service Water System makeup capability.

Due to similarities in materials and environment, the Intake Crib is included with the Discharge Structure scoping. The Intake Crib is a non-safety related structure that functionally supports the Ultimate Heat Sink function of maintaining adequate level in the Intake Structure.

The portions of the Discharge Structure containing components subject to an AMR include the Discharge Structure concrete walls (through which the Service Water discharge pipes pass), the Warm Water Recirculating Pump pump house, and the sluice gates. The Intake Crib is included to provide a section source to the Service Water Pumps.

System Function Listing'

A comprehensive listing of functions associated with the Discharge Structure, or specific components contained in the structures, is provided in the summary below.

Code S1200-NSAS-1	Cri 1	Cri 2	Cri 3				
Provides structural and/or functional support to safety related components			FP	EQ	PTS	AT	SB
		X					

Comment: Service water discharges to the Discharge Structure. The structural integrity of the Discharge Structure concrete walls is necessary to ensure the non-safety related SWS system discharge piping is not constricted or plugged so that it maintains required flow through the system. The Intake Crib must functionally pass flow to ensure a supply of lake water to the suction of the Service Water Pumps in the Intake Structure.

Code S1200-NSAS	Cri 1	Cri 2	Cri 3				
Provides warm water to the Intake Structure for de-icing to maintain operability of the Service Water System.			FP	EQ	PTS	AT	SB
		X					

Comment: The non-safety related Warm Water Recirculation Pump, located in a shed atop the Discharge Structure, pumps warm water from the Discharge Structure to a location upstream of the traveling screens in the Intake Structure to de-ice the screens. This is performed on an as-needed basis to maintain lake water flow to the safety related Service Water Pumps during icing conditions.

FSAR Reference

None.

Scoping Boundary Drawings

The scoping boundary drawings for the Discharge Structure are listed below:

LR-C-3

Components/Commodities Subject to an AMR

The component groups for the Discharge Structure that require aging management review are addressed in Table 2.4.5-1 along with each component group's intended function(s).

Table 2.4.5-1 Discharge Structure

Component Group	Intended Function
Building Framing - Cast Iron, Raw Water (sluice gates)	Structural Support for Non-Safety Related
Building Framing - Discharge/Intake Crib - Carbon Steel, Raw Water (sluice gate, column, bracket, beam, bracing, threaded fastener, connector, weld, etc.)	Structural Support for Non-Safety Related
Building Framing - Concrete, Below Grade (wall, footing, foundation, slab, grout, reinforcement, trenches, cable pits, tunnels, etc.)	Structural Support for Non-Safety Related
Building Framing - Concrete, Exposed (foundations, masonry & concrete wall, floor/roof slab, grout, reinforcements, steel shapes, concrete around expansion & grouted anchors)	Structural Support for Non-Safety Related
Building Framing - Concrete, Raw Water (wall, footing, foundation, slab, grout, reinforcements)	Structural Support for Non-Safety Related

2.4.6 Feedwater Purity Building

Description

The Feedwater Purity Building (condensate and makeup demineralizer building) was constructed during the feedwater purity modification. It houses the raw water filtration system, the reverse osmosis pretreatment system, the makeup demineralizer system, various components of the condensate demineralizer system, regeneration chemicals handling system, feedwater purity service and instrument air, chemical storage and a boiler room.

Because of a concern for steam generator contamination from resin leakage and sodium release, the condensate demineralizer system has been rendered inoperable and retired in place; however, the Water Purity Building is “in-scope” based on Fire Protection requirements to achieve safe shutdown. The Feed Water Purity Boiler Room in the south end of the building houses a diesel fuel oil transfer pump and piping, and this is the only area in this structure that is in-scope for License Renewal.

System Function Listing

A comprehensive listing of functions associated with the Feedwater Purity Building, or specific components contained in the structures, is provided in the summary below.

Code S0800-FP Provides structural and/or functional support to Fire Protection related components	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: Appendix R Safe Shutdown Analysis credits portions of the Fuel Oil transfer piping between Fuel Oil Tank & the EDG Day Tanks in the event of a fire that renders the normal FO transfer pumps inoperable. The credited piping is located in the FWP Building.

FSAR Reference

None.

Scoping Boundary Drawings

The scoping boundary drawings for the Feedwater Purity Building are listed below:

LR-C-3

Components/Commodities Subject to an AMR

The component groups for the Feedwater Purity Building that require aging management review are addressed in Table 2.4.6-1 along with each component group's intended function(s).

Table 2.4.6-1 Feedwater Purity Building

Component Group	Intended Function
Building Framing - Carbon Steel, Protected (column, beam, bracing, threaded fastener, concrete expansion bolt, column base plate, welds, etc.)	Structural Support for Regulated Events

Table 2.4.6-1 Feedwater Purity Building

Component Group	Intended Function
Building Framing - Concrete, Below Grade (pedestal, footing, foundation, slab, grout, reinforcement, trenches, cable pits, etc.)	Structural Support for Regulated Events
Building Framing - Concrete, Exposed (foundations, masonry/concrete wall, grout, reinforcement, concrete around expansion & grouted anchors, etc.)	Structural Support for Regulated Events
Building Framing - Concrete, Protected (foundations, masonry/concrete wall, column, pedestal, beam, slab, grout, reinforcement, concrete around expansion & grouted anchors, etc.)	Structural Support for Regulated Events

2.4.7 Intake Structure

Description

The portion of the Intake Structure above elevation 590 feet was designed to CP Co Design Class I standards. Major items of equipment housed in the area include the three safety-related service water pumps, two dilution pumps, two diesel engine-driven fire pumps, and two 480V MCCs providing electrical power to miscellaneous non- safety related equipment, including a motor-driven fire pump.

The pump room east wall adjacent to the turbine building has a three-hour fire rating with a single three-hour fire door installed. All other walls and access have outdoor exposure with a small section common with the diesel engine-driven fire pump day tank room. Fire ratings are in excess of three hours.

The portions of the Intake Structure containing components subject to AMR include the Design Class 1 portion (above elevation 590 feet), Design Class 3 walls and slab (below elevation 588 feet), including the triangular portion, sluice gates, trash racks, and the North and South chambers.

System Function Listing

A comprehensive listing of functions associated with the Intake Structure, or specific components contained in the structures, is provided in the summary below.

Code S0400-FP	Cri 1	Cri 2	Cri 3				
Provides structural and/or functional support to Fire Protection related component			FP	EQ	PTS	AT	SB
			X				

Comment: Portions of Intake Structure (Building) is a Consumers Design Class 3 (Non-Category 1) structure that support the Fire Protection function are in-scope of 10CFR 54.4. Structure contains Appendix R fire barriers.

Code S0400-NSAS	Cri 1	Cri 2	Cri 3				
Provides structural and/or functional support to safety related component			FP	EQ	PTS	AT	SB
		X					

Comment: Walls and slab below El 588' that are not part of the triangular portion (Class 1) are Consumers Design Class 3 portion of the structure. However, the Class 3 walls and slab are connected to the triangular portion and support the Class 1 slab and walls above El 588' that are in-scope of 10CFR 54.4. Failure of these Class 3 (non-safety related) walls or slab will affect the structural integrity of the Class 1 structures, which protect and support the safety related service water pumps, strainers, and associated piping and supports.

Code S0400A-FP	Cri 1	Cri 2	Cri 3				
Provides structural support to the fire pumps and rated fire barriers to confine or retard a fire from spreading to or from adjacent areas of the plant			FP	EQ	PTS	AT	SB
			X				

Comment: Structure contains Appendix R fire barriers and fire protection pumps.

Code S0400A-SR	Cri 1	Cri 2	Cri 3				
Provides shelter/protection to safety related components			FP	EQ	PTS	AT	SB
	X						

Comment: Provides structural support to safety related components. Safety related components include the Service Water Pumps, Fuel Oil (FO) Transfer Pumps. The walls that protect the FO pumps from flooding are included in this structure.

Code S0400B-FP	Cri 1	Cri 2	Cri 3				
Provides source of water for fire protection Pumps.			FP	EQ	PTS	AT	SB
			X				

Comment: The north and south chambers provide water into the triangular concrete reservoir through the north and south sluice gates. The triangular portion provides water supply suction for the safety related Service Water and Fire Water system pumps. See FWS system for Fire Protection Functions supported by this structure.

Code S0400B-SR Provides source of cooling water for service water pumps P-7A, P-7B & P-7C.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: The north and south chambers provide water into the triangular concrete reservoir through the north and south sluice gates. The triangular portion provides water supply suction for the safety related Service Water and Fire Water system pumps. See SWS system for Service Water Functions supported by this structure. For structural stability of the whole Intake Structure the adjacent walls, slabs below & above the “triangular” portion are considered in-scope.

FSAR Reference

None.

Scoping Boundary Drawings

The scoping boundary drawings for the Intake Structure are listed below:

LR-C-3

Components/Commodities Subject to an AMR

The component groups for the Intake Structure that require aging management review are addressed in Table 2.4.7-1 along with each component group’s intended function(s).

Table 2.4.7-1 Intake Structure

Component Group	Intended Function
Building Framing - Carbon Steel, Raw Water (gates, guides, and trash racks)	Structural Support for Regulated Events Structural Support for Safety Related
Building Framing - Cast Iron, Raw Water (Sluice Gates)	Structural Support for Regulated Events Structural Support for Safety Related

Table 2.4.7-1 Intake Structure

Component Group	Intended Function
Building Framing - Galvanized, Raw Water (fasteners and anchor bolts)	Structural Support for Regulated Events Structural Support for Safety Related
Building Framing - Concrete, Below Grade (wall, foundation, slab, grout, reinforcement, trenches, cable pits, tunnels, etc.)	Shelter / Protection Structural Support for Regulated Events Structural Support for Safety Related
Building Framing - Concrete, Exposed (wall, beam, floor slab, roof slab, grout, reinforcements, grouted anchors, etc.)	Fire Barrier Flood Protection Missile Barrier Shelter/ Protection Structural Support for Regulated Events Structural Support for Safety Related
Building Framing - Concrete, Protected (wall, beam, floor slab, grout, reinforcements, grouted anchors, etc.)	Direct Flow Flood Protection Shelter/ Protection Structural Support for Regulated Events Structural Support for Safety Related

Table 2.4.7-1 Intake Structure

Component Group	Intended Function
Building Framing - Concrete, Raw Water (wall, column, beam, footing, foundation slab, floor slab, grout, reinforcements, etc.)	Structural Support for Regulated Events Structural Support for Safety Related
Flood Barrier - Concrete, Protected (concrete interior wall in southeast corner)	Flood Protection
HELB/MELB Component - Carbon Steel, Protected (steel curbs, floor drains, shields, etc.)	Direct Flow Flood Protection Structural Support for Safety Related
HELB/MELB Component - Concrete, Protected (concrete/masonry walls, curbs, etc.)	Direct Flow Flood Protection

2.4.8 **Miscellaneous Structural and Bulk Commodities**

Description

This group includes miscellaneous component types (assets) of various materials that were identified as requiring aging management review, but which did not fit into the other structural commodity groups. Assets in this group include:

- Building crane bridges, trolleys, girders, and rails
- Elastomers for watertight door seals/gaskets, Emergency Diesel Generator vibration isolator elements, flood seals, seismic/expansion gap filler, waterstop, spray shield gasket, etc.
- Fire Protection related commodities (seals, fire wrap, fire doors, concrete and concrete block fire barrier floors, ceilings and walls, etc.)
- Architectural commodities (e.g., roofing systems, siding, control room ceiling, etc.)
- Soil and rip rap related to Water Control Structures - Intake Crib

- Insulation between the main steam and feedwater piping and containment concrete at the piping penetration.

For the most part, these assets consist of component types that are included under GALL Section IIIA6.4-a and under various portions of GALL Section VII, as well as non-GALL items.

System Function Listing

A comprehensive listing of functions associated with the Miscellaneous Structural and Bulk Commodities is provided in the summary below.

Code Architectural (BLA)-FP1 Provides shelter and support to Fire Protection components.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: Roofing over the Intake Structure protects the three fire pumps.

Code Architectural (BLA)-NSAS1 Provides support of non-safety related Operator Habitability components whose failure could affect safety related components.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: Architectural commodities provide ceiling framing to support lighting fixtures in the Control Room.

Code Architectural (BLA)-NSAS2 Provides shelter to safety-related component and personnel.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: The roofing of the Auxiliary, Intake Structure, Discharge Structure (over warm water recirc pump), Water Treatment Building, and Safeguard Bus Building; and the metal siding of the Auxiliary Building & Safeguard Bus Building are non-safety related commodities. However, leaky roofing (e.g., Control Room roof) or siding (metal) may prevent satisfactory accomplishment of safety related functions of Mechanical & Electrical safety related components. Therefore, the design function of architectural roofing & siding of the Auxiliary, Intake Structure, Discharge Structure, Water Treatment building, and Safeguard Bus buildings, as applicable, is in-scope of 10CFR 54.4. The roofing of the Containment Building is not in-scope because of the liner plate. The Turbine Building and Feedwater Purity Building roofing & metal siding; and Boiler Building roofing do not affect any safety related components due to leaky roof or siding.

Code Architectural (BLA)-SB Provides shelter and support to SBO related components.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
							X

Comment: The roofing of the Switchyard Relay House Building, Switchgear 1F & 1G Building (incoming breakers need to open to isolate the 1F & 1G loads from SBO required loads) and Safeguards Bus Building protects SBO related components. Rain dripping from leaky roof may prevent satisfactory accomplishment of SBO related components function.

Code Consumables (BLB)-EQ Provides seal.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
				X			

Comment: Consumables are used for protection of safety related electrical components from internal flooding (e.g., gasket or water seal used in internal flooding curbs).

Code Consumables (BLB)-NSAS Provides seal to prevent transmission of undesired fluid or/and debris between two adjacent areas	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: Some non-safety related Civil/Structural consumables perform supporting or protection functions for safety related components: 1) HVAC air seal (e.g., weatherstripping of Control Room doors to minimize air in-leakage) & duct seals. Control Room doors weatherstrippings are safety related, see commodity group BLO. 2) Thermal expansion / seismic separation joint filler between safety related building to safety or non-safety related building (e.g., between Containment and Auxiliary Building / Turbine Building). Aging of filler may allow hard material intrusion into the joint and prevent free movement of structures. 3) Water leak seal at flood doors/hatches, interior & exterior concrete wall penetrations, roofing, internal flood preventive curbs, etc. 4) Containment Personnel Air Lock, Escape Air Lock and Equipment Hatch seals are safety related and are addressed in commodity group BLS.

Code Cranes (BLC)-NSAS Provides handling for plant modification and maintenance activities	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: Some Cranes & Rigging commodities do not perform any safety related functions. However, depending on the load path and mode of operations, they may affect other safety related components due to potential accidental load drop. Cranes included in this function are Containment Building Boom Crane, Containment Jib Crane Hatch Area. Therefore BLC-NSAS function of Cranes & Rigging commodities is in-scope of 10CFR 54.4.

Code Cranes (BLC)-SR Provides handling of fuel assemblies, and lift and transport of loads in the Containment and Spent Fuel Pool areas without dropping.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Some Cranes & Rigging commodities perform safety related functions, such as handling of fuel assemblies in the Containment and Spent Fuel Pool of the Auxiliary Building. The handling equipment (Refueling Machine, Spent Fuel Handling Machine {SFHM}, New Fuel Inspection Elevator, SFHM Auxiliary Hoist, Tilt Machines, Transfer Carriage, and Spent Fuel Handling Machine Auxiliary Hoist) provide for retrieval, reorientation and transport of fuel and poisons (control rods) in and between the Spent Fuel Pool and the reactor cavity. Some cranes in this group handle Dry Fuel cask, sealed canister and transfer cask; Reactor Vessel components (e.g., Upper Guide Structure, Reactor Vessel Head, etc.); and other heavy material inside the Containment Building.

Code Soil (BLY)-NSAS Provides foundation, missile shield, and natural slopes stability.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: Provides soil stability for Intake Crib structure. The Intake Crib steel structure is surrounded by a sloped ring of stone riprap with sand fill inside the crib around the intake pipe elbow. The sand fill is covered and protected by three layers of concrete sack riprap. The riprap and sand for the Intake Crib provides a supporting function to deliver water to the safety related Service Water pumps.

FSAR Reference

None.

Scoping Boundary Drawings

None.

Components/Commodities Subject to an AMR

The component groups for the Miscellaneous Structural and Bulk Commodities that require aging management review are addressed in Table 2.4.8-1 along with each component group's intended function(s).

Table 2.4.8-1 Miscellaneous Structural and Bulk Commodities

Component Group	Intended Function
Built-Up Roofing - Auxiliary Bldg - Tarred, Exposed	Shelter/ Protection

Table 2.4.8-1 Miscellaneous Structural and Bulk Commodities

Component Group	Intended Function
Built-Up Roofing - Discharge Structure - Tarred, Exposed	Structural Support for Non-Safety Related
Built-Up Roofing - Intake Structure Bldg - Tarred, Exposed	Shelter/ Protection
Built-Up Roofing - Switch Yard Relay House Bldg - Tarred, Exposed	Structural Support for Regulated Events
Built-Up Roofing - Water Treatment Bldg - Tarred, Exposed	Structural Support for Non-Safety Related
Crane - Auxiliary Bldg - Carbon Steel, Protected	Structural Support for Safety Related
Crane - Containment Bldg - Carbon Steel, Protected	Structural Support for Safety Related
Crane Lift Device - Containment Bldg - Carbon Steel, Protected	Structural Support for Non-Safety Related
Crane Support - Auxiliary Bldg - Carbon Steel, Protected	Structural Support for Non-Safety Related Structural Support for Safety Related
Crane Support - Containment Bldg - Carbon Steel, Protected	Structural Support for Non-Safety Related Structural Support for Safety Related
Fire Barrier - Auxiliary Bldg - Carbon Steel, Protected	Fire Barrier
Fire Barrier - Auxiliary Bldg - Concrete, Exposed	Fire Barrier
Fire Barrier - Auxiliary Bldg - Concrete, Protected	Fire Barrier
Fire Barrier - Auxiliary Bldg - Fire Stop, Protected	Fire Barrier
Fire Barrier - Auxiliary Bldg - Fire Wrap, Protected	Fire Barrier

Table 2.4.8-1 Miscellaneous Structural and Bulk Commodities

Component Group	Intended Function
Fire Barrier - Containment Bldg - Carbon Steel, Protected	Fire Barrier
Fire Barrier - Containment Bldg - Concrete, Exposed	Fire Barrier
Fire Barrier - Intake Structure Bldg - Carbon Steel, Protected	Fire Barrier
Fire Barrier - Intake Structure Bldg - Concrete, Exposed	Fire Barrier
Fire Barrier - Intake Structure Bldg - Fire Stop, Protected	Fire Barrier
Fire Barrier - Intake Structure Bldg - Fire Wrap, Protected	Fire Barrier
Fire Barrier - Turbine Bldg - Carbon Steel, Protected	Fire Barrier
Fire Barrier - Turbine Bldg - Concrete, Exposed	Fire Barrier
Fire Barrier - Turbine Bldg - Concrete, Protected	Fire Barrier
Fire Barrier - Turbine Bldg - Fire Stop, Protected	Fire Barrier
Fire Barrier - Turbine Bldg - Fire Wrap, Protected	Fire Barrier
Fire Barrier - Turbine Bldg - Water Treatment Bldg - Concrete, Exposed	Fire Barrier
Fire Barrier - Turbine Bldg - Water Treatment Bldg - Concrete, Protected	Fire Barrier
Fire Barrier - Turbine Bldg - Water Treatment Bldg - Fire Stop, Protected	Fire Barrier
Fire Barrier - Turbine Bldg - Water Treatment Bldg - Fire Wrap, Protected	Fire Barrier
Flood Barrier - Auxiliary Bldg - Elastomer, Protected	Flood Protection
Flood Barrier - Turbine Building - Elastomer, Protected	Flood Protection

Table 2.4.8-1 Miscellaneous Structural and Bulk Commodities

Component Group	Intended Function
HELB/MELB Civil/ Structural Component - Auxiliary Bldg - Elastomer, Protected	Direct Flow Flood Protection
HELB/MELB Civil/ Structural Component - Intake Structure Bldg - Elastomer, Protected	Direct Flow Flood Protection
Mechanical General Component Support - Containment Bldg - Elastomer, Protected	Structural Support for Safety Related
Riprap - Yard - Soil, Submerged	Structural Support for Non-Safety Related
Roof Flashing - Auxiliary Bldg - Galvanized, Exposed	Shelter / Protection
Roof Flashing - Intake Structure Bldg - Galvanized, Exposed	Shelter / Protection
Roof Flashing - Switchyard Relay House - Galvanized, Exposed	Structural Support for Regulated Events
Seal, Gasket or Filler - Auxiliary Bldg - Elastomer, Exposed	Shelter/ Protection
Seal, Gasket or Filler - Auxiliary Bldg - Elastomer, Protected	Direct Flow Expansion / Separation Flood Protection Shelter/ Protection
Seal, Gasket or Filler - Containment Bldg - Elastomer, Protected	Expansion / Separation Flood Protection Shelter/ Protection
Seal, Gasket or Filler - Discharge Structure - Elastomer, Exposed	Structural Support for Non-Safety Related

Table 2.4.8-1 Miscellaneous Structural and Bulk Commodities

Component Group	Intended Function
Seal, Gasket or Filler - Discharge Structure - Elastomer, Protected	Structural Support for Non-Safety Related
Seal, Gasket or Filler - Intake Structure Bldg - Elastomer, Exposed	Shelter/ Protection
Seal, Gasket or Filler - Switchyard Relay House/ Switchgear/ Safeguard Bldg - Elastomer, Exposed	Shelter/ Protection Structural Support for Regulated Events
Seal, Gasket or Filler - Turbine Bldg - Elastomer, Exposed	Structural Support for Non-Safety Related
Seal, Gasket or Filler - Turbine Bldg - Elastomer, Protected	Flood Protection Shelter/ Protection

2.4.9 Switchyard and Yard Structures

Description

Switchyard Relay House Group

The Switch Yard Relay House group includes 1) the Switch Yard Relay House, 2) the Switchgear 1F & 1G Building, and 3) the Safeguards Bus Building. These buildings are Consumers Design Class 3 structures.

Switch Yard Foundations

This group includes reinforced concrete foundations for the Switch Yard Relay House, Safeguards Building, and Bus 1F/1G Building. It also includes foundations for startup and safeguards transformers; foundation and framing for the high voltage towers between the plant and switchyard, foundation and framing for the overhead lines from the plant to switchyard disconnect 24R2 (including the subject disconnect and takeoff towers); and foundation and framing for the underground line transitioning to the overhead bus, safeguards transformer, and disconnect 24 F1. Transformer foundations are reinforced concrete, slab on grade. Take-off towers have substantial, reinforced concrete spread footings, well below grade, and piers extending to 6" above grade. The switchgear housing and bus supports also are on spread footings with piers extending through backfill to above grade. These structures are in-scope based on SBO coping requirements.

Also included are cable trench commodities, consisting of precast reinforced concrete “Trenwa” underground utility trench system comprised of trench walls and cover, which are set on a bed of gravel, precast reinforced concrete cable pits/manholes, and underground, reinforced concrete duct bank, rigid steel or PVC conduit encased in concrete.

Tank Foundations

This includes reinforced concrete foundations for Primary Makeup Storage Tank, Safety Injection and Refueling (SIRW) Tank, Condensate Storage Tank including valve pit, Demineralized Water Storage Tank, Primary System Makeup Storage Tank, Utility Water Storage Tank, Diesel Generator Oil Storage Tank, and Fuel Oil Storage Tank.

The foundations for primary makeup storage tank, condensate storage tank including valve pit, demineralized water storage tank, primary system makeup storage tank, and utility water storage tank are reinforced concrete rings, with backfill compacted to 95% to support the tank bottoms. The SIRW Tank is supported on a concrete pad on top of the reinforced concrete roof of the Auxiliary Building. The Class 1 diesel generator fuel oil tank is housed in a below-grade vault, constructed of reinforced concrete (2' thick floor and walls, with an 18" thick roof) for missile protection. The fuel oil tank is credited in the Fire Protection Program.

The portions of the Switchyard and Yard Structures containing components subject to an AMR include the Switch Yard Relay House, Safeguards Bus Building, and bus 1F/1G building; foundations for startup and safeguards transformers; high voltage towers between plant and switchyard; foundation and framing for the overhead lines from the plant to switchyard disconnect 24R2 (including the subject disconnect and takeoff towers); foundation and framing for the underground line transitioning to the overhead bus, safeguards transformer and disconnect 24F1; cable trench commodities (including manholes, duct bank, conduit, etc.), and tank foundations.

System Function Listing

A comprehensive listing of functions associated with the Switchyard and Yard Structures, or specific components contained in the structure, is provided in the summary below.

Code S1100-SB Provides shelter to SBO related components	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
							X

Comment: The Switchyard Relay House, Switchgear 1F & 1G Building, and Safeguards Bus Building provide shelter and support to Station Blackout components. Transformer foundations and foundations and framing for overhead and buried lines from plant to switchyard (including disconnects) provide functional support to SBO components.

FSAR Reference

None

Scoping Boundary Drawings

The scoping boundary drawings for the Switchyard and Yard Structures are listed below:

LR-C-3 LR-C2

Components/Commodities Subject to an AMR

The component groups for the Switchyard and Yard Structures that require aging management review are addressed in Table 2.4.9-1 along with each component group's intended function(s).

Table 2.4.9-1 Switchyard and Yard Structures

Component Group	Intended Function
Building Framing - Safeguard Bus/Switchgear - Carbon Steel, Protected (floor beam, panels, welds, threaded fasteners, concrete expansion bolt, etc.)	Structural Support for Regulated Events
Building Framing - switchyard - concrete, Below Grade (grade beam, footing, trenches, slab, grout, reinforcement, trenches, cable pits, tunnels, etc.)	Structural Support for Regulated Events

Table 2.4.9-1 Switchyard and Yard Structures

Component Group	Intended Function
Building Framing - Switchyard - Concrete, Exposed (masonry/concrete wall, grout, reinforcements, foundations, concrete around expansion & grouted anchors, bus supports, etc.)	Structural Support for Regulated Events
Building Framing - Switchyard - Concrete, Protected (masonry roof bearing walls, reinforcements, concrete around expansion & grouted anchors, etc.)	Structural Support for Regulated Events
Metal Roofing & Siding - Switchgear & Safeguard Bus - Carbon Steel, Exposed (flashing)	Structural Support for Regulated Events
Missile Shield - Yard - Concrete, Exposed (Tank vault roof, pavement over buried piping)	Missile Barrier
Tank Foundations - Building & Yard - Concrete, Below Grade (concrete foundation)	Structural Support for Regulated Events Structure Functional Support
Tank Foundations - Building & Yard - Concrete, Exposed (concrete foundation)	Structural Support for Regulated Events Structural Support for Safety Related

2.4.10 **Turbine Building**

Description

The turbine building houses the turbine generator, condenser, feedwater heaters, condensate and feed water pumps, turbine auxiliaries and certain of the switchgear assemblies. The north end of the turbine building provides additional shop, laboratory and office space.

The following areas of the turbine building were designed to CP Co Design Class I standards:

- Portion of the turbine building basement forming the Auxiliary Feedwater Pump Room.
- Portion of the turbine building known as the South Electrical Penetration Room.

The remainder of the turbine building is CP Co Design Class 3. Areas include the following:

- Control Room Door Enclosure
- Other tornado missile enclosures
- Containment Escape Hatch Enclosure
- Boiler Buildings
- Water Treatment Building

The Turbine Building foundation is a series of concrete spread footings and piers for each steel building column or heavy equipment location with an interlocking grade beam arrangement. A grade slab fill is then provided to complete the Turbine Building floor. The Turbine Pedestal has its own concrete base mat, piers, and top deck that are independent of the remaining Turbine Building substructure.

The Turbine Building has a steel frame superstructure with siding and a concrete curb wall at the foundation line. In addition, the Turbine Building has concrete and masonry wall construction in the areas adjoining the Auxiliary and Containment Buildings. The floors within the Turbine Building are framed steel supporting formed concrete slabs, concrete metal decking slabs, and grating. The turbine building also contains fire barrier concrete commodities credited in Fire Protection requirements for achieving safe shutdown.

Boiler Buildings portions of the Turbine Building

The Boiler Building and Boiler Building Addition are Consumers Design Class 3 structures. They house the Evaporator Heating Boiler and Heating Boiler. The Boiler Buildings support and protect diesel fuel oil components and associated piping.

Water Treatment Area Portion of the Turbine Building

The Water Treatment Area, like the Turbine Building, has a steel frame superstructure with siding and a concrete curb wall at the foundation line. The foundation and floor slab are reinforced concrete. The Water Treatment Area is “in-scope” for NSAS and SBO due to the Condensate Storage Tank valves located within.

The portions of the Turbine Building containing components subject to an AMR include the Turbine Building framing (steel & concrete), the Water Treatment Building framing (steel & concrete), flood barriers, HELB components (steel and concrete), HVAC components (steel and concrete), missile shield and operators access components (steel and concrete).

System Function Listing

A comprehensive listing of functions associated with the Turbine Building, or specific components contained in the structure, is provided in the summary below.

Code S0500-FP	Cri 1	Cri 2	Cri 3				
Provides structural and/or functional support to Fire Protection related component.			FP	EQ	PTS	AT	SB
			X				

Comment: Turbine Building contains Appendix R Fire Barriers and Fire Protection related equipment or components.

Code S0500-NSAS	Cri 1	Cri 2	Cri 3				
Provides structural or functional support to safety related components			FP	EQ	PTS	AT	SB
		X					

Comment: Shelters or protects the Main Feed Regulator and Regulator Bypass valves that have a safety related closed function, and the turbine driven AFW pump steam supply valve and associated piping. Also contains whip restraints and seismic supports for the Main Steam and Feedwater lines.

Code S0500A,B-SB	Cri 1	Cri 2	Cri 3				
Provides shelter or support to SBO related components			FP	EQ	PTS	AT	SB
							X

Comment: Class 3 portion of the Turbine Building shelters AFW Pump P-8B steam supply & nitrogen supply to Atmospheric Steam Dump Valves. AFW Pump Room contains AFW Pump which is required during SBO.

Code S0500A,B-FP	Cri 1	Cri 2	Cri 3				
Provides rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant			FP	EQ	PTS	AT	SB
			X				

Comment: The Turbine Building AFW Pump Room is Fire Zone Area 24. The Turbine Building South Electrical Penetration Room is Fire Zone Area 26. These rooms contain fire detection and/or suppression equipment.

Code S0500A,B-SR	Cri 1	Cri 2	Cri 3				
Provides shelter/protection to safety related component.			FP	EQ	PTS	AT	SB
	X						

Comment: The Turbine Building Electrical Penetration Room and AFW Pump Rooms are Consumers Design Class 1 (Category 1) structures.

Code S0500C,D,E-NSAS Provides missile barrier.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: The concrete, steel and masonry enclosures provide tornado missile protection for selected safety related Class 1 equipment, including Control Room Door 52 and Containment Escape Lock.

Code S0600-NSAS Provides shelter, protection, and structural support of non-safety related components whose failure could affect the function of safety related components.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: Boiler Rooms contain Diesel Fuel Oil components and associated piping that are non-safety related, but are attached to safety related fuel oil piping. Thus, shelter, protection and support of the FOS piping is an in-scope function.

Code S0700-NSAS Provides shelter to safety related component or personnel	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: Turbine Bldg Water Treatment Area is Consumers Design Class 3 structure that is a non-safety related structure. The building provides shelter and support to safety related components and their associated piping.

Code S0700-SB Provides shelter or support to SBO related components.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
							X

Comment: Gravity feed of condensate to AFW suction are required to support SBO. Associated valves in this flow path and located in the Turbine Bldg Water Treatment Area.

Code Architectural (BLA)-FP2 Provides access to remote shutdown panels & components in the event of a fire	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: Provides access for the operator from the Control Room to Redundant Safety Injection Panel EC-33 in Auxiliary Building and Hot Shutdown Panels EC-150 & EC-150A in Turbine Building for alternate shutdown station. Other Auxiliary Building platforms, stairs, and similar commodities that are necessary to provide access to the operators that affects safety operations are in-scope.

FSAR Reference

None

Scoping Boundary Drawings

The scoping boundary drawings for the Turbine Building are listed below:

LR-C-3

Components/Commodities Subject to an AMR

The component groups for the Turbine Building that require aging management review are addressed in Table 2.4.10-1 along with each component group's intended function(s).

Table 2.4.10-1 Turbine Building

Component Group	Intended Function
Building Framing - Boiler Buildings Area - Carbon Steel, Protected (column, hanger, beam, truss, decking, platform, floor grating or plate, catwalk, threaded fastener, concrete expansion bolt, column base plate, welds, etc.)	Structural Support for Non-Safety Related
Building Framing - Boiler Buildings Area - Concrete, Below Grade (wall, pedestal, grade beam, footing, foundation, slab, grout, reinforcement, cable pits, tunnels, etc.)	Structural Support for Non-Safety Related
Building Framing - Boiler Buildings Area - Concrete, Exposed (masonry/concrete wall, grout, reinforcements, concrete around expansion & grouted anchors, etc.)	Structural Support for Non-Safety Related
Building Framing - Boiler Buildings Area - Concrete, Protected (foundations, concrete & masonry wall, column, beam, floor.roof slab, grout, reinforcements, concrete around expansion & grouted anchors, etc.)	Structural Support for Non-Safety Related
Building Framing - Water Treatment Area - Concrete, Protected (concrete/ masonry wall, beam, floor slab, grout, reinforcements, etc.)	Structural Support for Non-Safety Related Structural Support for Regulated Events

Table 2.4.10-1 Turbine Building

Component Group	Intended Function
Building Framing - Carbon Steel, Protected (column, beam, truss, decking, platform, floor grating or plate, catwalk, bracing, threaded fastener, concrete expansion bolt, column base plate, welds, etc.)	Flood Protection HELB Shielding Pipe Whip Restraint Shelter/ Protection Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related
Building Framing - Concrete, Below Grade (wall, pedestal, grade beam, footing, foundation, slab, grout, reinforcement, cable pits, tunnels, etc.)	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related
Building Framing - Concrete, Exposed (foundations, masonry/concrete wall, grout, reinforcements, concrete around expansion & grouted anchors, etc.)	Flood Protection Shelter/ Protection Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related

Table 2.4.10-1 Turbine Building

Component Group	Intended Function
Building Framing - Concrete, Protected (foundations, concrete & masonry wall, column, beam, floor/roof slab, grout, reinforcements, concrete around expansion & grouted anchors, etc.)	Flood Protection HELB Shielding Missile Barrier Shelter/ Protection Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related
Building Framing - Water Treatment Area - Carbon Steel, Protected (column, beam, bracing, threaded fastener, weld, etc.)	Structural Support for Regulated Events
Building Framing - Water Treatment Area - Concrete, Below Grade (wall, pedestal, grade beam, footing, foundation, slab, grout, reinforcement, trenches, cable pits, tunnels, etc.)	Structural Support for Non-Safety Related Structural Support for Regulated Events
Building Framing - Water Treatment Area - Concrete, Exposed (foundations, wall, grout, reinforcements, concrete around expansion & grouted anchors, etc.)	Structural Support for Non-Safety Related Structural Support for Regulated Events
Building Framing - Water Treatment Area - Concrete, Protected (foundations, concrete/masonry wall, beam, floor slab, grout, reinforcements, concrete around expansion & grouted anchors, etc.)	Structural Support for Non-Safety Related Structural Support for Regulated Events
Flood Barrier - Carbon Steel, Protected (flood doors, hatch, standpipe)	Flood Protection Structural Support for Safety Related

Table 2.4.10-1 Turbine Building

Component Group	Intended Function
HELB/MELB Component - Carbon Steel, Protected (curbs, floor drains, pipe whip restraints, spray shields, etc.)	Flood Protection HELB Shielding Pipe Whip Restraint Structural Support for Safety Related
HELB/MELB Component - Concrete, Protected (concrete/masonry wall, whip restraint grout, concrete around expansion & grouted anchors)	Flood Protection HELB Shielding Pipe Whip Restraint
HVAC Component - Carbon Steel, Protected (Control Room vestibule door)	Fluid Pressure Boundary Structural Support for Non-Safety Related
HVAC Component - Concrete, Protected (Control Room vestibules, concrete & masonry walls, floors, ceilings)	Fluid Pressure Boundary Structural Support for Non-Safety Related
Missile Shield - Concrete, Exposed (concrete and/or masonry walls protecting CCW room door and Containment escape hatch)	Missile Barrier
Missile Shield - Concrete, Protected (concrete.masonry walls, floor & roof protecting Control Room Door)	Missile Barrier
Operator Access Component - Carbon Steel, Protected (stairs, floors, platforms)	Structural Support for Non-Safety Related Structural Support for Regulated Events

Table 2.4.10-1 Turbine Building

Component Group	Intended Function
Operator Access Component - Concrete, Protected (stairs, floors, platforms, concrete at locations of expansion & grouted anchors, etc.)	Structural Support for Non-Safety Related Structural Support for Regulated Events
Operator Access Component - Galvanized, Protected (stairs, floors, platforms)	Structural Support for Non-Safety Related Structural Support for Regulated Events

2.5 Scoping and Screening Results: Electrical and Instrumentation and Controls

The following commodity groups and systems are addressed in this section:

- Commodity Group Descriptions (Section 2.5.1)
- Cables and Terminations Commodity (Section 2.5.2)
- Containment Isolation and Penetration System (Section 2.5.3)
- Control Rod Drive System (Section 2.5.4)
- Neutron Monitoring System (Section 2.5.5)
- Radiation Monitoring System (Section 2.5.6)
- Reactor Protective System (Section 2.5.7)
- Station Power System (Section 2.5.8
- Switchyard System (Section 2.5.9

During the Scoping Process, evaluation boundaries were established for each system or commodity group based on the system functions. The Screening Process identified the long-lived, passive, components from the in-scope systems and commodity groups that would be subject to an aging management review. For convenience, because of similarities across systems in materials, environments, and aging management strategies, the various system components that required aging management review were assigned to commodity groups. Aging management reviews were then performed on the electrical components as commodities, regardless of system association. This approach is consistent with the approach taken by most applicants for license renewal.

The electrical commodity groupings identified at Palisades are:

- Electrical cables and connections not subject to 10 CFR 50.49 EQ requirements
- Electrical cables and connections used in instrumentation circuits not subject to 10 CFR 50.49 EQ requirements that are sensitive to reduction in conductor IR (Nuclear Instrumentation & Radiation Monitoring Systems)
- Electrical Portions of the Non-EQ Electrical and I&C Penetration Assemblies (Cables and Connections)
- Fuse Holders (ISG-5)
- Non-Segregated Phase Bus and Connections
- High-Voltage Transmission Conductors (ISG-2)
- High-Voltage Switchyard Bus and Connections (ISG-2)
- Inaccessible Medium-Voltage (2kV to 15kV) Cables and Connections not subject to 10 CFR 50.49 requirements

- High-Voltage Insulators (ISG-2)

Section 2.5.1 describes the commodity groups that were used to evaluate all in-scope electrical and I&C components subject to aging management review.

Sections 2.5.2 through 2.5 8 provide descriptions of the electrical and I&C systems that were determined to be in-scope. Components within those electrical systems which were determined to require aging management review were addressed as part of the defined commodities. This system organization is not retained in later sections of this application.

2.5.1 Commodity Group Descriptions

Commodity: Electrical cables and connections not subject to 10 CFR 50.49 EQ requirements

Commodity: Electrical cables and connections used in instrumentation circuits not subject to 10 CFR 50.49 EQ requirements that are sensitive to reduction in conductor IR (Nuclear Instrumentation & Radiation Monitoring Systems)

The two commodity groups listed above include the same types of cables and components. They differ primarily in the programs selected to manage aging of those components. The determination of which cables and components are assigned to each commodity is self explanatory based on the commodity titles.

The component types which comprise these two commodity groups are described below:

Insulated Cables

Insulated cables encompass all in-scope cable types used in the plant. The cable insulation material used at Palisades is SR, EPR, XLP, XLPE, EPDM, PVC, BR, PE, polymide, and teflon. Cable insulation material groups are assessed on the basis of material similarity and their respective material aging characteristics. The insulated cable materials subject to aging are metal and insulation. The metals used are copper and tinned copper. The insulation materials are various elastomers and thermoplastics. The insulated cable conductor metal is protected from the environments that induce aging effects, as long as the cable insulation maintains its integrity (material properties). Therefore, the metal conductor portion of insulated electrical cables does not experience aging effects requiring aging management.

Electrical Connections - Splices

Palisades utilizes Raychem WCSF-N heat shrink tubing for all of its field splices. Raychem heat-shrinkable tubing consists of a proprietary material stated by Raychem to be made of a “modified polyolefin” similar to the properties of XLPE and XLPO. Since this material and its cable applications (P, C & I) are similar to the XLPE insulated cables, the cable splices are bounded by the XLPE cable insulation review.

Electrical Connections - Terminations (Terminal Blocks and Fuse Blocks)

Terminal blocks and fuse blocks, that are part of an active component, were excluded from further aging management review.

Terminal blocks used at Palisades, that are not part of an active component, were identified to be Buchanan (Model NQB), States (ZWM and NT Types), Cinch Jones (140 Series), and Westinghouse (542-245). The Palisades J-Box Drawings were reviewed to capture the prominent terminal block brands and models used at Palisades. All terminal blocks identified and reviewed, potentially in-scope of LR, had phenolic insulation material.

Electrical Connections - Connectors

Electrical connectors are used to connect the cable conductors to other cables or electrical devices. The three main types of connectors are compression, fusion, and plug-in (mated) connectors. A brief description of each is provided below:

Compression Connectors: Fittings (e.g. ring lugs or barrels) that are bolted, physically crimped or mechanically swaged to connect cable conductors.

Fusion Connectors: Cable connections made by welding, brazing or soldering where permanence of the conductor connection is desired.

Plug-in (Mated) Connectors: Connectors with one or more electrical contacts that plug or screw into a mating receptacle; useful where ease and frequency of separation of an electrical connection is desired, for ease of mating specific types of equipment, and where multiple simultaneous electrical disconnections / reconnections need to be made.

Insulated plug-in (mated) connectors were the main focus of the connector aging review.

Detailed drawings and vendor information may not be available to identify all insulating materials used in non-EQ connectors. Therefore, the identification of insulation materials on all potentially in-scope plug-in connectors at Palisades (both EQ and non-EQ) began with a review of other previous AMRs with references to NUREG/CR-6412. This review provided a listing of the predominantly-used connector insulation materials in the nuclear industry: EPDM, EPR, Kapton, and XLPE. The plant EQ Master Equipment List and the EQ File Reports were then used to validate completeness of the generic list, and identify any additional connector insulation materials used at Palisades. The EQ files identified additional materials, which were added to the generic materials list. This combined list of electrical connector insulation materials provides reasonable assurance that it is a bounding list of connector insulation materials installed at Palisades.

Commodity: Electrical Portions of the Non-EQ Electrical and I&C Penetration Assemblies (Cables and Connections)

There are thirty-seven (37) active Electrical Containment Penetrations (Canisters) at Palisades. Twenty-two (22) of the electrical penetrations fall within the EQ Program, and fifteen (15) are non-EQ.

Palisades uses two types of Electrical Containment Penetrations (Canisters) manufactured by two vendors: thirty-five (35) penetrations were manufactured by Viking Industries, and two (2) by Conax. The two manufactured by Conax are both EQ electrical penetrations. Fifteen (15) of the thirty-five (35) electrical penetrations manufactured by Viking Industries are non-EQ electrical penetrations.

The non-electrical portions of the Viking Industries electrical penetration assemblies, that support the LR pressure boundary or structural function, are addressed in the civil/structural area. The electrical portions of the electrical penetration assemblies that electrically support the License Renewal functions are the internal cable insulation materials and the connector insulation materials.

The internal cables used in the Viking Electrical Penetrations are:

- Coaxial Cable (PE)
- Rockbestos Firewall Cable (SR)
- Anaconda Cable (EPR)
- Ceramic Bushings
- Fiberglass

The cable insulation materials are included in the non-EQ insulated cable review.

Commodity: Fuse Holders (ISG-5)

This commodity includes all fuse blocks used at Palisades that are not part of an active component or larger active assembly. Those identified are all located inside junction boxes in a controlled environment and are not subject to any aging mechanism, including cycling more than once an refueling outage.

Commodity: Non-Segregated Phase Bus and Connections

The non-segregated phase bus, supporting the SBO restoration path for Palisades, is in-scope. The structural hardware and enclosure housing components are addressed in the civil/structural area, if applicable.

The Unibus Inc. metal-enclosed bus contains the following vital electrical components that provide the primary electrical conductivity and insulation functions:

- Copper Bus
- Glass-Reinforced Polyester Blocks (track-resistant)
- Porcelain Bus Sleeve, or Ceramic and Fiberglass
- Bus mounting boots (Material unknown)

The non-segregated bus is provided with uniform heating and temperature control to prevent condensation. The contact surfaces of all electrical joints [connections] are either silver or tin-plated. Per design, no electrical joint compounds are used or required. Per design, no retorquing of properly installed conductor connecting hardware is required on a routine basis. Flexible ground continuity connections are provided at bus expansion joints. The Unibus, Inc. metal enclosed bus system has been designed to eliminate components or consumables that might require periodic or routine replacement.

Recent concern among the industry, as discussed in ISG-17, has focused upon the non-segregated bus enclosure seals and loose connections from thermal cycling. Palisades has conservatively included the non-segregated bus duct into its aging management program due to moisture level concerns resulting from potential seal leakage and loose connections from thermal cycling.

Commodity: High-Voltage Transmission Conductors (ISG-2)

The switchyard is not included in the plant equipment defined in the CLB for mitigation of the SBO event. ISG-2, however, indicates that certain equipment which provides offsite power for restoration from a Station Blackout is to be included in the scope of license renewal. Therefore, to be consistent with the NRC guidance in ISG-2, Palisades has included in-scope, selected equipment associated with the switchyard and the two qualified offsite power circuits that can provide offsite power to the safety related Buses 1C and 1D following an SBO event. The equipment brought in to scope is addressed as a commodity consisting of the following component types.

- All in-scope 345 kV station transmission cables are “Bluebird” 2,156 MCM, 84/19 ACSR (aluminum conductor steel reinforced) and are constructed of the following materials:
    (a) Aluminum

(b) Steel

- The transmission conductor between the switchyard bus and transformers or tie breakers is within scope.

Commodity: High-Voltage Switchyard Bus and Connections (ISG-2)

The 345 kV switchyard bus is 5" and 3 ½” aluminum alloy 6061-T6, Schedule 40 seamless pipe. The aluminum bus bolted connection hardware is aluminum or stainless steel.

Commodity: Insulation for Inaccessible Medium-Voltage (2kV to 15kV) Cables and Connections

Medium-voltage cables, including those in the SBO restoration path for Palisades, are in-scope and were included with the non-EQ insulated cable commodity for review. Insulation materials are various elastomers and polymers.

Commodity: High-Voltage Insulators (ISG-2)

The Palisades high-voltage cable insulators are 5-3/4” x 10", 25,000 lbs. M&E Strength, ASA 52-5 suspension insulators with a porcelain electrical insulation material, Portland cement, on a steel post. The Switchyard Bus post insulators are made of porcelain and cement. Porcelain and cement do not experience aging effects from the bounding site radiation and temperature levels. The internal metal posts of the HV Cable Insulators are reviewed for potential wear from the effects of vibration. It was determined that no significant aging effects are induced by temperature or radiation on the high-voltage insulator materials.

Table 2.5-1 Electrical Commodity Groups

Commodity Group	Intended Function
Electrical cables and connections not subject to 10 CFR 50.49 EQ Requirements	Provide electrical connections to specified sections of an electrical circuit to deliver voltage, current, or signals
Electrical cables and connections used in instrumentation circuits not subject to 10 CFR 50.49 EQ Requirements that are sensitive to reduction in conductor IR. (ISG-15) (Nuclear Instrumentation and Radiation Monitoring Systems)	Provide electrical connections to specified sections of an electrical circuit to deliver voltage, current, or signals
Electrical Portion of the Non-EQ Electrical and I&C Penetration Assemblies (Cables & Connections)	Provide electrical connections to specified sections of an electrical circuit to deliver voltage, current, or signals
Fuse Holders (ISG-5)	Provide electrical connections to specified sections of an electrical circuit to deliver voltage, current, or signals.
Non-Segregated Phase Bus and Connections (ISG-17)	Provide electrical connections to specified sections of an electrical circuit to deliver voltage, current, or signals
High-Voltage Transmission Conductors (ISG-2)	Provide electrical connections to specified sections of an electrical circuit to deliver voltage, current, or signals
High-Voltage Switchyard Bus and Connections (ISG-2)	Provide electrical connections to specified sections of an electrical circuit to deliver voltage, current, or signals

Table 2.5-1 Electrical Commodity Groups

Commodity Group	Intended Function
Inaccessible medium-voltage (2kV to 15kV) cables and connections not subject to 10 CFR 50.49 EQ Requirements (ISG-18)	Provide electrical connections to specified sections of an electrical circuit to deliver voltage, current, or signals
High-Voltage Insulators (ISG-2)	Insulate and support an electrical conductor

2.5.2 Cables and Terminations Commodity

Description

The license renewal system Cables and Terminations (CBL) Commodity includes the cables and terminations in each Palisades system that includes components requiring electrical power or control. The purpose of establishing CBL is to treat all plant non-EQ cables and terminations as a commodity group.

Some SSCs in this system are considered in-scope in accordance with 10 CFR 54.4(a)(1). Since some SSCs in the Cables and Terminations Commodity are non-safety related, and their failure could affect the capability of SR SSCs to perform their safety function, they are considered to be in-scope in accordance with 10 CFR 54.4(a)(2). In addition, some SSCs are considered in-scope due to ATWS, Environmental Qualification, Fire Protection, and Station Blackout in accordance with 10 CFR 54.4(a)(3).

System Function Listing

A comprehensive listing of functions associated with the Cables and Terminations Commodity, or specific components contained in the commodity, is provided in the summary below.

System Function: CBL-01	Cri 1	Cri 2	Cri 3				
The CBL system includes safety related components.			FP	EQ	PTS	AT	SB
	X						

Comment: The system includes 1E components.

System Function: CBL-ATWS	Cri 1	Cri 2	Cri 3				
The system contains structures and/or components required by the current design basis for Anticipated Transient Without Scram.			FP	EQ	PTS	AT	SB
						X	

Comment: An example of a component with this system function is cabling used for ATWS detection.

System Function: CBL-EQ	Cri 1	Cri 2	Cri 3				
The system contains structures and/or components required by the current design basis for Environmental Qualification.			FP	EQ	PTS	AT	SB
				X			

Comment: EQ components are in the scope of license renewal but do not require an AMR since they are periodically replaced.

System Function: CBL-FP	Cri 1	Cri 2	Cri 3				
The system contains structures and/or components required by the current design basis for Fire Protection.			FP	EQ	PTS	AT	SB
			X				

Comment: Examples of components with this system function are fire detector cabling and fire alarm cabling.

System Function: CBL-NSAS	Cri 1	Cri 2	Cri 3				
The system contains structures and/or components whose failure could cause failure of safety related components due to spatial orientation.			FP	EQ	PTS	AT	SB
		X					

Comment: An example of a component with this system function is a non-safety related cable next to a safety related cable in a cable tray.

System Function: CBL-SBO	Cri 1	Cri 2	Cri 3				
The system contains structures and/or components required by the current design basis for Station Blackout (Loss of all AC power).			FP	EQ	PTS	AT	SB
							X

Comment: An example of a component with this system function is DC (direct current) cabling.

FSAR Reference

None.

Scoping Boundary Drawings

None.

Components Subject to an AMR

The components in this system that require aging management review are addressed as part of the commodity groups defined in Table 2.5-1.

2.5.3 Containment Isolation and Penetration System

System Description

The Containment Isolation and Penetration System (CIS) is designed to minimize the release of radioactivity from the containment building to the atmosphere in the event of an accident resulting in containment building high atmospheric pressure or radioactivity. All containment building penetrations are considered as potential sources of airborne radioactive leakage of the atmosphere in the event of a DBA or leakage of radioactive waste. Isolation valves are provided in all process systems penetrating the containment which serve to isolate the containment building atmosphere when required. The type and quantity of isolation valves for each pipe penetration are determined by the process system operational and physical characteristics in relation to the containment building atmosphere.

Included in the CIS is the Safety Injection System control circuits which are designed to automatically initiate the necessary engineered safeguards equipment upon a safety injection actuation signal (SIAS or SIS Signal - terms are used interchangeably) with or without offsite power available.

The Normal Shutdown and Design Basis Accident Sequencer is designed to sequentially load the safe shutdown equipment onto the emergency buses and/or diesel generators. Sequencing of loads ensures that the appropriate equipment is energized when needed while preventing excessive step loads from being on the diesel generator which could result in the loss of the generator.

The CIS also includes some components from the Plant Data Logger that provide an isolation function or have a seismic function only.

For license renewal purposes, the Containment Isolation and Penetration System is an electrical system. The components are active and not subject to an AMR. The mechanical portion of the system, (e.g., piping and valves) are addressed in their respective mechanical systems. The seals and penetrations are addressed in the civil / structural system.

The license renewal boundary of the CIS is described as follows: 1) from the containment pressure transmitters to the containment isolation relays that send signals to the process system containment isolation valves 2) from the containment isolation high radiation monitors to the containment isolation high radiation relays that send signals to the process system containment isolation valves and 3) the Normal Shutdown and Design Basis Accident Sequencers; 4) the SIS subsystem from the containment pressure transmitters to the safety injection relays that send signals to the process system safeguard equipment and containment isolation valves, and from the pressurizer low-low pressure transmitters to the safety injection relays that send signals to the process system; 5) the DTA subsystem, including the Multiplexer Input Modules for the Critical Function Monitor System (CFMS).

The description above results in some SSCs in this system being considered in-scope in accordance with 10 CFR 54.4(a)(1). Since some SSCs in the Containment Isolation and Penetration System are non-safety related and their failure could affect the capability of SR SSCs to perform their safety function, they are considered to be in-scope in accordance with 10 CFR 54.4(a)(2).

System Function Listing

A comprehensive listing of functions associated with the Containment Isolation and Penetration System, or specific components contained in the system, is provided in the summary following.

System Function: CIS-01 Process a signal on Containment High Radiation (CHR).	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Coincident two-out-of-four high-radiation signals from the auxiliary relays will trigger an alarm in the main control room, close all containment isolation valves not required for engineering safeguards except the component cooling line valves, and will isolate the control room ventilation system. High radiation detected by the refueling accident high-radiation monitors will also close all containment isolation valves not required for engineered safeguards when enabled by the respective refueling monitor key switches.

System Function: CIS-02 Process a signal on Containment High Pressure (CHP).	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Coincident two-out-of-four high-containment pressure signals will trigger an alarm in the main control room, close all containment isolation valves not required for engineered safeguards, and will isolate the control room ventilation system.

System Function: CIS-03 Provide containment operability in order to minimize the release of radioactive material to the environment.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: The containment isolation and penetration system is an electrical system for license renewal purposes. Components such as the personnel airlock, escape airlock, containment penetrations, and containment isolation valves are addressed in their individual system.

System Function: CIS-NSAS The system contains structures and/or components whose failure could cause failure of safety related components due to spatial interactions.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: in-scope of License Renewal to protect safety related components.

System Function: DTA-07 The Critical Functions Monitoring subsystem provides Safety Parameter Display System (SPDS) functions required by NUREG 0696 and NUTREG 0737 Supplement 1.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: The Critical Functions Monitoring node of the Data Logger contains equipment that is credited with a safety related function to isolate safety related 1E circuits from Non-1E circuits.

System Function: SEQ-01 Automatically sequence the connected Normal Shut Down (NSD) loads to bus 1C and/or bus 1D upon automatic closing of the associated diesel generator breaker with no SIS signal present.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: The sequencers are emergency generator support systems that are required to safely shutdown the plant after a loss of offsite power.

System Function: SEQ-02 Automatically sequence the connected Design Basis Accident (DBA) loads to bus 1C and/or bus 1D upon automatic closing of the associated diesel generator breaker with an SIS signal present.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: The sequencers are emergency generator support components that are required to safely shutdown the plant after a loss of offsite power. The design basis LOCA is assumed to be coincident with a loss of offsite power.

System Function: SIS-01 Process a signal on Containment High Pressure. (CHP).	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Two-out-of-four containment high-pressure signals initiate the SIS signal which, in turn, actuates two safety injection control circuits, each of which is supplied by a separate preferred ac source.

System Function: SIS-02 Process a signal on Pressurizer Low-Low Pressure.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Two-out-of-four pressurizer low-low pressure signals initiate the SIS signal which, in turn, actuates two safety injection control circuits, each of which is supplied by a separate preferred ac source.

FSAR Reference

Additional Containment Isolation and Penetration System details are provided in Section 7.3 and Section 8.4 of the FSAR.

Scoping Boundary Drawings

None

Components Subject to an AMR

There are no assets for this system that require aging management review.

2.5.4 Control Rod Drive System

System Description

The Control Rod Drive (CRD) System is designed to provide for gravity insertion of control rods on reactor scram and positive controlled insertion or withdrawal of control rods to provide definite shutdown margin, for short-term reactivity control and for control of axial xenon shifting. The major components are control rod drive mechanisms, and primary and secondary rod position indication. The control rod drive mechanism housing functions to provide a pressure boundary between primary coolant pressure and atmosphere are addressed by the Reactor Vessel System. For license renewal purposes, this system includes electrical functions only. Mechanical functions of the Control rod Drives are addressed with the Reactor Vessel (RVG) license renewal system.

The power to the CRD is supplied from the Control Rod Drive Power Supply subsystem. The CRD Power Supply subsystem provides both control power and motive power for the Control Rod Drive motors. The CRD System contains no electrical components which are required to remain operable during and after an accident to enable the operator to bring the plant to safe shutdown conditions. A loss of power from the CRD Power Supply System does not affect the position of the control rods. Power for the clutches, which (when energized) engage the control rods with the drives, is supplied from an independent source (Preferred AC) which is discussed under the Reactor Protective System. A loss of clutch power automatically causes a reactor safe shutdown, since the control rods drop into the reactor core by gravity when the clutches are de energized for any reason.

The license renewal boundary (electrical) of the CRD System begins at the load side of MCC breakers, step-down Transformers X45 and X46, manual bus transfer switch, the CRD Power Supply bus, and associated cables.

The description above results in some SSCs in this system being considered in-scope in accordance with 10 CFR 54.4(a)(1). In addition, some SSCs are considered in-scope due to ATWS, Environmental Qualification, and Station Blackout in accordance with 10 CFR 54.4(a)(3).

System Function Listing

A comprehensive listing of functions associated with the Control Rod Drive System is provided in the summary below.

System Function: CRD-01	Cri 1	Cri 2	Cri 3				
Provide instantaneous negative reactivity to the core on demand via gravity insertion of control rods on reactor scram.			FP	EQ	PTS	AT	SB
	X						

Comment: The Control Rod Drive System, in response to a Reactor Protection System trip signal, shuts down the reactor and maintains it in a safe shutdown condition. CRD is used to mitigate the consequences of numerous accidents and transients as described in FSAR Chapter 14. During a LOCA, high containment pressure provides the signal to shutdown the reactor and protect containment integrity.

System Function: CRD-02 Provide positive controlled insertion or withdrawal of control rods on receipt of a signal from the CRD control system.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: The ability to drive control rods is not required to shutdown the reactor and maintain it in a safe shutdown condition. In the event of an unsafe condition, the reactor trips and the rods are gravity inserted, i.e., they are not driven in by the control rod drive mechanism. As a contingency action to the reactor trip and a failure of all rods to fully insert, the control rods are selected and manually driven in per the Emergency Operating Procedures. Based on the above, this is a normal operating system function and is not required for safe shutdown. Therefore, this CRD system function is not within the boundary of License Renewal.

System Function: CRD-03 Provide control position indication and position related alarms.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Rod position indication is required to allow verification that the rods are positioned and aligned as assumed in the safety analysis. If no rod position indication exists for one or more control rods, continued operation is not allowed because the safety analysis assumptions of rod position cannot be ensured.

This is a normal operating system function and is not required for safe shutdown. Therefore, this CRD system function is not within the License Renewal boundary.

System Function: CRD-04 Maintain PCS pressure boundary.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: The pressure housings of the Control Rod Drive Mechanisms, including seals, are an extension of the reactor vessel and provide a primary coolant pressure boundary. This CRD system function is not within the electrical CRD license renewal system. The equipment which performs this function is in scope for License Renewal but will be evaluated in RVG, Reactor Vessel. See RVG System Function RVG-01.

System Function: CRD-05 Provide manual control of shutdown, regulating and part length control rods in either individual, group or group sequential modes.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: In the event of an unsafe condition, the reactor trips and the rods are gravity inserted, i.e., they are not driven in by the control rod drive mechanism. Following a reactor trip and a failure of all rods to fully insert, emergency boration is used to complete the reactor shutdown. As part of the reactivity control functional recovery and in parallel action with emergency boration, the control rods are selected and manually driven in per the Emergency Operating Procedures. This can only be performed if the non-Class 1E Control Rod Drive Power Supply is available. This availability and ability to drive control rod is not relied upon to shut down the reactor and maintain it in a safe shutdown condition.

Therefore, this CRD system function is not within the License Renewal boundary.

System Function: CRD-06 Provide a rundown signal to the shutdown and regulating rods on a reactor trip.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: The rundown feature of the Control Rod Drive System is a non-safety related regulating control. This is a backup action to the insertion of the regulating and shutdown control rods by gravity action on the receipt of a reactor trip signal. The reactor trip initiates rundown of the shutdown and regulating control rods through relays connected in parallel with the CRDM clutches. The rundown feature of the CRD system is not required for safe shutdown of the plant. Therefore, this CRD system function is not within the License Renewal boundary.

System Function: CRD-07 Provide interlocks to prohibit regulating group rod withdrawal under conditions that would put the reactor in an undesirable condition.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Interlocks to prohibit regulating group withdrawal are provided to prevent the reactor from reaching undesirable conditions. Operation outside of the limitations imposed by these control rod interlocks is not allowed and a reactor shutdown is required to enforce this. The limits on shutdown, regulating, and part-length rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on operability ensure that upon reactor trip, the full-length control rods will be available and will be inserted to provide enough negative reactivity to shut down the reactor. The operability requirements also ensure that the control rod banks maintain the correct alignment and that each full-length control rod is capable of being moved by its CRDM. The operability requirement for the part-length rods is that they are fully withdrawn. By maintaining rod alignments consistent with the safety analysis a safe shutdown of the reactor is ensured.

This CRD system function is within the boundary of License Renewal.

System Function: CRD-AT Provide instantaneous negative reactivity to the core on demand via gravity insertion of control rods during an Anticipated Transient Without Scram.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
						X	

Comment: The Control Rod Drive System, in response to a diverse and independent ATWS trip signal, interrupts power to the CRDM clutches which release to allow the control rods and connecting CRDM components to drop by gravity into the core. This action shuts down the reactor and maintains it in a safe shutdown condition. This CRD system function is within the boundary of License Renewal.

System Function: CRD-EQ SYSTEM Provide Full In or Not Full In analog position for any of 45 Control Rods.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
				X			

Comment: Full In or Not Full In Control Rod position indication is required for post accident monitoring. This indication is needed to verify safe shutdown of the reactor.

System Function: CRD-SB Provide instantaneous negative reactivity to the core on demand via gravity insertion of control rods during a Station Blackout.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
							X

Comment: The Control Rod Drive System, in response to a loss of all AC power (Station Blackout), receives an RPS low flow trip signal that deenergizes the CRDM clutches which release to allow the control rods and connecting CRDM components to drop by gravity into the core. This action shuts down the reactor and maintains it in a safe shutdown condition. This CRD system function is within the boundary of License Renewal.

FSAR Reference

Additional Control Rod Drive System details are provided in Section 7.2, Section 7.5, Section 7.6.2.3, Section 4, Section 4.7, Section 8.1.5, Section 8.4, and Appendix 7C of the FSAR.

Scoping Boundary Drawings

None.

Components Subject to an AMR

There are no assets for this system that require aging management review.

2.5.5 Neutron Monitoring System

Description

The Nuclear Monitoring System (NMS) instrumentation consists of excore and incore flux monitoring chambers. Eight channels of excore instrumentation monitor the neutron flux and six of the eight channels provide reactor protection signals during start-up and power operation. Two of the channels follow the neutron flux through the start-up range. The incore monitors consist of rhodium neutron detectors and a thermocouple. The incore monitoring system provides information on neutron flux and temperatures in the core.

The License Renewal Boundary for the Neutron Monitoring System extends from the neutron detector wells embedded in the Reactor Shield Wall which contains the source/wide range and power range neutron detector assemblies, through the signal cables from the detectors which are routed through Containment electrical penetrations to the Control Room and the remote shutdown panel. The system provides safety-related inputs to the Reactor Protective System (RPS).

Some SSCs in this system are considered in-scope in accordance with 10 CFR 54.4(a)(1). Since some SSCs in the Neutron Monitoring System are non-safety related and their failure could affect the capability of SR SSCs to perform their safety function, they are considered to be in-scope in accordance with 10 CFR 54.4(a)(2). In addition, some SSCs are considered in-scope due to Environmental Qualification and Fire Protection in accordance with 10 CFR 54.4(a)(3).

System Function Listing

A comprehensive listing of functions associated with the Neutron Monitoring System, or specific components contained in the system, is provided in the summary that follows.

System Function: NMS-01	Cri 1	Cri 2	Cri 3				
Provide accurate, self-powered neutron flux detection within the reactor core and provide the core power profile signals to the Plant Process Computer (PPC) for reporting and alarms.			FP	EQ	PTS	AT	SB

Comment: This function of the neutron monitoring system does not perform a license renewal intended function per 10CFR54.4 and is not within the scope of the license renewal rule.

System Function: NMS-02 Provide instantaneous multiple channel detection of neutron flux in the reactor core continuously from the source range through 200% of full power.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This normal operating function is not in scope. The similar Appendix R function is addressed as function NMS-FP.

System Function: NMS-03 Provide an instantaneous reactor power input signal to Thermal Margin Monitor from power range nuclear instrumentation for thermal margin/low pressure trips.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: This function of the Neutron Monitoring System (NMS) is a nuclear safety function. The NMS provides a safety-related signal to the Reactor Protection System (RPS).

System Function: NMS-04 Provide an instantaneous reactor power input signal to RPS from source/wide range Nuclear Instrumentation for high rate of power change trips.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: This function of the Neutron Monitoring System (NMS) is a nuclear safety function. The NMS provides a safety-related signal to the Reactor Protection System (RPS).

System Function: NMS-05 Provide an instantaneous reactor power input signal to Thermal Margin Monitor from power range Nuclear Instrumentation for variable high power trips.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: This function of the Neutron Monitoring System (NMS) is a nuclear safety function. The NMS provides a safety-related signal to the Reactor Protection System (RPS).

System Function: NMS-EQ The system contains structures and/or components required by the current licensing basis for Environmental Qualification.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
				X			

Comment: The Neutron Monitoring System contains components that are required to be environmentally qualified in accordance with 10CFR50.49.

System Function: NMS-FP Provide instantaneous multiple channel detection of neutron flux in the reactor core continuously from the source range through 200% of full power.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: The system contains structures and/or components required by the current licensing basis for fire protection.

System Function: NMS-NSAS	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
The system contains structures and/or components whose failure could cause failure of safety related components due to spatial interactions.		X					

Comment: The system contains some non-safety related components that are seismically supported to in order to preclude spatial interaction with other safety-related components (specifically, the Audible Count Rate Speaker).

FSAR Reference

Additional Neutron Monitoring System details are provided in Section 7.6, Section 7.2, and Table 5.2-5 of the FSAR.

Scoping Boundary Drawings

None

Components Subject to an AMR

The components in this system that require aging management review are addressed as part of the commodity groups described in Table 2.5-1.

2.5.6 Radiation Monitoring System

System Description

The Radiation Monitoring System (RIA) consists of monitors, instrumentation and alarms that warn plant personnel of increasing radiation levels in various areas of the plant. The alarm and/or control functions of these monitors allow action, either automatic or manual, to be taken to correct excessive radiation levels or control discharges of radioactive material to the environment. The major elements of this system are area monitors, process monitors and continuous air monitors. Thirty-seven area monitors are strategically locations throughout the plant. They provide both indication and warning of radiation levels in both normally radioactive and non-radioactive areas. Instrument ranges and sensitivities are chosen to enable monitoring within the requirements of 10 CFR 20. The process radiation and effluent radiological monitoring and sampling system is designed to assure that ionizing radiation levels are indicated and alarmed so that action, either automatic or manual, can be taken to prevent radioactive releases from exceeding the limits of 10 CFR 20. Continuous air monitors are used in areas of potential airborne radioactivity or air samples are taken with a portable sampler and analyzed for airborne radioactivity. Continuous air monitors are set to alarm when the airborne radioactivity reaches the applicable derived air concentrations.

For license renewal purposes, this system is an electrical system only. The pressure boundary components associated with the Radiation Monitoring System are included with the systems which they monitor.

The license renewal boundaries of the Radiation Monitoring System are described as follows: The boundaries are as follows: 1) The Containment radiation monitors, 2) area radiation elements in the Containment, the auxiliary building, the radwaste building, control building, turbine building, and support building, 3) plant system radiation elements for the radwaste ventilation, the engineered safeguards rooms ventilation, the Containment building gas, the waste gas surge tank outlet, the radwaste discharge, the gaseous effluent, the main steam line, and the service water discharge.

Some SSCs in this system are in-scope in accordance with 10 CFR 54.4(a)(1). Since some SSCs in the Radiation Monitoring System are non-safety related and their failure could affect the capability of SR SSCs to perform their safety function, they are considered to be in-scope in accordance with 10 CFR 54.4(a)(2). In addition, some SSCs are considered in-scope due to Environmental Qualification in accordance with 10 CFR 54.4(a)(3).

System Function Listing

A comprehensive listing of functions associated with the Radiation Monitoring System, or specific components contained in the system, is provided in the summary following.

System Function: RIA-01 Monitor condenser off-gas radiation for identification of a primary-to-secondary steam generator tube leak.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This function is not in-scope of LR because it does not meet any of the three LR Criteria. In particular, it does not meet Criterion 1 because it is not required to ensure the integrity of the reactor coolant pressure boundary, does not affect the capability to shut down the reactor and maintain it in a safe shutdown condition, and does not mitigate the consequences of accidents that could result in potential offsite exposure comparable to the 10CFR100 guidelines. It does not meet Criterion 2 because its failure could not prevent satisfactory accomplishment of any of the functions of Criterion 1. The monitor that supports this function is RE-0631.

System Function: RIA-02 Automatically align the component cooling surge tank vent to the gas collection header on high radiation.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This function is not in-scope of LR because it does not meet any of the three LR Criteria. In particular, it does not meet Criterion 1 because it is not required to ensure the integrity of the reactor coolant pressure boundary, does not affect the capability to shut down the reactor and maintain it in a safe shutdown condition, and does not mitigate the consequences of accidents that could result in potential offsite exposure comparable to the 10CFR100 guidelines. It does not meet Criterion 2 because its failure could not prevent satisfactory accomplishment of any of the functions of Criterion 1. The monitor that supports this function is RE-0915. However, this component was transferred to the Component Cooling System due to a pressure boundary function for that system.

System Function: RIA-03 Monitor the waste gas discharge, and automatically close the waste gas decay tank discharge valve when the release rate exceeds pre-determined limits.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This function is not in-scope of LR because it does not meet any of the three LR Criteria. In particular, it does not meet Criterion 1 because it is not required to ensure the integrity of the reactor coolant pressure boundary, does not affect the capability to shut down the reactor and maintain it in a safe shutdown condition, and does not mitigate the consequences of accidents that could result in potential offsite exposure comparable to the 10CFR100 guidelines. It does not meet Criterion 2 because its failure could not prevent satisfactory accomplishment of any of the functions of Criterion 1. The monitor that supports this function is RE-1113.

System Function: RIA-04 Automatically isolate the liquid radwaste discharge.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Monitor RE-1049 is Q-listed as safety related. Therefore, RE-1049 is in boundary of License Renewal.

System Function: RIA-05 Automatically isolate the engineered safeguards ventilation on high radiation.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Monitor RE-1810 and 1811 initiate the electrical signal to perform this function. They are Q-listed for seismic and, therefore, RE-1810 and 1811 are in boundary of License Renewal under System Function RIA-NSAS.

System Function: RIA-06	Cri 1	Cri 2	Cri 3				
Monitor the main steam safety and dump valve discharge.			FP	EQ	PTS	AT	SB

Comment: Monitor RE-2323 and RE-2324 perform this function. RE-2323 and -2324 are Q-Listed for seismic and, therefore, in boundary of License Renewal under RIA-NSAS.

System Function: RIA-07	Cri 1	Cri 2	Cri 3				
Monitor particulates in the main plant stack.			FP	EQ	PTS	AT	SB

Comment: Monitor RE-2325 is Q-listed for Technical specifications only. RE-2325 supports no license renewal criterion and, therefore, is not in the boundary of license renewal.

System Function: RIA-08	Cri 1	Cri 2	Cri 3				
Monitor for normal range and high range noble gases in the main plant stack.			FP	EQ	PTS	AT	SB

Comment: Monitors RE-2326 and -2327 are Q-listed for Tech. Spec. only. RE-2326 and -2327 support no LR criterion and, therefore, are not in the boundary of License Renewal.

System Function: RIA-09	Cri 1	Cri 2	Cri 3				
Monitor radiation levels in containment and provide high radiation signals to CHR.			FP	EQ	PTS	AT	SB
	X						

Comment: Monitors RE-1805, 1806, 1807, 1808, 2316, and 2317 perform this function. RE-1805, 1806, 1807, and 1808 are Q-listed as safety related. Therefore, RE-1805, 1806, 1807, and 1808 are in boundary of LR for this system function. RE-2316, and 2317 Q-listed as seismic only, therefore, they are in the boundary of License Renewal under RIA-NSAS.

System Function: RIA-10	Cri 1	Cri 2	Cri 3				
Monitor the CVC letdown line for fuel degradation via the failed fuel monitor.			FP	EQ	PTS	AT	SB

Comment: Monitor RE-0202 is retired in place. Therefore, RE-0202A is not in the boundary of License Renewal other than for function RIA-NSAS.

System Function: RIA-11 Automatically isolate the surface and bottom steam generator blowdown isolation valves on receipt of a high radiation signal.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This function is not in-scope of LR because it does not meet any of the three LR Criteria. In particular, it does not meet Criterion 1 because it is not required to ensure the integrity of the reactor coolant pressure boundary, does not affect the capability to shut down the reactor and maintain it in a safe shutdown condition, and does not mitigate the consequences of accidents that could result in potential offsite exposure comparable to the 10CFR100 guidelines. It does not meet Criterion 2 because its failure could not prevent satisfactory accomplishment of any of the functions of Criterion 1. The monitor that supports this function is RE-0707.

System Function: RIA-12 Monitor for containment gamma radiation.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Monitors RE-2321 and RE-2322 are Q-listed as safety related. Therefore, RE-2321 and RE-2322 are in the boundary of License Renewal.

System Function: RIA-13 Monitor for criticality in the Spent Fuel Pool.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Monitors RE-2313 and RE-5709 perform this function and are Q-listed as seismic only. Therefore, RE-2313 and RE-5709 are in the boundary of License Renewal under RIA-NSAS.

System Function: RIA-14 Automatically isolate Fuel Handling Area HVAC on high radiation.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This function is not in-scope of LR because it does not meet any of the three LR Criteria. In particular, it does not meet Criterion 1 because it is not required to ensure the integrity of the reactor coolant pressure boundary, does not affect the capability to shut down the reactor and maintain it in a safe shutdown condition, and does not mitigate the consequences of accidents that could result in potential offsite exposure comparable to the 10CFR100 guidelines. It does not meet Criterion 2 because its failure could not prevent satisfactory accomplishment of any of the functions of Criterion 1. The monitor that supports this function is RE-5712.

System Function: RIA-15 Automatically isolate Penetration/Fan Room HVAC on high radiation.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This function is not in-scope of LR because it does not meet any of the three LR Criteria. In particular, it does not meet Criterion 1 because it is not required to ensure the integrity of the reactor coolant pressure boundary, does not affect the capability to shut down the reactor and maintain it in a safe shutdown condition, and does not mitigate the consequences of accidents that could result in potential offsite exposure comparable to the 10CFR100 guidelines. It does not meet Criterion 2 because its failure could not prevent satisfactory accomplishment of any of the functions of Criterion 1. The monitor that supports this function is RE-5710.

System Function: RIA-16 Monitor radiation in the Auxiliary Building areas adjacent to piping which could be a source of loss of PCS coolant.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Monitors RE-2300, -2301 and -2302. perform this function and are Q-listed as seismic only. Therefore, RE-2300, RE-2301 and RE-2302 are in the boundary of License Renewal under RIA-NSAS.

System Function: RIA-17 Provide Continuous Air Monitoring (CAM) of the control room viewing gallery, and alarm on high radiation to provide for manual switching of control room HVAC to emergency mode.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: Monitors RE-1818A and RE-1818B are Q-listed for seismic only. Therefore, RE-1818A and RE-1818B are in the boundary of License Renewal under RIA-NSAS.

System Function: RIA-18 Monitor for noble gases in Containment Building.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This function is not in-scope of LR because it does not meet any of the three LR Criteria. The monitor that supports this function is RE-1817.

System Function: RIA-19 Monitors for radiation in other areas of the plant that do not perform a 10 CFR 54, Criteria 1, 2, or 3 function.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This function covers the radiation monitors in the plant that are not covered in the other functions for this system. This function is not in-scope of LR because it does not meet any of the three LR Criteria. The monitors that support this function are RE-1323, 1809, 1815, 2300, 2301, 2302, 2304, 2305, 2306, 2307, 2308, 2309, 2320, 2328, 5211, 5705, 5707, 5708, 5711, 5713, and 5714.

System Function: RIA-20 Monitors for radiation in other areas of the plant that perform a 10 CFR 54, Criteria 1 function.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: Monitors RE-0833, 2300, 2301, 2302, 2303, 2304, 2305, 2306, 2307, 2308, 2309, 2310, 2311, 2312, 2314, 2315, 2323, 2324, 2326, 2327, 5701, 5702, 5703, 5704, 5706, and 5710 perform this function and are required to be operable. Therefore, these REs are in boundary of License Renewal. RE-2303 is Q-listed for seismic only and is in License Renewal boundary under RIA-NSAS.

System Function: RIA-EQ The system contains structures and/or components required by the current licensing basis for Environmental Qualification.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
				X			

Comment: This function is in-scope of License Renewal. The monitors that support this function are RE/RIA-1805, -1806, -1807, -1808, -2321 and -2322.

System Function: RIA-NSAS The system contains structures and/or components whose failure could cause failure of safety related components due to spatial interactions.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: The following Radiation Detection Elements are in-scope of License Renewal with a Q-list of seismic only: RE-0202, 1810, 1818A, 1818B, and 5709.

FSAR Reference

Additional Radiation Monitoring System details are provided in Section 6.7, Section 7.4.5, Section 9.8, Section 10.2, Section 11.5, Section 11.6, Appendix 7C, Table 5.2-5, Section 14.15, and Section14.19 of the FSAR.

Scoping Boundary Drawings

LR-M-202-1
LR-M-207-1
LR-M-208-1A
LR-M-210-1C
LR-M-218-2
LR-M-218-4
LR-M-218-6
LR-M-223-1
LR-M-223-1A
LR-M-223-1B
LR-M-223-2
LR-M-223-3
LR-M-654

Components Subject to an AMR

The components in this system that require aging management review are addressed as part of the commodity groups listed in Table 2.5-1.

2.5.7 Reactor Protective System

Description

The Reactor Protective System (RPS) is a Class 1E system comprised of the sensor instrumentation, amplifiers, trip units, logic circuits, actuator circuits and other equipment as required to monitor selected nuclear steam supply system conditions is designed to reliably effect a rapid reactor shutdown (scram) if any one or combination of conditions deviates from a preselected operating range. The system functions to protect the reactor core.

The RPS is housed in four cabinets in the control room. The major components are: the bistable trip units and Auxiliary trip units; coincidence logic matrices; clutch power trip circuits; clutch power supplies; and clutch coil relays.

Components of the RPS in boundary for license renewal include 1E electrical components (circuit breakers, indicators, integrators, switches, transformers, isolation devices, relays, etc.) that were screened as active. Therefore, no aging management is required.

Some SSCs in this system are considered in-scope in accordance with 10 CFR 54.4(a)(1). Since some SSCs in the Reactor Protective System are non-safety related and their failure could affect the capability of SR SSCs to perform their safety function, they are considered to be in-scope in accordance with 10 CFR 54.4(a)(2). In addition, some SSCs are considered in-scope due to ATWS in accordance with 10 CFR 54.4(a)(3).

System Function Listing

A comprehensive listing of functions associated with the Reactor Protective System, or specific components contained in the system, is provided in the summary below.

System Function: RPS-01	Cri 1	Cri 2	Cri 3				
Provide automatic and manual trips via RPS clutch coils.			FP	EQ	PTS	AT	SB
	X						

Comment: RPS is a 1E electrical system.

System Function: RPS-02 Provide automatic and manual trips via RPS breakers.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: RPS is a 1E electrical system.

System Function: RPS-03 Provide manual and automatic bypass of RPS reactor trips for testing (at power and shutdown) and for the equipment protection trips.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: This function does not support a license renewal criterion.

System Function: RPS-04 Provide trip signals and setpoints and provide annunciated alarms to warn of unacceptable DNB margins.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
	X						

Comment: RPS is a 1E electrical system.

System Function: RPS-AT The system contains structures and/or components required by the current licensing basis for Anticipated Transients Without Scram.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
						X	

Comment: ATWS components include Matrix Relays, ATWS1 Relays, etc.

System Function: RPS-NSAS The system contains structures and/or components whose failure could cause failure of safety related components due to spatial interactions.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment:

FSAR Reference

Additional Reactor Protective System details are provided in Section 7.2 and Section 7.6 of the FSAR.

Scoping Boundary Drawings

None.

Components Subject to an AMR

The components in this system that require aging management review are addressed as part of the commodity groups listed in Table 2.5-1.

2.5.8 Station Power System

Description

The Station Power System consists of the transformers, switchgear and other components necessary to transform 345KV Power to three voltage levels used by the internal plant equipment. SPS includes four 4,160 volt buses, four 2,400 volt buses, and several 480 volt load centers and 480 volt motor control centers. The subsystems are 4,160 volt, 2,400 volt, and 480 volt.

The 4,160 volt subsystem is designed to reliably function and supply power during normal, abnormal and accident conditions to the 4,160 volt station auxiliaries. The normal power source for the 4,160 volt station auxiliaries is the site turbine generator which supplies the reactor plant primary coolant pumps and condensate pumps via 21 KV - 4,160 volt Station Power Transformer 1-1. Also, the cooling towers and associated equipment loads are supplied via the 345 kV - 4,160 volt Station Power Transformer 1-3. When the turbine generator is out of service, offsite power is supplied to the primary coolant pumps and condensate pumps via 345 kV - 4,160 volt Start-Up Transformers 1-1 and 1-3. The 4,160 volt subsystem provides no safety function and is non-Class 1E.

The 2,400 volt subsystem is a three-phase, ungrounded power distribution system intended to provide a safe, reliable and efficient means of supplying power to the plant auxiliary loads during startup, normal, off-normal, emergency and shutdown conditions. 2,400 volt reactor and turbine plant loads, including the engineered safeguards electric subsystem, are normally supplied from either the offsite power source 345 kV - 2,400 volt Safeguard Transformer 1-1 or the offsite power source 345 KV - 2,400 volt Start-Up Transformer 1-2. To ensure adequate voltage levels on the engineered safeguards buses, load shedding and sequencing is utilized to shed all loads and reestablish the Class 1E loads on Buses 1C and 1D when the emergency diesel generators are called on to supply power.

The 480 volt subsystem distributes power to the safety and non-safety related process loads and to non-process equipment during normal, abnormal and emergency conditions. Certain 480 volt load center buses receive power from load center transformers energized from the 2,400 volt buses. Other 480 volt load center buses receive power from load center transformers energized from 4,160 volt Buses 1A, 1F and 1G. The 480 volt motor control centers are connected to the 480 volt load center buses.

The boundaries of the in-scope portions of the Station Power System can be described as follows: 1) Startup Transformers, 2) Safeguards Transformer and Safeguards Bus, and 3) 2400 volt AC buses 1C and 1D, 4) 480 volt distribution system supplied from buses 1C and 1D, and 5) Preferred AC buses.

Some SSCs in this system are considered in-scope in accordance with 10 CFR 54.4(a)(1). Since some SSCs in the Station Power System are non-safety related and their failure could affect the capability of SR SSCs to perform their safety function, they are considered to be in-scope in accordance with 10 CFR 54.4(a)(2). In addition, some SSCs are considered in-scope due to Fire Protection and Station Blackout in accordance with 10 CFR 54.4(a)(3).

System Function Listing

A comprehensive listing of functions associated with the Station Power System, or specific components contained in the system, is provided in the following summary.

<table>
<tr><td rowspan="3">System Function: HAC-01
Start and load 4,160 volt power to the associated auxiliary loads under all modes of operation.</td><td rowspan="2">Cri 1</td><td rowspan="2">Cri 2</td><td colspan="5">Cri 3</td></tr>
<tr><td>FP</td><td>EQ</td><td>PTS</td><td>AT</td><td>SB</td></tr>
<tr><td></td><td></td><td></td><td></td><td></td><td></td><td></td></tr>
</table>

Comment: The 4,160 volt subsystem provides no safety function and is non-Class 1E. This system function does not support a license renewal criterion.

<table>
<tr><td rowspan="3">System Function: HAC-02
Provide transfer (automatically or manually) of the 4,160 volt loads to the available offsite power source (while all auxiliaries continue to run) upon turbine generator or reactor trips.</td><td rowspan="2">Cri 1</td><td rowspan="2">Cri 2</td><td colspan="5">Cri 3</td></tr>
<tr><td>FP</td><td>EQ</td><td>PTS</td><td>AT</td><td>SB</td></tr>
<tr><td></td><td></td><td></td><td></td><td></td><td></td><td></td></tr>
</table>

Comment: The 4,160 volt subsystem provides no safety function and is non-Class 1E. This system function does not support a license renewal criterion.

<table>
<tr><td rowspan="3">System Function: HAC-03
Provide the control room with electrical parameter indications and annunciation of important functional/parameter changes in the 4160 volt subsystem.</td><td rowspan="2">Cri 1</td><td rowspan="2">Cri 2</td><td colspan="5">Cri 3</td></tr>
<tr><td>FP</td><td>EQ</td><td>PTS</td><td>AT</td><td>SB</td></tr>
<tr><td></td><td></td><td></td><td></td><td></td><td></td><td></td></tr>
</table>

Comment: The 4,160 volt subsystem provides no safety function and is non-Class 1E. This system function does not support a license renewal criterion.

<table>
<tr><td rowspan="3">System Function: HAC-04
Raise the output voltage produced by the main generator from 22KV to 345KV for distribution to the power grid.</td><td rowspan="2">Cri 1</td><td rowspan="2">Cri 2</td><td colspan="5">Cri 3</td></tr>
<tr><td>FP</td><td>EQ</td><td>PTS</td><td>AT</td><td>SB</td></tr>
<tr><td></td><td></td><td></td><td></td><td></td><td></td><td></td></tr>
</table>

Comment: This system function is not a function of 4,160 volt subsystem. This function is supported by the Turbine Generator System (TGS-09, TGS-10) and the Switchyard System (SWY-02).

System Function: LAC-01	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
Start and load 480 volt AC power to the associated auxiliary loads under all modes of operation.	X						

Comment: The 480 volt system involves both safety and non-safety related equipment. Most of the nuclear transient analyses in FSAR Chapter 14 assumes at least some portion of the 480 volt system associated with safeguards equipment is operable. This SPS system function is within the boundary of License Renewal.

System Function: LAC-02	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
Provide AC power to safety related instruments and components which are supplied by instrument AC panel EY-01.							

Comment: This system function is not a function of 480 volt power. This function is supported by the Emergency Power System (EPS-01, PAC-01, EDC-02).

System Function: LAC-03	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
Provide the instrument AC panel EY-01 as a backup to any single preferred AC panel via the bypass regulator.							

Comment: This system function is not a function of 480 volt power. This function is supported by the Emergency Power System (EPS-01).

System Function: LAC-04	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
Supply power to Data Logger system components from 125 volt non-vital DC or 125 volt AC.							

Comment: Per FSAR Tables 5.2-4 and 5.2-5, the Plant Process Computer and associated DC/AC power supplies are non-Safety Class 1E. This system function does not support a license renewal criterion.

System Function: MAC-01	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
Start and load 2,400 volt AC power to the associated auxiliary loads under all modes of operation.	X						

Comment: The 2,400 volt system is designed to reliably function and supply power during normal, abnormal and accident conditions. Two of the four 2,400 volt buses are an integral part of the plant engineered safeguards electrical system and are identified as Class 1E components.

System Function: MAC-02	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
Provide automatic and manual transfer of the 2,400 volt loads to the redundant offsite power source, while attached loads continue to run on loss of the normal offsite power source.	X						

Comment: If power is lost to the safeguards transformer, the 2,400 volt loads will automatically transfer to startup transformer 1-2. Startup Transformer 1-2 can supply both safety related and non-safety related 2,400 volt loads. These safety related loads are required for safe shutdown of the reactor.

System Function: MAC-03	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
Provide the control room with electrical parameter indications and annunciation of important functional/parameter changes for 2400 volt AC.	X						

Comment: 2,400 volt buses, 1C and 1D, supply power to engineered safeguard loads and are part of the engineered safeguards electrical system. Although annunciators are not safety related and required for safe shutdown of the reactor, meters and other indicators are necessary for personnel in the Control Room to monitor this safeguards electrical system.

System Function: MAC-04	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
Provide automatic voltage protection of buses 1C and 1D from loss or degradation of offsite power by tripping the respective incoming bus breakers, starting both EDGs, and initiating bus load shed.	X						

Comment: Two of the 2,400 volt buses, 1C and 1D, supply power to engineered safeguard loads and are part of the engineered safeguards electrical system. If either 2,400 volt bus, 1C or 1D, experiences a sustained undervoltage, the associated EDG is started, the affected bus is separated from its offsite power sources, major loads are stripped from that bus and its supported buses, the EDGs are connected to the bus, and ECCS or shutdown loads are started by an automatic load sequencer. These safety related loads are required for safe shutdown of the reactor.

System Function: MAC-05	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
Provide load shed of bus 1E and selected nonessential loads on receipt of a Safety Injection Actuation Signal.	X						

Comment: The load shedding function is required to support post-accident operation of the SPS system and supply of essential power to safety related loads. This SPS system function is within the boundary of License Renewal.

System Function: SPS-AT	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
The system contains structures and/or components required by the current licensing basis for Anticipated Transients Without Scram.						X	

Comment: The SPS provides power to the ATWS Control circuitry.

System Function: SPS-FP The system contains structures and/or components required by the current licensing basis for Fire Protection.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
			X				

Comment: Safeguard busses 1C and 1D and associated load centers, MCCs, and inverters and breakers are required for Appendix R safe shutdown of the reactor.

System Function: SPS-NSAS The system contains structures and/or components whose failure could cause failure of safety related components due to spatial interactions.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: In-scope of License Renewal to protect safety related components.

System Function: SPS-SBO The system constrains structures and/or components required by the current licensing basis for Station Blackout (Loss of All AC Power).	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
							X

Comment: The normal supplies from the safeguards transformer in the switchyard to safeguard buses 1C and 1D have been included as part of the plant equipment required (for License Renewal) for mitigation of the SBO event. The inverter breakers supply power to equipment required for safe shutdown of the reactor.

FSAR Reference

Additional Station Power System details are provided in Section 5, Section 8.1, Section 8.3, and Section 8.6 of the FSAR.

Scoping Boundary Drawings

The scoping boundary drawings for the Station Power System are listed below:

LR-WD950-A

Components Subject to an AMR

The components in this system that require aging management review are addressed as part of the commodity groups listed in Table 2.5-1.

2.5.9 Switchyard System

Description

The Switchyard System (SWY) operates at 345 KV and is arranged to give maximum availability of the power system grid. The equipment is selected to have the capability of isolating system and substation faults with a minimum effect on stability of the power system grid.

The switchyard is designed in a breaker-and-one-half arrangement with two main buses (Buses F and R) and connections for the generator main power transformer, the Plant safeguard transformer, the Plant start-up transformers, and six outgoing lines. Two of the outgoing lines are connected to the American Electric Power system and the remaining lines are connected to the Consumers Energy Company system and the Michigan Power Pool. Each line has sufficient capacity to carry the entire output of the main turbine generator. A line also goes to the independent generator near the plant site.

The license renewal boundaries of the Switchyard license renewal system are described as follows: Switchyard components that provide power from 345KV bus F through the motor operated disconnect to the Safeguards Transformer, and from 345KV bus R through the motor operated disconnect to the Startup Transformers.

Since some SSCs in the Switchyard are non-safety related and their failure could affect the capability of SR SSCs to perform their safety function, they are considered to be in-scope in accordance with 10 CFR 54.4(a)(2). In addition, some SSCs are considered in-scope due to Station Blackout in accordance with 10 CFR 54.4(a)(3).

System Function Listing

A comprehensive listing of functions associated with the Switchyard, or specific components contained in the system, is provided in the following summary.

System Function: SWY-01 Provide offsite power to and from the front bus and the rear bus.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: The 345 KV Switchyard System provides no safety function and is non-Class 1E. All 345 KV Switchyard components are non-Class 1E. The 345 KV Switchyard is not taken credit for in any FSAR Chapter 14 event. This system function does not support a license renewal criterion.

System Function: SWY-02 Provide 345 kV connection between the switchyard and the high side of plant transformers.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: The 345 KV Switchyard System provides no safety function and is non-Class 1E. All 345 KV Switchyard components are non-Class 1E. The 345 KV Switchyard is not taken credit for in any FSAR Chapter 14 event. This system function does not support a license renewal criterion.

System Function: SWY-03 Provide reliable 240 volt AC power for air compressor operation for the switchyard ACB breakers.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: The 345 KV Switchyard System provides no safety function and is non-Class 1E. All 345 KV Switchyard components are non-Class 1E. The 345 KV Switchyard is not taken credit for in any FSAR Chapter 14 event. This system function does not support a license renewal criterion.

System Function: SWY-04 Provide reliable DC power for switchyard breaker control.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: The 345 KV Switchyard System provides no safety function and is non-Class 1E. All 345 KV Switchyard components are non-Class 1E. The 345 KV Switchyard is not taken credit for in any FSAR Chapter 14 event. This system function does not support a license renewal criterion.

System Function: SWY-05 Provide for reliable operation of switchyard relays and breakers.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB

Comment: The 345 KV Switchyard System provides no safety function and is non-Class 1E. All 345 KV Switchyard components are non-Class 1E. The 345 KV Switchyard is not taken credit for in any FSAR Chapter 14 event. This system function does not support a license renewal criterion.

System Function: SWY-NSAS The system contains structures and/or components whose failure could cause failure of safety related components due to spatial orientation.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
		X					

Comment: Switchyard-related instruments are located on control room panel.

System Function: SWY-SBO Provide power to the plant during recovery from a station blackout event.	Cri 1	Cri 2	Cri 3				
			FP	EQ	PTS	AT	SB
							X

Comment: The system contains structures and/or components relied on for recovery of offsite power following a Station Blackout (Loss of all AC power). The 345 KV Switchyard System provides no safety function and is non-Class 1E.

FSAR Reference

Additional Switchyard details are provided in Section 8.2 of the FSAR.

Scoping Boundary Drawings

The scoping boundary drawings for the Switchyard are listed below:

LR-WD950-A
LR-WD-1421-31

Components Subject to an AMR

The components in this system that require aging management review are addressed as part of the commodity groups listed in Table 2.5-1.

3.0 Aging Management Review Results

For those structures and components that are subject to aging management review, 10 CFR 54.21(a)(3) of the license renewal rule requires a demonstration that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation.

This section describes the results of the aging management reviews for those structures and components that were identified in Section 2.0, Scoping and Screening Methodology for Identifying Structures and Components Subject to Aging Management Review.

3.0.1 Aging Management Review Results Display Method

This section provides the results of the aging management review for those structures and components identified in Section 2.0 as being subject to aging management review.

Descriptions of the internal and external service environments which were used in the aging management review to determine aging effects requiring management are included in Table 3.0-1, Service Environments. The environments used in the aging management reviews are listed in the Environment column.

Most of the Aging Management Review (AMR) results information in Section 3 is presented in the following two tables:

- Table 3.x.1 - where '3' indicates the LRA section number, 'x' indicates the subsection number from NUREG-1801, Volume 1, and '1' indicates that this is the first table type in Section 3. For example, in the Reactor Coolant System subsection, this table would be number 3.1.1, in the Engineered Safety Features subsection, this table would be 3.2.1, and so on. For ease of discussion, this table will hereafter be referred to in this Section as “Table 1."
- Table 3.x.2-y - where '3' indicates the LRA section number, 'x' indicates the subsection number from NUREG-1801, Volume 1, and '2' indicates that this is the second table type in Section 3; and 'y' indicates the system table number. For example, for the Reactor Vessel, within the Reactor Coolant System subsection, this table would be 3.1.2-1 and for the Reactor Vessel Internals, it would be table 3.1.2-2. For the Containment Spray System, within the Engineered Safety Features subsection, this table would be 3.2.2-1. For the next system within the ESF subsection, it would be table 3.2.2-2. For ease of discussion, this table will hereafter be referred to in this section as “Table 2."

Table Description

NUREG-1801, “Generic Aging Lessons Learned (GALL) Report,” contains the staff's generic evaluation of the existing plant programs. It documents the technical basis for determining where existing programs are adequate without modification, and where existing programs should be augmented for the extended period of operation. The evaluation results documented in the report indicate that many of the existing programs are adequate to manage the aging effects for particular structures or components, within the scope of license renewal, without change. The report also contains recommendations on specific areas for which existing programs should be augmented for license renewal. In order to take full advantage of NUREG-1801, a comparison between the AMR results and the tables of NUREG-1801 has been made. The results of that comparison are provided in the two tables.

Table 1 (Figure 3.0-1)

The purpose of Table 1 is to provide a summary comparison of how the facility aligns with the corresponding tables of NUREG-1801, Volume 1. The table is essentially the same as Tables 1 through 6 provided in NUREG-1801, Volume 1, except that the “Type” column has been replaced by an “Item Number” column and the “Item Number in GALL” column has been replaced by a “Discussion” column.

The “Item Number” column provides the reviewer with a means to cross-reference from Table 2 to Table 1.

The “Discussion” column is used by the applicant to provide clarifying/amplifying information. The following are examples of information that might be contained within this column:

- “Further Evaluation Recommended” information or reference to where that information is located (including a hyperlink if possible)
- The name of a plant specific program being used (and a hyperlink to the program if possible)
- Exceptions to the NUREG-1801 assumptions
- A discussion of how the line item is consistent with the corresponding line item in NUREG-1801, Volume 1, when that may not be intuitively obvious
- A discussion of how the line item is different than the corresponding line item in NUREG-1801, Volume 1, when it may appear to be consistent (e.g., when there is exception taken to an aging management program that is listed in NUREG-1801, Volume 1)

The format of Table 1 provides the reviewer with a means of aligning a specific Table 1 row with the corresponding NUREG-1801, Volume 1 table row, thereby allowing for the ease of checking consistency.

Table 2 (Figure 3.0-2)

Table 2 provides the detailed results of the aging management reviews for those components identified in LRA Section 2 as being subject to aging management review. There will be a Table 2 for each of the sub-systems within a “system” grouping. For example, the Engineered Safety Features System Group contains tables specific to Engineered Safeguards system and Reactor Cavity Flood System.

Table 2 consists of the following nine columns:

- Component Type
- Intended Function
- Material
- Environment
- Aging Effect Requiring Management
- Aging Management Programs
- NUREG-1801 Volume 2 Item
- Table 1 Item
- Notes

Component Type

The first column identifies all of the component types from Section 2 of the LRA that are subject to aging management review. They are typically listed in alphabetical order.

Intended Function

The second column contains the license renewal intended functions (including abbreviations where applicable) for the listed component types. Definitions and abbreviations of intended functions are contained within the Intended Functions table of LRA Section 2.

Material

The third column lists the particular materials of construction for the component type.

Environment

The fourth column lists the environment to which the component types are exposed. Internal and external service environments are indicated and a list of these environments is provided in Table 3.0-1, Service Environments.

Aging Effect Requiring Management

As part of the aging management review process, the applicant determines any aging effects requiring management for the material and environment combination in order to maintain the intended function of the component type. These aging effects requiring management are listed in column five.

Aging Management Programs

The aging management programs used to manage the aging effects requiring management are listed in column six of Table 2.

In many cases, a particular Palisades Aging Management Program will implement more than one program defined in the GALL. To assist with NRC reviews, and to simplify the correlation with the associated table in GALL Volume 2, the Programs column will typically contain the GALL program number, and then, in parentheses, the title of the Palisades AMP that contains the GALL program.

NUREG-1801 Vol. 2 Item

Each combination of component type, material, environment, aging effect requiring management, and aging management program that is listed in Table 2, is compared to NUREG-1801, Volume 2, with consideration given to the standard notes, to identify consistencies. When they are identified, they are documented by noting the appropriate NUREG-1801, Volume 2 item number in column seven of Table 2. If there is no corresponding item number in NUREG-1801, Volume 2, this row in column seven is left blank. That way, a reviewer can readily identify where there is correspondence between the plant specific tables and the NUREG-1801, Volume 2 tables.

Table 1 Item

Each combination of component, material, environment, aging effect requiring management, and aging management program that has an identified NUREG-1801 Volume 2 item number must also have a Table 3.x.1 line item reference number. The corresponding line item from Table 1 (Figure 3.0-1) is listed in column eight of Table 2. If there is no corresponding item in NUREG-1801, Volume 1, this row in column eight is left blank. That way, the information from the two tables can be correlated.

Notes

In order to realize the full benefit of NUREG-1801, each applicant needs to identify how the information in Table 2 aligns with the information in NUREG-1801, Volume 2. This is accomplished through a series of notes. All note references with letters are standard notes that will be the same from application to application throughout the industry. Any notes the plant requires, which are in addition to the standard notes, will be identified by a number and deemed plant specific.

Table Usage

Table 1

The reviewer evaluates each row in Table 1 (Figure 3.0-1) by moving from left to right across the table. Since the Component, Aging Effect/Mechanism, Aging Management Programs and Further Evaluation Recommended information is taken directly from NUREG-1801, Volume 1, no further analysis of those columns is required. The information intended to help the reviewer the most in this table is contained within the Discussion column. Here the reviewer will be given information necessary to determine, in summary, how the applicant's evaluations and programs align with NUREG-1801, Volume 1. This may be in the form of descriptive information within the Discussion column or the reviewer may be referred to other locations within the LRA for further information (including hyperlinks where possible/practical).

Table 2

The Table 2 (Figure 3.0-2) contains all of the Aging Management Review information for each system, structure, or commodity, whether or not it aligns with NUREG-1801. For a given row within the table, the reviewer is able to see the intended function, material, environment, aging effect requiring management and aging management program combination for a particular component type within that system, structure, or commodity. In addition, if there is a correlation between the combination in Table 2 and a combination in NUREG-1801, Volume 2, this will be identified by a referenced item number in column seven, NUREG-1801, Volume 2 Item. The reviewer can refer to the item number in NUREG-1801, Volume 2, if desired, to verify the correlation. If the column is blank, the applicant was unable to locate an appropriately corresponding combination in NUREG-1801, Volume 2.

As the reviewer continues across the table from left to right, within a given row, the next column is labeled Table 1 Item. If there is a reference number in this column, the reviewer is able to use that reference number to locate the corresponding row in Table 1 and see how the aging management program for this particular combination aligns with NUREG-1801. Program details may be found in Appendix B. There may be a hyperlink directly to the corresponding row in Table 1 as well.

Table 2 provides the reviewer with a means to navigate from the components subject to Aging Management Review (AMR) in LRA Section 2, all the way through the evaluation of the programs that will be used to manage the effects of aging of those components.

A listing of the abbreviations used in this section is provided in Section 1.5.

Table 3.0-1 Service Environments

Category	Description
Air/Gas	The air/gas environment includes dry, filtered instrument air, nitrogen, and other vendor-supplied gases that may be used for analysis or calibration. Air conditions may include humidity, condensation, and contaminants.
Atmosphere/ Weather	The atmosphere/weather conditions that may affect aging of systems, structures, or components (SSC) include temperature, precipitation, and ultraviolet radiation. The December 1977 through November 1978 recorded Palisades atmosphere/weather environment is bounded by temperatures of -2.9°F to 88.3°F. The US Weather Bureau has reports of extremes for areas near Palisades of -25°F and 95°F. Periodic wetting and ultraviolet radiation must be considered for SSC exposed to atmosphere/weather. The atmosphere/weather environment includes atmospheric air outside covered structures and includes moist laden air, precipitation, and wind.
Borated Water Leakage	Systems, structures, and components may be located in the vicinity of primary systems containing boric acid solution that could leak, creating a boric acid leakage environment, and leading to boric acid corrosion. Borated water systems and the boric acid leakage may exist within the Auxiliary Building, the Auxiliary Building Addition, and the Containment. The borated water leakage is an external environment that includes all plant areas that contain borated water systems that leak on nearby components or structures.

Table 3.0-1 Service Environments

Category	Description
Containment Air	Temperature - The Containment air environment, outside the reactor cavity, is bounded during normal operation by temperatures of 50°F to 144°F. Summertime temperatures inside the Pressurizer Shed typically approach 140°F, and around the PCS hot and cold leg piping they approach 130°F. Even the cooler areas in containment, down on the 590' elevation have typical summer temperatures up to around 110°F. EQ Design Basis temperatures approach 136°F for 590' - 610' elevations and 144°F for 611' - 670' elevations. Thermal hot spots may exist within the Containment and should be specified in individual aging management review reports. For electrical commodities the average temperature for elevations 611' - 702' is 123 °F and for elevations 590' - 610' it is 96.2 °F. Relative Humidity - Normal relative humidity is approximately 50% but for aging management is assumed at 100%. Radiation - Plant radiation doses outside the reactor cavity are not of concern for aging management. Materials can be affected by cumulative radiation exposure. For concrete, neutron fluence above 10^{19} n/cm^2 (>1 Mev) or gamma dose >10^{10} rads is required to cause degradation. For metals, neutron fluence above 10^{17} n/cm^2 (>1 Mev) is required to cause degradation. For electrical commodities in containment the 60 year bounding value for radiation is $3X10^7$ rads.
Electrical Environment Evaluation Parameters	Each of the three significant stressors for insulated cables and connections could lead to a loss of material properties of the insulating material of cables and connections when exposed to environment more severe than their respective 60-year service-limiting values for temperature and radiation. Bounding electrical stressor conditions have been defined for temperature, radiation and moisture.
Oil	This category encompasses lubricating oil, diesel fuel oil, or hydraulic oil. Palisades Systems having this internal environment include, but not limited to, the Emergency Diesel Generator (EDG) Fuel Oil System, and the EDG Lube Oil System.

Table 3.0-1 Service Environments

Category	Description
Plant Indoor Air	Plant indoor air includes all heated, ventilated, or air conditioned plant areas sheltered from outside conditions. Temperature - A temperature range of 50°F to 120°F and relative humidity of up to100% bound plant indoor air conditions. For example, in the Component Cooling Water room el. 607', the average temperature is 110 °F, and for Component Cooling Water room el. 625', the average temperature is 120 °F. All other areas are less than or equal to an average temperature of 104 °F. Relative Humidity - Normal relative humidity is approximately 50% but for aging management is assumed at 100%. Radiation - Plant radiation doses outside the Containment are not of concern for aging management. For commodities outside containment the 60 year bounding radiation dose is $<7.5X10^7$ rads.
Raw Water	Raw water is defined as water that enters the plant from Lake Michigan, pond, or rain/ground water source that has not been demineralized or chemically treated to any significant extent. In general, the water is rough filtered to remove large particles. Biocides may be added to control micro-organisms or macro-organisms. Another designation of raw water is water that leaks from any system. The Raw Water category at Palisades constitutes the lake water used for the main condensers, intake cooling water, and fire protection.
Soil (Buried)	Portions of systems, structures, and components (SSC) such as foundations, buried piping, buried tanks, buried valves, buried cable, etc. may be exposed to soil/earth. These SSCs are exposed to soil/earth, fill, and ground water/runoff. Depending on the particular site groundwater table levels, buried components may also be exposed to the chemicals in the groundwater. The groundwater chemistry, which may be acidic or contain chlorides and sulfates, plays a major role in the determination of the degradation of below grade components. Examples of buried components are buried fuel oil tanks, the exterior surface of a foundation or basemat, those portions of a building's exterior walls below grade, buried concrete such as duct banks or concrete trenches. If the buried component is exposed to flowing groundwater or other aqueous liquids, then component surfaces exposed to raw water environment were considered.

Table 3.0-1 Service Environments

Category	Description
Submerged (Structural)	Components and piping that are exposed to raw water can be submerged in water or exposed to flowing water. The types of water may be raw water (lake or pond water), groundwater, or even treated water (e.g., wet cooling towers). The chemicals in the water source may be acidic or contain chlorides and sulfates and play a major role in the determination of the degradation of concrete. Examples of concrete exposed to raw water are the submerged potion of a water control structure (intake or pump structure) exterior and interior walls, intake system, and mechanical wet cooling towers.
Treated Water and/or Steam	Demineralized water or chemically purified water that is the source for water that may require further processing, such as for the primary or secondary coolant system. Treated water can be de-aerated, can include corrosion inhibitors, biocides, or boric acid, or can include a combination of treatments. Steam generated from treated water is included in this environment category.

Figure 3.0-1 Table 1 -
Table 3.x.1, Summary of Aging Management Evaluations in Chapter _____ of NUREG-1801 for ___________

Item Number	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.x.1-01					
3.x.1-02					
3.x.1-03					
3.x.1-04					
3.x.1-05					
3.x.1-06					

Figure 3.0-2 Table 2 -
Table 3.x.2-y, Section 3 Title - Plant Specific System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes

3.1 Aging Management of Reactor Coolant System

3.1.1 Introduction

This section provides the results of the aging management review for those components identified in Section 2.3.1, Reactor Vessel, Internals, and Reactor Coolant System, as being subject to aging management review. The systems, or portions of systems, which are addressed in this section, are described in the indicated sections.

- Primary Coolant System (Section 2.3.1.1)
- Reactor Vessel (Section 2.3.1.2)
- Reactor Vessel Internals (Section 2.3.1.3)
- Replacement Steam Generators (Section 2.3.1.4)

Table 3.1.1, Summary of Aging Management Evaluations in Chapter IV of NUREG-1801 for Reactor Coolant System, provides the summary of the programs evaluated in NUREG-1801 for the Reactor Coolant System component groups that are relied on for license renewal.

This table uses the format described in Figure 3.0-1 above. Note that this table only includes those component groups that are applicable to a PWR.

3.1.2 Results

The following tables summarize the results of the aging management review for systems in the Reactor Vessel, Reactor Internals, and Reactor Coolant System group:

Table 3.1.2-1, Reactor Coolant System - Primary Coolant System - Summary of Aging Management Evaluation

Table 3.1.2-2, Reactor Coolant System - Reactor Vessel - Summary of Aging Management Evaluation

Table 3.1.2-3, Reactor Coolant System - Reactor Vessel Internals - Summary of Aging Management Evaluation

Table 3.1.2-4, Reactor Coolant System - Replacement Steam Generators - Summary of Aging Management Evaluation

The materials that specific components are fabricated from, the environments to which components are exposed, the potential aging effects requiring management, and the aging management programs used to manage these aging effects are provided for each of the above systems in the following subsections of Section 3.1.2.1, Materials, Environment, Aging Effects Requiring Management and Aging Management Programs:

Section 3.1.2.1.1, Primary Coolant System

Section 3.1.2.1.2, Reactor Vessel

Section 3.1.2.1.3, Reactor Vessel Internals

Section 3.1.2.1.4, Replacement Steam Generators

3.1.2.1 Materials, Environment, Aging Effects Requiring Management and Aging Management Programs

3.1.2.1.1 Primary Coolant System

Materials

The materials of construction for the Primary Coolant System are:

- Alloy 600
- Carbon Steel
- Cast Austenitic SS
- Low Alloy Steel
- Stainless Steel
- Epoxy Coated Carbon Steel

Environment

The Primary Coolant System is exposed to the following environments:

- Containment Air (Ext)
- Plant Indoor Air (Ext)
- Treated Water (Ext)
- Treated Water (Int)

Aging Effects Requiring Management

The following aging effects, associated with the Primary Coolant System, require management:

- Cracking
- Loss of Material
- Loss of Preload
- Reduction of Fracture Toughness

Aging Management Programs

The following aging management programs manage the aging effects for the Primary Coolant System:

- Alloy 600 Program
- ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program
- Bolting Integrity Program
- Boric Acid Corrosion Program
- Closed Cycle Cooling Water Program
- One-Time Inspection Program
- System Monitoring Program
- Water Chemistry Program

3.1.2.1.2 Reactor Vessel

Materials

The materials of construction for the Reactor Vessel are:

- Carbon Steel
- Nickel-Based Alloys
- Stainless Steel

Environment

The Reactor Vessel components are exposed to the following environments:

- Containment Air (Ext)
- Treated Water (Ext)
- Treated Water (Int)

Aging Effects Requiring Management

The following aging effects, associated with the Reactor Vessel, require management:

- Crack Initiation and Growth
- Cracking
- Loss of Material
- Reduction in Fracture Toughness
- Loss of Preload

Aging Management Programs

The following aging management programs manage the aging effects for the Reactor Vessel components:

- Alloy 600 Program
- ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program
- Bolting Integrity Program
- Boric Acid Corrosion Program
- Reactor Vessel Integrity Surveillance Program
- Reactor Vessel Internals Inspection Program
- Water Chemistry Program

3.1.2.1.3 Reactor Vessel Internals

Materials

The materials of construction for the Reactor Vessel Internals are:

- Stainless Steel

Environment

The Reactor Vessel Internals components are exposed to the following environments:

- Treated Water (Ext)

Aging Effects Requiring Management

The following aging effects, associated with the Reactor Vessel Internals, require management:

- Changes in Dimensions
- Cracking
- Loss of Material
- Loss of Preload
- Reduction in Fracture Toughness

Aging Management Programs

The following aging management programs manage the aging effects for the Reactor Vessel Internals components:

- ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program

- Reactor Vessel Internals Inspection Program
- Water Chemistry Program

3.1.2.1.4 Replacement Steam Generators

Materials

The materials of construction for the Replacement Steam Generators are:

- Alloy 600/690
- Carbon Steel
- Carbon Steel w/SS Cladding
- Low-Alloy Steel
- Nickel Based Alloy
- Stainless Steel

Environment

The Replacement Steam Generators components are exposed to the following environments:

- Containment Air (Ext)
- Treated Water (Int)
- Treated Water (Int)

Aging Effects Requiring Management

The following aging effects, associated with the Replacement Steam Generators, require management:

- Cracking
- Crack Initiation and Growth
- Loss of Material
- Loss of Preload

Aging Management Programs

The following aging management programs manage the aging effects for the Replacement Steam Generators components:

- Alloy 600 Program
- ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program
- Bolting Integrity Program

- Boric Acid Corrosion Program
- Flow Accelerated Corrosion Program
- Steam Generator Tube Integrity Program
- Water Chemistry Program

3.1.2.2 Further Evaluation of Aging Management as Recommended by NUREG-1801

NUREG-1801 Volume 1 Tables provide the basis for identifying those programs that warrant further evaluation in the license renewal application. For the Reactor Vessel, Internals, and Reactor Coolant System group, those programs are addressed in the following sections.

3.1.2.2.1 Cumulative Fatigue Damage

NUREG-1800 states that cracking due to fatigue is an aging effect applicable to primary coolant system items subject to aging management review. Fatigue evaluations are TLAAs since they are based on design transients (cyclic loadings) defined for the life of the plant.

Fatigue evaluations were performed in the design of the Palisades Class 1 Primary Coolant System components in accordance with the requirements specified in applicable design codes. Fatigue is a TLAA as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c). The evaluation of this TLAA is addressed separately in Section 4.3.

3.1.2.2.2 Loss of Material Due to Pitting and Crevice Corrosion (PWR/BWR)

3.1.2.2.2.1 Loss of Material due to Pitting and Crevice Corrosion in Steam Generator Shell Assembly

NUREG-1800 states that Loss of material due to pitting and crevice corrosion could occur in the steam generator shell assembly. The existing program relies on control of chemistry to mitigate corrosion and ISI to detect loss of material. The extent and schedule of the existing steam generator inspections are designed to ensure that flaws cannot attain a depth sufficient to threaten the integrity of the welds. However, according to NRC Information Notice (IN) 90-04, if general corrosion pitting of the shell exists, the program may not be sufficient to detect pitting and corrosion. NUREG-1801 recommends augmented inspection to manage this aging effect.

NUREG-1801 refers to NRC Information Notice (IN) 90-04 and recommends augmented inspection to manage pitting and crevice corrosion. IN 90-04 states that if general corrosion pitting of the steam generator shell is known to exist the requirements of ASME Section XI may not be sufficient to differentiate isolated cracks for inherent geometric conditions.

The concerns of IN 90-04 are not applicable to Palisades since the steam generators were replaced in 1990 and pitting corrosion of the steam generator shell is not known to currently exist. Therefore, Palisades credits the water chemistry control program and the inservice inspection program for managing loss of material due to pitting and crevice corrosion on the internal surfaces of the steam generator shell, and recommends using the Steam Generator Tube Integrity Program to manage pitting and crevice corrosion. The Steam Generator Tube Integrity Program incorporates the guidance of NEI 97-06 “Steam Generator Program Guidelines” (January 2001) to verify the integrity of the secondary-side internal surfaces of the steam generators. The Program manages the aging effects of accessible SG secondary side internal components through a balance of mitigation, inspection, evaluation, repair, and leakage monitoring measures.

Therefore, a combination of the Water Chemistry Program , ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program , and the Steam Generator Tube Integrity Program are used to manage this aging effect.

3.1.2.2.2.2 Loss of Material due to Pitting and Crevice Corrosion

Applicable to BWR Only

3.1.2.2.3 Loss of Fracture Toughness due to Neutron Irradiation Embrittlement BWR/PWR)

3.1.2.2.3.1 Loss of Fracture Toughness due to Neutron Irradiation Embrittlement - TLAA Evaluation

NUREG-1800 states that certain aspects of neutron irradiation embrittlement are TLAAs as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c)(1).

Neutron irradiation embrittlement is a TLAA to be evaluated for the period of license renewal for ferritic materials that have a neutron fluence of greater than 1.0E17 n/cm^2 (E >1 MeV) at the end of the license renewal term. The beltline region, as described in Section 3.1.2.1.1, was verified to be the limiting region in evaluating loss of fracture toughness due to neutron irradiation embrittlement. This TLAA is discussed in Section 4.2

3.1.2.2.3.2 **Loss of Fracture Toughness due to Neutron Irradiation Embrittlement - Reactor Vessel Materials Surveillance Program**

NUREG-1800 states that loss of fracture toughness in reactor vessel beltline shell and weld materials due to neutron irradiation embrittlement has been identified as an aging effect requiring management during the period of extended operation. A reactor vessel materials surveillance program monitors neutron irradiation embrittlement of the reactor vessel. Reactor vessel surveillance programs are plant specific, depending on matters such as the composition of limiting materials, availability of surveillance capsules, and projected fluence levels. In accordance with 10 CFR Part 50, Appendix H, an applicant is required to submit its proposed withdrawal schedule for approval prior to implementation.

At Palisades, loss of fracture toughness in reactor vessel beltline materials due to neutron irradiation embrittlement is an aging effect requiring management during the period of extended operation. The limiting beltline materials in the Palisades reactor vessel are the plates and welds as indicated in Table 3.1.2-2 and Section 4.2. See Section 4.2 for additional information.

Reactor vessel upper shell and nozzles are not subject to significant neutron irradiation exposure because of their physical distance from the reactor core. For beltline materials, the Reactor Vessel Integrity Surveillance Program, in conjunction with TLAA analyses, effectively manages loss of fracture toughness. The Palisades Reactor Vessel Integrity Surveillance Program provides adequate material property and neutron dosimetry data to predict fracture toughness in beltline materials at the end of the period of extended operation.

3.1.2.2.3.3 **Loss of Fracture Toughness due to Neutron Irradiation Embrittlement**

NUREG-1800 states that loss of fracture toughness due to neutron irradiation embrittlement and void swelling could occur in Westinghouse and B&W baffle/former bolts. NUREG-1801 recommends further evaluation to ensure that this aging effect is adequately managed.

This issue is not applicable. Palisades is a Combustion Engineering PWR and does not have a baffle/former bolt configuration.

3.1.2.2.4 Crack Initiation and Growth due to Thermal and Mechanical Loading or Stress Corrosion Cracking (BWR/PWR)

3.1.2.2.4.1 Crack Initiation and Growth due to Thermal and Mechanical Loading or Stress Corrosion Cracking of Small-Bore Piping <4" NPS

NUREG-1800 states that crack initiation and growth due to thermal and mechanical loading or SCC (including intergranular stress corrosion cracking (IGSCC) could occur in small-bore reactor coolant system and connected system piping less than NPS 4". The existing program relies on ASME Section XI ISI and on control of water chemistry to mitigate SCC. NUREG-1801 recommends that a plant-specific destructive examination or a nondestructive examination (NDE) that permits inspection of the inside surfaces of the piping be conducted to ensure that cracking has not occurred and the component intended function will be maintained during the extended period. The AMPs should be augmented by verifying that service induced weld cracking is not occurring in the small-bore piping less than NPS 4, including pipe, fittings, and branch connections.

At Palisades, crack initiation and growth due to SCC was identified as an aging effect/mechanism requiring management in small-bore (<NPS 4) primary coolant system piping and branch lines. Aging management of service-induced cracking will be accomplished by a combination of the Water Chemistry Program and the ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program. Inspections of a sample of small bore PCS piping will be performed.

The proposed combination of the Water Chemistry Program , and the ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program , provide an effective means of managing service-induced cracking in small-bore primary coolant system piping and connected branch lines during the period of extended operation.

3.1.2.2.4.2 Crack Initiation and Growth due to Thermal and Mechanical Loading or Stress Corrosion Cracking (BWR Only)

Applicable to BWR Only

3.1.2.2.4.3 Crack Initiation and Growth due to Thermal and Mechanical Loading or Stress Corrosion Cracking (BWR Only)

Applicable to BWR Only

3.1.2.2.5 Crack Growth due to Cyclic Loading

NUREG-1800 states that crack growth due cyclic loading could occur in reactor vessel shell and reactor coolant system piping and fittings. Growth of intergranular separations (underclad cracks) in low-alloy or carbon steel heat affected zone under austenitic stainless steel cladding is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation for all the SA 508-Cl 2 forgings where the cladding was deposited with a high heat input welding process.

The Palisades reactor vessel items fabricated from SA-508 Class 2 material include the primary nozzles, reactor vessel flange, and the closure head flange. NUREG-1801 identifies this aging effect as applicable to reactor vessel items fabricated from SA-508, Class 2 materials exposed to a neutron fluence > 1.0E17 n/cm^2. The fluence at the end of the period of extended operation at the bottom of the nozzle to shell welds (highest fluence received by an SA-508 Class 2 Palisades item) has been determined to be 6 orders of magnitude less than 1.0E19 n/cm^2, approximately 1.0E13 n/cm^2. In addition, controls on the cladding chemical composition and processes during fabrication of the Palisades reactor vessel reduced the potential for cracking of the vessel cladding. There have been no cases of underclad cracking in any clad Combustion Engineering reactor vessel subcomponents. Therefore, this aging effect does not require management for the period of extended operation for Palisades.

3.1.2.2.6 Changes in Dimension due to Void Swelling

NUREG-1800 states that changes in dimension due to void swelling could occur in reactor internal components. NUREG-1801 recommends further evaluation to ensure that this aging effect is adequately managed. The reactor vessel internals receive a visual inspection (VT-3) according to Category B-N-3 of Subsection IXB, ASME Section XI. This inspection is not sufficient to detect the effects of changes in dimension due to void swelling. GALL recommends that a plant specific aging management program should be evaluated. The applicant provides a plant specific AMP or participates in industry programs to investigate aging effects and determine appropriate AMP. Otherwise, the applicant provides the basis for concluding that void swelling is not an issue for the component. The applicant should either provide the basis for concluding that void swelling is not an issue for the component or provide a program to manage the effects of changes in dimension due to void swelling and the loss of ductility associated with swelling.

Industry activities are underway to determine whether changes in dimension due to void swelling is an aging effect requiring management for license renewal. Palisades continues to participate in industry investigations of aging effects applicable to reactor vessel internals, as well as initiatives to develop and qualify methods for detection and management. Palisades will incorporate applicable results of industry initiatives related to void swelling in the Reactor Vessel Internals Inspection Program .

3.1.2.2.7 Crack Initiation and Growth due to Stress Corrosion Cracking or Primary Water Stress Corrosion Cracking (PWR)

3.1.2.2.7.1 Crack Initiation and Growth due to Stress Corrosion Cracking or Primary Water Stress Corrosion Cracking (PWR Components)

NUREG-1800 states that crack initiation and growth due to SCC and primary water stress corrosion cracking (PWSCC) could occur in PWR core support pads (or core guide lugs), instrument tubes (bottom head penetrations), pressurizer spray heads, and nozzles for the steam generator instruments and drains. NUREG-1801 recommends further evaluation to ensure that these aging effects are adequately managed. NUREG-1801 recommends that a plant specific aging management program be evaluated because existing programs may not be capable of mitigating or detecting crack initiation and growth due to SCC.

At Palisades, this grouping includes the surge nozzle thermal sleeve, safety injection nozzle thermal sleeve, charging inlet nozzle thermal sleeve, RTD nozzles, pressure measurement nozzle, sampling nozzle, and partial nozzle replacement. Reactor vessel items included in this grouping are the lower shell and bottom head cladding, surveillance capsule holders, core stabilizing lugs, core stop and support lugs, and the flow baffle and skirt., the reactor head O-ring, leakoff tubing and valves. Steam generator items included in this grouping are the tube plate cladding, channel head divider plate, and primary nozzle closure rings. Refer to item 3.1.1-44 of Table 3.1.1 for primary side steam generator items. EPRI Material Reliability Program (MRP) in conjunction with the PWR owners groups is developing a strategic plan to manage and mitigate cracking of nickel based alloy items. The guidance developed by the MRP will be used to identify critical locations for inspection and to augment existing ISI inspections at Palisades, as appropriate. The results of the strategic plan will be incorporated into the

Alloy 600 Program and, as applicable, the Water Chemistry Program and the ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program .

3.1.2.2.7.2 **Crack Initiation and Growth due to Stress Corrosion Cracking or Primary Water Stress Corrosion Cracking (CASS Components)**

NUREG-1800 states that crack initiation and growth due to SCC could occur in PWR cast austenitic stainless steel (CASS) reactor coolant system piping and fittings and pressurizer surge line nozzle. NUREG-1801 recommends further evaluation of piping that does not meet either the reactor water chemistry guidelines of TR-105714 or material guidelines of NUREG-0313.

At Palisades, the Primary Coolant Pump casing and the valve bodies of PORV isolation valves are fabricated of cast austenitic stainless steel (CASS). The Palisades Water Chemistry Program and the ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program are used to manage this aging effect. The Water Chemistry Program meets the EPRI reactor water chemistry guideline TR-105714. As discussed in the respective sections of Appendix B, each of these programs provides reasonable assurance that the aging effect will be managed such that the SSCs within the scope of each program will continue to perform their intended functions consistent with the current licensing bases for the period of extended operation.

3.1.2.2.7.3 **Crack Initiation and Growth due to Stress Corrosion Cracking or Primary Water Stress Corrosion Cracking (Ni Alloys)**

NUREG-1800 states that crack initiation and growth due to PWSCC could occur in PWR pressurizer instrumentation penetrations and heater sheaths and sleeves made of Ni alloys. The existing program relies on ASME Section XI ISI and on control of water chemistry to mitigate PWSCC. However, the existing program should be augmented to manage the effects of SCC on the intended function of Ni-alloy components. NUREG-1801 recommends that the applicant provide a plant-specific AMP or participate in industry programs to determine appropriate AMP for PWSCC of Inconel 182 weld.

At Palisades, nickel based alloy material is identified for the pressurizer instrumentation nozzles, heater sheaths and sleeves, and thermal sleeves. Palisades pressurizer components included in this grouping are the instrument nozzles, electric heaters (penetration nozzles and plugs, original heater sheath, heater sleeve, and end plugs). The programs credited for the management of PWSCC of these nickel based alloy items are the Alloy 600 Program and Water Chemistry Program , supplemented by the ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program . As described in Section 3.1.2.2.7.1, above, the Alloy 600 Program includes participation in industry programs to identify critical locations for inspection and augment existing ISI inspections at Palisades where appropriate.

3.1.2.2.8 Crack Initiation and Growth due to Stress Corrosion Cracking or Irradiation-Assisted Stress Corrosion Cracking

NUREG-1800 states that crack initiation and growth due to SCC or IASCC could occur in baffle / former bolts in Westinghouse and B&W reactors. Historically the VT-3 visual examinations have not identified baffle/former bolt cracking because cracking occurs at the juncture of the bolt head and shank, which is not accessible for visual inspection. However, recent UT examinations of the baffle/former bolts at several plants have identified cracking. The industry is currently addressing the issue of baffle bolt cracking in the PWR Materials Reliability Project, Issues Task Group (ITG) activities to determine, develop, and implement the necessary steps and plans to manage the applicable aging effects on a plant-specific basis. NUREG-1801 recommends further evaluation to ensure that these aging effects are adequately managed.

The Palisades reactor vessel internals do not include baffle / former bolts. The discussion in this paragraph of NUREG-1800 is not applicable to Palisades.

3.1.2.2.9 Loss of Preload due to Stress Relaxation

NUREG-1800 states that loss of preload due to stress relaxation could occur in baffle/former bolts in Westinghouse and B&W reactors. Visual inspection (VT-3) should be augmented to detect relevant conditions of stress relaxation because only the heads of the baffle/former bolts are visible, and a plant specific aging management program is thus required. NUREG-1801 recommends a plant specific aging management program to ensure that these aging effects are adequately managed. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of the standard review plan).

The Palisades reactor vessel internals do not include baffle / former bolts. The discussion in this paragraph of NUREG-1800 is not applicable to Palisades.

3.1.2.2.10 **Loss of Section Thickness due to Erosion**

NUREG-1800 states that loss of section thickness due to erosion could occur in steam generator feedwater impingement plates and supports. NUREG-1801 recommends further evaluation of a plant-specific aging management program to ensure that this aging effect is adequately managed.

The Palisades steam generators do not include impingement plates. Therefore, the discussion in this paragraph is not applicable.

3.1.2.2.11 **Crack Initiation and Growth due to PWSCC, ODSCC, or Intergranular Attack or Loss of Material due to Wastage and Pitting Corrosion or Loss of Section Thickness due to Fretting and Wear or Denting due to Corrosion of Carbon Steel Tube Support Plate**

NUREG-1800 states that crack initiation and growth due to PWSCC, ODSCC, or intergranular attack (IGA) or loss of material due to wastage and pitting corrosion or deformation due to corrosion could occur in alloy 600 components of the steam generator tubes, repair sleeves and plugs. All PWR licensees have committed voluntarily to a SG degradation management program described in NEI 97-06. NUREG-1801 recommends that an AMP based on the recommendations of staff-approved NEI 97-06 guidelines, or other alternate regulatory basis for SG degradation management, should be developed to ensure that this aging effect is adequately managed.

At Palisades, crack initiation and growth due to PWSCC, SCC, or intergranular attack (IGA) or loss of material due to wastage and pitting corrosion or deformation due to corrosion could occur in nickel based alloy components of the steam generator tube plugs. To manage these aging effects, Palisades credits the Steam Generator Tube Integrity Program supplemented by the Water Chemistry Program and the ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program. The steam generator tube integrity program assessment of tube integrity and plugging or repair criteria of flawed tubes is in accordance with the plant technical specifications and NEI 97-06 guidelines. For general and pitting corrosion, the acceptance criteria are in accordance with NEI 97-06 guidelines.

New, replacement recirculating steam generators were Installed at Palisades in 1990. These new steam generators incorporate many enhancements In design and materials of construction to minimize aging effects. However, cracking due to PWSCC and IGA/IGSCC and loss of material due to pitting and wear could occur in the Palisades steam generator tubes and plugs.

The Water Chemistry Program conforms to the guidelines in EPRI TR-105714 and TR-102134. The Water Chemistry Program mitigates aging effects such as cracking due to PWSCC and IGA/IGSCC and loss of material due to pitting and wear, by controlling the environment to which the steam generator tubes and plugs are exposed. These aging effects are minimized by controlling the chemical species that cause the underlying mechanisms that result in these aging effects. The program provides assurance that an elevated level of contaminants and oxygen does not exist in either the primary or secondary sides of the steam generators, and thus minimizes the occurrences of these aging effects. The Water Chemistry Program has been in effect since initial plant operation and has been effective at maintaining the desired primary and secondary water chemistry and detecting abnormal conditions, which have been corrected in an expedient manner. Therefore, the Water Chemistry Program effectively mitigates cracking due to PWSCC and IGA/IGSCC and loss of material due to pitting and wear in the steam generator tubes and plugs. Verification of the effectiveness of the program will be performed to ensure that these aging effects are not occurring.

The Steam Generator Tube Integrity Program is used to manage these aging effects for the steam generator tubes and plugs in order to confirm the effectiveness of the Water Chemistry Program. The Steam Generator Tube Integrity Program was developed to meet the guidelines of NEI 97-06 “Steam Generator Program Guidelines” (January 2001). The program manages these aging effects through a balance of prevention, inspection, evaluation, repair, and leakage monitoring measures. Eddy Current Testing is used to detect steam generator tube flaws and degradation. Steam generator tubes not meeting the Technical Specification limits for continued operation are removed from service by installation of tube plugs. Tube plugs installed in the steam generators are fabricated from heat-treated Alloy 690 material. Although these plugs have a high resistance to PWSCC, they are routinely inspected as a part of the program. A tube integrity assessment is performed following each steam generator tube inspection to ensure that the performance criteria have been met for the previous operating period and will continue to be met for the next period.

Therefore, a combination of the Water Chemistry Program and the Steam Generator Tube Integrity Program are used to manage these aging effects.

3.1.2.2.12 **Loss of Section Thickness due to Flow-accelerated Corrosion**

NUREG-1800 states that loss of section thickness due to flow-accelerated corrosion could occur in tube support lattice bars made of carbon steel. NUREG-1801 recommends that a plant-specific aging management program be evaluated and, on the basis of the guidelines of NRC Generic Letter 97-06, an inspection program for steam generator internals be developed to ensure that this aging effect is adequately managed.

The Palisades steam generators do not include carbon steel tube support lattice bars. Therefore, loss of section thickness of these bars is not an applicable aging effect for Palisades.

3.1.2.2.13 **Ligament Cracking due to Corrosion**

NUREG-1800 states that ligament cracking due to corrosion could occur in carbon steel components in the steam generator tube support plate. All PWR licensees have committed voluntarily to a SG degradation management program described in NEI 97-06. NUREG-1801 recommends that an AMP based on the recommendations of staff-approved NEI 97-06 guidelines, or other alternate regulatory basis for SG degradation management, be developed to ensure that this aging effect is adequately managed.

The Palisades steam generators have stainless steel eggcrate tube lattice support rings. Therefore, ligament cracking due to corrosion is not an applicable aging effect for Palisades. New, replacement recirculating steam generators were installed at Palisades in 1990. These new steam generators incorporate many enhancements in design and materials of construction, to minimize aging effects.

The Water Chemistry Program conforms to the guidelines In EPRI TR-105714, and TR-102134. The Water Chemistry Program mitigates aging effects such as cracking due to SCC, by controlling the environment to which the steam generator stainless steel tube support plates are exposed. This aging effect is minimized by controlling the chemical species that cause the underlying mechanisms that result in the aging effect. The program provides assurance that an elevated level of contaminants and oxygen does not exist in either the primary or secondary sides of the steam generators, and thus minimizes the occurrences of this aging effect. The Water Chemistry Program has been in effect since initial plant operation and has been effective at maintaining the desired primary and secondary water chemistry and detecting abnormal conditions, which have been corrected in an expedient manner.

Therefore, the Water Chemistry Program effectively mitigates cracking due to SCC in the steam generator stainless steel tube support plates. Verification of the effectiveness of the program will be performed to ensure that this aging effect is not occurring.

The Steam Generator Tube Integrity Program is used to manage this aging effect for the steam generator stainless steel tube support plates in order to confirm the effectiveness of the Water Chemistry Program. The Steam Generator Tube Integrity Program was developed to meet the guidelines of NEI 97-06 “Steam Generator Program Guidelines” (January 2001). The program manages this aging effect through a balance of prevention, inspection, evaluation, repair, and leakage monitoring measures. Periodic visual inspections of accessible areas are performed to verify the integrity of secondary-side components, including the steam generator stainless steel tube support plates.

Therefore, a combination of the Water Chemistry Program and the Steam Generator Tube Integrity Program are used to manage this aging effect.

3.1.2.2.14 **Loss of Material due to Flow-accelerated Corrosion**

NUREG-1800 states that loss of material due to flow-accelerated corrosion could occur in feedwater inlet ring and supports. As noted in Combustion Engineering (CE) Information Notice (IN) 90-04 and NRC IN 91-19 and LER 50-362/90-05-01, this form of degradation has been detected only in certain CE System 80 steam generators. NUREG-1801 recommends further evaluation to ensure that this aging effect is adequately managed. NUREG-1801 recommends that a plant-specific aging management program be evaluated because existing programs may not be capable of mitigating or detecting loss of material due to flow-accelerated corrosion.

The discussion in this paragraph of NUREG-1800 is applicable to CE System 80 steam generators only, whereas Palisades has Combustion Engineering Model 2530 steam generators.

3.1.2.2.15 **Quality Assurance for Aging Management of Non-Safety Related Components**

Quality Assurance Program applicability to non-safety-related components is addressed in Appendix B, Section 1.2.

3.1.2.3 Time-Limited Aging Analysis

The time-limited aging analyses (TLAA) identified below are associated with the Reactor Vessel, Internals, and Reactor Coolant System components. The section of the LRA that contains the TLAA review results is indicated in parenthesis.

- Reactor Vessel Neutron Embrittlement (Section 4.2)
- Metal Fatigue (Section 4.3)
- Alloy 600 Nozzle and Safe End Life Assessment Analyses (Section 4.7.2)
- ASME Code Case N-481 Relaxation of The Primary Coolant Pump Weld Category B-L-1 Inspection Interval from 10 Years to 40 Years (Section 4.7.3)

3.1.3 Conclusion

The Reactor Vessel, Internals, and Reactor Coolant System piping, fittings, and components that are subject to aging management review have been identified in accordance with the requirements of 10 CFR 54.4. The aging management programs selected to manage aging effects for the Reactor Vessel, Internals, and Reactor Coolant System components are identified Section 3.1.2.1.

A description of these aging management programs is provided in Appendix B, along with the demonstration that the identified aging effects will be managed for the period of extended operation.

Therefore, based on the conclusions provided in Appendix B, the effects of aging associated with the Reactor Vessel, Internals, and Reactor Coolant System components will be adequately managed so that there is reasonable assurance that the intended function(s) will be maintained consistent with the current licensing basis during the period of extended operation.

Table 3.1.1 Summary of Aging Management Evaluations in Chapter IV of NUREG-1801 for Reactor Coolant System

Item Number	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.1.1-01	Reactor coolant pressure boundary components	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA (see [SRP] subsection 3.1.2.2.1)	Further evaluation is documented in Section 3.1.2.2.1. The aging mechanism fatigue is potentially applicable to many component types in the Reactor Coolant System Supergroup, but only selected components or locations require explicit analysis as TLAAs and/or warrant aging management. The Palisades approach to identifying and managing the relevant locations and components is addressed in Section 4.3 and in Appendix B, Fatigue Monitoring Program. Therefore, cumulative fatigue damage is not identified as an aging effect in Tables 3.1.2-1 through 3.1.2-4.below.
3.1.1-02	Steam generator shell assembly	Loss of material due to pitting and crevice corrosion	Inservice inspection; water chemistry	Yes, detection of aging effects is to be further evaluated (see [SRP] subsection 3.1.2.2.2.1)	Further evaluation documented in Section 3.1.2.2.2.1.
3.1.1-03	BWR only				
3.1.1-04	Pressure vessel ferritic materials that have a neutron fluence greater than 10^{17} n/cm^2 (E>1 MeV)	Loss of fracture toughness due to neutron irradiation embrittlement	TLAA, evaluated in accordance with Appendix G of 10 CFR 50 and RG 1.99	Yes, TLAA (see [SRP] subsection 3.1.2.2.3.1)	Further evaluation documented in Section 3.1.2.2.3.1.
3.1.1-05	Reactor vessel beltline shell and welds	Loss of fracture toughness due to neutron irradiation embrittlement	Reactor vessel surveillance	Yes, plant specific (see [SRP] subsection 3.1.2.2.3.2)	Further evaluation documented in Section 3.1.2.2.3.2.

Table 3.1.1 Summary of Aging Management Evaluations in Chapter IV of NUREG-1801 for Reactor Coolant System

Item Number	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.1.1-06	Westinghouse and B&W baffle/former bolts	Loss of fracture toughness due to neutron irradiation embrittlement and void swelling	Plant specific	Yes, plant specific (see [SRP] subsection 3.1.2.2.3.3)	Further evaluation documented in Section 3.1.2.2.3.3.
3.1.1-07	Small-bore reactor coolant system and connected systems piping	Crack initiation and growth due to SCC, intergranular SCC, and thermal and mechanical loading	Inservice inspection; water chemistry; one-time inspection	Yes, parameters monitored/ inspected and detection of aging effects are to be further evaluated (see [SRP] subsection 3.1.2.2.4.1)	Further evaluation documented in Section 3.1.2.2.4.1.
3.1.1-08	BWR only				
3.1.1-09	BWR only				
3.1.1-10	Vessel shell	Crack growth due to cyclic loading	TLAA	Yes, TLAA (see [SRP] subsection 3.1.2.2.5)	Further evaluation documented in Section 3.1.2.2.5.
3.1.1-11	Reactor internals	Changes in dimension due to void swelling	Plant specific	Yes, plant specific (see [SRP] subsection 3.1.2.2.6)	Further evaluation documented in Section 3.1.2.2.6.
3.1.1-12	PWR core support pads, instrument tubes (bottom head penetrations), pressurizer spray heads, and nozzles for the steam generator instruments and drains	Crack initiation and growth due to SCC and/or primary water stress corrosion cracking (PWSCC)	Plant specific	Yes, plant specific (see [SRP] subsection 3.1.2.2.7.1)	Further evaluation documented in Section 3.1.2.2.7.1.

Table 3.1.1 Summary of Aging Management Evaluations in Chapter IV of NUREG-1801 for Reactor Coolant System

Item Number	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.1.1-13	Cast austenitic stainless steel (CASS) reactor coolant system piping	Crack initiation and growth due to SCC	Plant specific	Yes, plant specific (see [SRP] subsection 3.1.2.2.7.2)	Further evaluation documented in Section 3.1.2.2.7.2.
3.1.1-14	Pressurizer instrumentation penetrations and heater sheaths and sleeves made of Ni-alloys	Crack initiation and growth due to PWSCC	Inservice inspection; water chemistry	Yes, AMP for PWSCC of Inconel 182 weld is to be evaluated (see [SRP] subsection 3.1.2.2.7.3)	Further evaluation documented in Section 3.1.2.2.7.3.
3.1.1-15	Westinghouse and B&W baffle former bolts	Crack initiation and growth due to SCC and IASCC	Plant specific	Yes, plant specific (see [SRP] subsection 3.1.2.2.8)	Further evaluation documented in Section 3.1.2.2.8.
3.1.1-16	Westinghouse and B&W baffle former bolts	Loss of preload due to stress relaxation	Plant specific	Yes, plant specific (see [SRP] subsection 3.1.2.2.9)	Further evaluation documented in Section 3.1.2.2.9.
3.1.1-17	Steam generator feedwater impingement plate and support	Loss of section thickness due to erosion	Plant specific	Yes, plant specific (see [SRP] subsection 3.1.2.2.10)	Further evaluation documented in Section 3.1.2.2.10.

Table 3.1.1 Summary of Aging Management Evaluations in Chapter IV of NUREG-1801 for Reactor Coolant System

Item Number	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.1.1-18	(Alloy 600) Steam generator tubes, repair sleeves, and plugs	Crack initiation and growth due to PWSCC, outside diameter stress corrosion cracking (ODSCC), and/or intergranular attack (IGA) or loss of material due to wastage and pitting corrosion, and fretting and wear: or deformation due to corrosion at tube support plate intersections	Steam generator tubing integrity; water chemistry	Yes, effectiveness of a proposed AMP is to be evaluated (see [SRP] subsection 3.1.2.2.11)	Further evaluation documented in Section 3.1.2.2.11.
3.1.1-19	Tube support lattice bars made of carbon steel	Loss of section thickness due to FAC	Plant specific	Yes, plant specific (see [SRP] subsection 3.1.2.2.12)	Further evaluation documented in Section 3.1.2.2.12
3.1.1-20	Carbon steel tube support plate	Ligament cracking due to corrosion	Plant specific	Yes, effectiveness of a proposed AMP is to be evaluated (see [SRP] subsection 3.1.2.2.13)	Further evaluation documented in Section 3.1.2.2.13.
3.1.1-21	Steam generator feedwater inlet ring and supports	Loss of material due to flow accelerated corrosion	Combustion Engineering (CE) steam generator feedwater ring inspection	Yes, plant specific	Further evaluation documented in Section 3.1.2.2.14.
3.1.1-22	Reactor vessel closure studs and stud assembly	Crack initiation and growth due to SCC and/or IGSCC	Reactor head closure studs	No	See [Section B2.1.2] ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program for aging management program

Table 3.1.1 Summary of Aging Management Evaluations in Chapter IV of NUREG-1801 for Reactor Coolant System

Item Number	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.1.1-23	CASS pump casing and valve body	Loss of fracture toughness due to thermal aging embrittlement	Inservice inspection	No	See [Section B2.1.2] ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program for aging management program
3.1.1-24	CASS piping	Loss of fracture toughness due to thermal aging embrittlement	Thermal aging embrittlement of CASS	No	Palisades does not have CASS piping in the Primary Coolant System
3.1.1-25	BWR piping and fittings; steam generator components	Wall thinning due to flow-accelerated corrosion	Flow-accelerated corrosion	No	See [Section B2.1.11] Flow Accelerated Corrosion Program for aging management program
3.1.1-26	Reactor coolant pressure boundary (RCPB) valve closure bolting, manway and holding bolting, and closure bolting in high pressure and high temperature systems	Loss of material due to wear; loss of preload due to stress relaxation; crack initiation and growth due to cyclic loading and/or SCC	Bolting integrity	No	Bolting integrity is managed under the Bolting Integrity Program. See [Section B2.1.3]
3.1.1-27	BWR only				
3.1.1-28	BWR only				
3.1.1-29	BWR only				
3.1.1-30	BWR only				
3.1.1-31	BWR only				
3.1.1-32	BWR only				
3.1.1-33	BWR only				
3.1.1-34	BWR only				

Table 3.1.1 Summary of Aging Management Evaluations in Chapter IV of NUREG-1801 for Reactor Coolant System

Item Number	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.1.1-35	CRD nozzle	Crack initiation and growth due to PWSCC	Ni-alloy nozzles and penetrations; water chemistry	No	See [Section B2.1.1] Alloy 600 Program and [Section B2.1.21] Water Chemistry Program for aging management programs
3.1.1-36	Reactor vessel nozzles safe ends and CRD housing; reactor coolant system components (except CASS and bolting)	Crack initiation and growth due to cyclic loading, and/or SCC, and PWSCC	Inservice inspection; water chemistry	No	See [Section B2.1.1] Alloy 600 Program, [Section B2.1.21] Water Chemistry Program, and [Section B2.1.2] ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program for aging management programs
3.1.1-37	Reactor vessel internals CASS components	Loss of fracture toughness due to thermal aging, neutron irradiation embrittlement, and void swelling	Thermal aging and neutron irradiation embrittlement	No	Palisades does not have CASS reactor vessel internals
3.1.1-38	External surfaces of carbon steel components in reactor coolant system pressure boundary	Loss of material due to boric acid corrosion	Boric acid corrosion	No	See [Section B2.1.4] Boric Acid Corrosion Program for aging management program
3.1.1-39	Steam generator secondary manways and handholds (CS)	Loss of material due to erosion	Inservice inspection	No	This NUREG 1801 grouping which addresses erosion concerns in once-through steam generators is not applicable to Palisades
3.1.1-40	Reactor internals, reactor vessel closure studs, and core support pads	Loss of material due to wear	Inservice inspection	No	See [Section B2.1.2] ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program for aging management program
3.1.1-41	Pressurizer integral support	Crack initiation and growth due to cyclic loading	Inservice inspection	No	See [Section B2.1.2] ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program for aging management program

Table 3.1.1 Summary of Aging Management Evaluations in Chapter IV of NUREG-1801 for Reactor Coolant System

Item Number	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.1.1-42	Upper and lower internals assembly (Westinghouse)	Loss of preload due to stress relaxation	Inservice inspection; loose part and/or neutron noise monitoring	No	Not applicable to Palisades. Palisades is a CE unit.
3.1.1-43	Reactor vessel internals in fuel zone region (except Westinghouse and Babcock & Wilcox [B&W] baffle bolts)	Loss of fracture toughness due to neutron irradiation embrittlement, and void swelling	PWR vessel internals; water chemistry	No	See [Section B2.1.17] Reactor Vessel Internals Program for aging management program
3.1.1-44	Steam generator upper and lower heads; tubesheets; primary nozzles and safe ends	Crack initiation and growth due to SCC, PWSCC and IASCC	Inservice inspection; water chemistry	No	See [Section B2.1.2] ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program and [Section B2.1.21] Water Chemistry Program for aging management programs
3.1.1-45	Vessel internals (except Westinghouse and B&W baffle former bolts)	Crack initiation and growth due to SCC and IASCC	PWR vessel internals; water chemistry	No	See [Section B2.1.17] Reactor Vessel Internals Program and [Section B2.1.21] Water Chemistry Program for aging management programs
3.1.1-46	Reactor internals (B&W screws and bolts)	Loss of preload due to stress relaxation	Inservice inspection; loose part monitoring	No	Not applicable to Palisades. Palisades is a CE unit.
3.1.1-47	Reactor vessel closure studs and stud assembly	Loss of material due to wear	Reactor head closure studs	No	See [Section B2.1.2] ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program for aging management program
3.1.1-48	Reactor internals (Westinghouse upper and lower internal assemblies; CE bolts and tie rods)	Loss of preload due to stress relaxation	Inservice inspection; loose part monitoring	No	See [Section B2.1.2] ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program for aging management program

Table 3.1.2-1 Reactor Coolant System - Primary Coolant System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Alloy 600 Cladding	Shelter / Protection	Alloy 600	Treated Water (Int)	Cracking	Alloy 600 Program	IV.D1.1-i	3.1.1-44	C
					Water Chemistry Program	IV.D1.1-i	3.1.1-44	C
Alloy 600 Safe Ends	Fluid Pressure Boundary	Alloy 600	Treated Water (Int)	Cracking	Alloy 600 Program	IV.C2.2-f	3.1.1-36	A, 123
						IV.C2.5-h	3.1.1-36	A
					Water Chemistry Program	IV.C2.2-f	3.1.1-36	A, 123
						IV.C2.5-h	3.1.1-36	A
Alloy 600 Thermal Sleeves	Shelter / Protection	Alloy 600	Treated Water (Ext)	Cracking	Alloy 600 Program	IV.C2.5-k	3.1.1-14	C, 124
					Water Chemistry Program	IV.C2.5-k	3.1.1-14	C, 124
Bolting and Fasteners	Fluid Pressure Boundary	Low Alloy Steel	Containment Air (Ext)	Loss of Preload	Bolting Integrity Program	IV.C2.4-g	3.1.1-27	101, A
				Loss of Material	Bolting Integrity Program			F
					Boric Acid Corrosion Program	IV.C2.1-d	3.1.1-39	C
						IV.C2.2-d	3.1.1-39	C
			Plant Indoor Air (Ext)	Loss of Preload	Bolting Integrity Program	IV.C2.4-g	3.1.1-27	A
				Loss of Material	Bolting Integrity Program			F
					Boric Acid Corrosion Program	IV.C2.1-d	3.1.1-39	C
						IV.C2.2-d	3.1.1-39	C
Carbon Steel Nozzles	Fluid Pressure Boundary	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	IV.C2.2-d	3.1.1-38	C

Table 3.1.2-1 Reactor Coolant System - Primary Coolant System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Carbon Steel Pipe (30" and 42")	Fluid Pressure Boundary	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	IV.C2.1-d	3.1.1-38	A
Flow Element (PCP Controlled Bleed)	Fluid Pressure Boundary	Stainless Steel	Treated Water (Int)	Cracking	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.C2.2-f	3.1.1-36	C
					Water Chemistry Program	IV.C2.2-f	3.1.1-36	C
Non-CASS Valves in PCS and Connected Systems	Fluid Pressure Boundary	Stainless Steel	Treated Water (Int)	Cracking	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.C2.2-f	3.1.1-36	C
					Water Chemistry Program	IV.C2.2-f	3.1.1-36	C
						IV.C2.2-f	3.1.1-36	C
PCS Spray and Drain Nozzles	Fluid Pressure Boundary	Alloy 600	Treated Water (Int)	Cracking	Alloy 600 Program	IV.C2.5-k	3.1.1-14	C
						IV.C2.5-s	3.1.1-14	C
					Water Chemistry Program	IV.C2.5-k	3.1.1-14	C
						IV.C2.5-s	3.1.1-14	C
PORV Isolation, Quench Tank Spray Manual Valves	Fluid Pressure Boundary	Cast Austenitic SS	Treated Water (Int)	Cracking	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.C2.4-b	3.1.1-36	A
					Water Chemistry Program	IV.C2.4-b	3.1.1-36	A
PORV Isolation Valves	Fluid Pressure Boundary	Cast Austenitic SS	Treated Water (Int)	Reduction of Fracture Toughness	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.C2.4-c	3.1.1-23	A

Table 3.1.2-1 Reactor Coolant System - Primary Coolant System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Pressurizer Alloy 600 Instrument Penetrations	Fluid Pressure Boundary	Alloy 600	Treated Water (Int)	Cracking	Alloy 600 Program	IV.C2.5-k	3.1.1-14	A
					Water Chemistry Program	IV.C2.5-k	3.1.1-14	A
Pressurizer Heater Sleeves	Fluid Pressure Boundary	Alloy 600	Treated Water (Int)	Cracking	Alloy 600 Program	IV.C2.5-s	3.1.1-14	A
					Water Chemistry Program	IV.C2.5-s	3.1.1-14	A
Pressurizer Heaters	Fluid Pressure Boundary	Alloy 600	Treated Water (Int)	Cracking	Alloy 600 Program	IV.C2.5-s	3.1.1-14	A
					Water Chemistry Program	IV.C2.5-s	3.1.1-14	A
Pressurizer Integral Support Weld	Fluid Pressure Boundary	Carbon Steel	Containment Air (Ext)	Cracking	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.C2.5-v	3.1.1-41	A
				Loss of Material	Boric Acid Corrosion Program	IV.C2.5-u	3.1.1-38	A
Pressurizer Manway and Flange Bolting	Fluid Pressure Boundary	Low Alloy Steel	Containment Air (Ext), Borated Water or Steam Leakage	Loss of Material	Boric Acid Corrosion Program	IV.C2.5-o	3.1.1-38	A
			Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	IV.C2.5-o	3.1.1-38	A
				Loss of Preload	Bolting Integrity Program	IV.C2.5-p	3.1.1-26	101, A
Pressurizer Manway and Flanges	Fluid Pressure Boundary	Carbon Steel	Containment Air (Ext), Borated Water or Steam Leakage	Loss of Material	Boric Acid Corrosion Program	IV.C2.5-o	3.1.1-38	A

Table 3.1.2-1 Reactor Coolant System - Primary Coolant System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Pressurizer Quench Tank	Fluid Pressure Boundary	Epoxy Coated Carbon Steel	Containment Air (Ext)	Loss of Material	System Monitoring Program	IV.C2.6-b	3.1.1-38	E
			Treated Water (Int)	Loss of Material	One-Time Inspection Program			F
Pressurizer Quench Tank Shell and Heads	Fluid Pressure Boundary	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	IV.C2.6-b	3.1.1-38	A
Pressurizer Spray Head	Spray Pattern	Alloy 600	Treated Water (Int)	Cracking	One-Time Inspection Program	IV.C2.5-j	3.1.1-12	A
					Water Chemistry Program	IV.C2.5-j	3.1.1-12	A
Primary Coolant Pump Casing	Fluid Pressure Boundary	Cast Austenitic SS	Treated Water (Int)	Cracking	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.C2.3-b	3.1.1-36	A
					Water Chemistry Program	IV.C2.3-b	3.1.1-36	A
				Reduction of Fracture Toughness	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.C2.3-c	3.1.1-23	A
Primary Coolant Pump Closure Bolting	Fluid Pressure Boundary	Carbon Steel	Containment Air (Ext)	Loss of Preload	Bolting Integrity Program	IV.C2.3-g	3.1.1-27	101, A
				Loss of Material	Boric Acid Corrosion Program	IV.C2.3-f	3.1.1-39	A

Table 3.1.2-1 Reactor Coolant System - Primary Coolant System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Primary Coolant Sample Heat Exchanger Shell	Fluid Pressure Boundary	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	IV.C2.2-d	3.1.1-38	A
					System Monitoring Program	VII.I.1-b	3.1.1-05	A
			Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	V.D1.5-a	3.1.1-13	C
Reactor Head Vent	Fluid Pressure Boundary	Stainless Steel	Treated Water (Int)	Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.C2.1-g	3.1.1-07	C
					Water Chemistry Program	IV.C2.1-g	3.1.1-07	C
Reactor Head Vent Orifice	Fluid Pressure Boundary	Stainless Steel	Treated Water (Int)	Cracking	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.C2.1-g	3.1.1-07	A
					Water Chemistry Program	IV.C2.1-g	3.1.1-07	A
Sample Point (Quench Tank Liquid, Loop 2 Hot Leg, Pressurizer Surge Line)	Fluid Pressure Boundary	Stainless Steel	Treated Water (Int)	Cracking	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.C2.1-g	3.1.1-07	C
						IV.C2.2-h	3.1.1-07	C
					Water Chemistry Program	IV.C2.1-g	3.1.1-07	C
						IV.C2.2-h	3.1.1-07	C
Small Bore Stainless Steel Pipe (PCS and Connected Systems)	Fluid Pressure Boundary	Stainless Steel	Treated Water (Int)	Cracking	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.C2.1-g	3.1.1-07	A
						IV.C2.2-h	3.1.1-07	A
					Water Chemistry Program	IV.C2.1-g	3.1.1-07	A
						IV.C2.2-h	3.1.1-07	A

Table 3.1.2-1 Reactor Coolant System - Primary Coolant System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
SS Cladding	Shelter/ Protection	Stainless Steel	Treated Water (Int)	Cracking	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.C2.1-c	3.1.1-36	A
						IV.C2.5-c	3.1.1-36	A
						IV.C2.5-g	3.1.1-36	A
						IV.C2.5-h	3.1.1-36	A
						IV.C2.5-m	3.1.1-36	A
					Water Chemistry Program	IV.C2.1-c	3.1.1-36	A
						IV.C2.5-c	3.1.1-36	A
						IV.C2.5-g	3.1.1-36	A
						IV.C2.5-h	3.1.1-36	A
						IV.C2.5-m	3.1.1-36	A
Stainless Steel Pipe (PCS and Connected Systems)	Fluid Pressure Boundary	Stainless Steel	Treated Water (Int)	Cracking	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.C2.1-c	3.1.1-36	A
						IV.C2.2-f	3.1.1-36	A
					Water Chemistry Program	IV.C2.1-c	3.1.1-36	A
						IV.C2.2-f	3.1.1-36	A
Stainless Steel Safe Ends (Pressurizer and Connected Systems)	Fluid Pressure Boundary	Stainless Steel	Treated Water (Int)	Cracking	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.C2.2-f	3.1.1-36	A
						IV.C2.5-h	3.1.1-36	A
					Water Chemistry Program	IV.C2.2-f	3.1.1-36	A
						IV.C2.5-h	3.1.1-36	A

Table 3.1.2-1 Reactor Coolant System - Primary Coolant System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Stainless Steel Thermal Sleeves	Shelter / Protection	Stainless Steel	Treated Water (Ext)	Cracking	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.C2.5-h	3.1.1-36	C
					Water Chemistry Program	IV.C2.5-h	3.1.1-36	C
			Treated Water (Int)	Cracking	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.C2.5-h	3.1.1-36	C
					Water Chemistry Program	IV.C2.5-h	3.1.1-36	C
Stainless Steel Tubing	Fluid Pressure Boundary	Stainless Steel	Treated Water (Int)	Cracking	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.C2.2-f	3.1.1-36	A
					Water Chemistry Program	IV.C2.2-f	3.1.1-36	A
Vessels, Pressure (Pressurizer)	Fluid Pressure Boundary	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	IV.C2.5-b	3.1.1-38	A

Table 3.1.2-2 Reactor Coolant System - Reactor Vessel - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
CRDM Upper Pressure Housing & Flange	Pressure Boundary/ Fission Product Retention	Stainless Steel	Treated Water (Int)	Cracking	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.A2.2-b	3.1.1-36	A
					Water Chemistry Program	IV.A2.2-b	3.1.1-36	A
CRDM/Incore Instrument Bolting	Pressure Boundary/ Fission Product Retention	Stainless Steel	Containment Air (Ext)	Loss of Material	Bolting Integrity Program	IV.A2.2-f	3.1.1-26	A
				Loss of Preload	Bolting Integrity Program	IV.A2.2-g	3.1.1-26	101, A
				Cracking	Bolting Integrity Program	IV.A2.2-e	3.1.1-26	A
Incore Instrument Closure Flanges	Pressure Boundary/ Fission Product Retention	Stainless Steel	Containment Air (Ext)	Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.A2.2-f	3.1.1-26	C
			Treated Water (Int)	Cracking	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.A2.2-b	3.1.1-36	C
					Water Chemistry Program	IV.A2.2-b	3.1.1-36	C
Internal SS Cladding	Pressure Boundary/ Fission Product Retention	Stainless Steel	Treated Water (Int)	Cracking	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.A2.4-b	3.1.1-36	C
					Water Chemistry Program	IV.A2.4-b	3.1.1-36	C
Reactor Vessel Column Support	Component Structural Support	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	IV.A2.8-b	3.1.1-38	119, 122, A
Reactor Vessel Bottom Head	Pressure Boundary/ Fission Product Retention	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	IV.A2.5-e	3.1.1-38	C

Table 3.1.2-2 Reactor Coolant System - Reactor Vessel - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Reactor Vessel Closure Head	Component Structural Support Pressure Boundary/ Fission Product Retention	Carbon Steel	Containment Air (Ext)	Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.A2.5-f	3.1.1-40	C
					Boric Acid Corrosion Program	IV.A2.1-a	3.1.1-38	A
Reactor Vessel Closure Head Lifting Lugs	Component Structural Support	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	IV.A2.1-a	3.1.1-38	C
Reactor Vessel Core Stabilizer Lugs	Structure Functional Support	Nickel-Based Alloys	Treated Water (Ext)	Cracking	Water Chemistry Program	IV.A2.6-a	3.1.1-12	A
					Reactor Vessel Internals Inspection Program	IV.A2.6-a	3.1.1-12	A
				Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program			110, H
Reactor Vessel CRDM Nozzles	Structure Functional Support Pressure Boundary/ Fission Product Retention	Nickel-Based Alloys	Treated Water (Int)	Cracking	Alloy 600 Program	IV.A2.2-a	3.1.1-35	A
					Water Chemistry Program	IV.A2.2-a	3.1.1-35	A
Reactor Vessel Flow Skirt	Reduce Flow Inequalities	Nickel-Based Alloys	Treated Water (Ext)	Cracking	Water Chemistry Program	IV.A2.6-a	3.1.1-12	E
					Reactor Vessel Internals Inspection Program	IV.A2.6-a	3.1.1-12	A

Table 3.1.2-2 Reactor Coolant System - Reactor Vessel - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Reactor Vessel Head O-ring Leakage Monitor Tube	Pressure Boundary/ Fission Product Retention	Nickel-Based Alloys	Treated Water (Int)	Cracking	Alloy 600 Program	IV.A2.1-f	3.1.1-12	123, A
Reactor Vessel Head Vent Nozzle	Pressure Boundary/ Fission Product Retention	Nickel-Based Alloys	Treated Water (Int)	Cracking	Alloy 600 Program	IV.A2.7-b	3.1.1-35	A
					Water Chemistry Program	IV.A2.7-b	3.1.1-35	A
Reactor Vessel Incore Instrument Nozzles	Pressure Boundary/ Fission Product Retention	Nickel-Based Alloys	Treated Water (Int)	Cracking	Alloy 600 Program	IV.A2.7-b	3.1.1-35	121, A
					Water Chemistry Program	IV.A2.7-b	3.1.1-35	A
Reactor Vessel Intermediate Shell	Pressure Boundary/ Fission Product Retention	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	IV.A2.5-e	3.1.1-38	C
				Reduction in Fracture Toughness	Reactor Vessel Integrity Surveillance Program	IV.A2.5-c	3.1.1-05	109, A
Reactor Vessel Lower Shell	Pressure Boundary/ Fission Product Retention	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	IV.A2.5-e	3.1.1-38	C
				Reduction in Fracture Toughness	Reactor Vessel Integrity Surveillance Program	IV.A2.5-c	3.1.1-05	109, A
Reactor Vessel Nozzle Safe Ends	Pressure Boundary/ Fission Product Retention	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	IV.A2.5-e	3.1.1-38	C

Table 3.1.2-2 Reactor Coolant System - Reactor Vessel - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Reactor Vessel Primary Coolant Nozzles	Pressure Boundary/ Fission Product Retention Structure Functional Support	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	IV.A2.5-e	3.1.1-38	C
Reactor Vessel Seal Ledge Ring	Pressure Boundary/ Fission Product Retention	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	IV.A2.5-e	3.1.1-38	C
Reactor Vessel Stop Lugs	Prevent Core Displacement	Nickel-Based Alloys	Treated Water (Ext)	Cracking	Water Chemistry Program	IV.A2.6-a	3.1.1-12	A
					Reactor Vessel Internals Inspection Program	IV.A2.6-a	3.1.1-12	A
Reactor Vessel Studs, Nuts, Washers	Pressure Boundary/ Fission Product Retention	Carbon Steel	Containment Air (Ext)	Crack Initiation and Growth	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.A2.1-c	3.1.1-22	A
				Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.A2.1-d	3.1.1-47	C
					Boric Acid Corrosion Program	IV.A2.1-a	3.1.1-38	A
Reactor Vessel Surveillance Capsule Holder	Structure Functional Support	Nickel-Based Alloys	Treated Water (Ext)	Cracking	Reactor Vessel Internals Inspection Program	IV.A2.6-a	3.1.1-12	A
					Water Chemistry Program	IV.A2.6-a	3.1.1-12	C

Table 3.1.2-2 Reactor Coolant System - Reactor Vessel - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Reactor Vessel Upper Shell	Pressure Boundary/ Fission Product Retention	Carbon Steel	Containment Air (Ext)	Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.A2.5-f	3.1.1-40	A
					Boric Acid Corrosion Program	IV.A2.5-e	3.1.1-38	A
Reactor Vessel Upper Shell Flange	Pressure Boundary/ Fission Product Retention	Carbon Steel	Containment Air (Ext)	Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.A2.5-f	3.1.1-40	A
					Boric Acid Corrosion Program	IV.A2.5-e	3.1.1-38	A
Under Head CRDM Support	Structure Functional Support	Stainless Steel	Treated Water (Ext)	Cracking	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.A2.2-b	3.1.1-36	C
					Water Chemistry Program	IV.A2.2-b	3.1.1-36	C

Table 3.1.2-3 Reactor Coolant System - Reactor Vessel Internals - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Control Rod Shroud Assembly = IV.B3.2 - CEA Shroud Assemblies								
Control Rod Shroud	Structure Functional Support	Stainless Steel	Treated Water (Ext)	Changes in Dimensions	Reactor Vessel Internals Inspection Program	IV.B3.2-c	3.1.1-11	A, 113
Control Rod Shroud, Shroud Top Support, Shroud Support Lug	Structure Functional Support	Stainless Steel	Treated Water (Ext)	Cracking	Reactor Vessel Internals Inspection Program	IV.B3.2-a	3.1.1-45	A
					Water Chemistry Program	IV.B3.2-a	3.1.1-45	A
Control Rod Support Lug, Fuel Guide Pin, Fuel Guide Pin Nuts	Structure Functional Support	Stainless Steel	Treated Water (Ext)	Changes in Dimensions	Reactor Vessel Internals Inspection Program	IV.B3.2-c	3.1.1-11	A, 113
Fuel Guide Pin, Fuel Guide Pin Nuts	Structure Functional Support	Stainless Steel	Treated Water (Ext)	Cracking	Reactor Vessel Internals Inspection Program	IV.B3.2-b	3.1.1-45	A
					Water Chemistry Program	IV.B3.2-b	3.1.1-45	A
Fuel Guide Pin, Fuel Guide Pin Nuts, Fuel Plate Cap Screw	Structure Functional Support	Stainless Steel	Treated Water (Ext)	Loss of Preload	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	V.B3.2-g	3.1.1-48	101, A
Core Shroud Assembly = IV.B3.4 - Core Shroud Assembly								

Table 3.1.2-3 Reactor Coolant System - Reactor Vessel Internals - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Anchor Block, Centering Plate, Core Shroud Plate	Structure Functional Support	Stainless Steel	Treated Water (Ext)	Changes in Dimensions	Reactor Vessel Internals Inspection Program	IV.B3.4-b	3.1.1-11	A, 113
				Cracking	Reactor Vessel Internals Inspection Program	IV.B3.4-a	3.1.1-45	A
					Water Chemistry Program	IV.B3.4-a	3.1.1-45	A
				Reduction in Fracture Toughness	Reactor Vessel Internals Inspection Program	IV.B3.4-c	3.1.1-43	A, 113
Anchor Screw & Pin, Centering Screw & Pin, Positioning Screw, Shroud Bolt & Pin	Structure Functional Support	Stainless Steel	Treated Water (Ext)	Cracking	Reactor Vessel Internals Inspection Program	IV.B3.4-e	3.1.1-45	A
					Water Chemistry Program	IV.B3.4-e	3.1.1-45	A
				Changes in Dimensions	Reactor Vessel Internals Inspection Program	IV.B3.4-f	3.1.1-11	A, 113
				Reduction in Fracture Toughness	Reactor Vessel Internals Inspection Program	IV.B3.4-g	3.1.1-43	A, 113
				Loss of Preload	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.B3.4-h	3.1.1-48	101, A
Core Support Barrel Assembly = IV.B3.3 - Core Support Barrel Assembly								
Core Support Barrel	Structure Functional Support	Stainless Steel	Treated Water (Ext)	Cracking	Reactor Vessel Internals Inspection Program	IV.B3.3-a	3.1.1-45	A
					Water Chemistry Program	IV.B3.3-a	3.1.1-45	A

Table 3.1.2-3 Reactor Coolant System - Reactor Vessel Internals - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Core Support Barrel Integral Upper Flange	Structure Functional Support	Stainless Steel	Treated Water (Ext)	Cracking	Reactor Vessel Internals Inspection Program	IV.B3.3-a	3.1.1-45	A
					Water Chemistry Program	IV.B3.3-a	3.1.1-45	A, 113
				Changes in Dimensions	Reactor Vessel Internals Inspection Program	IV.B3.3-b	3.1.1-11	A, 113
Incore Instrument Guide Tube (This group of components is not addressed in GALL IV.B3)								
Guide Tube Plug Screw	Structure Functional Support	Stainless Steel	Treated Water (Ext)	Loss of Preload	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.B3.2-g	3.1.1-48	E, 103
Instrument Guide Tube, Guide Tube Bracket	Structure Functional Support	Stainless Steel	Treated Water (Ext)	Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.B3.1-c	3.1.1-40	C
Instrument Guide Tube, Guide Tube Bracket, Guide Tube Plugs, Guide Tube Plug Screw, Guide Tube Support	Structure Functional Support	Stainless Steel	Treated Water (Ext)	Changes in Dimensions	Reactor Vessel Internals Inspection Program	IV.B3.1-b	3.1.1-11	C, 113
				Cracking	Reactor Vessel Internals Inspection Program	IV.B3.1-a	3.1.1-45	C
					Water Chemistry Program	IV.B3.1-a	3.1.1-45	C
				Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.B3.1-c	3.1.1-40	C
				Reduction in Fracture Toughness	Reactor Vessel Internals Inspection Program	IV.B3.3-a	3.1.1-45	C, 113
Lower Internal Assembly (Integral with Core Barrel Assembly) = IV.B3.5 - Lower Internal Assembly								

Table 3.1.2-3 Reactor Coolant System - Reactor Vessel Internals - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Core Support Barrel Cap Screws	Structure Functional Support	Stainless Steel	Treated Water (Ext)	Changes in Dimensions	Reactor Vessel Internals Inspection Program	IV.B3.5-c	3.1.1-11	A, 113
				Cracking	Reactor Vessel Internals Inspection Program	IV.B3.5-b	3.1.1-45	A
					Water Chemistry Program	IV.B3.5-b	3.1.1-45	A
				Reduction in Fracture Toughness	Reactor Vessel Internals Inspection Program	IV.B3.5-d	3.1.1-11	A
Core Support Barrel Snubber Lug	Structure Functional Support	Stainless Steel	Treated Water (Ext)	Changes in Dimensions	Reactor Vessel Internals Inspection Program	IV.B3.5-c	3.1.1-11	A, 113
				Cracking	Reactor Vessel Internals Inspection Program	IV.B3.5-a	3.1.1-45	A
					Water Chemistry Program	IV.B3.5-a	3.1.1-45	A
				Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.B3.5-e	3.1.1-40	A
				Reduction in Fracture Toughness	Reactor Vessel Internals Inspection Program	IV.B3.5-d	3.1.1-11	A
Core Support Column	Structure Functional Support	Stainless Steel	Treated Water (Ext)	Changes in Dimensions	Reactor Vessel Internals Inspection Program	IV.B3.5-c	3.1.1-11	A, 113
				Cracking	Reactor Vessel Internals Inspection Program	IV.B3.5-a	3.1.1-45	A
					Water Chemistry Program	IV.B3.5-a	3.1.1-45	A

Table 3.1.2-3 Reactor Coolant System - Reactor Vessel Internals - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Core Support Column Support Beams and Tie Rods	Structure Functional Support	Stainless Steel	Treated Water (Ext)	Changes in Dimensions	Reactor Vessel Internals Inspection Program	IV.B3.5-c	3.1.1-11	A, 113
				Cracking	Reactor Vessel Internals Inspection Program	IV.B3.5-a	3.1.1-45	A
					Water Chemistry Program	IV.B3.5-a	3.1.1-45	A
				Reduction in Fracture Toughness	Reactor Vessel Internals Inspection Program	IV.B3.5-d	3.1.1-11	A
Core Support Plate	Structure Functional Support	Stainless Steel	Treated Water (Ext)	Changes in Dimensions	Reactor Vessel Internals Inspection Program	IV.B3.5-c	3.1.1-11	A, 113
				Cracking	Reactor Vessel Internals Inspection Program	IV.B3.5-a	3.1.1-45	A
					Water Chemistry Program	IV.B3.5-a	3.1.1-45	A
				Reduction in Fracture Toughness	Reactor Vessel Internals Inspection Program	IV.B3.5-d	3.1.1-11	A
Upper Guide Structure - Not in GALL								
Instrument Sleeve	Structure Functional Support	Stainless Steel	Treated Water (Ext)	Changes in Dimensions	Reactor Vessel Internals Inspection Program	IV.B3.1-b	3.1.1-11	C, 113
				Reduction in Fracture Toughness	Reactor Vessel Internals Inspection Program	IV.B3.2-e	3.1.1-43	F

Table 3.1.2-3 Reactor Coolant System - Reactor Vessel Internals - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Spacer Shim, Instrument Sleeve	Structure Functional Support	Stainless Steel	Treated Water (Ext)	Cracking	Reactor Vessel Internals Inspection Program	IV.B3.1-a	3.1.1-45	C
				Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.B3.1-c	3.1.1-40	C
Upper Internal Assembly = GALL IV.B3.1 - Upper Internal Assembly								
Brace Grid Beam, Cross Brace Screw, Shroud Grid Ring	Structure Functional Support	Stainless Steel	Treated Water (Ext)	Changes in Dimensions	Reactor Vessel Internals Inspection Program	IV.B3.1-b	3.1.1-45	A, 113
				Cracking	Reactor Vessel Internals Inspection Program	IV.B3.1-a	3.1.1-45	A
					Water Chemistry Program	IV.B3.1-a	3.1.1-45	A
Fuel Alignment Plate	Structure Functional Support	Stainless Steel	Treated Water (Ext)	Changes in Dimensions	Reactor Vessel Internals Inspection Program	IV.B3.1-b	3.1.1-45	A, 113
				Cracking	Reactor Vessel Internals Inspection Program	IV.B3.1-a	3.1.1-45	A
					Water Chemistry Program	IV.B3.1-a	3.1.1-45	A
				Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.B3.1-c	3.1.1-40	A

Table 3.1.2-3 Reactor Coolant System - Reactor Vessel Internals - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Fuel Plate Align Lug, Fuel Plate Cap Screw, Fuel Plate Guide Pin	Structure Functional Support	Stainless Steel	Treated Water (Ext)	Changes in Dimensions	Reactor Vessel Internals Inspection Program	IV.B3.1-b	3.1.1-45	A, 113
				Cracking	Reactor Vessel Internals Inspection Program	IV.B3.1-a	3.1.1-45	A
					Water Chemistry Program	IV.B3.1-a	3.1.1-45	A
				Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.B3.1-c	3.1.1-40	A
Holddown Ring Plunger, Holddown Ring Strap, Holddown Ring	Structure Functional Support	Stainless Steel	Treated Water (Ext)	Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.B3.1-c	3.1.1-40	A

Table 3.1.2-4 Reactor Coolant System - Replacement Steam Generators - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Alloy 690 Tube Plugs	Fluid Pressure Boundary	Alloy 690	Treated Water (Ext)	Cracking	Steam Generator Tube Integrity Program	IV.D1.2-i	3.1.1-18	A
					Water Chemistry Program	IV.D1.2-i	3.1.1-18	A
			Treated Water (Int)	Cracking	Steam Generator Tube Integrity Program	IV.D1.2-i	3.1.1-18	A
					Water Chemistry Program	IV.D1.2-i	3.1.1-18	A
Handhole Cover	Fluid Pressure Boundary	Low-Alloy Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	IV.D1.1-g	3.1.1-38	C
			Treated Water (Int)	Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.D1.1-c	3.1.1-02	C
					Water Chemistry Program	IV.D1.1-c	3.1.1-02	C
Tube Bundle Support Assembly	Structural Support for Safety Related	Stainless Steel	Treated Water (Ext)	Cracking	Water Chemistry Program			F
					Steam Generator Tube Integrity Program			118, F
		Carbon Steel	Treated Water (Ext)	Loss of Material	Water Chemistry Program	IV.D1.2-k	3.1.1-20	A
					Steam Generator Tube Integrity Program	IV.D1.2-k	3.1.1-20	118, A
Tube Bundle Wrapper	Direct Flow	Low Alloy Steel	Treated Water (Int)	Loss of Material	Water Chemistry Program			F

Table 3.1.2-4 Reactor Coolant System - Replacement Steam Generators - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Lower Head	Fluid Pressure Boundary	Carbon Steel w/SS cladding	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	IV.D1.1-g	3.1.1-38	A
			Treated Water (Int)	Cracking	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.D1.1-i	3.1.1-44	C
					Water Chemistry Program	IV.D1.1-i	3.1.1-44	C
Primary Manway Cover	Fluid Pressure Boundary	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	IV.D1.1-k	3.1.1-38	A
Manway Cover Diaphragm	Fluid Pressure Boundary	Stainless Steel	Treated Water (Int)	Cracking	Water Chemistry Program	IV.D1.1-i	3.1.1-44	E
Nozzle Safe Ends	Fluid Pressure Boundary	Carbon Steel w/ SS Cladding	Treated Water (Int)	Cracking	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.D1.1-i	3.1.1-44	A
					Water Chemistry Program	IV.D1.1-i	3.1.1-44	A
			Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	IV.D1.1-g	3.1.1-38	C
Primary Divider Plate	Fluid Pressure Boundary	Stainless Steel	Treated Water (Ext)	Cracking	Water Chemistry Program	IV.D1.1-i	3.1.1-44	C

Table 3.1.2-4 Reactor Coolant System - Replacement Steam Generators - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Primary Inlet and Outlet Nozzles	Fluid Pressure Boundary	Carbon Steel w/SS cladding	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	IV.D1.1-g	3.1.1-38	C
			Treated Water (Int)	Cracking	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.D1.1-i	3.1.1-44	A
					Water Chemistry Program	IV.D1.1-i	3.1.1-44	A
Feedwater Inlet Nozzles and Thermal Sleeves	Fluid Pressure Boundary	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	IV.D1.1-g	3.1.1-38	C
			Treated Water (Int)	Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.D1.1-c	3.1.1-02	C
					Water Chemistry Program	IV.D1.1-c	3.1.1-02	C
					Flow Accelerated Corrosion Program	IV.D1.1-d	3.1.1-25	117, A
Steam Outlet Nozzle and Flow Limiter, Blowdown Nozzle	Fluid Pressure Boundary	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	IV.D1.1-g	3.1.1-38	C
			Treated Water (Int)	Loss of Material	Water Chemistry Program	IV.D1.1-c	3.1.1-02	E
					Flow Accelerated Corrosion Program	IV.D1.1-d	3.1.1-25	117, A

Table 3.1.2-4 Reactor Coolant System - Replacement Steam Generators - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Secondary Side 2" Inspection Port Cover	Fluid Pressure Boundary	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	IV.D1.1-k	3.1.1-38	C
			Treated Water (Int)	Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.D1.1-c	3.1.1-02	A
					Water Chemistry Program	IV.D1.1-c	3.1.1-02	A
Wide and Narrow Range Water level Nozzles, Sampling and Instrument Nozzles	Fluid Pressure Boundary	Low Alloy Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	IV.D1.1-g	3.1.1-38	C
			Treated Water (Int)	Loss of Material	Water Chemistry Program	IV.D1.1-c	3.1.1-02	C
				Cracking	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.D1.1-c	3.1.1-02	E
					Water Chemistry Program	IV.D1.1-c	3.1.1-02	E
Shells (Lower, Upper, Transition)	Fluid Pressure Boundary	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	IV.D1.1-g	3.1.1-38	A
			Treated Water (Int)	Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.D1.1-c	3.1.1-02	A
					Water Chemistry Program	IV.D1.1-c	3.1.1-02	A
					Steam Generator Tube Integrity Program	IV.D1.1-c	3.1.1-02	B

Table 3.1.2-4 Reactor Coolant System - Replacement Steam Generators - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Tubesheet	Fluid Pressure Boundary	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	IV.D1.1-g	3.1.1-38	114, C
			Treated Water (Int) (Secondary)	Loss of Material	Steam Generator Tube Integrity Program	IV.D1.1-c	3.1.1-02	C
					Water Chemistry Program	IV.D1.1-c	3.1.1-02	C
		Nickel Based Alloy	Treated Water (Int) (Primary)	Cracking	Steam Generator Tube Integrity Program	IV.D1.2-a	3.1.1-18	C
					Water Chemistry Program	IV.D1.2-a	3.1.1-18	C
Upper Head	Fluid Pressure Boundary	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	IV.D1.1-g	3.1.1-38	C
			Treated Water (Int)	Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.D1.1-c	3.1.1-02	A
				Loss of Material	Water Chemistry Program	IV.D1.1-c	3.1.1-02	A

Table 3.1.2-4 Reactor Coolant System - Replacement Steam Generators - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
U-Tubes	Fluid Pressure Boundary Heat Transfer	Alloy 600	Treated Water (Ext)	Cracking	Steam Generator Tube Integrity Program	IV.D1.2-b	3.1.1-18	A
						IV.D1.2-c	3.1.1-18	A
					Water Chemistry Program	IV.D1.2-b	3.1.1-18	A
						IV.D1.2-c	3.1.1-18	A
				Loss of Material	Water Chemistry Program	IV.D1.2-e	3.1.1-18	A
					Steam Generator Tube Integrity Program	IV.D1.2-e	3.1.1-18	A
			Treated Water (Int)	Cracking	Steam Generator Tube Integrity Program	IV.D1.2-a	3.1.1-18	A
					Water Chemistry Program	IV.D1.2-a	3.1.1-18	A
Manway Fasteners	Fluid Pressure Boundary	Carbon Steel / Low Alloy Steel	Containment Air (Ext)	Crack Initiation and Growth	Bolting Integrity Program	IV.D1.1-l	3.1.1-26	A
				Loss of Material	Boric Acid Corrosion Program	IV.D1.1-k	3.1.1-38	A
				Loss of Preload	Bolting Integrity Program	IV.D1.1-f	3.1.1-26	101, A

Notes for Tables 3.1.2-1 through 3.1.2-4

A Consistent with NUREG-1801 item for component, material, environment, and aging management program. AMP is consistent with NUREG-1801 AMP.

B Consistent with NUREG-1801 item for component, material, environment, and aging management program. AMP has exceptions to NUREG-1801 AMP.

C Component is different, but consistent with NUREG-1801 for material, environment, aging effect and AMP. AMP is consistent with NUREG-1801 AMP.

D Component is different, but consistent with NUREG-1801 for material, environment, aging effect and AMP. AMP has exceptions to NUREG-1801 AMP.

E Consistent with NUREG-1801 for material, environment, aging effect but a different AMP is credited.

F Material not in NUREG-1801 for this component.

G Environment not in NUREG-1801 for this component and material.

H Aging effect not in NUREG-1801 for this component, material and environment combination.

I Aging effect in NUREG-1801 for this component, material and environment combination is not applicable.

J Neither the component nor the material and environment combination is evaluated in NUREG-1801.

Plant-specific notes:

101 Loss of preload is included here in response to recent NRC RAIs on non-primary system, high temperature bolting that may experience loss of preload. The Palisades Bolting Integrity Program manages potential bolting AERMs and event driven degradation. GALL reconciliation is based on Loss of Material.

102 Not used

103 Palisades will not credit Loose Parts Monitoring for managing Loss of Preload.

104 Not used

105 Not used

106 Not used

107 SCC not applicable since temperature is less than 500F.

108 Not Used

109 Cladding addressed separately, therefore, environment does not match NUREG-1801, IV A2.5-c. The low alloy steel plate at the lower and intermediate shells is monitored at (1/4)Thickness and (3/4)Thickness for neutron fluence.

110 Plant-specific OE identified this aging effect/mechanism.

111 Fuel alignment pins are integral with fuel assemblies and fit into holes in fuel alignment plate. No CASS or nickel alloys.

112 The material and environment combination is in NUREG-1801 but neither the plant component, nor a reasonable substitute exists.

113 NMC will participate with the nuclear industry to determine if void swelling is an aging effect/mechanism requiring management, and help develop a program if needed.

114 Denting and corrosion of tube support plate

115 Loss of section thickness due to flow accelerated corrosion

116 Ligament cracking / Corrosion

117 The flow-accelerated corrosion program will be supplemented by the water chemistry control program for the management of loss of material of this component.

118 Focus of the credited steam generator integrity program is on maintaining the integrity of the steam generator tubes which additionally serves to supplement the water chemistry control program for management of cracking and loss of material of pertinent secondary side components, including internal supports.

119 Reactor Vessel column support is evaluated in civil/structural aging management review. Only evaluated Boric Acid Corrosion to match GALL item.

120 Safety Injection nozzles connect to the primary coolant piping not the reactor vessel so GALL items IV.A2.3-a and IV.A2.3-b are N/A.

121 There are no instrument tubes in bottom head so GALL item IV.A2.7-a is N/A

122 There is no skirt support so GALL item IV.A2.8-a is N/A.

123 Based on EPRI Mechanical Tools, Alloy 600 and Stainless Steel have similar properties with respect to Cracking / SCC.

124 All Alloy 600 Thermal Sleeves were evaluated under this item.

3.2 Aging Management of Engineered Safety Features

3.2.1 Introduction

This section provides the results of the aging management review for those components identified in Section 2.3.2, Engineered Safety Features, as being subject to aging management review. The systems, or portions of systems, which are addressed in this section, are described in the indicated section.

- Engineering Safeguards System (Section 2.3.2.1)

Table 3.2.1, Summary of Aging Management Evaluations in Chapter V of NUREG-1801 for Engineered Safety Features, provides the summary of the programs evaluated in NUREG-1801 for the Engineered Safety Features component groups that are utilized in license renewal.

This table uses the format described in Figure 3.0-1 above. Note that this table only includes those component groups that are applicable to a PWR.

3.2.2 Results

The following table summarizes the results of the aging management review for systems in the Engineered Safety Features system group.

Table 3.2.2-1, Engineered Safety Features - Engineering Safeguards System - Summary of Aging Management Evaluation

The materials that specific components are fabricated from, the environments to which components are exposed, the potential aging effects requiring management, and the aging management programs used to manage these aging effects are provided for the above system in the following subsection of Section 3.2.2.1, Materials, Environment, Aging Effects Requiring Management and Aging Management Programs:

Section 3.2.2.1.1, Engineering Safeguards System

3.2.2.1 Materials, Environment, Aging Effects Requiring Management and Aging Management Programs

3.2.2.1.1 Engineering Safeguards System

Materials

The materials of construction for the Engineering Safeguards System are:

- Aluminum
- Carbon Steel
- Carbon Steel w/SS clad lining

- Cast Austenitic SS
- Cast Iron
- Stainless Steel

Environment

The Engineering Safeguards System components are exposed to the following environments:

- Air (Int)
- Atmosphere / Weather (Ext)
- Containment Air (Ext)
- Gas (Int)
- Plant Indoor Air (Ext)
- Steam (Ext)
- Steam (Int)
- Treated Water (Ext)
- Treated Water (Int)

Aging Effects Requiring Management

The following aging effects, associated with the Engineering Safeguards System, require management:

- Cracking
- Heat Transfer Degradation
- Loss of Material
- Loss of Preload
- Reduction of Fracture Toughness

Aging Management Programs

The following aging management programs manage the aging effects for the Engineering Safeguards System components:

- ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program
- Bolting Integrity Program
- Boric Acid Corrosion Program
- Closed Cycle Cooling Water Program

- System Monitoring Program
- Water Chemistry Program
- One-Time Inspection Program

3.2.2.2 Further Evaluation of Aging Management as Recommended by NUREG-1801

NUREG-1801 Volume 1 Tables provide the basis for identifying those programs that warrant further evaluation by the reviewer in the license renewal application. For the Engineered Safety Features, those programs are addressed in the following sections.

3.2.2.2.1 Cumulative Fatigue Damage

Fatigue is a TLAA as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c). The evaluation of this TLAA is addressed separately in Section 4.3.

3.2.2.2.2 Loss of Material Due to General Corrosion

3.2.2.2.2.1 Loss of Material Due to General Corrosion (BWR Only)

Applicable to BWR Only

3.2.2.2.2.2 Loss of Material Due to General Corrosion (PWR Components)

NUREG-1800 states that loss of material due to general corrosion could occur in the PWR containment spray system header and spray nozzle components, containment isolation valves and associated piping, and the external surfaces of PWR carbon steel components. NUREG-1801 recommends further evaluation on a plant specific basis to ensure that the aging effect is adequately managed.

This aging effect of carbon steel in air does not apply to stainless steel components in Containment Spray and Emergency Core Cooling systems. The external surface of carbon steel and cast iron components in the Containment Spray and Emergency Core Cooling systems are susceptible to general corrosion in an air environment. The System Monitoring Program is credited with managing this aging effect.

Containment isolation components are addressed with their individual systems. Programs credited for aging management are identified in the 3.x.2 table for the system containing the penetration.

The aging effect/mechanism for containment isolation in aqueous systems is managed by the Water Chemistry Program and supplemented by the One-Time Inspection Program.

The aging effect/mechanism for containment isolation components in gaseous systems, the containment sump level instrumentation, and radwaste systems is managed by the One-Time Inspection Program.

All containment isolation valves and associated piping are currently tested on a set frequency by the Containment Leakage Testing Program. The testing will be continued in the extended period of operation.

3.2.2.2.3 Local Loss of Material due to Pitting and Crevice Corrosion

3.2.2.2.3.1 Local Loss of Material due to Pitting and Crevice Corrosion (BWR Only)

Applicable to BWR Only

3.2.2.2.3.2 Local Loss of Material due to Pitting and Crevice Corrosion (PWR Components)

NUREG-1800 states that local loss of material from pitting and crevice corrosion could occur in the PWR containment spray components, containment isolation valves and associated piping, and the buried portion of the refueling water tank external surface. NUREG-1801 recommends further evaluation to ensure that the aging effect is adequately managed.

The Palisades Containment Spray and Emergency Core Cooling system components are stainless steel, susceptible to pitting and crevice corrosion and are managed by the Water Chemistry Program supplemented by the One-Time Inspection Program. Loss of material due to pitting and crevice corrosion is a potential aging effect/mechanism that require management, even though the component has O_2<100 ppb supporting the effectiveness of the existing water chemistry program.

Containment isolation valve bodies and connecting piping are addressed with their individual systems. Programs credited for aging management are identified in the 3.x.2 table for the system containing the penetration.

The stainless steel containment isolation components are exposed to borated water, susceptible to pitting and crevice corrosion, and managed by the Water Chemistry Program supplemented by the One-Time Inspection Program. All containment isolation valves and associated piping are currently tested on a set frequency by the Containment Leakage Testing Program. The testing will be continued in the extended period of operation.

The SIRW tank bottom is not buried. It is located on the Auxiliary Building Roof. The bottom edge is sealed around its circumference from exposure to the weather. Supply piping enters and exits the bottom of the tank through the roof of the Auxiliary Building. This arrangement exposes the bottom surface of the tank to a plant indoor air environment. Loss of material due to crevice and pitting is not a potential aging effect/mechanism for aluminum in air.

3.2.2.2.4 Local Loss of Material due to Microbiologically Influenced Corrosion

NUREG-1800 states that local loss of material due to microbiologically influenced corrosion (MIC) could occur in BWR and PWR containment isolation valves and associated piping in systems that are not addressed in other chapters of NUREG-1801.

Loss of material due to MIC is considered a potential aging effect/mechanism that requires management, even though the component has no potential source of MIC contamination in treated water. The treated water environment is effectively controlled by the existing water chemistry program.

Containment isolation valve bodies and connecting piping are addressed with their individual systems. Programs credited for aging management are identified in the 3.x.2 table for the system containing the penetration. All containment isolation valves and associated piping are currently tested on a set frequency by the Containment Leakage Testing Program. The testing will be continued in the extended period of operation.

The Water Chemistry Program is credited for managing this aging effect/mechanism on containment isolation valves and associated piping in aqueous systems. This is supplemented by the One-Time Inspection Program to determine if the aging effect/mechanism exists, and, if it exists, how rapidly it is progressing.

The high temperature (>210ºF) of the Main Steam, Main Feedwater and Steam Generator Blowdown containment isolation components make them not susceptible to MIC.

The One-Time Inspection Program is credited for managing this aging effect/mechanism in gaseous systems, the containment sump level instrumentation, and radwaste systems containment isolation components.

3.2.2.2.5 Changes in Properties due to Elastomer Degradation

Applicable to BWR Only

3.2.2.2.6 Local Loss of Material due to Erosion

NUREG-1800 states that local loss of material due to erosion could occur in the high pressure safety injection pump miniflow orifice. This aging mechanism and effect will apply only to pumps that are normally used as charging pumps in the chemical and volume control systems (PWRs).

The high pressure safety injection pumps are not used for normal charging at Palisades. Loss of material due to erosion of miniflow orifices is not applicable to Palisades.

3.2.2.2.7 Buildup of Deposits due to Corrosion

Applicable to BWR Only

3.2.2.2.8 Quality Assurance for Aging Management of Non-Safety Related Components

See Section 3.1.2.2.15

3.2.2.3 Time-Limited Aging Analysis

The time-limited aging analyses (TLAA) identified below are associated with the Engineered Safety Features system components. The section of the LRA that contains the TLAA review results is indicated in parenthesis.

- Metal Fatigue (Section 4.3)

3.2.3 Conclusion

The Engineered Safety Features piping, fittings, and components that are subject to aging management review have been identified in accordance with the requirements of 10 CFR 54.4. The aging management programs selected to manage aging effects for the Engineered Safety Features components are identified in the summaries in Section 3.2.2.1.

A description of these aging management programs is provided in Appendix B, along with the demonstration that the identified aging effects will be managed for the period of extended operation.

Therefore, based on the demonstrations provided in Appendix B, the effects of aging associated with the Engineered Safety Features components will be adequately managed so that there is reasonable assurance that the intended function(s) will be maintained consistent with the current licensing basis during the period of extended operation.

Table 3.2.1 Summary of Aging Management Evaluations in Chapter V of NUREG-1801 for Engineered Safety Features

Item Number	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.2.1-01	Piping, fittings, and valves in emergency core cooling system	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA (see [SRP] subsection 3.2.2.2.1)	Further evaluation is documented in Section 3.2.2.2.1. The aging mechanism fatigue is potentially applicable to many component types in the Engineered Safety Features Supergroup, but only selected components or locations require explicit analysis as TLAAs and/or warrant aging management. The Palisades approach to identifying and managing the relevant locations and components for fatigue damage is addressed in Section 4.3 and in Appendix B, Fatigue Monitoring Program. Therefore, cumulative fatigue damage is not identified as an aging effect in Table 3.2.2-1 below.
3.2.1-02	BWR only				
3.2.1-03	Components in containment spray (PWR only), standby gas treatment (BWR only), containment isolation, and emergency core cooling systems	Loss of material due to general corrosion	Plant specific	Yes, plant specific (see [SRP] subsection 3.2.2.2.2.2)	Further evaluation is documented in Section 3.2.2.2.2.2.
3.2.1-04	BWR only				
3.2.1-05	Components in containment spray (PWR only), standby gas treatment (BWR only), containment isolation, and emergency core cooling systems	Loss of material due to pitting and crevice corrosion	Plant specific	Yes, plant specific (see [SRP] subsection 3.2.2.2.3.2)	Further evaluation is documented in Section 3.2.2.2.3.2.
3.2.1-06	Containment isolation valves and associated piping	Loss of material due to microbiologically influenced corrosion	Plant specific	Yes, plant specific (see [SRP] subsection 3.2.2.2.4)	Further evaluation is documented in Section 3.2.2.2.4.

Table 3.2.1 Summary of Aging Management Evaluations in Chapter V of NUREG-1801 for Engineered Safety Features

Item Number	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.2.1-07	BWR only				
3.2.1-08	High pressure safety injection (charging) pump miniflow orifice	Loss of material due to erosion	Plant specific	Yes, plant specific (see [SRP] subsection 3.2.2.2.6)	Further evaluation is documented in Section 3.2.2.2.6.
3.2.1-09	BWR only				
3.2.1-10	External surface of carbon steel components	Loss of material due to general corrosion	Plant specific	Yes, plant specific	Further evaluation is documented in Section 3.2.2.2.2.2
3.2.1-11	Piping and fittings of CASS in emergency core cooling system	Loss of fracture toughness due to thermal aging embrittlement	Thermal aging embrittlement of CASS	No	There are no CASS piping and fittings in ESF systems. ESF CASS valves are managed as described in [Section B2.1.2] ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program
3.2.1-12	Components serviced by open-cycle cooling system	Local loss of material due to corrosion and/or buildup of deposit due to biofouling	Open-cycle cooling water system	No	The components in the ESF system are not cooled by an open cycle cooling water system. Therefore, this line item is not applicable to Palisades.
3.2.1-13	Components serviced by closed-cycle cooling system	Loss of material due to general, pitting, and crevice corrosion	Closed-cycle cooling water system	No	A closed cycle cooling system cools the ESF system components. See [Section B2.1.6] Closed Cycle Cooling Water Program and [Section B2.1.21] Water Chemistry Program for the aging management programs
3.2.1-14	BWR only				

Table 3.2.1 Summary of Aging Management Evaluations in Chapter V of NUREG-1801 for Engineered Safety Features

Item Number	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.2.1-15	Pumps, valves, piping, and fittings in containment spray and emergency core cooling systems	Crack initiation and growth due to SCC	Water chemistry	No	The line items in NUREG-1801 Volume 2 that refer to this row number specify a temperature less than 93°C (200°F). The aging management reviews consider a threshold for SCC of 140°F. Environments for the systems are either less than 120°F such that cracking is not an aging effect requiring management or at 582°F PCS temperature, which is outside the range of the NUREG 1801 listed environment. The items from the following tables referring to this row number have a temperature between 120°F and 582°F. The Water Chemistry Program [Section B2.1.21] is credited with managing stress corrosion cracking for stainless steel in borated water at temperatures above the 140°F threshold for SCC.
3.2.1-16	BWR only				
3.2.1-17	Carbon steel components	Loss of material due to boric acid corrosion	Boric acid corrosion	No	See [Section B2.1.4] Boric Acid Corrosion Program for the aging management program
3.2.1-18	Closure bolting in high pressure or high temperature systems	Loss of material due to general corrosion, loss of preload due to stress relaxation, and crack initiation and growth due to cyclic loading or SCC	Bolting integrity	No	Aging effects are managed by the Bolting integrity Program [Section B2.1.3].

Table 3.2.2-1 Engineered Safety Features - Engineering Safeguards System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
SIRW Tank	Fluid Pressure Boundary	Aluminum	Treated Water (Int)	Cracking	Water Chemistry Program	V.D1.8-a	3.2.1-15	F, 210
					One-Time Inspection Program	V.D1.8-a	3.2.1-15	F, 210
Safety Injection Tank	Fluid Pressure Boundary	Carbon Steel w/SS clad lining	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	V.D1.7-a	3.2.1-17	A
					System Monitoring Program	V.E1-b	3.2.1-10	A
SDC HX Shell	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	V.D1.5-b	3.2.1-17	A
			Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	V.D1.5-a	3.2.1-13	B
SIRWT HX Shell	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	V.D1.6-d	3.2.1-17	A
			Steam (Int)	Loss of Material	Water Chemistry Program			G, 211
					One-Time Inspection Program			G, 211
SDC, SIRWT HX Shell	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	V.E1-b	3.2.1-10	A
SDC HX Channel Head	Fluid Pressure Boundary	Carbon Steel w/SS clad lining	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	V.D1.5-b	3.2.1-17	A
					System Monitoring Program	V.E1-b	3.2.1-10	A
SDC HX Tube Sheet shell side	Fluid Pressure Boundary	Carbon Steel w/SS clad lining	Treated Water (Ext)	Loss of Material	Closed Cycle Cooling Water Program	V.D1.5-a	3.2.1-13	B
SDC HX Channel Head shell side	Fluid Pressure Boundary	Carbon Steel w/SS clad lining	Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	V.D1.5-a	3.2.1-13	B

Table 3.2.2-1 Engineered Safety Features - Engineering Safeguards System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Cont. Spray Pump HX shell, LPSI Pump HX shell	Fluid Pressure Boundary	Cast Iron	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	V.E1-a	3.2.1-17	A
					System Monitoring Program	V.E1-b	3.2.1-10	A
			Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	V.D1.5-a	3.2.1-13	H, 212
					One-Time Inspection Program	V.D1.5-a	3.2.1-13	H, 212
SIRWT HX Tubes	Fluid Pressure Boundary	Stainless Steel	Steam (Ext)	Cracking	Water Chemistry Program	V.D1.6-a	3.2.1-13	G, 211
				Heat Transfer Degradation Loss of Material	One-Time Inspection Program	V.D1.6-a	3.2.1-13	G, 211
Cont. Spray, LPSI Pump coils	Fluid Pressure Boundary	Stainless Steel	Treated Water (Ext)	Cracking	Closed Cycle Cooling Water Program	V.D1.5-a	3.2.1-13	H
PCP Seal Cooler Coils, Cont. Spray Pump coils, LPSI pump Coils, SDC HX Tubes	Fluid Pressure Boundary Heat Transfer	Stainless Steel	Treated Water (Ext)	Heat Transfer Degradation	Closed Cycle Cooling Water Program	V.D1.5-a	3.2.1-13	H
				Loss of Material	Closed Cycle Cooling Water Program	V.D1.5-a	3.2.1-13	B
SDC HX Tubes	Fluid Pressure Boundary Heat Transfer	Stainless Steel	Treated Water (Ext)	Loss of Material	Closed Cycle Cooling Water Program	V.D1.5-a	3.2.1-13	B
PCP Seal Cooler Coils, SIRWT HX Tubes	Fluid Pressure Boundary Heat Transfer	Stainless Steel	Treated Water (Int)	Cracking	Water Chemistry Program	V.D1.5-a	3.2.1-13	H

Table 3.2.2-1 Engineered Safety Features - Engineering Safeguards System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
PCP Seal Cooler Coils	Fluid Pressure Boundary	Stainless Steel	Treated Water (Int)	Cracking	Closed Cycle Cooling Water Program	V.D1.5-a	3.2.1-13	H
				Loss of Material	Closed Cycle Cooling Water Program	V.D1.5-a	3.2.1-13	B
SIRWT HX Tubes	Fluid Pressure Boundary Heat Transfer	Stainless Steel	Treated Water (Int)	Cracking	One-Time Inspection Program	V.D1.6-a	3.2.1-13	G
Cont. Spray Pump Coils, SDC HX Tubes	Fluid Pressure Boundary Heat Transfer	Stainless Steel	Treated Water (Int)	Heat Transfer Degradation	Water Chemistry Program	V.D1.5-a	3.2.1-13	H
				Loss of Material	Water Chemistry Program	V.D1.5-a	3.2.1-13	A, 204
SIRWT HX Tubes, Cont. Spray Pump coils, LPSI Pump coils, SDC HX Tubes	Fluid Pressure Boundary	Stainless Steel	Treated Water (Int)	Heat Transfer Degradation	One-Time Inspection Program	V.D1.5-a	3.2.1-13	H
				Loss of Material	One-Time Inspection Program	V.D1.6-a	3.2.1-13	G
Fasteners	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext) Containment Air (Ext)	Loss of Preload	Bolting Integrity Program			206, F
				Loss of Material	Bolting Integrity Program	V.E.2-a	3.2.1-18	A
		Stainless Steel	Plant Indoor Air (Ext) Containment Air (Ext)	Loss of Preload	Bolting Integrity Program			206, F

Table 3.2.2-1 Engineered Safety Features - Engineering Safeguards System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Containment Spray System Fasteners	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext) Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	V.A.1-b	3.2.1-17	A
Containment Spray Pump Bolting	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	V.A.3-b	3.2.1-17	A
Containment Spray System Valves Bolting	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext) Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	V.A.4-b	3.2.1-17	A
Containment Spray System Valves Header Bolting	Fluid Pressure Boundary	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	V.A.5-b	3.2.1-15	A
HPSI, LPSI Pumps Bolting	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	V.D1.2-b	3.2.1-17	A
SIRWT HX Bolting	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	V.D1.6-d	3.2.1-17	A
SIRWT Bolting	Fluid Pressure Boundary	Carbon Steel	Atmosphere/ Weather (Ext)	Loss of Material	Boric Acid Corrosion Program	V.D1.8-b	3.2.1-17	A
HPSI Check Valves, SDC from PCS MOVs	Fluid Pressure Boundary	Cast Austenitic SS	Treated Water (Int)	Cracking	Water Chemistry Program			G, 207
					One-Time Inspection Program			G, 207
				Reduction of Fracture Toughness	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	V.D1.1-b	3.2.1-11	E, 209

Table 3.2.2-1 Engineered Safety Features - Engineering Safeguards System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
HPSI Check Valves	Fluid Pressure Boundary	Cast Austenitic SS	Treated Water (Int)	Loss of Material	Water Chemistry Program			G, 207
					One-Time Inspection Program			G, 207
Hot Leg Injection Check Valves	Fluid Pressure Boundary	Cast Austenitic SS	Treated Water (Int)	Cracking	Water Chemistry Program			G, 207
					One-Time Inspection Program			G, 207
HPSI Check Valves Loops 1A, 2A	Fluid Pressure Boundary	Stainless Steel	Treated Water (Int)	Loss of Material	Water Chemistry Program			G, 207
					One-Time Inspection Program			G, 207

Notes for Table 3.2.2-1

A Consistent with NUREG-1801 item for component, material, environment, and aging management program. AMP is consistent with NUREG-1801 AMP.

B Consistent with NUREG-1801 item for component, material, environment, and aging management program. AMP has exceptions to NUREG-1801 AMP.

C Component is different, but consistent with NUREG-1801 for material, environment, aging effect and AMP. AMP is consistent with NUREG-1801 AMP.

D Component is different, but consistent with NUREG-1801 for material, environment, aging effect and AMP. AMP has exceptions to NUREG-1801 AMP.

E Consistent with NUREG-1801 for material, environment, aging effect but a different AMP is credited.

F Material not in NUREG-1801 for this component.

G Environment not in NUREG-1801 for this component and material.

H Aging effect not in NUREG-1801 for this component, material and environment combination.

I Aging effect in NUREG-1801 for this component, material and environment combination is not applicable.

J Neither the component nor the material and environment combination is evaluated in NUREG-1801.

Plant-specific notes:

201 Not used

202 Not used

203 Not used

204 GALL addresses crevice and pitting corrosion; Palisades included crevice, pitting and MIC.

205 GALL has crevice and pitting corrosion. Palisades included also general corrosion

206 Loss of preload is included here in response to recent NRC RAIs on non-primary system, high temperature bolting that may experience loss of preload. The Palisades Bolting Integrity Program manages potential bolting AERMs and event driven degradation. GALL reconciliation is based on Loss of Material.

207 The GALL assumes a temperature for valves in this system to be less than 93 degrees C (<200 degrees F) whereas these valves have the potential to be >93 degrees C (>200 degrees F)

208 GALL has crevice and pitting corrosion. Palisades has included crevice, pitting, fretting, and MIC.

209 GALL credits XI.M12 Thermal Aging of CASS. Palisades credits ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program

210 GALL evaluates RWT as stainless steel only. Palisades SIRW Tank is aluminum.

211 GALL evaluates open or closed cooling water only. Palisades has steam environment.

212 GALL addresses pitting and crevice corrosion. Palisades includes general corrosion and selective leaching.

3.3 Aging Management of Auxiliary Systems

3.3.1 Introduction

This section provides the results of the aging management review for those components identified in Section 2.3.3, Auxiliary Systems, as being subject to aging management review. The systems, or portions of systems, which are addressed in this section, are described in the indicated sections.

- Chemical Addition System (Section 2.3.3.17)
- Chemical Volume and Control System (Section 2.3.3.1)
- Circulating Water System (Section 2.3.3.2)
- Component Cooling Water System (Section 2.3.3.3)
- Compressed Air System (Section 2.3.3.4)
- Containment Air Recirculation and Cooling System (Section 2.3.3.5)
- Domestic Water System (Section 2.3.3.16)
- Emergency Power System (Section 2.3.3.6)
- Fire Protection System (Section 2.3.3.7)
- Fuel Oil System (Section 2.3.3.8)
- Heating, Ventilation, and Air Conditioning System (Section 2.3.3.9)
- Miscellaneous Gas System (Section 2.3.3.10)
- Radwaste System (Section 2.3.3.11)
- Service Water System (Section 2.3.3.12)
- Shield Cooling System (Section 2.3.3.13)
- Spent Fuel Pool Cooling System (Section 2.3.3.14)
- Waste Gas System (Section 2.3.3.15)

Table 3.3.1, Summary of Aging Management Evaluations in Chapter VII of NUREG-1801 for Auxiliary Systems, provides the summary of the programs evaluated in NUREG-1801 for the Auxiliary Systems component groups that are relied on for license renewal.

This table uses the format described in Figure 3.0-1 above. Note that this table only includes those component groups that are applicable to a PWR.

3.3.2 Results

The following tables summarize the results of the aging management review for systems in the Auxiliary Systems group:

Table 3.3.2-1, Auxiliary Systems - Chemical and Volume Control System - Summary of Aging Management Evaluation

Table 3.3.2-2, Auxiliary Systems - Circulating Water System - Summary of Aging Management Evaluation

Table 3.3.2-3, Auxiliary Systems - Component Cooling Water System - Summary of Aging Management Evaluation

Table 3.3.2-4, Auxiliary Systems - Compressed Air System - Summary of Aging Management Evaluation

Table 3.3.2-5, Auxiliary Systems - Containment Air Recirculation and Cooling System - Summary of Aging Management Evaluation

Table 3.3.2-6, Auxiliary Systems - Emergency Power System - Summary of Aging Management Evaluation

Table 3.3.2-7, Auxiliary Systems - Fire Protection System - Summary of Aging Management Evaluation

Table 3.3.2-8, Auxiliary Systems - Fuel Oil System - Summary of Aging Management Evaluation

Table 3.3.2-9, Auxiliary Systems - Heating, Ventilation, and Air Conditioning System - Summary of Aging Management Evaluation

Table 3.3.2-10, Auxiliary Systems - Miscellaneous Gas - Summary of Aging Management Evaluation

Table 3.3.2-11, Auxiliary Systems - Radwaste System - Summary of Aging Management Evaluation

Table 3.3.2-12, Auxiliary Systems - Service Water System - Summary of Aging Management Evaluation

Table 3.3.2-13, Auxiliary Systems - Shield Cooling System - Summary of Aging Management Evaluation

Table 3.3.2-14, Auxiliary Systems - Spent Fuel Pool Cooling System - Summary of Aging Management Evaluation

Table 3.3.2-15, Auxiliary Systems - Waste Gas System - Summary of Aging Management Evaluation

Table 3.3.2-16, Auxiliary Systems - Domestic Water System - Summary of Aging Management Evaluation

Table 3.3.2-17, Auxiliary Systems - Chemical Addition System - Summary of Aging Management Evaluation

The materials that specific components are fabricated from, the environments to which components are exposed, the aging effects requiring management, and the aging management programs used to manage these aging effects are provided for each of the above systems in the following subsections of Section 3.3.2.1, Materials, Environment, Aging Effects Requiring Management and Aging Management Programs:

Section 3.3.2.1.1, Chemical and Volume Control System

Section 3.3.2.1.2, Circulating Water System

Section 3.3.2.1.3, Component Cooling Water System

Section 3.3.2.1.4, Compressed Air System

Section 3.3.2.1.5, Containment Air Recirculation and Cooling System

Section 3.3.2.1.6, Emergency Power System

Section 3.3.2.1.7, Fire Protection System

Section 3.3.2.1.8, Fuel Oil System

Section 3.3.2.1.9, Heating, Ventilation, and Air Conditioning System

Section 3.3.2.1.10, Miscellaneous Gas System

Section 3.3.2.1.11, Radwaste System

Section 3.3.2.1.12, Service Water System

Section 3.3.2.1.13, Shield Cooling System

Section 3.3.2.1.14, Spent Fuel Pool Cooling System

Section 3.3.2.1.15, Waste Gas System

Section 3.3.2.1.16, Domestic Water System

Section 3.3.2.1.17, Chemical Addition System

3.3.2.1 Materials, Environment, Aging Effects Requiring Management and Aging Management Programs

3.3.2.1.1 Chemical and Volume Control System

Materials

The materials of construction for the Chemical and Volume Control System components are:

- Brass
- Carbon Steel
- Cast Austenitic SS
- Copper Nickel
- Low-Alloy Steel
- Stainless Steel

Environment

The Chemical and Volume Control System components are exposed to the following environments:

- Air (Ext)
- Air (Int)
- Containment Air (Ext)
- Gas (Ext)
- Oil (Int)
- Plant Indoor Air (Ext)
- Treated Water (Ext)
- Treated Water (Int)

Aging Effects Requiring Management

The following aging effects, associated with the Chemical and Volume Control System, require management:

- Cracking
- Loss of Material
- Loss of Preload
- Reduction of Fracture Toughness

Aging Management Programs

The following aging management programs manage the aging effects for the Chemical and Volume Control System components:

- ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program
- Boric Acid Corrosion Program
- Bolting Integrity Program
- Closed Cycle Cooling Water Program
- One-Time Inspection Program
- System Monitoring Program
- Water Chemistry Program

3.3.2.1.2 Circulating Water System

Materials

The materials of construction for the Circulating Water System components are:

- Carbon Steel
- Cast Iron
- Low-Alloy Steel

Environment

The Circulating Water System components are exposed to the following environments:

- Plant Indoor Air (Ext)
- Raw Water (Int)

Aging Effects Requiring Management

The following aging effects, associated with the Circulating Water System, require management:

- Loss of Material
- Loss of Preload

Aging Management Programs

The following aging management programs manage the aging effects for the Circulating Water System components:

- Bolting Integrity Program

- One-Time Inspection Program
- Open Cycle Cooling Water Program
- System Monitoring Program

3.3.2.1.3 Component Cooling Water System

Materials

The materials of construction for the Component Cooling Water System components are:

- Aluminum Bronze
- Brass
- Bronze
- Carbon Steel
- Cast Iron
- Copper Alloys
- Copper Nickel
- Nickel-Based Alloys
- Stainless Steel

Environment

The Component Cooling Water System components are exposed to the following environments:

- Air (Int)
- Containment Air (Ext)
- Gas (Int)
- Plant Indoor Air (Ext)
- Raw Water (Ext)
- Raw Water (Int)
- Treated Water (Ext)
- Treated Water (Int)

Aging Effects Requiring Management

The following aging effects, associated with the Component Cooling Water System, require management:

- Buildup of Deposit
- Cracking
- Heat Transfer Degradation
- Loss of Material
- Loss of Preload

Aging Management Programs

The following aging management programs manage the aging effects for the Component Cooling Water System components:

- Bolting Integrity Program
- Boric Acid Corrosion Program
- Closed Cycle Cooling Water Program
- One-Time Inspection Program
- Open Cycle Cooling Water Program
- System Monitoring Program
- Water Chemistry Program

3.3.2.1.4 Compressed Air System

Materials

The materials of construction for the Compressed Air System components are:

- Aluminum
- Brass
- Bronze
- Carbon Steel
- Cast Iron
- Copper Alloys
- Galvanized
- Stainless Steel

Environment

The Compressed Air System components are exposed to the following environments:

- Air (Ext)
- Air (Int)
- Containment Air (Ext)
- Oil (Int)
- Plant Indoor Air (Ext)
- Raw Water (Ext)
- Raw Water (Int)

Aging Effects Requiring Management

The following aging effects, associated with the Compressed Air System, require management:

- Cracking
- Loss of Material
- Loss of Preload

Aging Management Programs

The following aging management programs manage the aging effects for the Compressed Air System components:

- Bolting Integrity Program
- Boric Acid Corrosion Program
- One-Time Inspection Program
- Open Cycle Cooling Water Program
- System Monitoring Program

3.3.2.1.5 Containment Air Recirculation and Cooling System

Materials

The materials of construction for the Containment Air Recirculation and Cooling System components are:

- Carbon Steel
- Copper Alloys

- Galvanized
- Low-Alloy Steel
- Stainless Steel

Environment

The Containment Air Recirculation and Cooling System components are exposed to the following environments:

- Air (Ext)
- Air (Int)
- Containment Air (Ext)
- Raw Water (Int)

Aging Effects Requiring Management

The following aging effects, associated with the Containment Air Recirculation and Cooling System, require management:

- Buildup of Deposit
- Heat Transfer Degradation
- Loss of Material
- Loss of Preload

Aging Management Programs

The following aging management programs manage the aging effects for the Containment Air Recirculation and Cooling System components:

- Bolting Integrity Program
- Boric Acid Corrosion Program
- One-Time Inspection Program
- Open Cycle Cooling Water Program
- System Monitoring Program

3.3.2.1.6 Emergency Power System

Materials

The materials of construction for the Emergency Power System components are:

- Brass

- Bronze
- Carbon Steel
- Cast Iron
- Copper Alloys
- Galvanized
- Low-Alloy Steel
- Stainless Steel

Environment

The Emergency Power System components are exposed to the following environments:

- Air (Ext)
- Air (Int)
- Atmosphere/ Weather (Ext)
- Oil (Ext)
- Oil (Int)
- Plant Indoor Air (Ext)
- Raw Water (Int)
- Treated Water (Ext)
- Treated Water (Int)

Aging Effects Requiring Management

The following aging effects, associated with the Emergency Power System, require management:

- Buildup of Deposit
- Cracking
- Heat Transfer Degradation
- Loss of Material
- Loss of Preload

Aging Management Programs

The following aging management programs manage the aging effects for the Emergency Power System components:

- Bolting Integrity Program
- Closed Cycle Cooling Water Program
- Diesel Fuel Monitoring and Storage Program
- Fire Protection Program
- One-Time Inspection Program
- Open Cycle Cooling Water Program
- System Monitoring Program

3.3.2.1.7 Fire Protection System

Materials

The materials of construction for the Fire Protection System components are:

- Bare Copper
- Brass
- Bronze
- Carbon Steel
- Cast Iron
- Copper Alloys
- Stainless Steel

Environment

The Fire Protection System components are exposed to the following environments:

- Air (Ext)
- Air (Int)
- Atmosphere/Weather (Ext)
- Containment Air (Ext)
- Oil (Int)
- Plant Indoor Air (Ext)
- Raw Water (Ext)

- Raw Water (Int)
- Soil (Ext)

Aging Effects Requiring Management

The following aging effects, associated with the Fire Protection System, require management:

- Buildup of Deposit
- Loss of Material
- Loss of Preload

Aging Management Programs

The following aging management programs manage the aging effects for the Fire Protection System components:

- Bolting Integrity Program
- Boric Acid Corrosion Program
- Fire Protection Program
- One-Time Inspection Program
- System Monitoring Program

3.3.2.1.8 Fuel Oil System

Materials

The materials of construction for the Fuel Oil System are:

- Brass
- Bronze
- Carbon Steel
- Cast Iron
- Copper Alloys
- Low-Alloy Steel
- Stainless Steel

Environment

The Fuel Oil System components are exposed to the following environments:

- Air (Int)

- Atmosphere/Weather (Ext)
- Oil (Int)
- Plant Indoor Air (Ext)
- Raw Water (Ext)

Aging Effects Requiring Management

The following aging effects, associated with the Fuel Oil System, require management:

- Cracking
- Loss of Material
- Loss of Preload

Aging Management Programs

The following aging management programs manage the aging effects for the Fuel Oil System components:

- Bolting Integrity Program
- Buried Services Corrosion Monitoring Program
- Diesel Fuel Monitoring and Storage Program
- Fire Protection Program
- One-Time Inspection Program
- System Monitoring Program

3.3.2.1.9 Heating, Ventilation, and Air Conditioning System

Materials

The materials of construction for the Heating, Ventilation, and Air Conditioning System components are:

- Bronze
- Carbon Steel
- Cast Iron
- Copper Alloys
- Elastomers
- Galvanized

- Low-Alloy Steel
- Stainless Steel

Environment

The Heating, Ventilation, and Air Conditioning System components are exposed to the following environments:

- Air (Ext)
- Air (Int)
- Containment Air (Ext)
- Gas (Ext)
- Oil (Int)
- Plant Indoor Air (Ext)
- Raw Water (Int)
- Steam (Int)

Aging Effects Requiring Management

The following aging effects, associated with the Heating, Ventilation, and Air Conditioning System, require management:

- Buildup of Deposit
- Change in Material Properties
- Cracking
- Heat Transfer Degradation
- Loss of Material
- Loss of Preload

Aging Management Programs

The following aging management programs manage the aging effects for the Heating, Ventilation, and Air Conditioning System components:

- Bolting Integrity Program
- Boric Acid Corrosion Program
- One-Time Inspection Program
- Open Cycle Cooling Water Program
- System Monitoring Program

3.3.2.1.10 Miscellaneous Gas System

Materials

The materials of construction for the Miscellaneous Gas System are:

- Brass
- Bronze
- Carbon Steel
- Copper Alloys
- Galvanized
- Low-Alloy Steel
- Stainless Steel

Environment

The Miscellaneous Gas System components are exposed to the following environments:

- Air (Int)
- Atmosphere/Weather (Ext)
- Containment Air (Ext)
- Plant Indoor Air (Ext)
- Soil (Ext)

Aging Effects Requiring Management

The following aging effects, associated with the Miscellaneous Gas System, require management:

- Loss of Material
- Loss of Preload

Aging Management Programs

The following aging management programs manage the aging effects for the Miscellaneous Gas System components:

- Bolting Integrity Program
- Boric Acid Corrosion Program
- Buried Services Corrosion Monitoring Program

- One-Time Inspection Program
- System Monitoring Program

3.3.2.1.11 Radwaste System

Materials

The materials of construction for the Radwaste System components are:

- Bronze
- Carbon Steel
- Cast Austenitic SS
- Cast Iron
- Copper Alloys
- Stainless Steel

Environment

The Radwaste System components are exposed to the following environments:

- Air (Int)
- Atmosphere/Weather (Ext)
- Containment Air (Ext)
- Gas (Int)
- Plant Indoor Air (Ext)
- Raw Water (Ext)
- Raw Water (Int)
- Treated Water (Ext)
- Treated Water (Int)

Aging Effects Requiring Management

The following aging effects, associated with the Radwaste System, require management:

- Loss of Material
- Loss of Preload

Aging Management Programs

The following aging management programs manage the aging effects for the Radwaste System components:

- Bolting Integrity Program
- Boric Acid Corrosion Program
- Closed Cycle Cooling Water Program
- One-Time Inspection Program
- System Monitoring Program

3.3.2.1.12 Service Water System

Materials

The materials of construction for the Service Water System components are:

- Aluminum Bronze
- Brass
- Bronze
- Carbon Steel
- Cast Iron
- Copper Alloys
- Low-Alloy Steel
- Rubber
- Stainless Steel

Environment

The Service Water System components are exposed to the following environments:

- Air (Int)
- Containment Air (Ext)
- Soil (Ext)
- Gas (Int)
- Oil (Int)
- Plant Indoor Air (Ext)
- Raw Water (Ext)

- Raw Water (Int)
- Treated Water (Int)

Aging Effects Requiring Management

The following aging effects, associated with the Service Water System, require management:

- Buildup of Deposit
- Cracking (of rubber)
- Loss of Material
- Loss of Preload

Aging Management Programs

The following aging management programs manage the aging effects for the Service Water System components:

- Bolting Integrity Program
- Boric Acid Corrosion Program
- Buried Services Corrosion Monitoring Program
- One-Time Inspection Program
- Open Cycle Cooling Water Program
- System Monitoring Program

3.3.2.1.13 Shield Cooling System

Materials

The materials of construction for the Shield Cooling System are:

- Carbon Steel

Environment

The Shield Cooling System components are exposed to the following environments:

- Containment Air (Ext)
- Plant Indoor Air (Ext)
- Treated Water (Int)

Aging Effects Requiring Management

The following aging effects, associated with the Shield Cooling System, require management:

- Loss of Material
- Loss of Preload

Aging Management Programs

The following aging management programs manage the aging effects for the Shield Cooling System components:

- Bolting Integrity Program
- Boric Acid Corrosion Program
- Closed Cycle Cooling Water Program
- System Monitoring Program

3.3.2.1.14 Spent Fuel Pool Cooling System

Materials

The materials of construction for the Spent Fuel Pool Cooling System components are:

- Carbon Steel
- Stainless Steel

Environment

The Spent Fuel Pool Cooling System components are exposed to the following environments:

- Plant Indoor Air (Ext)
- Treated Water (Ext)
- Treated Water (Int)

Aging Effects Requiring Management

The following aging effects, associated with the Spent Fuel Pool Cooling System, require management:

- Heat Transfer Degradation
- Loss of Material
- Loss of Preload

Aging Management Programs

The following aging management programs manage the aging effects for the Spent Fuel Pool Cooling System components:

- Bolting Integrity Program
- Boric Acid Corrosion Program
- Closed Cycle Cooling Water Program
- One-Time Inspection Program
- System Monitoring Program
- Water Chemistry Program

3.3.2.1.15 Waste Gas System

Materials

The materials of construction for the Waste Gas System are:

- Carbon Steel
- Low-Alloy Steel

Environment

The Waste Gas System components are exposed to the following environments:

- Containment Air (Ext)
- Gas (Int)
- Plant Indoor Air (Ext)
- Raw Water (Int)

Aging Effects Requiring Management

The following aging effects, associated with the Waste Gas System, require management:

- Loss of Material
- Loss of Preload

Aging Management Programs

The following aging management programs manage the aging effects for the Waste Gas System components:

- Bolting Integrity Program

- Boric Acid Corrosion Program
- One-Time Inspection Program
- System Monitoring Program

3.3.2.1.16 Domestic Water System

Materials

The materials of construction for the Domestic Water System are:

- Carbon Steel
- Cast Iron
- Copper Alloy
- Low Alloy Steel
- Stainless Steel

Environment

The Domestic Water System components are exposed to the following environments:

- Air (Int)
- Plant Indoor Air (Ext)
- Raw Water (Int)
- Steam (Int)
- Treated Water (Int)

Aging Effects Requiring Management

The following aging effects, associated with the Domestic Water System, require management:

- Loss of Material

Aging Management Programs

The following aging management programs manage the aging effects for the Domestic Water System components:

- Bolting Integrity Program
- One-Time Inspection Program
- System Monitoring Program

3.3.2.1.17 Chemical Addition System

Materials

The materials of construction for the Chemical Addition System are:

- Carbon Steel
- Stainless Steel

Environment

The Chemical Addition System components are exposed to the following environments:

- Air (Int)
- Plant Indoor Air (Ext)
- Treated Water (Int)

Aging Effects Requiring Management

The following aging effects, associated with the Chemical Addition System, require management:

- Loss of Material

Aging Management Programs

The following aging management programs manage the aging effects for the Chemical Addition System components:

- One-Time Inspection Program
- System Monitoring Program

3.3.2.2 Further Evaluation of Aging Management as Recommended by NUREG-1801

NUREG-1801 Volume 1 Tables provide the basis for identifying those programs that warrant further evaluation by the reviewer in the license renewal application. For the Auxiliary Systems, those programs are addressed in the following sections.

3.3.2.2.1 Loss of Material due to General, Pitting, and Crevice Corrosion

Paragraph 1 of this NUREG-1800 item states that loss of material due to general, pitting, and crevice corrosion could occur in the channel head and access cover, tubes, and tubesheets of the heat exchanger in the spent fuel pool cooling and cleanup. The water chemistry program relies on monitoring and control of reactor water chemistry based on EPRI guidelines of TR-105714 for primary water chemistry in PWRs, and TR-102134 for secondary water chemistry in PWRs to manage the effects of loss of material from general, pitting or crevice corrosion. However, high concentrations of impurities at crevices and locations of stagnant flow conditions could cause general, pitting, or crevice corrosion. Therefore, verification of the effectiveness of the chemistry control program should be performed to ensure that corrosion is not occurring. NUREG-1801 recommends further evaluation of programs to manage loss of material from general, pitting, and crevice corrosion to verify the effectiveness of the water chemistry program.A one-time inspection of select components at susceptible locations is an acceptable method to ensure that corrosion is not occurring and the component's intended function will be maintained during the period of extended operation.

Paragraph 2 states that loss of material due to pitting and crevice corrosion could occur in the filter housing, valve bodies, and nozzles of the ion exchanger in the spent fuel pool cooling and cleanup system (PWR). The water chemistry program relies on monitoring and control of reactor water chemistry based on EPRI guidelines of TR-105714 (Ref. 4) for primary water chemistry in PWRs, and TR-102134 (Ref. 5) for secondary water chemistry in PWRs to manage the effects of loss of material from pitting or crevice corrosion. However, high concentrations of impurities at crevices and locations of stagnant flow conditions could cause pitting, or crevice corrosion. Therefore, verification of the effectiveness of the chemistry control program should be performed to ensure that corrosion is not occurring. NUREG-1801 recommends further evaluation of programs to manage loss of material from pitting and crevice corrosion to verify the effectiveness of the water chemistry program. A one-time inspection of select components at susceptible locations is an acceptable method to ensure that corrosion is not occurring and that the component's intended function will be maintained during the period of extended operation.

The materials of the spent fuel pool cooling components evaluated in NUREG-1801 are carbon steels with elastomer linings, which differ from the stainless steel Spent Fuel Pool (SFP) Cooling System components of Palisades. This aging effect is not applicable to the Palisades spent Fuel Pool Cooling System.

3.3.2.2.2 Hardening and Cracking or Loss of Strength due to Elastomer Degradation or Loss of Material due to Wear

NUREG-1800 states that hardening and cracking due to elastomer degradation could occur in elastomer linings of the filter, valve, and ion exchangers in spent fuel pool cooling and cleanup systems (BWR and PWR). Hardening and loss of strength due to elastomer degradation could occur in the collars and seals of the duct and in the elastomer seals of the filters in the control room area, auxiliary and radwaste area, and primary containment heating ventilation systems and in the collars and seals of the duct in the diesel generator building ventilation system. Loss of material due to wear could occur in the collars and seals of the duct in the ventilation systems. NUREG-1801 recommends further evaluation to ensure that these aging effects are adequately managed.

This paragraph of NUREG 1800 describes the potential for degradation of elastomers in collars and seals in spent fuel cooling systems and ventilation systems. The materials of the spent fuel pool cooling components subject to aging management review have no elastomers. For the ventilation systems, elastomers are evaluated for cracking and changes of material properties due to thermal and radiation exposure. The System Monitoring Program manages degradation of elastomers at the seals and flexible connections. Elastomers are used in other systems. For those systems, management of elastomer degradation is provided by the System Monitoring Program.

3.3.2.2.3 Cumulative Fatigue Damage

NUREG-1800 states that fatigue of Reactor Pressure Boundary components is a TLAA as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c). The evaluation of this TLAA is addressed separately in Section 4.3.

3.3.2.2.4 Crack Initiation and Growth due to Cracking or Stress Corrosion Cracking

NUREG-1800 states that crack initiation and growth due to SCC could occur due to cracking in the high-pressure pump in the chemical and volume control system (PWR). NUREG-1801 recommends further evaluation to ensure that these aging effects are managed adequately.

This paragraph of NUREG -1800, which discusses cracking in the high-pressure pump is not applicable to Palisades because the pump temperature is <140°F, which is below the temperature threshold required to support cracking.

3.3.2.2.5 Loss of Material due to General, Microbiologically Influenced, Pitting, and Crevice Corrosion

NUREG-1800 states that loss of material due to general, pitting, and crevice corrosion could occur in the piping and filter housing and supports in the control room area, the auxiliary and radwaste area, the primary containment heating and ventilation systems, in the piping of the diesel generator building ventilation system, in the aboveground piping and fittings, valves, and pumps in the diesel fuel oil system and in the diesel engine starting air, combustion air intake, and combustion air exhaust subsystems in the emergency diesel generator system. Loss of material due to general, pitting, crevice, and microbiologically influenced corrosion (MIC) could occur in the duct fittings, access doors, and closure bolts, equipment frames and housing of the duct, due to pitting and crevice corrosion could occur in the heating/cooling coils of the air handler heating/cooling, and due to general corrosion could occur on the external surfaces of all carbon steel structures and components, including bolting exposed to operating temperatures less than 212°F in the ventilation systems. NUREG-1801 recommends further evaluation to ensure that these aging effects are adequately managed.

This paragraph of NUREG -1800 discusses the loss of material from corrosion that could occur on internal and external surfaces of components exposed to plant indoor air and atmosphere and weather. Specifically included are the ventilation systems, diesel fuel oil, emergency diesel starting air, and combustion air intake and exhaust systems, and the external carbon steel surfaces of auxiliary systems.

For the internal environments of applicable Auxiliary Systems, the Open Cycle Cooling Water Program, Diesel Fuel Monitoring and Storage Program, One-Time Inspection Program (including tank internal inspection), and Fire Protection Program are credited for managing the aging effect of loss of material.

For the external surfaces of all carbon steel components in Auxiliary Systems, the System Monitoring Program is credited for managing the aging effect of loss of material. The Open Cycle Cooling Water Program and the Fire Protection Program are credited to augment the System Monitoring Program for managing external aging effects in Service Water and Fire Protection

System, respectively. Closure bolting is managed by the Bolting Integrity Program.

As discussed in the respective program sections of Appendix B, each of these programs provides reasonable assurance that aging effects will be managed such that SSCs within the scope of each program will continue to perform their intended functions consistent with the current licensing bases for the period of extended operation.

3.3.2.2.6 Loss of Material due to General, Galvanic, Pitting, and Crevice Corrosion

NUREG-1800 states that loss of material due to general, galvanic, pitting, and crevice corrosion could occur in tanks, piping, valve bodies, and tubing in the reactor coolant pump oil collection system in fire protection. The fire protection program relies on a combination of visual and volumetric examinations in accordance with the guidelines of 10 CFR Part 50 Appendix R and Branch Technical Position 9.5-1 to manage loss of material from corrosion. However, corrosion may occur at locations where water from wash downs may accumulate. Therefore, verification of the effectiveness of the program should be performed to ensure that corrosion is not occurring. NUREG-1801 recommends further evaluation of programs to manage loss of material due to general, galvanic, pitting, and crevice corrosion to verify the effectiveness of the program. A one-time inspection of the bottom half of the interior surface of the tank of the reactor coolant pump oil collection system is an acceptable method to ensure that corrosion is not occurring and that the component's intended function will be maintained during the period of extended operation.

The Palisades Fire Protection Program has a separate oil collection system for each Primary Coolant Pump (PCP). NUREG-1801 assumes that the reactor coolant pump oil collection tanks and piping is constructed from carbon steel, copper and stainless steel alloys. However, the Palisades Plant oil collection tank, drip pans and oil lift pump enclosures are stainless steel. Annealed copper tubing is used to connect these components and direct the path of any oil leakage from the PCP motor. These materials are located inside containment. They are normally exposed to a Containment Air internal and external environment. Any oil that is collected will be lubricating oil without any water entrainment. Loss of material due to general and galvanic corrosion is not applicable to this oil collection system. If water condensed from air or spray from some water source were to enter the system then crevice and pitting corrosion could be an aging effect. A One-Time Inspection of the tank and piping is credited to ensure that loss of material due to crevice and pitting corrosion are not occurring.

3.3.2.2.7 Loss of Material due to General, Pitting, Crevice, and Microbiologically Influenced Corrosion and Biofouling

NUREG-1800 states that loss of material due to general, pitting, and crevice corrosion, MIC, and biofouling could occur in the internal surface of tanks in the diesel fuel oil system and due to general, pitting, and crevice corrosion and MIC in the tanks of the diesel fuel oil system in the emergency diesel generator system. The existing aging management program relies on the fuel oil chemistry program for monitoring and control of fuel oil contamination in accordance with the guidelines of ASTM Standards D4057, D1796, D2709 and D2276 to manage loss of material due to corrosion or biofouling. Corrosion or biofouling may occur at locations where contaminants accumulate. Verification of the effectiveness of the chemistry control program should be performed to ensure that corrosion is not occurring. NUREG-1801 recommends further evaluation of programs to manage corrosion/biofouling to verify the effectiveness of the program. A one-time inspection of selected components at susceptible locations is an acceptable method to ensure that corrosion is not occurring and that the component's intended function will be maintained during the period of extended operation.

The Diesel Fuel Monitoring and Storage Program manages components in the Fuel Oil System associated with the Emergency Diesel Generators and Diesel Fire Pumps, including storage tanks, day tanks, piping, valve bodies, and other passive components rely on the Diesel Fuel Monitoring and Storage Program to minimize potential for degradation and loss of intended function. The program manages the conditions that would cause general, pitting and microbiological influenced corrosion (MIC) of the diesel fuel tank internal surfaces.

The One-Time Inspection Program provides reasonable assurance that aging effects will be managed such that SSCs within the scope of this program will continue to perform their intended functions consistent with the current licensing bases for the period of extended operation.

3.3.2.2.8 Quality Assurance for Aging Management of Non-Safety Related Components

Quality Assurance Program applicability to non-safety-related components is addressed in Appendix B, Section 1.2.

3.3.2.2.9 Crack Initiation and Growth due to Stress Corrosion Cracking and Cyclic Loading

NUREG-1800 states that crack initiation and growth due to SCC and cyclic loading could occur in the channel head and access cover, tube sheet, tubes, shell and access cover, and closure bolting of the regenerative heat exchanger and in the channel head and access cover, tube sheet, and tubes of the letdown heat exchanger in the chemical and volume control system (PWR). The water chemistry program relies on monitoring and control of water chemistry based on the guidelines of TR-105714 for primary water chemistry in PWRs to manage the effects of crack initiation and growth due to SCC and cyclic loading. Verification of the effectiveness of the chemistry control program should be performed to ensure that crack initiation and growth are not occurring. NUREG-1801 recommends further evaluation to manage crack initiation and growth from SCC and cyclic loading for these systems to verify the effectiveness of the water chemistry program. A one-time inspection of select components and susceptible locations is an acceptable method to ensure that crack initiation and growth are not occurring and that the components' intended function will be maintained during extended operations.

The Water Chemistry Program is credited for managing aging effects such as loss-of-material due to general, pitting and crevice corrosion, MIC, cracking due to SCC, and fouling due to corrosion product buildup in stagnant and low flow regions, by controlling the environment to which internal surfaces of systems and components are exposed. The aging effects are minimized by controlling the chemical species that cause the underlying mechanisms that result in these aging effects. The program provides assurance that an elevated level of contaminants and oxygen does not exist in the systems and components covered by the program, thus minimizing the occurrences of aging effects, and maintaining components ability to perform their intended functions. The program is based on the guidelines in EPRI TR-105714, Rev. 5 and TR-102134, Rev. 5.

The One-Time Inspection Program is credited by the Water Chemistry Program and its corresponding GALL Report section XI.M2 for managing the effects of aging in stagnant or low-flow areas of components. The Water Chemistry Control Program may be credited for managing aging effects for components where the water flow is low or stagnant conditions exist. However water chemistry sampling points may not be indicative of conditions in stagnant or low-flow locations. Therefore, confirmatory inspections are appropriate. Accordingly, to ensure that significant degradation is not occurring and to ensure that the component will continue to perform its intended function during the period of extended operation, a one-time inspection of selected components will be performed.

3.3.2.2.10 Reduction of Neutron-Absorbing Capacity and Loss of Material due to General Corrosion

NUREG-1800 states that reduction of neutron-absorbing capacity and loss of material due to general corrosion could occur in the neutron-absorbing sheets of the spent fuel storage rack in the spent fuel storage. NUREG-1801 recommends further evaluation to ensure that these aging effects are adequately managed.

Palisades FSAR Section 9.11.3.2, “Modified Spent Fuel Storage,” provides a discussion of the spent fuel storage racks in the Palisades spent fuel pool. It describes two regions, as follows: “Region 1 contains racks in the spent fuel pool having a 10.25-inch center-to-center spacing and a single rack in the North Tilt Pit having 11.25-inch x 10.69-inch center-to-center spacing. Region 2 contains racks in both the spent fuel pool and North Tilt Pit having 9.17-inch center-to-center spacing. Because of the larger center-to-center spacing, and the poison (B_{10}) concentration of Region 1 cells, Region I (NUS) spent fuel storage racks can accommodate fuel assemblies having a maximum planar average U-235 enrichment of 4.95 weight percent. This assures the fuel enrichment limit assumed in the spent fuel analyses will not be exceeded.

“The Region 2 racks contain a neutron absorbing material, boraflex, manufactured by the Brand Industrial Services, Inc., and fabricated to the Nuclear Criteria of 10 CFR 50, Appendix B. Boraflex is a silicone-based polymer containing fine particles of boron carbide in a homogeneous matrix. The boraflex used in the Region 2 racks contains a minimum B_{10} areal density of 0.006 gm/cm^2. Since the long-term stability of boraflex has not yet been resolved, the Region 2 criticality calculations were revised to eliminate any credit taken for criticality control due to the presence of Boraflex.”

FSAR Section 9.11.3.4, “Prevention of Criticality During Transfer and Storage,” further clarifies that: “The Region 2 racks are designed for a 9.17 inch center-to-center spacing with boraflex sheets used as a neutron absorbing material to compensate for the closer spacing. As discussed earlier the most recent criticality calculation for Region 2 did not take credit for boraflex. The omission of boraflex from the calculation is compensated for by taking credit for the presence of soluble boron in the spent fuel pool water and the radioactive decay time of the spent fuel being stored.

“The precedent of crediting soluble boron to provide criticality control aside from normal reactor operations has already been established. Soluble boron credit was used in the “Westinghouse Spent Fuel Rack Criticality Analysis Methodology” described in WCAP-14416-NP-A. That methodology was accepted for use by an NRC Safety Evaluation dated October 25, 1996. Additional guidance outlining the requirements for use of soluble boron has been issued by the NRC and documented in 'Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants,' Laurence I. Kopp, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Reactor Systems Branch, February 1998."

This section of the FSAR continues to discuss that: “In order to maintain a k-effective less than or equal to 0.95 for the tilt machine and fuel elevator, 850 ppm boron is required. This is maintained by meeting normal requirements outlined in Technical Specification LCO 3.7.15 and associated procedures. Credit is taken for boron to maintain k-effective less than or equal to 0.95 with a margin of 870 ppm being available due to the Technical Specification requirement of 1720 ppm.”

Thus, Boraflex in Region 2 of the Spent Fuel Pool is not credited in the Spent Fuel Pool criticality analysis since the required boron is ensured by the Technical Specifications requirements. Accordingly, it is not included in-scope of License Renewal and no age managing of it is required.

For Region 1, FSAR Section 9.11.3.4, “Prevention of Criticality During Transfer and Storage,” states that B_4C (boron carbide) is credited in criticality calculations:

“The Region 1 racks in the main pool are designed for a 10-1/4-inch center-to-center spacing with B_4C plates around each assembly. Borated water surrounds the spent fuel storage racks in the same concentration and to a level common to the refueling cavity and pool. The center-to-center distance of the Region 1 storage racks is such that a k_{eff} of less than 0.95 is maintained even in the event that unborated water was used to fill the storage areas.”

The B_4C plates in the Region 1 racks are sheathed between stainless plates, unlike Boral that has the boron carbide sandwiched between aluminum alloy. There has been some industry experience with blistering and swelling of Boral as a result of a chemical reaction between the boron carbide and aluminum alloy. This has not been observed with stainless cladding. In addition, per NRC memorandum entitled “Resolution of Spent Fuel Storage Pool Action Plan Issue” dated July 26, 1996, section 3.3.1 states “…degradation in neutron absorption performance has not been observed in materials other than Boraflex. Some neutron absorbing panels have been observed to swell due to gas accumulation within the cladding material, but this effect has not degraded neutron absorption performance.” Thus, reduction in neutron-absorbing capacity and loss of material due to general corrosion is not an aging effect requiring management for boron carbide (B_4C). The stainless steel sheathing for the boron carbide will be age managed for loss of material due to pitting or crevice corrosion cracking via the water chemistry program, consistent with NUREG-1801 item VIIA2.1c.

In conclusion, the Palisades Spent Fuel Storage Racks contain B_4C (Boron Carbide) and Boraflex neutron absorbing material. Due to industry concerns on degradation of boraflex, soluble boron is maintained at 1720 ppm per Technical Specification LCO 3.7.15 to maintain k-effective less than or equal to 0.95. Criticality calculations take credit for the soluble boron in the spent fuel pool water and conclude that boraflex material does not need to be credited.

For the storage racks containing boron carbide, no credit is taken for soluble boron. There is no industry experience that boron carbide sheathed in stainless steel has experienced loss of material due to corrosion. With regards to reduction of neutron absorbing capability due to corrosion, NRC memorandum entitled “Resolution of Spent Fuel Storage Pool Action Plan Issue” dated July 26, 1996, section 3.3.1 states “…degradation in neutron absorption performance has not been observed in materials other than Boraflex.”

Therefore, no plant specific aging management program is required to manage reduction of neutron-absorbing capacity and loss of material due to general corrosion, as addressed in NUREG-1801 Item VIIA2.1-b, in VIIA2, Spent Fuel Storage. The stainless steel sheathing for the boron carbide will be age managed for loss of material due to pitting or crevice corrosion cracking via the Water Chemistry Program, consistent with NUREG-1801 item VIIA2.1c. Boraflex, as discussed in NUREG-1801 item VIIA2.1a is not in-scope of license renewal since it performs no intended function.

3.3.2.2.11 Loss of Material due to General, Pitting, Crevice, and Microbiologically Influenced Corrosion

NUREG-1800 states that loss of material due to general, pitting, and crevice corrosion and MIC could occur in the underground piping and fittings in the open-cycle cooling water system (service water system) and in the diesel fuel oil system. The buried piping and tanks inspection program relies on industry practice, frequency of pipe excavation, and operating experience to manage the effects of loss of material from general, pitting, and crevice corrosion and MIC. The effectiveness of the buried piping and tanks inspection program should be verified to evaluate an applicant's inspection frequency and operating experience with buried components, ensuring that loss of material is not occurring.

The Buried Services Corrosion Monitoring Program manages aging effects on the external surfaces of carbon steel, low-alloy steel, and stainless steel components (e.g. tanks, piping) that are buried in soil or sand. This program includes (a) visual inspections of external surfaces of buried components for evidence of coating damage and substrate degradation to manage the effects of aging, (b) visual inspection of the external surfaces of buried stainless steel components for evidence of crevice corrosion, pitting, and MIC. The periodicity of these inspections for carbon, low-alloy, and stainless steel will be based on opportunities for inspection such as scheduled maintenance work.

3.3.2.3 Time-Limited Aging Analysis

The time-limited aging analyses (TLAA) identified below are associated with the Auxiliary Systems components. The section of the LRA that contains the TLAA review results is indicated in parenthesis.

- Fatigue (Section 4.3, Metal Fatigue)

3.3.3 Conclusion

The Auxiliary System piping, fittings, and components that are subject to aging management review have been identified in accordance with the requirements of 10 CFR 54.4. The aging management programs selected to manage aging effects for the Auxiliary Systems components are identified in the summaries in Section 3.3.2.1 above.

A description of these aging management programs is provided in Appendix B, along with the demonstration that the identified aging effects will be managed for the period of extended operation.

Therefore, based on the demonstrations provided in Appendix B, the effects of aging associated with the Auxiliary System components will be adequately managed so that there is reasonable assurance that the intended function(s) will be maintained consistent with the current licensing basis during the period of extended operation.

Table 3.3.1 Summary of Aging Management Evaluations in Chapter VII of NUREG-1801 for Auxiliary Systems

Item Number	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.3.1-01	Components in spent fuel pool cooling and cleanup	Loss of material due to general, pitting, and crevice corrosion	Water chemistry and one-time inspection	Yes, detection of aging effects is to be further evaluated (see [SRP] subsections 3.3.2.2.1.1 and 3.3.2.2.1.2)	Further evaluation documented in Section 3.3.2.2.1 Paragraph 1 and Paragraph 2.
3.3.1-02	Linings in spent fuel pool cooling and cleanup system; seals and collars in ventilation systems	Hardening, cracking and loss of strength due to elastomer degradation; loss of material due to wear	Plant specific	Yes, plant specific (see [SRP] subsection 3.3.2.2.2)	Further evaluation documented in Section 3.3.2.2.2.
3.3.1-03	Components in load handling, chemical and volume control system (PWR), and reactor water cleanup and shutdown cooling systems (older BWR)	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA (see [SRP] subsection 3.3.2.2.3)	Further evaluation documented in Section 3.3.2.2.3. The aging mechanism fatigue is potentially applicable to many component types in the Auxiliary Systems Supergroup, but only selected components or locations require explicit analysis as TLAAs and/or warrant aging management. The Palisades approach to identifying and managing the relevant locations and components for fatigue damage is addressed in Section 4.3 and in Appendix B, Fatigue Monitoring Program. Therefore, cumulative fatigue damage is not identified as an aging effect in Tables 3.3.2-1 through 3.3.2-15.below.
3.3.1-04	Heat exchangers in reactor water cleanup system (BWR); high pressure pumps in chemical and volume control system (PWR)	Crack initiation and growth due to SCC or cracking	Plant specific	Yes, plant specific (see [SRP] subsection 3.3.2.2.4)	Further evaluation documented in Section 3.3.2.2.4.

Table 3.3.1 Summary of Aging Management Evaluations in Chapter VII of NUREG-1801 for Auxiliary Systems

Item Number	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.3.1-05	Components in ventilation systems, diesel fuel oil system, and emergency diesel generator systems; external surfaces of carbon steel components	Loss of material due to general, pitting, and crevice corrosion, and MIC	Plant specific	Yes, plant specific (see [SRP] subsection 3.3.2.2.5)	Further evaluation documented in Section 3.3.2.2.5.
3.3.1-06	Components in reactor coolant pump oil collect system of fire protection	Loss of material due to galvanic, general, pitting, and crevice corrosion	One-time inspection	Yes, detection of aging effects is to be further evaluated (see [SRP] subsection 3.3.2.2.6)	Further evaluation documented in Section 3.3.2.2.6.
3.3.1-07	Diesel fuel oil tanks in diesel fuel oil system and emergency diesel generator system	Loss of material due to general, pitting, and crevice corrosion, MIC, and biofouling	Fuel oil chemistry and one-time inspection	Yes, detection of aging effects is to be further evaluated (see [SRP] subsection 3.3.2.2.7)	Further evaluation documented in Section 3.3.2.2.7.
3.3.1-08	BWR only				
3.3.1-09	Heat exchangers in chemical and volume control system	Crack initiation and growth due to SCC and cyclic loading	Water chemistry and a plant-specific verification program	Yes, plant specific (see [SRP] subsection 3.3.2.2.9)	Further evaluation documented in Section 3.3.2.2.9.
3.3.1-10	Neutron absorbing sheets in spent fuel storage racks	Reduction of neutron absorbing capacity and loss of material due to general corrosion (Boral, boron steel)	Plant specific	Yes, plant specific (see [SRP] subsection 3.3.2.2.10)	Further evaluation documented in Section 3.3.2.2.10.
3.3.1-11	New fuel rack assembly	Loss of material due to general, pitting, and crevice corrosion	Structures monitoring	No	See [Section B2.1.4] Boric Acid Corrosion Program for aging management program.

Table 3.3.1 Summary of Aging Management Evaluations in Chapter VII of NUREG-1801 for Auxiliary Systems

Item Number	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.3.1-12	Neutron absorbing sheets in spent fuel storage racks	Reduction of neutron absorbing capacity due to Boraflex degradation	Boraflex monitoring	No	Not applicable
3.3.1-13	Spent fuel storage racks and valves in spent fuel pool cooling and cleanup	Crack initiation and growth due to stress corrosion cracking	Water chemistry	No	See [Section B2.1.21] Water Chemistry Program for aging management program.
3.3.1-14	Closure bolting and external surfaces of carbon steel and low-alloy steel components	Loss of material due to boric acid corrosion	Boric acid corrosion	No	See [Section B2.1.4] Boric Acid Corrosion Program for aging management program.
3.3.1-15	Components in or serviced by closed-cycle cooling water system	Loss of material due to general, pitting, and crevice corrosion, and MIC	Closed-cycle cooling water system	No	See [Section B2.1.6] Closed Cycle Cooling Water Program, [Section B2.1.13] One Time Inspection Program, and [Section B2.1.21] Water Chemistry Program for aging management program.
3.3.1-16	Cranes including bridge and trolleys and rail system in load handling system	Loss of material due to general corrosion and wear	Overhead heavy load and light load handling systems	No	See [Section B2.1.15] Overhead Load Handling System Inspection Program for aging management program.
3.3.1-17	Components in or serviced by open-cycle cooling water systems	Loss of material due to general, pitting, crevice, and galvanic corrosion, MIC, and biofouling; buildup of deposit due to biofouling	Open-cycle cooling water system	No	See [Section B2.1.6] Closed Cycle Cooling Water Program, [Section B2.1.13] One Time Inspection Program, and [Section B2.1.14] Open Cycle Cooling Water Program for aging management program.

Table 3.3.1 Summary of Aging Management Evaluations in Chapter VII of NUREG-1801 for Auxiliary Systems

Item Number	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.3.1-18	Buried piping and fittings	Loss of material due to general, pitting, and crevice corrosion, and MIC	Buried piping and tanks surveillance or Buried piping and tanks inspection	No Yes, detection of aging effects and operating experience are to be further evaluated (see [SRP] subsection 3.3.2.2.11)	Further evaluation documented in Section 3.3.2.2.11.
3.3.1-19	Components in compressed air system	Loss of material due to general and pitting corrosion	Compressed air monitoring	No	See [Section B2.1.13] One Time Inspection Program for aging management program.
3.3.1-20	Components (doors and barrier penetration seals) and concrete structures in fire protection	Loss of material due to wear; hardening and shrinkage due to weathering	Fire protection	No	See [Section B2.1.10] Fire Protection Program for aging management program.
3.3.1-21	Components in water-based fire protection	Loss of material due to general, pitting, crevice, and galvanic corrosion, MIC, and biofouling	Fire water system	No	See [Section B2.1.10] Fire Protection Program and [Section B2.1.13] One Time Inspection Program for aging management programs.
3.3.1-22	Components in diesel fire system	Loss of material due to galvanic, general, pitting, and crevice corrosion	Fire protection and fuel oil chemistry	No	See [Section B2.1.10] Fire Protection Program and [Section B2.1.9] Diesel Fuel Monitoring and Storage Program for aging management programs.
3.3.1-23	Tanks in diesel fuel oil system	Loss of material due to general, pitting, and crevice corrosion	Above ground carbon steel tanks	No	See [Section B2.1.9] Diesel Fuel Monitoring and Storage Program, [Section B2.1.13] One Time Inspection Program, and [Section B2.1.20] System Monitoring Program for aging management program.

Table 3.3.1 Summary of Aging Management Evaluations in Chapter VII of NUREG-1801 for Auxiliary Systems

Item Number	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.3.1-24	Closure bolting	Loss of material due to general corrosion; crack initiation and growth due to cyclic loading and SCC	Bolting integrity	No	See [Section B2.1.4] Boric Acid Corrosion Program, [Section B2.1.13] One Time Inspection Program, and [Section B2.1.20] System Monitoring Program for aging management programs.
3.3.1-25	BWR only				
3.3.1-26	BWR only				
3.3.1-27	BWR only				
3.3.1-28	BWR only				
3.3.1-29	Components (aluminum bronze, brass, cast iron, cast steel) in open-cycle and closed-cycle cooling water systems, and ultimate heat sink	Loss of material due to selective leaching	Selective leaching of materials	No	See [Section B2.1.13] One Time Inspection Program for aging management program.
3.3.1-30	Fire barriers, walls, ceilings and floors in fire protection	Concrete cracking and spalling due to freeze-thaw, aggressive chemical attack, and reaction with aggregates; loss of material due to corrosion of embedded steel	Fire protection and structures monitoring	No	See [Section B2.1.10] Fire Protection Program and [Section B2.1.19] Structural Monitoring Program for aging management programs.

Table 3.3.2-1 Auxiliary Systems - Chemical and Volume Control System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Boric Acid Storage Tanks	Fluid Pressure Boundary	Stainless Steel	Air (Int)	Loss of Material	One-Time Inspection Program			G
Oil Cooler Shell	Fluid Pressure Boundary	Brass	Oil (Int)	Loss of Material	One-Time Inspection Program			J
			Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program			F
Oil Cooler Tubes	Fluid Pressure Boundary	Copper Nickel	Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program			J
Letdown Heat Exchanger Shell	Fluid Pressure Boundary	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	VII.E1.8-c	3.3.1-15	B
Letdown Heat Exchanger Channel Head, Tubes, Tube Sheet	Fluid Pressure Boundary	Stainless Steel	Treated Water (Ext)	Cracking	Closed Cycle Cooling Water Program	VII.E1.8-b	3.3.1-09	B
Letdown Heat Exchanger Tubes, Tube Sheet	Fluid Pressure Boundary	Stainless Steel	Treated Water (Ext)	Loss of Material	Closed Cycle Cooling Water Program			H
Letdown Heat Exchanger Tubes	Fluid Pressure Boundary	Stainless Steel	Treated Water (Ext)	Loss of Material	One-Time Inspection Program			H

Table 3.3.2-1 Auxiliary Systems - Chemical and Volume Control System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Letdown Heat Exchanger Channel Head, Tubes, Tube Sheet	Fluid Pressure Boundary	Stainless Steel	Treated Water (Int)	Cracking	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	VII.E1.8-b	3.3.1-09	A
					Water Chemistry Program	VII.E1.8-b	3.3.1-09	A
Fasteners	Fluid Pressure Boundary	Low-Alloy Steel	Plant Indoor Air (Ext)	Loss of Preload	Bolting Integrity Program	VII.I.2-a	3.3.1-24	324, A
				Loss of Material	Bolting Integrity Program	VII.I.2-a	3.3.1-24	A
					Boric Acid Corrosion Program	VII.E1.1-b	3.3.1-14	A
						VII.E1.2-a	3.3.1-14	A
						VII.E1.5-b	3.3.1-14	A
						VII.E1.6-a	3.3.1-14	A
						VII.E1.7-b	3.3.1-14	A
						VII.E1.8-d	3.3.1-14	A
						VII.E1.10-a	3.3.1-14	A
		Stainless Steel	Plant Indoor Air (Ext)	Loss of Preload	Bolting Integrity Program			324, F
			Containment Air (Ext)	Loss of Preload	Bolting Integrity Program			324, F

Table 3.3.2-1 Auxiliary Systems - Chemical and Volume Control System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Pipe - CVC Cooler	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	VII.C2.1-a	3.3.1-15	366, B
Nozzle - CVC Spray	Fluid Pressure Boundary	Stainless Steel	Gas (Ext)	Loss of Material	One-Time Inspection Program			G
Tubing - CVC Oil	Fluid Pressure Boundary	Stainless Steel	Oil (Int)	Loss of Material	One-Time Inspection Program			G
Pipe and Pressure Test Fittings	Fluid Pressure Boundary	Stainless Steel	Treated Water (Int)	Cracking	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	IV.C2.2-f	3.1.1-36	A
						IV.C2.2-h	3.1.1-07	A
					One-Time Inspection Program	IV.C2.2-h	3.1.1-07	B
					Water Chemistry Program	IV.C2.2-f	3.1.1-36	A
						IV.C2.2-h	3.1.1-07	A
Flow Elements	Fluid Pressure Boundary	Stainless Steel	Treated Water (Int)	Loss of Material	One-Time Inspection Program	IV.C2.2-h	3.1.1-07	367, H
Regenerative Heat Exchanger Tubes, Tube Sheet	Fluid Pressure Boundary	Stainless Steel	Treated Water (Ext)	Cracking	Water Chemistry Program	VII.E1.7-c	3.3.1-09	A
					One-Time Inspection Program	VII.E1.7-c	3.3.1-09	B

Table 3.3.2-1 Auxiliary Systems - Chemical and Volume Control System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Regenerative Heat Exchanger Channel Head, Shell, Tubes, Tube Sheet	Fluid Pressure Boundary	Stainless Steel	Treated Water (Int)	Cracking	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	VII.E1.7-c	3.3.1-09	A
					Water Chemistry Program	VII.E1.7-c	3.3.1-09	A
Control Valves	Fluid Pressure Boundary	Cast Austenitic SS	Treated Water (Int)	Cracking	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	VII.E1.7-c	3.3.1-09	C
					One-Time Inspection Program	VII.E1.7-c	3.3.1-09	D
					Water Chemistry Program	VII.E1.7-c	3.3.1-09	C
Letdown Stop Valve CV-2001	Fluid Pressure Boundary	Cast Austenitic SS	Treated Water (Int)	Reduction of Fracture Toughness	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program			H
Check, Control, Manual & Relief valves; Instrument Assemblies	Fluid Pressure Boundary	Stainless Steel	Treated Water (Int)	Cracking	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	VII.E1.5-a	3.3.1-04	C
					One-Time Inspection Program	VII.E1.5-a	3.3.1-04	D
					Water Chemistry Program	VII.E1.5-a	3.3.1-04	C

Table 3.3.2-2 Auxiliary Systems - Circulating Water System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Fasteners	Fluid Pressure Boundary	Low-Alloy Steel	Plant Indoor Air (Ext)	Loss of Material	Bolting Integrity Program	VII.I.2-a	3.3.1-24	A
				Loss of Preload	Bolting Integrity Program	VII.I.2-a	3.3.1-24	324, A
Pipe & Fittings	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Raw Water (Int)	Loss of Material	Open Cycle Cooling Water Program	VII.C1.1-a	3.3.1-17	A
Pumps	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Raw Water (Int)	Loss of Material	Open Cycle Cooling Water Program	VII.C1.5-a	3.3.1-17	A
Valves & Dampers	Fluid Pressure Boundary	Cast Iron	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Raw Water (Int)	Loss of Material	Open Cycle Cooling Water Program	VII.C1.5-a	3.3.1-17	A
					One-Time Inspection Program	VII.C1.5-a	3.3.1-17	B, 301

Table 3.3.2-3 Auxiliary Systems- Component Cooling Water System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Accumulators	Fluid Pressure Boundary	Carbon Steel	Air (Int)	Loss of Material	Closed Cycle Cooling Water Program			319, G
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	VII.C2.4-a	3.3.1-15	B
Bistable/Switch (In-line Flow Indicator)	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	VII.C2.2-a	3.3.1-15	D
Component Cooling Heat Exchanger	Fluid Pressure Boundary	Aluminum Bronze	Raw Water (Ext)	Buildup of Deposit	Open Cycle Cooling Water Program	VII.C1.3-b	3.3.1-17	315, C
				Loss of Material	Open Cycle Cooling Water Program	VII.C1.3-a	3.3.1-17	A
				Loss of Material - Selective Leaching	One-Time Inspection Program	VII.C1.3-a	3.3.1-29	301, B
		Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Raw Water (Int)	Buildup of Deposit	Open Cycle Cooling Water Program	VII.C1.3-a	3.3.1-17	380, A

Table 3.3.2-3 Auxiliary Systems- Component Cooling Water System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Component Cooling Heat Exchanger	Fluid Pressure Boundary	Carbon Steel	Raw Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	VII.C1.3-a	3.3.1-17	E
					Open Cycle Cooling Water Program	VII.C1.3-a	3.3.1-17	A
			Treated Water (Ext)	Loss of Material	Closed Cycle Cooling Water Program	VII.C1.3-a	3.3.1-17	E
			Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	VII.C1.3-a	3.3.1-17	E
	Fluid Pressure Boundary Heat Transfer	Copper Alloys	Raw Water (Int)	Buildup of Deposit	Open Cycle Cooling Water Program	VII.C1.3-b	3.3.1-17	A
				Heat Transfer Degradation	Closed Cycle Cooling Water Program	VII.C1.3-b	3.3.1-17	315, E
					Open Cycle Cooling Water Program	VII.C1.3-b	3.3.1-17	313, 315, 316, A
				Loss of Material	Closed Cycle Cooling Water Program	VII.C1.3-a	3.3.1-17	313, 315, E
					Open Cycle Cooling Water Program	VII.C1.3-a	3.3.1-17	313, 315, A
				Loss of Material - Selective Leaching	One-Time Inspection Program	VII.C1.3-a	3.3.1-29	301, 313, 315, B

Table 3.3.2-3 Auxiliary Systems- Component Cooling Water System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Component Cooling Heat Exchanger	Fluid Pressure Boundary Heat Transfer	Copper Alloys	Treated Water (Ext)	Heat Transfer Degradation	Closed Cycle Cooling Water Program	VII.C1.3-b	3.3.1-17	313, 315, 316, E
				Loss of Material	Closed Cycle Cooling Water Program	VII.C1.3-a	3.3.1-17	313, 315, 323, E
				Loss of Material - Selective Leaching	One-Time Inspection Program	VII.C1.3-a	3.3.1-29	301, 313, B
Cooler	Fluid Pressure Boundary	Nickel-Based Alloys	Treated Water (Ext)	Loss of Material	Closed Cycle Cooling Water Program			308, J
					One-Time Inspection Program			308, J
			Treated Water (Int)	Loss of Material	One-Time Inspection Program			308, J
					Water Chemistry Program			308, J
		Stainless Steel	Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	VII.C2.2-a	3.3.1-15	D

Table 3.3.2-3 Auxiliary Systems- Component Cooling Water System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Heat Exchanger	Fluid Pressure Boundary	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	VII.C2.4-a	3.3.1-15	C
		Copper Alloys	Treated Water (Ext)	Loss of Material	Closed Cycle Cooling Water Program	VII.C2.4-a	3.3.1-15	323, C
				Loss of Material - Selective Leaching	One-Time Inspection Program	VII.C2.3-a	3.3.1-15	301, C
			Treated Water (Int)	Cracking	Closed Cycle Cooling Water Program			J
Fasteners	Fluid Pressure Boundary	Carbon Steel	Containment Air (Ext)	Loss of Preload	Bolting Integrity Program	VII.I.2-a	3.3.1-24	324, C
				Loss of Material	Bolting Integrity Program	VII.I.2-a	3.3.1-24	324, C
					Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
			Plant Indoor Air (Ext)	Loss of Preload	Bolting Integrity Program	VII.I.2-a	3.3.1-24	324, C
				Loss of Material	Bolting Integrity Program	VII.I.2-a	3.3.1-24	324, C
					Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
		Stainless Steel	Containment Air (Ext)	Loss of Preload	Bolting Integrity Program			324, F
			Plant Indoor Air (Ext)	Loss of Preload	Bolting Integrity Program			324, F

Table 3.3.2-3 Auxiliary Systems- Component Cooling Water System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Primary Coolant Pump Motor Oil Cooler	Fluid Pressure Boundary	Brass	Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	VII.C1.3-a	3.3.1-17	325, E
				Loss of Material - Selective Leaching	One-Time Inspection Program	VII.C1.3-a	3.3.1-29	325, D
		Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	VII.C2.2-a	3.3.1-15	D
		Copper Nickel	Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	VII.C1.3-a	3.3.1-17	325, E
Pipe & Fittings	Fluid Pressure Boundary	Stainless Steel	Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	VII.C2.2-a	3.3.1-15	D
		Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	VII.C2.1-a	3.3.1-15	B

Table 3.3.2-3 Auxiliary Systems- Component Cooling Water System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Pipe & Fittings	Fluid Pressure Boundary	Copper Alloys	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program			317, F
			Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	VII.C1.3-a	3.3.1-17	325, E
Pumps	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	VII.C2.3-a	3.3.1-15	B
Valves & Dampers	Fluid Pressure Boundary	Bronze	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program			317, F
			Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	VII.C1.3-a	3.3.1-17	325, E
		Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	VII.C2.2-a	3.3.1-15	B

Table 3.3.2-3 Auxiliary Systems- Component Cooling Water System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Valves & Dampers	Fluid Pressure Boundary	Cast Iron	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	VII.C2.2-a	3.3.1-15	390, B
				Loss of Material - Selective Leaching	One-Time Inspection Program	VII.C2.3-a	3.3.1-29	D
Waste Gas Compressor Cooler	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	VII.C2.2-a	3.3.1-15	318, 320, D
		Stainless Steel	Gas (Int)	Cracking	Closed Cycle Cooling Water Program			308, J
				Loss of Material	Closed Cycle Cooling Water Program			308, J
			Treated Water (Ext)	Loss of Material	Closed Cycle Cooling Water Program	VII.C2.2-a	3.3.1-15	323, D

Table 3.3.2-4 Auxiliary Systems - Compressed Air System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Accumulators	Fluid Pressure Boundary	Carbon Steel	Air (Int)	Loss of Material	One-Time Inspection Program	VII.D.3-a	3.3.1-19	E
			Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
Air Dryers	Fluid Pressure Boundary	Carbon Steel	Air (Int)	Loss of Material	One-Time Inspection Program	VII.D.6-a	3.3.1-19	E
			Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
Blowers, Fans, Compressors, Vacuum	Fluid Pressure Boundary	Carbon Steel	Air (Int)	Loss of Material	One-Time Inspection Program	VII.D.6-a	3.3.1-19	E
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A

Table 3.3.2-4 Auxiliary Systems - Compressed Air System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Filters/Strainers	Fluid Pressure Boundary Filtration	Aluminum	Air (Int)	Cracking	One-Time Inspection Program			F
				Loss of Material	One-Time Inspection Program			F
		Brass	Air (Int)	Cracking	One-Time Inspection Program			F
				Loss of Material	One-Time Inspection Program			F
			Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program			F
		Bronze	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program			F
		Carbon Steel	Air (Int)	Loss of Material	One-Time Inspection Program	VII.D.5-a	3.3.1-19	E
			Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
		Cast Iron	Air (Int)	Loss of Material	One-Time Inspection Program	VII.D.5-a	3.3.1-19	E
			Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.D.5-a	3.3.1-19	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	335, A

Table 3.3.2-4 Auxiliary Systems - Compressed Air System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Filters/Strainers	Fluid Pressure Boundary Filtration	Galvanized	Air (Int)	Loss of Material	One-Time Inspection Program	VII.D.5-a	3.3.1-19	E
			Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
		Stainless Steel	Air (Int)	Loss of Material	One-Time Inspection Program			F
			Oil (Int)	Loss of Material	One-Time Inspection Program			F
Heat Exchangers	Fluid Pressure Boundary	Bronze	Air (Int)	Loss of Material	Open Cycle Cooling Water Program			F
			Raw Water (Ext)	Loss of Material	Open Cycle Cooling Water Program			F
		Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Raw Water (Int)	Loss of Material	Open Cycle Cooling Water Program			F
		Copper Alloys	Air (Int)	Loss of Material	One-Time Inspection Program			F
			Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program			F

Table 3.3.2-4 Auxiliary Systems - Compressed Air System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Fasteners	Fluid Pressure Boundary	Carbon Steel	Containment Air (Ext)	Loss of Preload	Bolting Integrity Program			324, F
			Plant Indoor Air (Ext)	Loss of Material	Bolting Integrity Program	VII.I.1-b	3.3.1-05	E
					Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
		Stainless Steel	Plant Indoor Air (Ext)	Loss of Preload	Bolting Integrity Program			324, F
		Copper Alloys	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program			F
			Plant Indoor Air (Ext)	Loss of Preload	Bolting Integrity Program			324, F
Pipe & Fittings	Fluid Pressure Boundary	Brass	Air (Int)	Cracking	One-Time Inspection Program			F
				Loss of Material	One-Time Inspection Program			F
		Carbon Steel	Air (Int)	Loss of Material	One-Time Inspection Program	VII.D.1-a	3.3.1-19	E
			Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
			Plant Indoor Air (Ext)		System Monitoring Program	VII.I.1-b	3.3.1-05	A
		Copper Alloys	Air (Int)	Cracking	One-Time Inspection Program			F
				Loss of Material	One-Time Inspection Program			F

Table 3.3.2-4 Auxiliary Systems - Compressed Air System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Pipe & Fittings	Fluid Pressure Boundary	Copper Alloys	Containment Air (Ext) Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program			F
		Stainless Steel	Air (Int)	Loss of Material	One-Time Inspection Program			F
Pumps	Fluid Pressure Boundary	Cast Iron	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	335, A
Traps (Steam)	Fluid Pressure Boundary	Aluminum	Air (Int)	Cracking	One-Time Inspection Program			F
				Loss of Material	One-Time Inspection Program			F
		Cast Iron	Air (Int)	Loss of Material	One-Time Inspection Program	VII.D.2-a	3.3.1-19	E
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	335, A
Valves & Dampers	Fluid Pressure Boundary	Brass	Air (Int)	Cracking	One-Time Inspection Program			F
				Loss of Material	One-Time Inspection Program			F
			Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program			F
			Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program			F

Table 3.3.2-4 Auxiliary Systems - Compressed Air System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Valves & Dampers	Fluid Pressure Boundary	Brass	Oil	Cracking	One-Time Inspection Program			F
				Loss of Material	One-Time Inspection Program			F
		Bronze	Air (Int)	Cracking	One-Time Inspection Program			F
				Loss of Material	One-Time Inspection Program			F
			Containment Air (Ext) Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program			F
		Carbon Steel	Air (Int)	Loss of Material	One-Time Inspection Program	VII.D.2-a	3.3.1-19	E
			Containment Air (Ext) Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
		Cast Iron	Air (Int)	Loss of Material	One-Time Inspection Program	VII.D.2-a	3.3.1-19	E
			Containment Air (Ext) Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	335, A

Table 3.3.2-4 Auxiliary Systems - Compressed Air System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Valves & Dampers	Fluid Pressure Boundary	Copper Alloys	Containment Air (Ext) Air (Ext)	Loss of Material	Boric Acid Corrosion Program			F
		Stainless Steel	Air (Int)	Loss of Material	One-Time Inspection Program			F

Table 3.3.2-5 Auxiliary Systems- Containment Air Recirculation and Cooling System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Containment Air Cooler Coils	Fluid Pressure Boundary	Copper Alloys	Air (Ext)	Heat Transfer Degradation	System Monitoring Program	VII.F3.2-a	3.3.1-05	387, H
				Loss of Material	One-Time Inspection Program	VII.F3.2-a	3.3.1-05	B
			Raw Water (Int)	Buildup of Deposit	Open Cycle Cooling Water Program	VII.C1.3-b	3.3.1-17	325, C
				Heat Transfer Degradation	Open Cycle Cooling Water Program	VII.C1.3-b	3.3.1-17	316, 325, C
				Loss of Material	Open Cycle Cooling Water Program	VII.C1.3-a	3.3.1-17	325, 394, C
	Heat Transfer	Copper Alloys	Air (Ext)	Heat Transfer Degradation	System Monitoring Program	VII.F3.2-a	3.3.1-05	387, H
				Loss of Material	One-Time Inspection Program	VII.F3.2-a	3.3.1-05	B
			Raw Water (Int)	Buildup of Deposit	Open Cycle Cooling Water Program	VII.C1.3-b	3.3.1-17	325, C
				Heat Transfer Degradation	Open Cycle Cooling Water Program	VII.C1.3-b	3.3.1-17	316, 325, C
				Loss of Material	Open Cycle Cooling Water Program	VII.C1.3-a	3.3.1-17	325, 394, C

Table 3.3.2-5 Auxiliary Systems- Containment Air Recirculation and Cooling System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Containment Air Cooler Filter	Filtration	Stainless Steel	Air (Int)	Loss of Material	One-Time Inspection Program	VII.F3.4-a	3.3.1-05	308, B
	Fluid Pressure Boundary	Stainless Steel	Air (Int)	Loss of Material	One-Time Inspection Program	VII.F3.4-a	3.3.1-05	308, B
Containment Air Cooler Flow Element	Flow Measurement	Stainless Steel	Air (Int)	Loss of Material	One-Time Inspection Program	VII.F3.4-a	3.3.1-05	308, E
	Fluid Pressure Boundary	Stainless Steel	Air (Int)	Loss of Material	One-Time Inspection Program	VII.F3.4-a	3.3.1-05	308, E
Containment Air Cooler Housing	Fluid Pressure Boundary	Carbon Steel	Air (Int)	Loss of Material	One-Time Inspection Program	VII.F3.1-a	3.3.1-05	391, B
			Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
Containment Air Cooler Recirculation Fans	Fluid Pressure Boundary	Carbon Steel	Air (Int)	Loss of Material	One-Time Inspection Program	VII.F3.1-a	3.3.1-05	391, B
			Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A

Table 3.3.2-5 Auxiliary Systems- Containment Air Recirculation and Cooling System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Dampers	Fluid Pressure Boundary	Carbon Steel	Air (Int)	Loss of Material	One-Time Inspection Program	VII.F3.1-a	3.3.1-05	391, B
			Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
Drip Pans	Fluid Pressure Boundary	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Raw Water (Int)	Loss of Material	One-Time Inspection Program	VII.F3.1-a	3.3.1-05	B
Duct	Pressure Boundary	Galvanized	Air (Int)	Loss of Material	One-Time Inspection Program	VII.F3.1-a	3.3.1-05	B
			Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	386, A
Fasteners	Fluid Pressure Boundary Structure Functional Support	Stainless Steel	Containment Air (Ext)	Loss of Preload	Bolting Integrity Program			324, F
		Low-Alloy Steel	Containment Air (Ext)	Loss of Preload	Bolting Integrity Program	VII.I.2-a	3.3.1-24	324, A
				Loss of Material	Bolting Integrity Program	VII.I.2-a	3.3.1-24	324, D
					Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A

Table 3.3.2-5 Auxiliary Systems- Containment Air Recirculation and Cooling System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Manual and Instrumentation Valves	Fluid Pressure Boundary	Stainless Steel	Air (Int)	Loss of Material	One-Time Inspection Program	VII.F3.4-a	3.3.1-05	308, E

Table 3.3.2-6 Auxiliary Systems- Emergency Power System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Accumulators	Fluid Pressure Boundary	Carbon Steel	Air (Int)	Loss of Material	One-Time Inspection Program	VIII.H.2-a	3.4.1-08	A
						VII.H2.2-a	3.3.1-05	B
			Oil (Int)	Loss of Material	Diesel Fuel Monitoring and Storage Program	VII.H2.5-a	3.3.1-07	B
					One-Time Inspection Program	VII.H2.5-a	3.3.1-07	B
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	VII.H2.1-a	3.3.1-15	D
Blowers Fans Compressor Vacuum	Fluid Pressure Boundary	Cast Iron	Air (Int)	Loss of Material	One-Time Inspection Program	VII.H2.3-a	3.3.1-05	399, A
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A

Table 3.3.2-6 Auxiliary Systems- Emergency Power System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Cooler	Fluid Pressure Boundary	Carbon Steel	Oil (Ext)	Heat Transfer Degradation	One-Time Inspection Program			J
				Loss of Material	One-Time Inspection Program	VII.H2.5-a	3.3.1-07	D
			Oil (Int)	Loss of Material	One-Time Inspection Program	VII.H2.5-a	3.3.1-07	D
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Raw Water (Int)	Buildup of Deposit	Open Cycle Cooling Water Program			J
				Heat Transfer Degradation	Open Cycle Cooling Water Program	VII.C1.3-b	3.3.1-17	A
				Loss of Material	Open Cycle Cooling Water Program	VII.C1.3-a	3.3.1-17	A
	Heat Transfer	Carbon Steel	Oil (Ext)	Heat Transfer Degradation	One-Time Inspection Program			J
				Loss of Material	One-Time Inspection Program	VII.H2.5-a	3.3.1-07	D
			Raw Water (Int)	Buildup of Deposit	Open Cycle Cooling Water Program			J
				Heat Transfer Degradation	Open Cycle Cooling Water Program	VII.C1.3-b	3.3.1-17	A
				Loss of Material	Open Cycle Cooling Water Program	VII.C1.3-a	3.3.1-17	A

Table 3.3.2-6 Auxiliary Systems- Emergency Power System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Fasteners	Fluid Pressure Boundary	Low-Alloy Steel	Plant Indoor Air (Ext)	Loss of Preload	Bolting Integrity Program	VII.I.2-a	3.3.1-24	324, A
				Loss of Material	Bolting Integrity Program	VII.I.2-a	3.3.1-24	A
		Stainless Steel	Plant Indoor Air (Ext)	Loss of Preload	Bolting Integrity Program			324, F
Filters/Strainers	Filtration	Galvanized	Air (Int)	Loss of Material	One-Time Inspection Program	VII.H2.3-a	3.3.1-05	386, A
			Atmosphere/ Weather (Ext)	Loss of Material	System Monitoring Program			F
	Fluid Pressure Boundary	Carbon Steel	Oil (Int)	Loss of Material	Diesel Fuel Monitoring and Storage Program	VII.H2.5-a	3.3.1-07	D
					One-Time Inspection Program	VII.H2.5-a	3.3.1-07	D
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
		Galvanized	Air (Int)	Loss of Material	One-Time Inspection Program	VII.H2.3-a	3.3.1-05	386, A
			Atmosphere/ Weather (Ext)	Loss of Material	System Monitoring Program			F
Heat Exchangers	Fluid Pressure Boundary	Carbon Steel	Air (Int)	Loss of Material	One-Time Inspection Program	VII.H2.4-a	3.3.1-05	A
			Oil (Int)	Loss of Material	One-Time Inspection Program	VII.G.7-b	3.3.1-06	D
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A

Table 3.3.2-6 Auxiliary Systems- Emergency Power System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Heat Exchangers	Fluid Pressure Boundary	Carbon Steel	Raw Water (Int)	Buildup of Deposit	Open Cycle Cooling Water Program			F
				Heat Transfer Degradation	Open Cycle Cooling Water Program	VII.C1.3-a	3.3.1-17	H
				Loss of Material	Open Cycle Cooling Water Program	VIII.D1.3-a	3.4.1-02	C
			Treated Water (Int)	Heat Transfer Degradation	Closed Cycle Cooling Water Program	VII.H2.1-a	3.3.1-15	H
				Loss of Material	Closed Cycle Cooling Water Program	VII.H2.1-a	3.3.1-15	B
		Copper Alloys	Air (Ext)	Loss of Material	One-Time Inspection Program			F
			Oil (Ext)	Heat Transfer Degradation	One-Time Inspection Program	VII.G.7-b	3.3.1-06	H
				Loss of Material - Selective Leaching	One-Time Inspection Program	VII.G.7-b	3.3.1-06	301, H
				Loss of Material	One-Time Inspection Program	VII.G.7-b	3.3.1-06	D
			Raw Water (Int)	Buildup of Deposit	Open Cycle Cooling Water Program	VII.C1.3-b	3.3.1-17	A
				Heat Transfer Degradation	Open Cycle Cooling Water Program	VII.C1.3-a	3.3.1-17	396, A
						VII.C1.3-b	3.3.1-17	396, A
				Loss of Material - Selective Leaching	One-Time Inspection Program	VII.C1.3-a	3.3.1-29	301, B

Table 3.3.2-6 Auxiliary Systems- Emergency Power System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Heat Exchangers	Fluid Pressure Boundary	Copper Alloys	Raw Water (Int)	Loss of Material	Open Cycle Cooling Water Program	VII.C1.3-a	3.3.1-17	396, A
			Treated Water (Ext)	Heat Transfer Degradation	Closed Cycle Cooling Water Program	VII.H2.1-a	3.3.1-15	H
				Loss of Material	Closed Cycle Cooling Water Program	VII.C1.3-a	3.3.1-17	B
				Loss of Material - Selective Leaching	One-Time Inspection Program	VII.C1.3-a	3.3.1-29	301, B
			Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	VII.C1.3-a	3.3.1-17	B
				Loss of Material - Selective Leaching	One-Time Inspection Program	VII.C1.3-a	3.3.1-29	301, B
	Heat Transfer	Copper Alloys	Air (Ext)	Loss of Material	One-Time Inspection Program			F
			Oil (Ext)	Heat Transfer Degradation	One-Time Inspection Program	VII.G.7-b	3.3.1-06	H
				Loss of Material - Selective Leaching	One-Time Inspection Program	VII.G.7-b	3.3.1-06	301, H
				Loss of Material	One-Time Inspection Program	VII.G.7-b	3.3.1-06	D
			Raw Water (Int)	Buildup of Deposit	Open Cycle Cooling Water Program	VII.C1.3-b	3.3.1-17	A
				Heat Transfer Degradation	Open Cycle Cooling Water Program	VII.C1.3-a	3.3.1-17	396, A
						VII.C1.3-b	3.3.1-17	396, A

Table 3.3.2-6 Auxiliary Systems- Emergency Power System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Heat Exchangers	Heat Transfer	Copper Alloys	Raw Water (Int)	Loss of Material - Selective Leaching	One-Time Inspection Program	VII.C1.3-a	3.3.1-29	301, B
				Loss of Material	Open Cycle Cooling Water Program	VII.C1.3-a	3.3.1-17	396, A
			Treated Water (Ext)	Heat Transfer Degradation	Closed Cycle Cooling Water Program	VII.H2.1-a	3.3.1-15	H
				Loss of Material	Closed Cycle Cooling Water Program	VII.C1.3-a	3.3.1-17	B
				Loss of Material - Selective Leaching	One-Time Inspection Program	VII.C1.3-a	3.3.1-29	301, B
			Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	VII.C1.3-a	3.3.1-17	B
				Loss of Material - Selective Leaching	One-Time Inspection Program	VII.C1.3-a	3.3.1-29	301, B
Heaters, Electric	Fluid Pressure Boundary	Carbon Steel	Oil (Int)	Loss of Material	One-Time Inspection Program	VII.G.7-b	3.3.1-06	D
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Raw Water (Int)	Buildup of Deposit	Open Cycle Cooling Water Program	VII.H2.1-b	3.3.1-17	380, C
				Loss of Material	Open Cycle Cooling Water Program	VII.H2.1-b	3.3.1-17	C
		Stainless Steel	Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	VII.C2.2-a	3.3.1-15	D

Table 3.3.2-6 Auxiliary Systems- Emergency Power System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Muffler	Fluid Pressure Boundary	Carbon Steel	Air (Int)	Loss of Material	One-Time Inspection Program	VII.H2.3-a	3.3.1-05	A
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
Instrument Valve Assemblies & Tubing	Fluid Pressure Boundary	Bronze	Air (Int)	Cracking	One-Time Inspection Program			J
				Loss of Material	One-Time Inspection Program			J
				Loss of Material - Selective Leaching	One-Time Inspection Program			J
		Carbon Steel	Oil (Int)	Loss of Material	Diesel Fuel Monitoring and Storage Program	VII.H2.5-a	3.3.1-07	D
					One-Time Inspection Program	VII.G.7-b	3.3.1-06	D
						VII.H2.5-a	3.3.1-07	D
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
		Stainless Steel	Oil (Int)	Loss of Material	Diesel Fuel Monitoring and Storage Program			F
					One-Time Inspection Program			F
			Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program			F

Table 3.3.2-6 Auxiliary Systems- Emergency Power System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Motors	Fluid Pressure Boundary	Carbon Steel	Air (Int)	Loss of Material	One-Time Inspection Program	VII.H2.4-a	3.3.1-05	A
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
Pipe & Fittings	Fluid Pressure Boundary	Carbon Steel	Air (Int)	Loss of Material	One-Time Inspection Program	VII.H2.2-a	3.3.1-05	A
						VII.H2.3-a	3.3.1-05	A
						VII.H2.4-a	3.3.1-05	A
			Atmosphere/ Weather (Ext)	Loss of Material	System Monitoring Program	VII.H1.1-a	3.3.1-05	A
			Oil (Int)	Loss of Material	Diesel Fuel Monitoring and Storage Program	VII.H2.5-a	3.3.1-07	D
					One-Time Inspection Program	VII.H2.5-a	3.3.1-07	D
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	VII.H2.1-a	3.3.1-15	B
		Copper Alloys	Air (Int)	Cracking	One-Time Inspection Program			F
				Loss of Material	One-Time Inspection Program			F

Table 3.3.2-6 Auxiliary Systems- Emergency Power System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Pipe & Fittings	Fluid Pressure Boundary	Copper Alloys	Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program			F
		Stainless Steel	Air (Int)	Loss of Material	One-Time Inspection Program			F
Pumps	Fluid Pressure Boundary	Carbon Steel	Oil (Int)	Loss of Material	Diesel Fuel Monitoring and Storage Program	VII.G.8-a	3.3.1-22	B
					Fire Protection Program	VII.G.8-a	3.3.1-22	B
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
		Cast Iron	Oil (Int)	Loss of Material	Diesel Fuel Monitoring and Storage Program	VII.H2.5-a	3.3.1-07	399, D
					One-Time Inspection Program			J
				Loss of Material - Selective Leaching	One-Time Inspection Program			J
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	399, A
			Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	VII.C2.3-a	3.3.1-15	B
				Loss of Material - Selective Leaching	One-Time Inspection Program	VII.C2.3-a	3.3.1-29	301, B

Table 3.3.2-6 Auxiliary Systems- Emergency Power System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Traps (Steam)	Fluid Pressure Boundary	Cast Iron	Air (Int)	Loss of Material	One-Time Inspection Program	VII.H2.2-a	3.3.1-05	399, A
				Loss of Material - Selective Leaching	One-Time Inspection Program	VII.H2.2-a	3.3.1-05	399, H
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.2-b	3.3.1-24	399, A
Valves & Dampers	Fluid Pressure Boundary	Brass	Air (Int)	Cracking	One-Time Inspection Program			F
				Loss of Material	One-Time Inspection Program			F
				Loss of Material - Selective Leaching	One-Time Inspection Program			F
			Oil (Int)	Cracking	Diesel Fuel Monitoring and Storage Program			F
					One-Time Inspection Program			F
				Loss of Material	Diesel Fuel Monitoring and Storage Program			F
					One-Time Inspection Program			F
		Bronze	Air (Int)	Cracking	One-Time Inspection Program			F
				Loss of Material	One-Time Inspection Program			F

Table 3.3.2-6 Auxiliary Systems- Emergency Power System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Valves & Dampers	Fluid Pressure Boundary	Bronze	Oil (Int)	Cracking	Diesel Fuel Monitoring and Storage Program			F
					One-Time Inspection Program			F
				Loss of Material	Diesel Fuel Monitoring and Storage Program			F
					One-Time Inspection Program			F
			Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program			F
		Carbon Steel	Air (Int)	Loss of Material	Diesel Fuel Monitoring and Storage Program			G
					One-Time Inspection Program	VII.H2.2-a	3.3.1-05	B
						VIII.H2.2-a	3.4.1-05	A
			Atmosphere/ Weather (Ext)	Loss of Material	System Monitoring Program	VII.H1.2-a	3.3.1-05	A
				Loss of Material	Diesel Fuel Monitoring and Storage Program	VII.H2.5-a	3.3.1-07	D
					One-Time Inspection Program	VII.H2.5-a	3.3.1-07	D
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A

Table 3.3.2-6 Auxiliary Systems- Emergency Power System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Valves & Dampers	Fluid Pressure Boundary	Carbon Steel	Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	VII.H2.1-a	3.3.1-15	D
		Copper Alloys	Oil (Int)	Cracking	Diesel Fuel Monitoring and Storage Program			F
					Fire Protection Program			F
				Loss of Material	Diesel Fuel Monitoring and Storage Program			F
					Fire Protection Program			F
		Stainless Steel	Air (Int)	Loss of Material	One-Time Inspection Program			F
			Oil (Int)	Loss of Material	Diesel Fuel Monitoring and Storage Program			F
					One-Time Inspection Program			F

Table 3.3.2-7 Auxiliary Systems- Fire Protection System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Accumulators	Fluid Pressure Boundary	Stainless Steel	Oil (Int)	Loss of Material	One-Time Inspection Program			371, 372, F
Fasteners	Fluid Pressure Boundary	Carbon Steel	Atmosphere/ Weather (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	347, A
				Loss of Preload	Bolting Integrity Program	VII.I.2-a	3.3.1-24	324, E
			Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
				Loss of Preload	Bolting Integrity Program	VII.I.2-a	3.3.1-24	324, I
			Raw Water (Ext)	Loss of Material	Fire Protection Program	VII.G.6-b	3.3.1-21	348, D
				Loss of Preload	Bolting Integrity Program			324, J
			Soil (Ext)	Loss of Material	Fire Protection Program	VII.G.6-b	3.3.1-21	336, 349, D
				Loss of Preload	Bolting Integrity Program			324, J
		Stainless Steel	Plant Indoor Air (Ext)	Loss of Preload	Bolting Integrity Program			324, F

Table 3.3.2-7 Auxiliary Systems- Fire Protection System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Filters/Strainers	Filtration	Carbon Steel	Raw Water (Ext)	Loss of Material	Fire Protection Program	VII.G.6-b	3.3.1-21	B
			Raw Water (Int)	Loss of Material	Fire Protection Program	VII.G.6-b	3.3.1-21	B
	Fluid Pressure Boundary	Bronze	Raw Water (Int)	Buildup of Deposit	Fire Protection Program	VII.G.6-b	3.3.1-21	B
				Loss of Material	Fire Protection Program	VII.G.6-b	3.3.1-21	370, B
		Cast Iron	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program			335, G
			Raw Water (Int)	Buildup of Deposit	Fire Protection Program	VII.G.6-b	3.3.1-21	B
				Loss of Material	Fire Protection Program	VII.G.6-b	3.3.1-21	338, 378, 379, B
				Loss of Material - Selective Leaching	One-Time Inspection Program	VII.G.6-b	3.3.1-21	338, H
Pipe & Fittings	Fluid Pressure Boundary	Bare Copper	Raw Water (Int)	Buildup of Deposit	Fire Protection Program	VII.G.6-b	3.3.1-21	D
				Loss of Material	Fire Protection Program	VII.G.6-b	3.3.1-21	D
		Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	385, A
			Raw Water (Int)	Buildup of Deposit	Fire Protection Program	VII.G.6-a	3.3.1-21	B
				Loss of Material	Fire Protection Program	VII.G.6-a	3.3.1-21	B
			Soil (Ext)	Loss of Material	Fire Protection Program	VII.G.6-a	3.3.1-21	334, 350, B

Table 3.3.2-7 Auxiliary Systems- Fire Protection System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Pipe & Fittings	Fluid Pressure Boundary	Cast Iron	Raw Water (Ext)	Loss of Material	Fire Protection Program	VII.G.6-a	3.3.1-21	351, 352, B
				Loss of Material - Selective Leaching	One-Time Inspection Program	VII.G.6-a	3.3.1-21	352, H
			Raw Water (Int)	Buildup of Deposit	Fire Protection Program	VII.G.6-a	3.3.1-21	B
				Loss of Material	Fire Protection Program	VII.G.6-a	3.3.1-21	351, 352, B
				Loss of Material - Selective Leaching	One-Time Inspection Program	VII.G.6-a	3.3.1-21	352, H
		Copper Alloys	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program			353, F
			Oil (Int)	Loss of Material	One-Time Inspection Program	VII.G.7-b	3.3.1-06	373, 374, B
		Stainless Steel	Raw Water (Int)	Buildup of Deposit	Fire Protection Program	VII.G.6-a	3.3.1-21	B
				Loss of Material	Fire Protection Program	VII.G.6-a	3.3.1-21	B
Pumps	Fluid Pressure Boundary	Carbon Steel	Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Raw Water (Int)	Buildup of Deposit	Fire Protection Program	VII.G.6-b	3.3.1-21	B
				Loss of Material	Fire Protection Program	VII.G.6-b	3.3.1-21	339, B

Table 3.3.2-7 Auxiliary Systems- Fire Protection System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Pumps	Fluid Pressure Boundary	Cast Iron	Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Raw Water (Ext)	Loss of Material	Fire Protection Program	VII.G.6-b	3.3.1-21	340, B
				Loss of Material - Selective Leaching	One-Time Inspection Program	VII.G.6-b	3.3.1-21	340, H
			Raw Water (Int)	Buildup of Deposit	Fire Protection Program	VII.G.6-b	3.3.1-21	B
				Loss of Material	Fire Protection Program	VII.G.6-b	3.3.1-21	340, B
				Loss of Material - Selective Leaching	One-Time Inspection Program	VII.G.6-b	3.3.1-21	340, H
Sprinkler Heads	Fluid Pressure Boundary Spray Pattern	Brass	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program			346, F
			Raw Water (Int)	Buildup of Deposit	Fire Protection Program	VII.G.6-b	3.3.1-21	376, 377, H
				Loss of Material	Fire Protection Program	VII.G.6-b	3.3.1-21	375, B
Valves & Dampers	Fluid Pressure Boundary	Brass	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program			F
					System Monitoring Program			357, 383, F
			Raw Water (Int)	Buildup of Deposit	Fire Protection Program			F
				Loss of Material	Fire Protection Program			341, F

Table 3.3.2-7 Auxiliary Systems- Fire Protection System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Valves & Dampers	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Raw Water (Int)	Buildup of Deposit	Fire Protection Program	VII.G.6-b	3.3.1-21	B
				Loss of Material	Fire Protection Program	VII.G.6-b	3.3.1-21	342, B
		Cast Iron	Air (Int)	Loss of Material	Fire Protection Program			G
			Atmosphere/ Weather (Ext)	Loss of Material	Fire Protection Program			F
					System Monitoring Program			337, F
			Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program			F
					System Monitoring Program			384, F
			Raw Water (Int)	Buildup of Deposit	Fire Protection Program	VII.G.6-b	3.3.1-21	B
				Loss of Material	Fire Protection Program	VII.G.6-b	3.3.1-21	343, 344, B
					One-Time Inspection Program	VII.G.6-b	3.3.1-21	344, H
			Soil (Ext)	Loss of Material	Fire Protection Program	VII.G.6-b	3.3.1-21	358, 359, B
				Loss of Material - Selective Leaching	One-Time Inspection Program	VII.G.6-b	3.3.1-21	359, 384, H
		Stainless Steel	Raw Water (Int)	Buildup of Deposit	Fire Protection Program	VII.G.6-b	3.3.1-21	H
				Loss of Material	Fire Protection Program	VII.G.6-b	3.3.1-21	346, B

Table 3.3.2-8 Auxiliary Systems - Fuel Oil System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Accumulators	Fluid Pressure Boundary	Carbon Steel	Air (Int)	Loss of Material	Diesel Fuel Monitoring and Storage Program			319, G
			Oil (Int)	Loss of Material	Diesel Fuel Monitoring and Storage Program	VII.H1.4-a	3.3.1-07	305, B
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.H1.4-a	3.3.1-23	310, A
Fasteners	Fluid Pressure Boundary	Low-Alloy Steel	Plant Indoor Air (Ext)	Loss of Material	Bolting Integrity Program	VII.I.2-a	3.3.1-24	A
				Loss of Preload	Bolting Integrity Program	VII.I.2-a	3.3.1-24	324, A
Filters/Strainers	Fluid Pressure Boundary	Carbon Steel	Oil (Int)	Loss of Material	Diesel Fuel Monitoring and Storage Program	VII.H1.4-a	3.3.1-07	D
					One-Time Inspection Program	VII.H1.4-a	3.3.1-07	D
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
Indicators/ Recorders	Fluid Pressure Boundary	Carbon Steel	Oil (Int)	Loss of Material	Diesel Fuel Monitoring and Storage Program	VII.H1.4-a	3.3.1-07	D
					One-Time Inspection Program	VII.H1.4-a	3.3.1-07	D
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	C
Pipe & Fittings	Fire Barrier	Carbon Steel	Air (Int)	Loss of Material	Diesel Fuel Monitoring and Storage Program			307, 319, G
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.H1.1-b	3.3.1-18	A

Table 3.3.2-8 Auxiliary Systems - Fuel Oil System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Pipe & Fittings	Flow Restriction	Carbon Steel	Oil (Int)	Loss of Material	Diesel Fuel Monitoring and Storage Program	VII.H1.4-a	3.3.1-07	D
					One-Time Inspection Program	VII.H1.4-a	3.3.1-07	D
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.H1.1-b	3.3.1-18	A
	Fluid Pressure Boundary	Carbon Steel	Air (Int)	Loss of Material	Diesel Fuel Monitoring and Storage Program			307, 319, G
			Atmosphere/ Weather (Ext)	Loss of Material	System Monitoring Program	VII.H1.1-a	3.3.1-05	A
			Oil (Int)	Loss of Material	Diesel Fuel Monitoring and Storage Program	VII.H1.4-a	3.3.1-07	D
					Fire Protection Program	VII.G.8-a	3.3.1-22	B
					One-Time Inspection Program	VII.H1.4-a	3.3.1-07	D
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.H1.1-b	3.3.1-18	A
			Raw Water (Ext)	Loss of Material	Buried Services Corrosion Monitoring Program	VII.H1.1-b	3.3.1-18	A
		Copper Alloys	Oil (Int)	Cracking	Diesel Fuel Monitoring and Storage Program			317, F
					One-Time Inspection Program			317, F

Table 3.3.2-8 Auxiliary Systems - Fuel Oil System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Pipe & Fittings	Fluid Pressure Boundary	Copper Alloys	Oil (Int)	Loss of Material	Diesel Fuel Monitoring and Storage Program			317, F
					One-Time Inspection Program			317, F
		Stainless Steel	Oil (Int)	Loss of Material	Diesel Fuel Monitoring and Storage Program			317, F
					Fire Protection Program			317, F
					One-Time Inspection Program			317, F
Pumps	Fluid Pressure Boundary	Cast Iron	Oil (Int)	Loss of Material	Diesel Fuel Monitoring and Storage Program	VII.H1.4-b	3.3.1-23	D
					One-Time Inspection Program	VII.H1.4-b	3.3.1-23	D
				Loss of Material - Selective Leaching	One-Time Inspection Program	VII.H1.4-b	3.3.1-23	301, D
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.H1.4-b	3.3.1-23	A
Valves & Dampers	Fluid Pressure Boundary	Brass	Oil (Int)	Cracking	Diesel Fuel Monitoring and Storage Program			317, F
					One-Time Inspection Program			317, F
				Loss of Material	Diesel Fuel Monitoring and Storage Program			317, F
					One-Time Inspection Program			317, F

Table 3.3.2-8 Auxiliary Systems - Fuel Oil System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Valves & Dampers	Fluid Pressure Boundary	Bronze	Oil (Int)	Cracking	Diesel Fuel Monitoring and Storage Program			317, F
					One-Time Inspection Program			317, F
				Loss of Material	Diesel Fuel Monitoring and Storage Program			317, F
					One-Time Inspection Program			317, F
		Carbon Steel	Oil (Int)	Loss of Material	Diesel Fuel Monitoring and Storage Program	VII.H1.4-a	3.3.1-07	D
					Fire Protection Program	VII.G.8-a	3.3.1-22	D
					One-Time Inspection Program	VII.H1.4-a	3.3.1-07	D
			Plant Indoor Air (Ext)	Loss of Material	Fire Protection Program	VII.I.1-b	3.3.1-05	B
					System Monitoring Program	VII.H1.1-b	3.3.1-18	A
		Stainless Steel	Oil (Int)	Loss of Material	Diesel Fuel Monitoring and Storage Program			317, F
					One-Time Inspection Program			317, F

Table 3.3.2-9 Auxiliary Systems- Heating, Ventilation, and Air Conditioning System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Blowers Fans Compressor Vacuum	Fluid Pressure Boundary	Carbon Steel	Air (Int)	Loss of Material	One-Time Inspection Program	VII.F1.1-a	3.3.1-05	391, B
						VII.F2.1-a	3.3.1-05	391, B
						VII.F4.1-a	3.3.1-05	391, B
			Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
		Galvanized	Air (Int)	Loss of Material	One-Time Inspection Program	VII.F1.1-a	3.3.1-05	391, B
			Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	386, A
Ductwork	Filtration Fluid Pressure Boundary	Galvanized	Air (Int)	Loss of Material	One-Time Inspection Program	VII.F1.1-a	3.3.1-05	386, B
						VII.F2.1-a	3.3.1-05	386, B
						VII.F4.1-a	3.3.1-05	386, B
Filters/Strainers	Filtration Fluid Pressure Boundary	Carbon Steel	Air (Int)	Loss of Material	One-Time Inspection Program	VII.F1.4-a	3.3.1-05	B
						VII.F2.4-a	3.3.1-05	B
			Oil (Int)	Loss of Material	One-Time Inspection Program	VII.G.7-b	3.3.1-06	392, D

Table 3.3.2-9 Auxiliary Systems- Heating, Ventilation, and Air Conditioning System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Filters/Strainers	Filtration Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
		Galvanized	Air (Int)	Loss of Material	One-Time Inspection Program	VII.F2.4-a	3.3.1-05	386, B
			Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	386, A
Heat Exchangers	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Raw Water (Int)	Buildup of Deposit	Open Cycle Cooling Water Program	VII.C1.3-a	3.3.1-17	380, A
				Loss of Material	Open Cycle Cooling Water Program	VII.C1.3-a	3.3.1-17	C
			Steam (Int)	Loss of Material	One-Time Inspection Program			393, G

Table 3.3.2-9 Auxiliary Systems- Heating, Ventilation, and Air Conditioning System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Heat Exchangers	Fluid Pressure Boundary Heat Transfer	Copper Alloys	Air (Ext)	Heat Transfer Degradation	One-Time Inspection Program	VII.F1.2-a	3.3.1-05	387, H
						VII.F2.2-a	3.3.1-05	387, H
				Loss of Material	One-Time Inspection Program	VII.F1.2-a	3.3.1-05	323, B
						VII.F2.2-a	3.3.1-05	323, B
			Gas (Ext)	Loss of Material	One-Time Inspection Program			323, 388, G
			Raw Water (Int)	Buildup of Deposit	Open Cycle Cooling Water Program	VII.C1.3-b	3.3.1-17	C
				Heat Transfer Degradation	Open Cycle Cooling Water Program	VII.C1.3-b	3.3.1-17	316, 387, C
				Loss of Material - Erosion	One-Time Inspection Program	VII.C1.3-a	3.3.1-17	394, H
				Loss of Material - Selective Leaching	One-Time Inspection Program	VII.C1.3-a	3.3.1-29	301, D
				Loss of Material	Open Cycle Cooling Water Program	VII.C1.3-a	3.3.1-17	394, C
	Fluid Pressure Boundary	Copper Alloys	Steam (Int)	Loss of Material	One-Time Inspection Program			393, G

Table 3.3.2-9 Auxiliary Systems- Heating, Ventilation, and Air Conditioning System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Heaters, Electric	Fluid Pressure Boundary	Carbon Steel	Air (Int)	Loss of Material	One-Time Inspection Program	VII.F1.1-a	3.3.1-05	391, B
						VII.F2.1-a	3.3.1-05	391, B
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
		Galvanized	Air (Int)	Loss of Material	One-Time Inspection Program	VII.F1.1-a	3.3.1-05	391, B
Muffler (CRHVAC Refrig. condensing units)	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
Elastomers in Flex. Connections and Seals inside/outside of Containment	Fluid Pressure Boundary	Elastomers	Air (Int)	Change in Mat. Properties	One-Time Inspection Program	VII.F1.1-b	3.3.1-02	389, B
						VII.F1.4-b	3.3.1-02	389, B
						VII.F2.1-b	3.3.1-02	389, B
						VII.F2.4-b	3.3.1-02	389, B
						VII.F3.1-b	3.3.1-02	389, B
						VII.F3.4-b	3.3.1-02	389, B
						VII.F4.1-b	3.3.1-02	389, B

Table 3.3.2-9 Auxiliary Systems- Heating, Ventilation, and Air Conditioning System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Elastomers in Flex. Connections and Seals inside/outside of Containment	Fluid Pressure Boundary	Elastomers	Air (Int)	Cracking	One-Time Inspection Program	VII.F1.1-b	3.3.1-02	389, B
						VII.F1.4-b	3.3.1-02	389, B
						VII.F2.1-b	3.3.1-02	389, B
						VII.F2.4-b	3.3.1-02	389, B
						VII.F3.1-b	3.3.1-02	389, B
						VII.F3.4-b	3.3.1-02	389, B
						VII.F4.1-b	3.3.1-02	389, B
			Containment Air (Ext)	Change in Mat. Properties	System Monitoring Program	VII.F3.1-b	3.3.1-02	389, A
						VII.F3.4-b	3.3.1-02	389, A
				Cracking	System Monitoring Program	VII.F3.1-b	3.3.1-02	389, A
						VII.F3.4-b	3.3.1-02	389, A
			Plant Indoor Air (Ext)	Change in Mat. Properties	System Monitoring Program	VII.F1.1-b	3.3.1-02	389, A
						VII.F1.4-b	3.3.1-02	389, A
						VII.F2.1-b	3.3.1-02	389, A
						VII.F2.4-b	3.3.1-02	389, A

Table 3.3.2-9 Auxiliary Systems- Heating, Ventilation, and Air Conditioning System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Elastomers in Flex. Connections and Seals inside/outside of Containment	Fluid Pressure Boundary	Elastomers	Plant Indoor Air (Ext)	Cracking	System Monitoring Program	VII.F1.1-b	3.3.1-02	389, A
						VII.F1.4-b	3.3.1-02	389, A
						VII.F2.1-b	3.3.1-02	389, A
						VII.F2.4-b	3.3.1-02	389, A
CRHVAC Duct Silencer	Fluid Pressure Boundary	Galvanized	Air (Int)	Loss of Material	One-Time Inspection Program	VII.F1.1-a	3.3.1-05	D
Fasteners	Fluid Pressure Boundary	Low-Alloy Steel	Plant Indoor Air (Ext)	Loss of Preload	Bolting Integrity Program	VII.I.2-a	3.3.1-24	324, C
				Loss of Material	Bolting Integrity Program	VII.I.2-a	3.3.1-24	324, C
					Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
HVAC Dampers	Fluid Pressure Boundary	Galvanized	Air (Int)	Loss of Material	One-Time Inspection Program	VII.F1.1-a	3.3.1-05	391, B
						VII.F2.1-a	3.3.1-05	391, B
						VII.F4.1-a	3.3.1-05	391, B

Table 3.3.2-9 Auxiliary Systems- Heating, Ventilation, and Air Conditioning System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Pipe & Fittings	Fluid Pressure Boundary	Carbon Steel	Air (Int)	Loss of Material	One-Time Inspection Program	VII.F1.1-a	3.3.1-05	D
						VII.F2.1-a	3.3.1-05	D
						VII.F4.1-a	3.3.1-05	D
			Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Raw Water (Int)	Loss of Material	One-Time Inspection Program			395, G
			Steam (Int)	Loss of Material	One-Time Inspection Program			393, G
		Copper Alloys	Air (Int)	Loss of Material	One-Time Inspection Program	VII.F1.2-a	3.3.1-05	D
						VII.F2.2-a	3.3.1-05	D
						VII.F4.2-a	3.3.1-05	D
			Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program			317, F
		Stainless Steel	Air (Int)	Loss of Material	One-Time Inspection Program			308, F
	Containment Isolation	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.C.1-a	3.3.1-05	A

Table 3.3.2-9 Auxiliary Systems- Heating, Ventilation, and Air Conditioning System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Traps (Steam)	Fluid Pressure Boundary	Cast Iron	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	390, A
			Steam (Int)	Loss of Material	One-Time Inspection Program			390, 393, G
			Raw Water (Int)	Loss of Material	One-Time Inspection Program			390, 393, G
Valves & Dampers	Fluid Pressure Boundary	Bronze	Air (Int)	Loss of Material	One-Time Inspection Program	VII.F1.2-a	3.3.1-05	D
						VII.F2.2-a	3.3.1-05	D
						VII.F4.2-a	3.3.1-05	D
			Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program			317, F
			Steam (Int)	Loss of Material	One-Time Inspection Program			393, G
		Carbon Steel	Air (Int)	Loss of Material	One-Time Inspection Program	VII.F1.1-a	3.3.1-05	D
						VII.F2.1-a	3.3.1-05	D
						VII.F4.1-a	3.3.1-05	D
			Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A

Table 3.3.2-9 Auxiliary Systems- Heating, Ventilation, and Air Conditioning System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Valves & Dampers	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Raw Water (Int)	Loss of Material	One-Time Inspection Program			395, G
			Steam	Loss of Material	One-Time Inspection Program			G
					System Monitoring Program			G
		Galvanized	Air (Int)	Loss of Material	One-Time Inspection Program	VII.F1.1-a	3.3.1-05	391, B
						VII.F2.1-a	3.3.1-05	391, B
						VII.F4.1-a	3.3.1-05	391, B
			Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	386, A
		Stainless Steel	Air (Int)	Loss of Material	One-Time Inspection Program			308, F
			Raw Water (Int)	Loss of Material	One-Time Inspection Program			308, F

Table 3.3.2-10 Auxiliary Systems - Miscellaneous Gas System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Accumulators	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
Filters/Strainers	Fluid Pressure Boundary	Galvanized	Air (Int)	Loss of Material	One-Time Inspection Program	VII.D.5-a	3.3.1-19	E
			Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
Misc. Mechanical (fasteners, manifold, monitor)	Fluid Pressure Boundary	Carbon Steel	Containment Air (Ext) Plant Indoor Air (Ext)	Loss of Preload	Bolting Integrity Program	VII.D.3-a	3.3.1-19	324, E
				Loss of Material	Bolting Integrity Program	VII.D.1-a	3.3.1-19	E
						VII.D.2-a	3.3.1-19	E
						VII.D.3-a	3.3.1-19	E
						VII.D.5-a	3.3.1-19	E
						VII.D.6-a	3.3.1-19	E
					Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
		Copper Alloys	Containment Air (Ext) Plant Indoor Air (Ext)	Loss of Preload	Bolting Integrity Program			324, F
				Loss of Material	Bolting Integrity Program			F

Table 3.3.2-10 Auxiliary Systems - Miscellaneous Gas System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Misc. Mechanical (fasteners, manifold, monitor)	Fluid Pressure Boundary	Stainless Steel	Plant Indoor Air (Ext)	Loss of Preload	Bolting Integrity Program			324, F
Pipe & Fittings	Fluid Pressure Boundary	Carbon Steel	Atmosphere/ Weather (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Containment Air (Ext) Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Soil (Ext)	Loss of Material	Buried Services Corrosion Monitoring Program	VII.C1.1-b	3.3.1-18	A
		Copper Alloys	Containment Air (Ext) Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program			F
Tanks	Fluid Pressure Boundary	Carbon Steel	Atmosphere/ Weather (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A

Table 3.3.2-10 Auxiliary Systems - Miscellaneous Gas System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Valves & Dampers	Fluid Pressure Boundary	Brass	Containment Air (Ext) Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program			F
		Bronze	Containment Air (Ext) Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program			F
		Carbon Steel	Atmosphere/ Weather (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Containment Air (Ext) Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
		Low-Alloy Steel	Containment Air (Ext) Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A

Table 3.3.2-11 Auxiliary Systems- Radwaste System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Accumulators	Fluid Pressure Boundary	Carbon Steel	Air (Int)	Loss of Material	One-Time Inspection Program			331, J
			Atmosphere / Weather (Ext)	Loss of Material	System Monitoring Program	VII.H1.4-b	3.3.1-23	331, E
	Flood Protection Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Raw Water (Int)	Loss of Material	One-Time Inspection Program	VII.C3.1-a	3.3.1-29	331, E
		Stainless Steel	Treated Water (Int)	Loss of Material	One-Time Inspection Program	VII.C2.2-a	3.3.1-15	331, 381, E
Demineralizer	Fluid Pressure Boundary	Stainless Steel	Treated Water (Int)	Loss of Material	One-Time Inspection Program	VII.C2.2-a	3.3.1-15	331, 381, E
Filters/Strainers	Fluid Pressure Boundary	Stainless Steel	Raw Water (Int)	Loss of Material	One-Time Inspection Program	VII.C1.6-a	3.3.1-17	331, E
			Treated Water (Int)	Loss of Material	One-Time Inspection Program	VII.C2.2-a	3.3.1-15	331, 381, E

Table 3.3.2-11 Auxiliary Systems- Radwaste System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Heat Exchangers	Fluid Pressure Boundary	Bronze	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program			308, F
			Treated Water (Ext)	Loss of Material	One-Time Inspection Program	VII.F2.2-a	3.3.1-05	331, 368, 381, E
			Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program			331, 368, J
					One-Time Inspection Program	VII.F2.2-a	3.3.1-05	331, 368, 381, E
		Carbon Steel	Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-03	A
			Steam (Int)	Loss of Material	One-Time Inspection Program			331, J
		Copper Alloys	Treated Water (Ext)	Loss of Material	One-Time Inspection Program	VII.F2.2-a	3.3.1-05	331, 381, E
			Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program			331, 368, J
					One-Time Inspection Program	VII.F2.2-a	3.3.1-05	331, 381, E
		Stainless Steel	Treated Water (Int)	Loss of Material	One-Time Inspection Program	VII.C2.2-a	3.3.1-15	331, 381, E
Fasteners	Fluid Pressure Boundary	Carbon Steel	Containment Air (Ext)	Loss of Material	Bolting Integrity Program	VII.I.2-a	3.3.1-24	A
					Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A

Table 3.3.2-11 Auxiliary Systems- Radwaste System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Fasteners	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Bolting Integrity Program	VII.I.2-a	3.3.1-24	B
					Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
				Loss of Preload	Bolting Integrity Program	VII.I.2-a	3.3.1-24	324, A
		Stainless Steel	Plant Indoor Air (Ext)	Loss of Preload	Bolting Integrity Program			324, F
Pipe & Fittings	Flood Protection Fluid Pressure Boundary	Stainless Steel	Treated Water (Int)	Loss of Material	One-Time Inspection Program	VII.C2.2-a	3.3.1-15	331, 381, E
	Fluid Pressure Boundary	Carbon Steel	Gas (Int)	Loss of Material	One-Time Inspection Program			331, 381, G
			Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Raw Water (Int)	Loss of Material	One-Time Inspection Program	VII.C3.1-a	3.3.1-29	331, E
		Cast Iron	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	399, A
					System Monitoring Program	VII.I.1-b	3.3.1-05	399, A
			Treated Water (Int)	Loss of Material	One-Time Inspection Program			331, 381, F

Table 3.3.2-11 Auxiliary Systems- Radwaste System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Pumps	Flood Protection Fluid Pressure Boundary	Bronze	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program			331, G
			Treated Water (Int)	Loss of Material	One-Time Inspection Program	VII.F2.2-a	3.3.1-05	331, 381, E
		Stainless Steel	Treated Water (Int)	Loss of Material	One-Time Inspection Program	VII.C2.2-a	3.3.1-15	331, 381, E
	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Raw Water (Ext)	Loss of Material	One-Time Inspection Program			331, I
			Raw Water (Int)	Loss of Material	One-Time Inspection Program	VII.C1.5-a	3.3.1-17	331, E
		Stainless Steel	Treated Water (Int)	Loss of Material	One-Time Inspection Program	VII.C2.2-a	3.3.1-15	331, 381, E
Valves & Dampers	Flood Protection Fluid Pressure Boundary	Cast Austenitic SS	Treated Water (Int)	Loss of Material	One-Time Inspection Program	VII.C2.2-a	3.3.1-15	331, 381,E
		Cast Iron	Air (Int)	Loss of Material	One-Time Inspection Program	VII.F2.1-a	3.3.1-05	331, E
			Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	399, A
					System Monitoring Program	VII.I.1-b	3.3.1-05	399, A

Table 3.3.2-11 Auxiliary Systems- Radwaste System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Valves & Dampers	Fluid Pressure Boundary	Carbon Steel	Air (Int)	Loss of Material	One-Time Inspection Program	VII.F2.1-a	3.3.1-05	331, 369, E
			Gas (Int)	Loss of Material	One-Time Inspection Program	VII.F2.1-a	3.3.1-05	331, E
			Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Raw Water (Int)	Loss of Material	One-Time Inspection Program	VII.C1.2-a	3.3.1-17	331, E
		Stainless Steel	Treated Water (Int)	Loss of Material	One-Time Inspection Program	VII.C2.2-a	3.3.1-15	331, 381, E

Table 3.3.2-12 Auxiliary Systems- Service Water System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Accumulator	Fluid Pressure Boundary	Stainless Steel	Raw Water (Int)	Loss of Material	One-Time Inspection Program	VII.C1.1-a	3.3.1-17	E
Fasteners in Containment	Fluid Pressure Boundary	Low-Alloy Steel	Containment Air (Ext)	Loss of Material	Bolting Integrity Program	VII.I.2-a	3.3.1-24	A
					Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
				Loss of Preload	Bolting Integrity Program			324, F
Fasteners Not in Containment	Fluid Pressure Boundary	Low-Alloy Steel	Plant Indoor Air (Ext)	Loss of Material	Bolting Integrity Program	VII.I.1-b	3.3.1-05	A, 300
					Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
				Loss of Preload	Bolting Integrity Program			324, F
Filters/Strainers	Filtration	Stainless Steel	Raw Water (Ext)	Loss of Material	Open Cycle Cooling Water Program	VII.C1.6-a	3.3.1-17	A
	Fluid Pressure Boundary	Cast Iron	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Raw Water (Int)	Buildup of Deposit	Open Cycle Cooling Water Program	VII.C1.6-a	3.3.1-17	A, 399
				Loss of Material	One-Time Inspection Program	VII.C1.5-a	3.3.1-17	D, 301
					Open Cycle Cooling Water Program	VII.C1.6-a	3.3.1-17	A, 399
		Stainless Steel	Raw Water (Ext)	Loss of Material	Open Cycle Cooling Water Program	VII.C1.6-a	3.3.1-17	A

Table 3.3.2-12 Auxiliary Systems- Service Water System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Flex Connections & Instrument Valve Assemblies	Fluid Pressure Boundary	Stainless Steel	Raw Water (Int)	Buildup of Deposit	Open Cycle Cooling Water Program	VII.C1.2-a	3.3.1-17	A
				Loss of Material	Open Cycle Cooling Water Program	VII.C1.4-a	3.3.1-17	C
Heat Exchanger	Fluid Pressure Boundary	Carbon Steel	Air/Gas (Int)	Loss of Material	One-Time Inspection Program			G
			Oil (Int)	Loss of Material	One-Time Inspection Program			G
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Raw Water (Int)	Loss of Material	One-Time Inspection Program	VII.C1.1-a	3.3.1-17	E
		Stainless Steel	Treated Water (Int)	Loss of Material	One-Time Inspection Program	VII.I.3-a	3.3.1-17	E

Table 3.3.2-12 Auxiliary Systems- Service Water System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Pipe & Fittings	Flow Restriction	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Raw Water (Int)	Buildup of Deposit	Open Cycle Cooling Water Program	VII.C1.1-a	3.3.1-17	A
						VII.C1.2-a	3.3.1-17	A
						VII.C3.1-a	3.3.1-29	A
				Loss of Material	Open Cycle Cooling Water Program	VII.C1.1-a	3.3.1-17	A, 302
						VII.C3.1-a	3.3.1-29	A, 302
	Fluid Pressure Boundary	Bronze	Raw Water (Int	Loss of Material	Open Cycle Cooling Water Program	VII.C1.1-a	3.3.1-17	C
					One-Time Inspection Program	VII.C1.1-a	3.3.1-17	D
		Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Soil (Ext)	Loss of Material	Buried Services Corrosion Monitoring Program	VII.C1.1-b	3.3.1-18	A

Table 3.3.2-12 Auxiliary Systems- Service Water System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Pipe & Fittings	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Raw Water (Int)	Buildup of Deposit	Open Cycle Cooling Water Program	VII.C1.1-a	3.3.1-17	A
						VII.C1.2-a	3.3.1-17	A
						VII.C3.1-a	3.3.1-29	A
				Loss of Material	Open Cycle Cooling Water Program	VII.C1.1-a	3.3.1-17	A, 302
						VII.C3.1-a	3.3.1-29	A, 302
				Loss of Material - Erosion	Open Cycle Cooling Water Program	VII.C1.1-a	3.3.1-17	A, 302
						VII.C3.1-a	3.3.1-29	A, 302
		Cast Iron	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A, 399
			Raw Water (Int)	Buildup of Deposit	Open Cycle Cooling Water Program	VII.C1.1-a	3.3.1-17	A
				Loss of Material	One-Time Inspection Program	VII.C1.6-a	3.3.1-17	B, 399
					Open Cycle Cooling Water Program	VII.C1.6-a	3.3.1-17	A, 399

Table 3.3.2-12 Auxiliary Systems- Service Water System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Pipe & Fittings	Fluid Pressure Boundary	Copper Alloys	Raw Water (Int)	Buildup of Deposit	Open Cycle Cooling Water Program	VII.C1.1-a	3.3.1-17	A
				Loss of Material	Open Cycle Cooling Water Program	VII.C1.1-a	3.3.1-17	A
		Stainless Steel	Raw Water (Int)	Buildup of Deposit	Open Cycle Cooling Water Program	VII.C1.1-a	3.3.1-17	A, 302
				Loss of Material	Open Cycle Cooling Water Program	VII.C1.1-a	3.3.1-17	A
				Loss of Material - Erosion	Open Cycle Cooling Water Program	VII.C1.1-a	3.3.1-17	A,
		Rubber	Plant Indoor Air (Ext)	Cracking	System Monitoring Program			F
Pumps	Fluid Pressure Boundary	Cast Iron	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A, 399
			Raw Water (Int)	Buildup of Deposit	Open Cycle Cooling Water Program	VII.C1.5-a	3.3.1-17	A, 399
						VII.C3.3-a	3.3.1-17	A, 399
				Loss of Material	One-Time Inspection Program	VII.C1.5-a	3.3.1-29	C, 301
						VII.C3.3-a	3.3.1-17	C, 301
					Open Cycle Cooling Water Program	VII.C1.5-a	3.3.1-17	A

Table 3.3.2-12 Auxiliary Systems- Service Water System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Traveling Screen spray Nozzles	Spray Pattern	Aluminum Bronze	Raw Water (Int)	Buildup of Deposit	Open Cycle Cooling Water Program	VII.C1.3-b	3.3.1-17	C
				Loss of Material	One-Time Inspection Program	VII.C1.1-a	3.3.1-29	D, 301
					Open Cycle Cooling Water Program	VII.C1.1-a	3.3.1-17	C, 302
Valves & Dampers	Fluid Pressure Boundary	Brass	Raw Water (Int)	Buildup of Deposit	Open Cycle Cooling Water Program	VII.C1.2-a	3.3.1-17	A
				Loss of Material	Open Cycle Cooling Water Program	VII.C1.2-a	3.3.1-17	A
		Bronze	Raw Water (Int)	Buildup of Deposit	Open Cycle Cooling Water Program	VII.C1.2-a	3.3.1-17	A
				Loss of Material	Open Cycle Cooling Water Program	VII.C1.2-a	3.3.1-17	A
		Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A

Table 3.3.2-12 Auxiliary Systems- Service Water System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Valves & Dampers	Fluid Pressure Boundary	Carbon Steel	Raw Water (Int)	Buildup of Deposit	Open Cycle Cooling Water Program	VII.C1.1-a	3.3.1-17	C
						VII.C3.2-a	3.3.1-29	C
				Loss of Material	Open Cycle Cooling Water Program	VII.C1.2-a	3.3.1-17	A, 302
						VII.C3.2-a	3.3.1-29	A, 302
		Cast Iron	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A, 399
					System Monitoring Program	VII.I.1-b	3.3.1-05	A, 399
			Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A, 399
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A, 399
			Raw Water (Int)	Buildup of Deposit	Open Cycle Cooling Water Program	VII.C1.2-a	3.3.1-17	A, 399
				Loss of Material	One-Time Inspection Program	VII.C1.2-a	3.3.1-17	B, 301, 399
					Open Cycle Cooling Water Program	VII.C1.2-a	3.3.1-17	A, 302, 399
				Loss of Material - Erosion	Open Cycle Cooling Water Program	VII.C1.2-a	3.3.1-17	A, 302, 399
		Copper Alloys	Raw Water (Int)	Buildup of Deposit	Open Cycle Cooling Water Program	VII.C1.2-a	3.3.1-17	A
				Loss of Material	Open Cycle Cooling Water Program	VII.C1.2-a	3.3.1-17	A

Table 3.3.2-12 Auxiliary Systems- Service Water System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Valves & Dampers	Fluid Pressure Boundary	Stainless Steel	Raw Water (Int)	Buildup of Deposit	Open Cycle Cooling Water Program	VII.C1.2-a	3.3.1-17	A
				Loss of Material	Open Cycle Cooling Water Program	VII.C1.2-a	3.3.1-17	A

Table 3.3.2-13 Auxiliary Systems - Shield Cooling System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Accumulators	Fluid Pressure Boundary	Carbon Steel	Containment Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	C
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	C
			Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	VII.C2.4-a	3.3.1-15	327, B
Fasteners	Fluid Pressure Boundary	Carbon Steel	Containment Air (Ext)	Loss of Preload	Bolting Integrity Program			324, F
				Loss of Material	Bolting Integrity Program	VII.I.2-a	3.3.1-24	324, C
					Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
			Plant Indoor Air (Ext)	Loss of Preload	Bolting Integrity Program			324, F
				Loss of Material	Bolting Integrity Program	VII.I.2-a	3.3.1-24	324, C
					Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
Filters/Strainers	Fluid Pressure Boundary	Carbon Steel	Containment Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	VII.C2.4-a	3.3.1-15	327, D

Table 3.3.2-13 Auxiliary Systems - Shield Cooling System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Pipe & Fittings	Fluid Pressure Boundary	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	VII.C2.1-a	3.3.1-15	327, B
Valves & Dampers	Fluid Pressure Boundary	Carbon Steel	Containment Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	VII.C2.2-a	3.3.1-15	327, B

Table 3.3.2-14 Auxiliary Systems- Spent Fuel Pool Cooling System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Accumulators	Fluid Pressure Boundary	Stainless Steel	Treated Water (Int)	Loss of Material	One-Time Inspection Program	VII.C2.2-a	3.3.1-15	329, E
					Water Chemistry Program	VII.C2.2-a	3.3.1-15	329, E
Demineralizers	Fluid Pressure Boundary	Stainless Steel	Treated Water (Int)	Loss of Material	One-Time Inspection Program	VII.C2.2-a	3.3.1-15	329, E
					Water Chemistry Program	VII.C2.2-a	3.3.1-15	329, E
Filters/Strainers	Fluid Pressure Boundary	Stainless Steel	Treated Water (Int)	Loss of Material	One-Time Inspection Program	VII.C2.2-a	3.3.1-15	329, E
					Water Chemistry Program	VII.C2.2-a	3.3.1-15	329, E
Fasteners	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Preload	Bolting Integrity Program	VII.I.2-a	3.3.1-24	324, C
				Loss of Material	Bolting Integrity Program	VII.I.2-a	3.3.1-24	324, C
					Boric Acid Corrosion Program	VII.A3.1-a	3.3.1-14	A
						VII.A3.2-c	3.3.1-14	A
						VII.A3.3-c	3.3.1-14	A
						VII.A3.4-b	3.3.1-14	A
						VII.A3.5-b	3.3.1-14	A
						VII.A3.6-a	3.3.1-14	A
		Stainless Steel	Plant Indoor Air (Ext)	Loss of Preload	Bolting Integrity Program			324, F
			Containment Air (Ext)	Loss of Preload	Bolting Integrity Program			324, F

Table 3.3.2-14 Auxiliary Systems- Spent Fuel Pool Cooling System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Pipe & Fittings	Fluid Pressure Boundary	Stainless Steel	Treated Water (Int)	Loss of Material	One-Time Inspection Program	VII.C2.2-a	3.3.1-15	329, E
					Water Chemistry Program	VII.C2.2-a	3.3.1-15	329, E
Pumps	Fluid Pressure Boundary	Stainless Steel	Treated Water (Int)	Loss of Material	One-Time Inspection Program	VII.C2.2-a	3.3.1-15	329, E
					Water Chemistry Program	VII.C2.2-a	3.3.1-15	329, E
SFP Heat Exchanger Shell and Channel Head	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.A3.4-b	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	VII.A3.4-a	3.3.1-15	320, B
SFP Heat Exchanger Shell and Channel Head	Fluid Pressure Boundary	Stainless Steel	Treated Water (Int)	Loss of Material	Closed Cycle Cooling Water Program	VII.C2.2-a	3.3.1-15	318, 322, 329, D
					Water Chemistry Program	VII.C2.2-a	3.3.1-15	329, E
SFP Heat Exchanger Tube and Tubesheet	Fluid Pressure Boundary	Carbon Steel	Treated Water (Ext)	Loss of Material	Closed Cycle Cooling Water Program	VII.A3.4-a	3.3.1-15	328, D

Table 3.3.2-14 Auxiliary Systems- Spent Fuel Pool Cooling System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
SFP Heat Exchanger Tube and Tubesheet	Fluid Pressure Boundary Heat Transfer	Stainless Steel	Treated Water (Ext)	Heat Transfer Degradation	Closed Cycle Cooling Water Program			328, J
				Loss of Material	Closed Cycle Cooling Water Program	VII.C2.2-a	3.3.1-15	323, 328, D
			Treated Water (Int)	Heat Transfer Degradation	Closed Cycle Cooling Water Program			328, J
					Water Chemistry Program			328, J
				Loss of Material	Closed Cycle Cooling Water Program	VII.C2.2-a	3.3.1-15	323, 328, D
					Water Chemistry Program	VII.C2.2-a	3.3.1-15	E
Valves & Dampers	Fluid Pressure Boundary	Stainless Steel	Treated Water (Int)	Loss of Material	One-Time Inspection Program	VII.C2.2-a	3.3.1-15	E
					Water Chemistry Program	VII.C2.2-a	3.3.1-15	E

Table 3.3.2-15 Auxiliary Systems - Waste Gas System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Accumulator	Fluid Pressure Boundary	Carbon Steel	Gas (Int)	Loss of Material	One-Time Inspection Program			332, J
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Raw Water (Int)	Loss of Material	One-Time Inspection Program	VII.C3.1-a	3.3.1-29	332, E
Fasteners	Fluid Pressure Boundary	Low-Alloy Steel	Containment Air (Ext)	Loss of Material	Bolting Integrity Program	VII.I.2-a	3.3.1-24	A
					Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
			Plant Indoor Air (Ext)	Loss of Material	Bolting Integrity Program	VII.I.2-a	3.3.1-24	A
				Loss of Preload	Bolting Integrity Program	VII.I.2-a	3.3.1-24	324, A
Filter/Strainer	Fluid Pressure Boundary	Carbon Steel	Gas (Int)	Loss of Material	One-Time Inspection Program			332, J
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A

Table 3.3.2-15 Auxiliary Systems - Waste Gas System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Pipe & Fittings	Fluid Pressure Boundary	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Gas (Int)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Raw Water (Int)	Loss of Material	One-Time Inspection Program	VII.C3.1-a	3.3.1-29	332, E
Valves & Dampers	Fluid Pressure Boundary	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Gas (Int)	Loss of Material	One-Time Inspection Program			332, G
			Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.I.1-a	3.3.1-14	A
					System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Raw Water (Int)	Loss of Material	One-Time Inspection Program	VII.C1.2-a	3.3.1-17	332, E

Table 3.3.2-16 Auxiliary Systems - Domestic Water System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Accumulator	Fluid Pressure Boundary	Carbon Steel	Air (Int)	Loss of Material	One-Time Inspection Program			J
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Raw Water (Int)	Loss of Material	One-Time Inspection Program	VII.C1.1-a	3.3.1-17	E
		Cast Iron	Air (Int)	Loss of Material	One-Time Inspection Program			J
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Treated Water (Int)	Loss of Material	One-Time Inspection Program			J
		Stainless Steel	Raw Water (Int)	Loss of Material	One-Time Inspection Program	VII.C1.1-a	3.3.1-17	E
Fasteners	Fluid Pressure Boundary	Low-Alloy Steel	Plant Indoor Air (Ext)	Loss of Material	Bolting Integrity Program	VII.I.1-b	3.3.1-05	A
Heat Exchanger	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Raw Water (Int)	Loss of Material	One-Time Inspection Program			J
			Steam (Int)	Loss of Material	One-Time Inspection Program			J

Table 3.3.2-16 Auxiliary Systems - Domestic Water System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Pipe & Fittings	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Raw Water (Int)	Loss of Material	One-Time Inspection Program	VII.C1.1-a	3.3.1-17	E
		Copper Alloy	Raw Water (Int)	Loss of Material	One-Time Inspection Program	VII.C1.1-a	3.3.1-17	E
Pumps	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Raw Water (Int)	Loss of Material	One-Time Inspection Program	VII.C1.5-a	3.3.1-17	E
Valves & Dampers	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Raw Water (Int)	Loss of Material	One-Time Inspection Program	VII.C1.2-a	3.3.1-17	E
		Copper Alloy	Raw Water (Int)	Loss of Material	One-Time Inspection Program	VII.C1.2-a	3.3.1-17	E

Table 3.3.2-17 Auxiliary Systems - Chemical Addition System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Accumulator	Fluid Pressure Boundary	Carbon Steel	Air (Int)	Loss of Material	One-Time Inspection Program			G
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Treated Water (Int)	Loss of Material	One-Time Inspection Program	VII.C2-4a	3.3.1-15	E
		Stainless Steel	Treated Water (Int)	Loss of Material	One-Time Inspection Program	VII.C2.2-a	3.3.1-15	E
Fasteners	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
Pipe & Fittings	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Treated Water (Int)	Loss of Material	One-Time Inspection Program	VII.C2-4A	3.3.1-15	E
		Stainless Steel	Treated Water (Int)	Loss of Material	One-Time Inspection Program	VII.C2.2-a	3.3.1-15	E
Pumps	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Treated Water (Int)	Loss of Material	One-Time Inspection Program	VII.C2-4A	3.3.1-15	E
		Stainless Steel	Treated Water (Int)	Loss of Material	One-Time Inspection Program	VII.C2.2-a	3.3.1-15	E

Table 3.3.2-17 Auxiliary Systems - Chemical Addition System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Valves & Dampers	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VII.I.1-b	3.3.1-05	A
			Treated Water (Int)	Loss of Material	One-Time Inspection Program	VII.C2-4a	3.3.1-15	E
		Stainless Steel	Treated Water (Int)	Loss of Material	One-Time Inspection Program	VII.C2.2-a	3.3.1-15	E

Notes for Tables 3.3.2-1 through 3.3.2-17

A Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP.

B Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP takes some exceptions to NUREG-1801 AMP.

C Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP.

D Component is different, but consistent with NUREG-1801 for material, environment, and aging effect. AMP takes some exceptions to NUREG-1801 AMP.

E Consistent with NUREG-1801 for material, environment, aging effect but a different AMP is credited.

F Material not in NUREG-1801 for this component.

G Environment not in NUREG-1801 for this component and material.

H Aging effect not in NUREG-1801 for this component, material and environment combination.

I Aging effect in NUREG-1801 for this component, material and environment combination is not applicable.

J Neither the component nor the material and environment combination is evaluated in NUREG-1801.

Plant-specific notes:

300 Plant Indoor Air includes Indoor Uncontrolled Air and Condensation

301 The One-Time Inspection Program at Palisades includes the Selective Leaching of Materials Program of NUREG-1801.

302 Erosion has been identified through OE review as a plant-specific aging mechanism.

303 Not used

304 Not used

305 The Diesel Fuel Monitoring and Storage Program includes internal inspection.

306 Biofouling in fuel oil systems only occurs at fuel oil storage tanks, during initial fuel oil delivery. Therefore, biofouling is not a potential aging mechanism for this component.

307 These AERMs are to be inspected during fuel oil tank internal inspections.

308 This component/material group is not included in GALL for this system.

309 Not used

310 The System Monitoring Program at Palisades includes the requirements (for external tank inspections) of the Above ground Carbon Steel Tanks Program of NUREG-1801.

311 Not used

312 Not used

313 For CCW heat exchanger, the tubes are Admiralty Brass and tubesheets are Carbon Steels with Aluminum Bronze overlay for raw water side.

314 Not used

315 For the purpose of this GALL item, the material of CCW HX tubes (copper-nickel and aluminum brass in GALL) is considered to be equivalent to Admiralty Brass or Aluminum Bronze.

316 For the purpose of this GALL item, the aging effect of Buildup of deposit/Biofouling is considered to be equivalent to Heat Transfer Degradation/Fouling.

317 Copper alloys for this component type is not addressed in GALL.

318 Use GALL line item of Valves in VII.C2 for this evaluation.

319 Air space of the tank internal is not addressed in GALL.

320 For carbon steels in treated water, Galvanic Corrosion and/or MIC are conservatively included as aging mechanisms for Loss of Material.

321 Not used

322 For stainless steels or copper alloys, MIC is conservatively included as aging mechanisms for Loss of Material.

323 Fretting is included as a potential external aging effect for stainless steel or copper alloys tubes of heat exchangers.

324 Loss of preload is included here in response to recent NRC RAIs on non-primary system, high temperature bolting that may experience loss of preload. The Palisades Bolting Integrity Program manages potential bolting AERMs and event driven degradation. GALL reconciliation is based on Loss of Material.

325 Use copper alloys of this GALL line of CCW heat exchanger tube for this evaluation.

326 Not used

327 Shield Cooling System (SCS) is considered to be in the category of Closed-Cycle Cooling that is addressed in NUREG-1801(GALL), Chapter VII.C2.

328 The tubes and tubesheets of SFP heat exchangers are not evaluated in NUREG-1801(GALL) VII.A3.

329 The material of this SFP component addressed in NUREG-1801 (GALL) is carbon steel instead of stainless steel.

330 This component does not have temperatures > 140°F. Therefore, cracking due to stress corrosion cracking is not a potential aging mechanism.

331 The Radwaste System is not addressed in NUREG-1801.

332 The Waste Gas System is not addressed in NUREG-1801.

333 Not used

334 This component is buried in soil and exposed to Raw water.

335 Cast Iron is not a material in VII.I.1-b.

336 Component is bolting in Soil. Soil Environment is assumed to be raw water.

337 Material is cast iron. It does not match VII.I.1-b.

338 Component matches GALL. Palisades also identifies Selective Leaching.

339 Palisades matches GALL except Palisades does not identify galvanic corrosion for the Jockey Pump.

340 Palisades matches GALL except Palisades identifies Selective Leaching as an additional AERM for Fire Pumps.

341 GALL does not identify Brass as a material in Section VII.G.6-b.

342 Palisades matches GALL except Palisades does not have galvanic corrosion in carbon steel valves.

343 Palisades matches GALL except Palisades does not have galvanic corrosion in cast iron valves.

344 PAL matches GALL except (see Note 343). In addition, PAL Identifies Selective Leaching of cast iron valves.

345 Not used

346 Brass is not a material in GALL Section VII.I.1-a. But the Palisades Boric Acid Corrosion Program is credited.

347 Palisades matches GALL for general corrosion. PAL also identifies crevice & pitting.

348 Submerged bolting is not identified in GALL Section VII.G.6-b. PAL identifies general, crevice, pitting& MIC.

349 Buried bolting is not identified in GALL Section VII.G.6-b. PAL identifies general, crevice, pitting & MIC.

350 Palisades matches GALL for general, crevice, pitting, MIC. However Palisades does not have galvanic or biofouling.

351 Palisades matches GALL for general, crevice, pitting & MIC. However, Palisades does not identify galvanic or biofouling.

352 GALL does not identify Selective Leaching of cast iron. Palisades identifies Selective Leaching of cast iron.

353 GALL does not identify brass in Section VII.I.1-a. Palisades identifies BAW of brass in borated water.

354 Not used

355 Not used

356 Not used

357 GALL does not identify brass as a material in Section VII.I.1-b.

358 GALL matches Palisades for general, crevice, pitting and MIC. Palisades does not have galvanic or biofouling on buried pipe.

359 Palisades identifies Selective leaching of cast iron. GALL does not identify selective leaching.

360 Not used

361 Not used

362 Not used

363 Not used

364 Not used

365 Not used

366 The carbon steel piping is managed for MIC using the Closed Cycle Cooling Water Program.

367 Piping, pumps, flow elements, valves and restricting orifices are managed for crevice corrosion and pitting corrosion using the ASME Section XI, IWB, IWC, IWD, IWF Inservice Inspection Program; the Water Chemistry Program; and/or, the One-Time Inspection Program.

368 The system is Component Cooling Water and all AERMs are managed using the Closed Cycle Cooling Water Program and/or the One-Time Inspection Program.

369 Boric acid has the potential to get into floor drains; therefore, boric acid wastage is a potential internal aging mechanism that is managed by the One-Time Inspection Program.

370 This component is not connected to material higher in the Galvanic Series.

371 General & Galvanic Corrosion are not applicable to this Stainless Steel Tank.

372 GALL identified crevice, general, galvanic & pitting corrosion. Palisades identified crevice & pitting corrosion. See note 371.

373 General corrosion is not applicable to copper tubing.

374 GALL & Palisades identified crevice, pitting & galvanic corrosion in oil with water environment.

375 This component is a Palisades wet pipe sprinkler head made from brass. GALL identifies bronze not brass.

376 This component is a Palisades wet pipe sprinkler head made from brass. GALL identifies bronze not Brass.

377 PAL identifies corrosion product buildup due to bio-fouling. GALL identifies loss of material due to bio-fouling.

378 Flow in this component is zero; therefore, Loss of Material due to erosion is not applicable.

379 This component is not connected to a material higher in the Galvanic Series.

380 Palisades identifies corrosion product buildup due to bio-fouling. GALL identifies Loss of material due to bio-fouling.

381 For Radwaste, “Treated Water” is not controlled by Water Chemistry, and can be considered as “Warm, Moist Air” for GALL reconciliation.

382 Not used

383 This valve (MV-FP168) is installed on the Jockey Pump Discharge Piping. It is subject to sweating.

384 In addition to general corrosion, valves are subject to sweating. They are on the Jockey Pump discharge piping.

385 GALL identifies general corrosion. Palisades identifies general, crevice, pitting & galvanic for Jockey Pump pipe.

386 Galvanized steels are considered equivalent to carbon steels in terms of AERM evaluations, except for external environment of indoor air where General Corrosion is not an applicable aging mechanism for galvanized steels.

387 Heat Transfer Degradation is not evaluated in GALL VII.F1 through VII.F4 for HVAC heating/cooling coils.

388 The AERM evaluation in the environment of Freon is not included in GALL.

389 The aging effect of Hardening and Loss of Strength/Elastomer Degradation evaluated in GALL is considered to be equivalent to Cracking/Thermal Exposure and Change of Material Properties/Thermal Exposure.

390 Cast Irons are considered equivalent to carbon steels in terms of AERM evaluations, except for Selective Leaching which is only applicable for Cast Irons.

391 This line evaluates AERM of the equipment frames and/or housing, which is addressed in GALL lines of DUCT, VII.F1.1-a, VII.F2.1-a, VII.F3.1-a or VII.F4.1-a.

392 The AERM in the environment of Oil is not addressed in GALL for HVAC components. Use GALL line of Fire Protection in VII.G.7 for this evaluation.

393 The AERM in the environment of Steam is not addressed in GALL Chapter VII components.

394 Erosion is conservatively included as an applicable aging effect for heat exchanger tubes in service water and steam environments.

395 The AERM in the environment of HVAC condensation as raw water is not addressed in GALL Chapter VII components.

396 This is a raw water heat exchanger. Fouling is an AERM needing management. Standard note H applies to this roll up line item.

397 Not used

398 Not used

399 At Palisades, cast iron aging mechanisms are evaluated as carbon steel except for selective leaching.

3.4 Aging Management of Steam and Power Conversion System

3.4.1 Introduction

This section provides the results of the aging management review for those components identified in Section 2.3.4, Steam and Power Conversion Systems, as being subject to aging management review. The systems, or portions of systems, which are addressed in this section, are described in the indicated sections.

- Condensate and Condenser System (Section 2.3.4.1)
- Demineralized Makeup Water System (Section 2.3.4.2)
- Feedwater System (Section 2.3.4.3)
- Heater Extraction and Drain System (Section 2.3.4.4)
- Main Air Ejection and Gland Seal System (Section 2.3.4.5)
- Main Steam System (Section 2.3.4.6)
- Turbine Generator and Crane System (Section 2.3.4.7)

Table 3.4.1, Summary of Aging Management Evaluations in Chapter VII of NUREG-1801 for Steam and Power Conversion Systems, provides the summary of the programs evaluated in NUREG-1801 for the Steam and Power Conversion Systems component groups that are relied on for license renewal.

This table uses the format described in Figure 3.0-1 above. Note that this table only includes those component groups that are applicable to a PWR.

3.4.2 Results

The following tables summarize the results of the aging management review for systems in the Steam and Power Conversion Systems group:

Table 3.4.2-1, Steam and Power Conversion Systems - Condensate and Condenser System - Summary of Aging Management Evaluation

Table 3.4.2-2, Steam and Power Conversion Systems - Demineralized Makeup Water System - Summary of Aging Management Evaluation

Table 3.4.2-3, Steam and Power Conversion Systems - Feedwater System - Summary of Aging Management Evaluation

Table 3.4.2-4, Steam and Power Conversion Systems - Heater Extraction and Drain System - Summary of Aging Management Evaluation

Table 3.4.2-5, Steam and Power Conversion Systems - Main Air Ejection and Gland Seal System - Summary of Aging Management Evaluation

Table 3.4.2-6, Steam and Power Conversion Systems - Main Steam System - Summary of Aging Management Evaluation

Table 3.4.2-7, Steam and Power Conversion Systems - Turbine Generator and Crane System - Summary of Aging Management Evaluation

The materials that specific components are fabricated from, the environments to which components are exposed, the aging effects requiring management, and the aging management programs used to manage these aging effects are provided for each of the above systems in the following subsections of Section 3.4.2.1, Materials, Environment, Aging Effects Requiring Management and Aging Management Programs:

Section 3.4.2.1.1, Condensate and Condenser System

Section 3.4.2.1.2, Demineralized Makeup Water System

Section 3.4.2.1.3, Feedwater System

Section 3.4.2.1.4, Heater Extraction and Drain System

Section 3.4.2.1.5, Main Air Ejection and Gland Seal System

Section 3.4.2.1.6, Main Steam System

Section 3.4.2.1.7, Turbine Generator and Crane System

3.4.2.1 Materials, Environment, Aging Effects Requiring Management and Aging Management Programs

3.4.2.1.1 Condensate and Condenser System

Materials

The materials of construction for the Condensate and Condenser System components are:

- Carbon Steel
- Cast Iron
- Stainless Steel

Environment

The Condensate and Condenser System components are exposed to the following environments:

- Atmosphere/Weather (Ext)
- Plant Indoor Air (Ext)
- Soil (Ext)

- Steam (Int)
- Steam (Ext)
- Treated Water (Int)

Aging Effects Requiring Management

The following aging effects, associated with the Condensate and Condenser System, require management:

- Loss of Material
- Loss of Preload

Aging Management Programs

The following aging management programs manage the aging effects for the Condensate and Condenser System components:

- Bolting Integrity Program
- Buried Services Corrosion Monitoring Program
- Flow Accelerated Corrosion Program
- One-Time Inspection Program
- System Monitoring Program
- Water Chemistry Program

3.4.2.1.2 **Demineralized Makeup Water System**

Materials

The materials of construction for the Demineralized Makeup Water System components are:

- Carbon Steel
- Cast Austenitic SS
- Stainless Steel

Environment

The Demineralized Makeup Water System components are exposed to the following environments:

- Air (Int)
- Atmosphere/Weather (Ext)
- Plant Indoor Air (Ext)

- Soil (Ext)
- Steam (Int)
- Treated Water (Int)

Aging Effects Requiring Management

The following aging effects, associated with the Demineralized Makeup Water System, require management:

- Loss of Material
- Loss of Preload

Aging Management Programs

The following aging management programs manage the aging effects for the Demineralized Makeup Water System components:

- Bolting Integrity Program
- Buried Services Corrosion Monitoring Program
- One-Time Inspection Program
- System Monitoring Program
- Water Chemistry Program

3.4.2.1.3 Feedwater System

Materials

The materials of construction for the Feedwater System components are:

- Carbon Steel
- Cast Austenitic SS
- Cast Iron
- Stainless Steel

Environment

The Feedwater System components are exposed to the following environments:

- Atmosphere/Weather (Ext)
- Air (Int)
- Containment Air (Ext)
- Oil (Int)

- Plant Indoor Air (Ext)
- Raw Water (Int)
- Soil (Ext)
- Steam (Int)
- Treated Water (Int)

Aging Effects Requiring Management

The following aging effects, associated with the Feedwater System, require management:

- Cracking
- Loss of Material
- Loss of Preload

Aging Management Programs

The following aging management programs manage the aging effects for the Feedwater System components:

- Bolting Integrity Program
- Boric Acid Corrosion Program
- Buried Services Corrosion Monitoring Program
- Flow Accelerated Corrosion Program
- One-Time Inspection Program
- System Monitoring Program
- Water Chemistry Program

3.4.2.1.4 **Heater Extraction and Drain System**

Materials

The materials of construction for the Heater Extraction and Drain System components are:

- Carbon Steel
- Cast Iron
- Stainless Steel

Environment

The Heater Extraction and Drain System components are exposed to the following environments:

- Plant Indoor Air (Ext)
- Steam (Int)
- Treated Water (Int)

Aging Effects Requiring Management

The following aging effects, associated with the Heater Extraction and Drain System, require management:

- Cracking
- Loss of Material
- Loss of Preload

Aging Management Programs

The following aging management programs manage the aging effects for the Heater Extraction and Drain System components:

- Bolting Integrity Program
- Flow Accelerated Corrosion Program
- One-Time Inspection Program
- System Monitoring Program
- Water Chemistry Program

3.4.2.1.5 **Main Air Ejection and Gland Seal System**

Materials

The materials of construction for the Main Air Ejection and Gland Seal System components are:

- Carbon Steel
- Cast Iron

Environment

The Main Air Ejection and Gland Seal System components are exposed to the following environments:

- Air (Int)

- Plant Indoor Air (Ext)
- Raw Water (Int)
- Steam (Int)
- Treated Water (Int)

Aging Effects Requiring Management

The following aging effects, associated with the Main Air Ejection and Gland Seal System, require management:

- Loss of Material
- Loss of Preload

Aging Management Programs

The following aging management programs manage the aging effects for the Main Air Ejection and Gland Seal System components:

- Bolting Integrity Program
- Flow Accelerated Corrosion Program
- One-Time Inspection Program
- System Monitoring Program
- Water Chemistry Program

3.4.2.1.6 Main Steam System

Materials

The materials of construction for the Main Steam System components are:

- Carbon Steel
- Copper Alloys
- Stainless Steel

Environment

The Main Steam System components are exposed to the following environments:

- Containment Air (Ext)
- Plant Indoor Air (Ext)
- Steam (Int)
- Treated Water (Int)

Aging Effects Requiring Management

The following aging effects, associated with the Main Steam System, require management:

- Cracking
- Loss of Material
- Loss of Preload

Aging Management Programs

The following aging management programs manage the aging effects for the Main Steam System components:

- Bolting Integrity Program
- Boric Acid Corrosion Program
- Flow Accelerated Corrosion Program
- One-Time Inspection Program
- System Monitoring Program
- Water Chemistry Program

3.4.2.1.7 Turbine Generator System

Materials

The materials of construction for the Turbine Generator and Crane System components are:

- Carbon Steel
- Cast Iron
- Stainless Steel

Environment

The Turbine Generator and Crane System components are exposed to the following environments:

- Air/Gas (Int)
- Oil (Int)
- Plant Indoor Air (Ext)
- Raw Water (Int)

- Steam (Int)
- Treated Water (Int)

Aging Effects Requiring Management

The following aging effects, associated with the Turbine Generator and Crane System, require management:

- Loss of Material
- Loss of Preload

Aging Management Programs

The following aging management programs manage the aging effects for the Turbine Generator and Crane System components:

- One-Time Inspection Program
- System Monitoring Program
- Bolting Integrity Program

3.4.2.2 Further Evaluation of Aging Management as Recommended by NUREG-1801

NUREG-1801 provides the basis for identifying those programs that warrant further evaluation by the reviewer in the license renewal application. For the Steam and Power Conversion Systems, those programs are addressed in the following sections.

3.4.2.2.1 Cumulative Fatigue Damage

Fatigue is a TLAA as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c). The evaluation of this TLAA is addressed in Section 4.3.

3.4.2.2.2 Loss of Material due to General, Pitting, and Crevice Corrosion

NUREG-1800 states that the management of loss of material due to general, pitting, and crevice corrosion should be evaluated further for carbon steel piping and fittings, valve bodies and bonnets, pump casings, pump suction and discharge lines, tanks, tubesheets, channel heads, and shells except for main steam system components and for loss of material due to pitting and crevice corrosion for stainless steel tanks and heat exchanger/cooler tubes. The water chemistry program relies on monitoring and control of water chemistry based on the EPRI guidelines of TR-102134 for secondary water chemistry in PWRs to manage the effects of loss of material due to general, pitting, or crevice corrosion. However, corrosion may occur at locations of stagnant flow conditions. Therefore, the effectiveness of the chemistry control program should be verified to ensure that corrosion is not occurring. NUREG-1801 recommends further evaluation of programs to manage loss of material due to general, pitting, and crevice corrosion to verify the effectiveness of the water chemistry program. A one-time inspection of select components and susceptible locations is an acceptable method to ensure that corrosion is not occurring and that the component's intended function will be maintained during the period of extended operation.

At Palisades, under the One-Time Inspection Program, a representative sample of the component population will be chosen for inspection. The focus, when practical, will be placed on bounding or lead components. Factors that will be considered when choosing components for inspection are time in service, severity of operating conditions, and operating experience. The examination techniques will be visual, volumetric, or other appropriately established NDE methods that are capable of management of the aging effect loss of material due to galvanic and general corrosion, MIC, pitting and crevice corrosion, and selective leaching.

3.4.2.2.3 Loss of Material due to General, Pitting, and Crevice Corrosion, Microbiologically Influenced Corrosion, and Biofouling

NUREG-1800 states that loss of material due to general corrosion, pitting and crevice corrosion, microbiologically influenced corrosion (MIC), and biofouling could occur in carbon steel piping and fittings for untreated water from the backup water supply in the PWR auxiliary feedwater system. NUREG-1801 recommends further evaluation to ensure that these aging effects are adequately managed.

At Palisades, the portion of the lines from the Service Water System (SWS) to the auxiliary feedwater system is addressed as part of the SWS system (item 3.3.1-17 of Table 3.3.1 of NUREG-1800). The aging effect of loss of material is managed by Open Cycle Cooling Water Program. The program is consistent with the NUREG-1801, Section XI.M20, “Open-Cycle Cooling Water System”, and aging effects will be managed such that SSCs within the scope of this program will continue to perform their intended functions.

The portion of the lines from the fire protection water lines to the auxiliary feedwater system is addressed as part of the Fire Protection system (item

3.3.1-21 of Table 3.3.1 of NUREG-1800). The aging effect of loss of material is managed by the Fire Protection Program . The program is consistent with, but includes exceptions to, NUREG-1801, Section XI.M27, “Fire Water System.”

3.4.2.2.4 General Corrosion

NUREG-1800 states that loss of material due to general corrosion could occur on the external surfaces of all carbon steel structures and components, including closure boltings, exposed to operating temperature less than 212°F. NUREG-1801 recommends further evaluation to ensure that this aging effect is adequately managed.

At Palisades, the external aging effect of general corrosion is managed by the System Monitoring Program . The external surfaces of various component types (e.g., pump casings, valve bodies, piping, etc.) are visually inspected for leakage and evidence of material degradation, such as loss of material due to corrosion. The external aging effect of general corrosion of closure boltings is managed by the Bolting Integrity Program.

The attributes of System Monitoring Program are consistent with the criteria described in the Branch Technical Position RLSB-1 which is included in Appendix A of NUREG-1800, and this aging effect will be managed such that the subject SSCs will continue to perform their intended functions.

The Bolting Integrity Program is consistent with the NRC GALL Report, Section XI.M18, “Bolting Integrity.”

3.4.2.2.5 Loss of Material due to General, Pitting, Crevice, and Microbiologically Influenced Corrosion

3.4.2.2.5.1 Loss of Material due to General, Pitting, Crevice, and Microbiologically Influenced Corrosion in AFW Pump Bearing Oil Cooler

NUREG-1800 states that loss of material due to general corrosion (carbon steel only), pitting and crevice corrosion, and MIC could occur in stainless steel and carbon steel shells, tubes, and tubesheets within the bearing oil coolers (for steam turbine pumps) in the PWR auxiliary feedwater system. NUREG-1801 recommends further evaluation to ensure that these aging effects are adequately managed.

This paragraph of NUREG -1800 is not applicable to Palisades. The bearing oil coolers for the AFW pump turbine, addressed in NUREG-1801 VIII.G.5-a through VIII.G.5-d, are not applicable for Palisades.

3.4.2.2.5.2 Loss of Material due to General, Pitting, Crevice, and Microbiologically Influenced Corrosion in Underground Condensate Piping, Tanks, and Fittings

NUREG-1800 states that loss of material due to general corrosion, pitting and crevice corrosion, and MIC could occur in underground piping and fittings and emergency condensate storage tank in the auxiliary feedwater system and the underground condensate storage tank in the condensate system. The buried piping and tanks inspection program relies on industry practice, frequency of pipe excavation, and operating experience to manage the effects of loss of material from general corrosion, pitting and crevice corrosion, and MIC. The effectiveness of the buried piping and tanks inspection program should be verified to evaluate an applicant's inspection frequency and operating experience with buried components, ensuring that loss of material is not occurring.

Palisades does not have buried storage tanks. Loss of material in buried carbon steel and stainless steel piping is managed by the Buried Services Corrosion Monitoring Program . The program is consistent with NUREG-1801, Section XI.M34, “Buried Piping and Tanks Inspection,” and the Buried Services Corrosion Program will be effective in maintaining the intended functions of these underground piping systems.

3.4.2.2.6 Quality Assurance for Aging Management of Non-Safety Related Components

Quality Assurance Program applicability to non-safety-related components is addressed in Appendix B, Section 1.2.

3.4.2.3 Time-Limited Aging Analysis

The time-limited aging analyses (TLAA) identified below are associated with the Steam and Power Conversion Systems components. The section of the LRA that contains the TLAA review results is indicated in parenthesis.

- Fatigue (Section 4.3, Metal Fatigue)

3.4.3 Conclusion

The Steam and Power Conversion Systems piping, fittings, and components that are subject to aging management review have been identified in accordance with the requirements of 10 CFR 54.4. The aging management programs selected to manage aging effects for the Steam and Power Conversion Systems components are identified in the summaries in Section 3.4.2.1 above.

A description of these aging management programs is provided in Appendix B, along with the demonstration that the identified aging effects will be managed for the period of extended operation.

Therefore, based on the demonstrations provided in Appendix B, the effects of aging associated with the Steam and Power Conversion Systems components will be adequately managed so that there is reasonable assurance that the intended function(s) will be maintained consistent with the current licensing basis during the period of extended operation.

Table 3.4.1 Summary of Aging Management Evaluations in Chapter VIII of NUREG-1801 for Steam and Power Conversion System

Item Number	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.4.1-01	Piping and fittings in main feedwater line, steam line and AFW piping (PWR only)	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA (see [SRP] subsection 3.4.2.2.1)	Further evaluation documented in Section 3.4.2.2.1. The aging mechanism fatigue is potentially applicable to many component types in the Steam and Power Conversion Supergroup, but only selected components or locations require explicit analysis as TLAAs and/or warrant aging management. The Palisades approach to identifying and managing the relevant locations and components for fatigue damage is addressed in Section 4.3 and in Appendix B, Fatigue Monitoring Program. Therefore, cumulative fatigue damage is not identified as an aging effect in Tables 3.4.2-1 through 3.4.2-7 below.
3.4.1-02	Piping and fittings, valve bodies and bonnets, pump casings, tanks, tubes, tubesheets, channel head and shell (except main steam system)	Loss of material due to general (carbon steel only), pitting, and crevice corrosion	Water chemistry and one-time inspection	Yes, detection of aging effects is to be further evaluated (see [SRP] subsection 3.4.2.2.2)	Further evaluation documented in Section 3.4.2.2.2.
3.4.1-03	Auxiliary feedwater (AFW) piping	Loss of material due to general, pitting, and crevice corrosion, MIC, and biofouling	Plant specific	Yes, plant specific (see [SRP] subsection 3.4.2.2.3)	Further evaluation documented in Section 3.4.2.2.3.
3.4.1-04	Oil coolers in AFW system (lubricating oil side possibly contaminated with water)	Loss of material due to general (carbon steel only), pitting, and crevice corrosion and MIC	Plant specific	Yes, plant specific (see [SRP] subsection 3.4.2.2.5.1)	Further evaluation documented in Section 3.4.2.2.5.

Table 3.4.1 Summary of Aging Management Evaluations in Chapter VIII of NUREG-1801 for Steam and Power Conversion System

Item Number	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.4.1-05	External surface of carbon steel components	Loss of material due to general corrosion	Plant specific	Yes, plant specific (see [SRP] subsection 3.4.2.2.4)	Further evaluation documented in Section 3.4.2.2.4.
3.4.1-06	Carbon steel piping and valve bodies	Wall thinning due to flow-accelerated corrosion	Flow-accelerated corrosion	No	See [Section B2.1.11] Flow Accelerated Corrosion Program for Aging Management Program for Main Steam and Main Feedwater systems.
3.4.1-07	Carbon steel piping and valve bodies in main steam system	Loss of material due to pitting and crevice corrosion	Water chemistry	No	See [Section B2.1.13] One Time Inspection Program and [Section B2.1.21] Water Chemistry Program for aging management programs
3.4.1-08	Closure bolting in high-pressure or high-temperature systems	Loss of material due to general corrosion; crack initiation and growth due to cyclic loading and/or SCC	Bolting integrity	No	See [Section B2.1.3] Bolting Integrity Program and [Section B2.1.20] System Monitoring Program for aging management programs
3.4.1-09	Heat exchangers and coolers/ condensers serviced by open-cycle cooling water	Loss of material due to general (carbon steel only), pitting, and crevice corrosion, MIC, and biofouling; buildup of deposit due to biofouling	Open-cycle cooling water system	No	Heat exchangers were divided into subcomponents as indicated in [Section 2.1.2.2]
3.4.1-10	Heat exchangers and coolers/ condensers serviced by closed-cycle cooling water	Loss of material due to general (carbon steel only), pitting, and crevice corrosion	Closed-cycle cooling water system	No	Heat exchangers were divided into subcomponents as indicated in [Section 2.1.2.2]

Table 3.4.1 Summary of Aging Management Evaluations in Chapter VIII of NUREG-1801 for Steam and Power Conversion System

Item Number	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.4.1-11	External surface of above ground condensate storage tank	Loss of material due to general (carbon steel only), pitting, and crevice corrosion	Above ground carbon steel tanks	No	See [Section B2.1.13] One Time Inspection Program and [Section B2.1.20] System Monitoring Program for aging management programs
3.4.1-12	External surface of buried condensate storage tank and AFW piping	Loss of material due to general, pitting, and crevice corrosion and MIC	Buried piping and tanks surveillance or	No	Not credited for Aging Management
			Buried piping and tanks inspection	Yes, detection of aging effects and operating experience are to be further evaluated (see [SRP] subsection 3.4.2.2.5.2)	Further evaluation documented in Section 3.4.2.2.5.2.
3.4.1-13	External surface of carbon steel components	Loss of material due to boric acid corrosion	Boric acid corrosion	No	See [Section B2.1.4] Boric Acid Corrosion Program for Aging Management Program

Table 3.4.2-1 Steam and Power Conversion System - Condensate and Condenser System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Accumulators	Fluid Pressure Boundary	Carbon Steel	Atmosphere/ Weather (Ext)	Loss of Material	One-Time Inspection Program	VIII.E.5-c	3.4.1-11	418, 420, B
					System Monitoring Program	VIII.E.5-c	3.4.1-11	418, 420, A
			Treated Water (Int)	Loss of Material	Water Chemistry Program	VIII.E.5-a	3.4.1-02	411, A
					One-Time Inspection Program	VIII.E.5-a	3.4.1-02	411, B
CST Heater Shell	Fluid Pressure Boundary	Carbon Steel	Steam (Int)	Loss of Material	Water Chemistry Program	VIII.A.2-b	3.4.1-02	413, C
					One-Time Inspection Program	VIII.A.2-b	3.4.1-02	413, D
CST Heater Tubes	Fluid Pressure Boundary	Carbon Steel	Steam (Ext)	Loss of Material	Water Chemistry Program	VIII.A.2-b	3.4.1-02	413, C
					One-Time Inspection Program	VIII.A.2-b	3.4.1-02	413, D
				Loss of Material - Fretting	One-Time Inspection Program	VIII.A.2-b	3.4.1-02	413, D
			Treated Water (Int)	Loss of Material	One-Time Inspection Program	VIII.E.2-b	3.4.1-02	411, 413, D
					Water Chemistry Program	VIII.E.2-b	3.4.1-02	411, 413, C
Fasteners	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Preload	Bolting Integrity Program	VIII.H.2-a	3.4.1-08	406, C
				Loss of Material	Bolting Integrity Program	VIII.H.2-a	3.4.1-08	406, C

Table 3.4.2-1 Steam and Power Conversion System - Condensate and Condenser System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Filters/Strainers	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	A
			Treated Water (Int)	Loss of Material	One-Time Inspection Program	VIII.E.2-b	3.4.1-02	411, D
					Water Chemistry Program	VIII.E.2-b	3.4.1-02	411, C
FW Heater Shell and Channel Head	Fluid Pressure Boundary	Carbon Steel	Treated Water (Int)	Loss of Material	One-Time Inspection Program	VIII.E.2-b	3.4.1-02	411, 413, D
					Water Chemistry Program	VIII.E.2-b	3.4.1-02	411, 413, C
Heat Exchangers	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	A
			Steam (Int)	Loss of Material	One-Time Inspection Program	VIII.A.1-b	3.4.1-02	D
			Treated Water (Int)	Loss of Material	One-Time Inspection Program	VIII.E.4-a	3.4.1-02	A
Pipe & Fittings	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H1-b	3.4.1-05	A
			Soil (Ext)	Loss of Material	Buried Services Corrosion Monitoring Program	VIII.G1-e	3.4.1-12	408, C

Table 3.4.2-1 Steam and Power Conversion System - Condensate and Condenser System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Pipe & Fittings	Fluid Pressure Boundary	Carbon Steel	Treated Water (Int)	Loss of Material	Flow Accelerated Corrosion Program	VIII.E.1-a	3.4.1-06	423, A
					One-Time Inspection Program	VIII.E.1-b	3.4.1-02	411, B
					Water Chemistry Program	VIII.E.1-b	3.4.1-02	411, A
		Stainless Steel	Soil (Ext)	Loss of Material	Buried Services Corrosion Monitoring Program			407, F
			Treated Water (Int)	Loss of Material	One-Time Inspection Program	VIII.E.5-b	3.4.1-02	405, 419, D
					Water Chemistry Program	VIII.E.5-b	3.4.1-02	405, 419, C
Pumps	Fluid Pressure Boundary	Cast Iron	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	422, A
			Treated Water (Int)	Loss of Material	One-Time Inspection Program	VIII.E.3-a	3.4.1-02	411, 422, B
					Water Chemistry Program	VIII.E.3-a	3.4.1-02	411, 422, A
				Loss of Material - Selective Leaching	One-Time Inspection Program			412, 422, F

Table 3.4.2-1 Steam and Power Conversion System - Condensate and Condenser System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Valves & Dampers	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	A
			Treated Water (Int)	Loss of Material	Water Chemistry Program	VIII.E.2-b	3.4.1-02	411, A
					One-Time Inspection Program	VIII.E.2-b	3.4.1-02	411, B
					Flow Accelerated Corrosion Program	VIII.E.2-a	3.4.1-06	423, A

Table 3.4.2-2 Steam and Power Conversion System - Demineralized Makeup Water System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Accumulators	Fluid Pressure Boundary	Carbon Steel	Air (Int)	Loss of Material	One-Time Inspection Program	VII.H2.3-a	3.3.1-05	416, E
			Atmosphere/ Weather (Ext)	Loss of Material	One-Time Inspection Program	VIII.G.4-c	3.4.1-11	417, 418, 420, D
					System Monitoring Program	VIII.G.4-c	3.4.1-11	417, 418, 420, C
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	A
			Treated Water (Int)	Loss of Material	Water Chemistry Program	VIII.G.4-a	3.4.1-02	411, 417, C
					One-Time Inspection Program	VIII.G.4-a	3.4.1-02	411, 417, D
Fasteners	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Preload	Bolting Integrity Program	VIII.H.2-a	3.4.1-08	406, C
				Loss of Material	Bolting Integrity Program	VIII.H.2-a	3.4.1-08	406, C
Filters	Fluid Pressure Boundary	Stainless Steel	Treated Water (Int)	Loss of Material	One-Time Inspection Program	VIII.E.5-b	3.4.1-02	417, E

Table 3.4.2-2 Steam and Power Conversion System - Demineralized Makeup Water System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Heat Exchangers	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	A
			Steam (Int)	Loss of Material	One-Time Inspection Program	VIII.C.1-a	3.4.1-02	E
			Treated Water (Int)	Loss of Material	One-Time Inspection Program	VIII.E.1-b	3.4.1-02	E
Pipe & Fittings	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	A
			Treated Water (Int)	Loss of Material	One-Time Inspection Program	VIII.E.1-b	3.4.1-02	411, 417, D
					Water Chemistry Program	VIII.E.1-b	3.4.1-02	411, 417, C
		Stainless Steel	Soil (Ext)	Loss of Material	Buried Services Corrosion Monitoring Program			407, F
			Treated Water (Int)	Loss of Material	One-Time Inspection Program	VIII.E.5-b	3.4.1-02	417, 419, D
					Water Chemistry Program	VIII.E.5-b	3.4.1-02	417, 419, C
Pumps	Fluid Pressure Boundary	Stainless Steel	Treated Water (Int)	Loss of Material	One-Time Inspection Program	VIII.E.5-b	3.4.1-02	417, E

Table 3.4.2-2 Steam and Power Conversion System - Demineralized Makeup Water System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Transmitter/ Element	Fluid Pressure Boundary	Stainless Steel	Treated Water (Int)	Loss of Material	One-Time Inspection Program	VIII.E.5-b	3.4.1-02	417, 419, D
					Water Chemistry Program	VIII.E.5-b	3.4.1-02	417, 419, C
Valves & Dampers	Fluid Pressure Boundary	Cast Austenitic SS	Treated Water (Int)	Loss of Material	One-Time Inspection Program	VIII.E.5-b	3.4.1-02	417, 419, D
					Water Chemistry Program	VIII.E.5-b	3.4.1-02	417, 419, C
		Stainless Steel	Treated Water (Int)	Loss of Material	One-Time Inspection Program	VIII.E.5-b	3.4.1-02	417, 419, D
					Water Chemistry Program	VIII.E.5-b	3.4.1-02	417, 419, C

Table 3.4.2-3 Steam and Power Conversion System - Feedwater System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Accumulators	Fluid Pressure Boundary	Carbon Steel	Air/Gas (Int)	Loss of Material	One-Time Inspection Program	VII.H2.3-a	3.3.1-05	409, E
			Oil (Int)	Loss of Material	One-Time Inspection Program	VIII.G.5-d	3.3.1-06	C
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	A
Filters/Strainers	Fluid Pressure Boundary	Carbon Steel	Oil (Int)	Loss of Material	One-Time Inspection Program	VIII.G.5-d	3.3.1-06	C
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	A
			Treated Water (Int)	Loss of Material	One-Time Inspection Program	VIII.D1.1-c	3.4.1-02	408, 411, D
					Water Chemistry Program	VIII.D1.1-c	3.4.1-02	408, 411, C
Heat Exchangers	Fluid Pressure Boundary	Carbon Steel	Oil (Int)	Loss of Material	One-Time Inspection Program	VII.G.7-b	3.3.1-06	C
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	A
			Raw Water (Int)	Loss of Material	One-Time Inspection Program	VIII.G.5-b	3.4.1-03	411, 421, E
			Treated Water (Int)	Loss of Material	One-Time Inspection Program	VIII.D1.1-c	3.4.1-02	408, 411, D
					Water Chemistry Program	VIII.D1.1-c	3.4.1-02	408, 411, C

Table 3.4.2-3 Steam and Power Conversion System - Feedwater System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Indicators/ Recorders	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	A
			Treated Water (Int)	Loss of Material	One-Time Inspection Program	VIII.D1.1-c	3.4.1-02	408, 411, D
					Water Chemistry Program	VIII.D1.1-c	3.4.1-02	408, 411, C
Fasteners	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext Containment Air (Ext))	Loss of Preload	Bolting Integrity Program	VIII.H.2-a	3.4.1-08	406, C
				Loss of Material	Boric Acid Corrosion Program	VIII.H.1-a	3.4.1-13	A
					Bolting Integrity Program	VIII.H.2-a	3.4.1-08	406, C
Pipe & Fittings	Flow Restriction	Carbon Steel	Treated Water (Int)	Loss of Material	One-Time Inspection Program	VIII.D1.1-c	3.4.1-02	411, B
						VIII.D1.3-a	3.4.1-02	411, B
						VIII.G.1-c	3.4.1-02	411, B
						VIII.G.2-a	3.4.1-02	411, B
					Water Chemistry Program	VIII.D1.1-c	3.4.1-02	411, A
						VIII.D1.3-a	3.4.1-02	411, A
						VIII.G.1-c	3.4.1-02	411, A
						VIII.G.2-a	3.4.1-02	411, A

Table 3.4.2-3 Steam and Power Conversion System - Feedwater System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Pipe & Fittings	Fluid Pressure Boundary	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VIII.H.1-a	3.4.1-13	A
					System Monitoring Program	VIII.H.1-b	3.4.1-05	A
			Oil (Int)	Loss of Material	One-Time Inspection Program	VIII.G.5-d		C
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	A
					Boric Acid Corrosion Program	VIII.H.1-a	3.4.1-13	A
			Treated Water (Int)	Loss of Material	Flow Accelerated Corrosion Program	VIII.D1.1-a	3.4.1-06	A
						VIII.D1.3-b	3.4.1-06	A
						VIII.G.1-a	3.4.1-06	A
			Treated Water (Int)	Loss of Material	One-Time Inspection Program	VIII.D1.1-c	3.4.1-02	411, B
						VIII.D1.3-a	3.4.1-02	411, B
						VIII.G.1-c	3.4.1-02	411, B
						VIII.G.2-a	3.4.1-02	411, B
					Water Chemistry Program	VIII.D1.1-c	3.4.1-02	411, A
						VIII.D1.3-a	3.4.1-02	411, A
						VIII.G.1-c	3.4.1-02	411, A
						VIII.G.2-a	3.4.1-02	411, A

Table 3.4.2-3 Steam and Power Conversion System - Feedwater System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Pipe & Fittings	Fluid Pressure Boundary	Stainless Steel	Air (Int)	Loss of Material	One-Time Inspection Program			409, J
			Atmosphere/ Weather (Ext)	Loss of Material	System Monitoring Program			G
			Oil (Int)	Loss of Material	One-Time Inspection Program			F
			Soil (Ext)	Loss of Material	Buried Services Corrosion Monitoring Program			407, F
			Treated Water (Int)	Cracking	Water Chemistry Program			410, F
					One-Time Inspection Program			410, F
				Loss of Material	One-Time Inspection Program	VIII.E.5-b	3.4.1-02	405, 419, D
					Water Chemistry Program	VIII.E.5-b	3.4.1-02	405, 419, C

Table 3.4.2-3 Steam and Power Conversion System - Feedwater System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Pumps	Fluid Pressure Boundary	Carbon Steel	Oil (Int)	Loss of Material	One-Time Inspection Program	VIII.G.5-d	3.3.1-06	C
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	A
			Treated Water (Int)	Loss of Material	Flow Accelerated Corrosion Program	VIII.D1.3-b	3.4.1-06	A
					One-Time Inspection Program	VIII.D1.3-a	3.4.1-02	411, B
						VIII.G.2-a	3.4.1-02	411, B
					Water Chemistry Program	VIII.D1.3-a	3.4.1-02	411, A
						VIII.G.2-a	3.4.1-02	411, A
		Stainless Steel	Treated Water (Int)	Loss of Material	One-Time Inspection Program	VIII.E.5-b	3.4.1-02	405, 419, D
					Water Chemistry Program	VIII.E.5-b	3.4.1-02	405, 419, C
Traps (Steam)	Fluid Pressure Boundary	Carbon Steel	Air (Int)	Loss of Material	One-Time Inspection Program			409, J
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	A
Turbines	Fluid Pressure Boundary	Carbon Steel	Air (Int)	Loss of Material	One-Time Inspection Program			409, J
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	A

Table 3.4.2-3 Steam and Power Conversion System - Feedwater System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Valves & Dampers	Fluid Pressure Boundary	Carbon Steel	Air (Int)	Loss of Material	One-Time Inspection Program			409, G
			Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VIII.H.1-a	3.4.1-13	A
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	A
					Boric Acid Corrosion Program	VIII.H.1-a	3.4.1-13	A
			Raw Water (Int)	Loss of Material	One-Time Inspection Program	VIII.G.1-d	3.4.1-03	411, 421, D
			Steam (Int)	Loss of Material	Water Chemistry Program	VIII.A.2-b	3.4.1-02	A
					One-Time Inspection Program	VIII.A.2-b	3.4.1-02	411, B
			Treated Water (Int)	Loss of Material	Flow Accelerated Corrosion Program	VIII.D1.2-a	3.4.1-06	A
					One-Time Inspection Program	VIII.D1.2-b	3.4.1-02	411, B
						VIII.G.3-a	3.4.1-02	411, B
					Water Chemistry Program	VIII.D1.2-b	3.4.1-02	411, A
						VIII.G.3-a	3.4.1-02	411, A

Table 3.4.2-3 Steam and Power Conversion System - Feedwater System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Valves & Dampers	Fluid Pressure Boundary	Cast Austenitic SS	Treated Water (Int)	Loss of Material	One-Time Inspection Program	VIII.E.5-b	3.4.1-02	405, 419, D
					Water Chemistry Program	VIII.E.5-b	3.4.1-02	405, 419, C
		Cast Iron	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	422, A
			Treated Water (Int)	Loss of Material	One-Time Inspection Program	VIII.G.3-a	3.4.1-02	411, 422, B
					Water Chemistry Program	VIII.G.3-a	3.4.1-02	411, 422, A
				Loss of Material - Selective Leaching	One-Time Inspection Program			412, 422, F
		Stainless Steel	Air (Int)	Loss of Material	One-Time Inspection Program			409, F
			Steam (Int)	Cracking	Water Chemistry Program			404, F
					One-Time Inspection Program			404, F
				Loss of Material	Water Chemistry Program			404, F
					One-Time Inspection Program			404, F

Table 3.4.2-3 Steam and Power Conversion System - Feedwater System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Valves & Dampers	Fluid Pressure Boundary	Stainless Steel	Treated Water (Int)	Cracking	Water Chemistry Program			F
					One-Time Inspection Program			F
				Loss of Material	One-Time Inspection Program	VIII.E.5-b	3.4.1-02	405, 419, D
					Water Chemistry Program	VIII.E.5-b	3.4.1-02	405, 419, C

Table 3.4.2-4 Steam and Power Conversion System - Heater Extraction and Drain System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Accumulators	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-08	A
			Treated Water (Int)	Loss of Material	One-Time Inspection Program	VIII.E.1-a	3.4.1-02	E
Fasteners	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Bolting Integrity Program	VIII.H.2-a	3.4.1-08	406, C
				Loss of Preload	Bolting Integrity Program	VIII.H.2-a	3.4.1-08	406, C
Heat Exchangers	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-08	A
			Treated Water (Int)	Loss of Material	One-Time Inspection Program	VIII.E.1-a	3.4.1-02	E
Pipe & Fittings	Fluid Pressure Boundary	Carbon Steel	Steam (Int)	Loss of Material	Flow Accelerated Corrosion Program	VIII.C.1-a	3.4.1-06	423, A
					Water Chemistry Program	VIII.C.1-b	3.4.1-02	A
					One-Time Inspection Program	VIII.C.1-b	3.4.1-02	411, B
			Treated Water (Int)	Loss of Material	Flow Accelerated Corrosion Program	VIII.E.1-a	3.4.1-06	414, 423, C
					One-Time Inspection Program	VIII.E.1-b	3.4.1-02	411, 414, D
					Water Chemistry Program	VIII.E.1-b	3.4.1-02	411, 414, C

Table 3.4.2-4 Steam and Power Conversion System - Heater Extraction and Drain System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Pipe & Fittings	Fluid Pressure Boundary	Stainless Steel	Steam (Int)	Cracking	Water Chemistry Program			404, F
					One-Time Inspection Program			404, F
				Loss of Material	Water Chemistry Program			404, F
					One-Time Inspection Program			404, F
Pumps	Fluid Pressure Boundary	Carbon Steel	Treated Water (Int)	Loss of Material	One-Time Inspection Program	VIII.E.3-a	3.4.1-02	411, 414, D
					Water Chemistry Program	VIII.E.3-a	3.4.1-02	411, 414, C
Transmitter/ Element	Fluid Pressure Boundary	Stainless Steel	Treated Water (Int)	Cracking	Water Chemistry Program			410, F
					One-Time Inspection Program			410, F
				Loss of Material	Water Chemistry Program			F
					One-Time Inspection Program			F
Traps (Steam)	Fluid Pressure Boundary	Carbon Steel	Steam (Int)	Loss of Material	Water Chemistry Program	VIII.C.2-b	3.4.1-02	C
					One-Time Inspection Program	VIII.C.2-b	3.4.1-02	D

Table 3.4.2-4 Steam and Power Conversion System - Heater Extraction and Drain System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Valves & Dampers	Fluid Pressure Boundary	Carbon Steel	Steam (Int)	Loss of Material	Flow Accelerated Corrosion Program	VIII.C.2-a	3.4.1-06	423, A
					Water Chemistry Program	VIII.C.2-b	3.4.1-02	A
					One-Time Inspection Program	VIII.C.2-b	3.4.1-02	B
			Treated Water (Int)	Loss of Material	Flow Accelerated Corrosion Program	VIII.E.2-a	3.4.1-06	414, 423, C
					One-Time Inspection Program	VIII.E.2-b	3.4.1-02	411, 414, D
					Water Chemistry Program	VIII.E.2-b	3.4.1-02	411, 414, C
		Cast Iron	Plant Indoor Air (Ext)	Loss of Material	None Required	VIII.H.1-b	3.4.1-05	401, 422, I
			Steam (Int)	Loss of Material	Flow Accelerated Corrosion Program	VIII.C.2-a	3.4.1-06	422, 423, A
					Water Chemistry Program	VIII.C.2-b	3.4.1-02	422, A
					One-Time Inspection Program	VIII.C.2-b	3.4.1-02	422, B
				Loss of Material - Selective Leaching	One-Time Inspection Program			412, 422, F

Table 3.4.2-4 Steam and Power Conversion System - Heater Extraction and Drain System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Valves & Dampers	Fluid Pressure Boundary	Stainless Steel	Steam (Int)	Cracking	Water Chemistry Program			404, F
					One-Time Inspection Program			404, F
				Loss of Material	Water Chemistry Program			404, F
					One-Time Inspection Program			404, F

Table 3.4.2-5 Steam and Power Conversion System - Main Air Ejection and Gland Seal System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Blowers Fans Compressor Vacuum	Fluid Pressure Boundary	Carbon Steel	Air (Int)	Loss of Material	One-Time Inspection Program			415, J
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	A
			Steam (Int)	Loss of Material	Water Chemistry Program	VIII.A.1-b	3.4.1-02	415, C
					One-Time Inspection Program	VIII.A.1-b	3.4.1-02	415, D
		Cast Iron	Steam (Int)	Loss of Material	Water Chemistry Program	VIII.A.1-b	3.4.1-02	415, 422, C
					One-Time Inspection Program	VIII.A.1-b	3.4.1-02	415, 422, D
				Loss of Material - Selective Leaching	One-Time Inspection Program			412, 415, J
Filters/Strainers	Fluid Pressure Boundary	Carbon Steel	Air (Int)	Loss of Material	One-Time Inspection Program			415, J
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	A
			Steam (Int)	Loss of Material	Flow Accelerated Corrosion Program	VIII.A.1-a	3.4.1-06	415, C
					Water Chemistry Program	VIII.A.1-b	3.4.1-02	415, C
					One-Time Inspection Program	VIII.A.1-b	3.4.1-02	415, D

Table 3.4.2-5 Steam and Power Conversion System - Main Air Ejection and Gland Seal System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Heat Exchangers	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	A
			Steam (Int)	Loss of Material	Water Chemistry Program	VIII.A.1-b	3.4.1-02	415, C
					One-Time Inspection Program	VIII.A.1-b	3.4.1-02	415, D
			Treated Water (Int)	Loss of Material	One-Time Inspection Program	VIII.D1.2-b	3.4.1-02	411, 415, D
Fasteners	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Bolting Integrity Program	VIII.H.2-a	3.4.1-08	406, C
				Loss of Preload	Bolting Integrity Program	VIII.H.2-a	3.4.1-08	406, C
Pipe & Fittings	Fluid Pressure Boundary	Carbon Steel	Air (Int)	Loss of Material	One-Time Inspection Program			416, G
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	A
			Steam (Int)	Loss of Material	Water Chemistry Program	VIII.A.1-b	3.4.1-02	415, C
					One-Time Inspection Program	VIII.A.1-b	3.4.1-02	415, D

Table 3.4.2-5 Steam and Power Conversion System - Main Air Ejection and Gland Seal System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Traps (Steam)	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	A
			Raw Water (Int)	Loss of Material	One-Time Inspection Program	VII.C1.2-a	3.3.1-17	E
			Steam (Int)	Loss of Material	Water Chemistry Program	VIII.A.2-b	3.4.1-02	415, C
					One-Time Inspection Program	VIII.A.2-b	3.4.1-02	415, D
Valves & Dampers	Fluid Pressure Boundary	Carbon Steel	Air (Int)	Loss of Material	One-Time Inspection Program			416, G
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	A
			Steam (Int)	Loss of Material	Flow Accelerated Corrosion Program	VIII.A.2-a	3.4.1-06	415, C
					One-Time Inspection Program	VIII.A.2-b	3.4.1-02	415, D
					Water Chemistry Program	VIII.A.2-b	3.4.1-02	415, C
			Treated Water (Int)	Loss of Material	One-Time Inspection Program	VIII.D1.2-b	3.4.1-02	411, 415, D
					Water Chemistry Program	VIII.D1.2-b	3.4.1-02	411, 415, C

Table 3.4.2-5 Steam and Power Conversion System - Main Air Ejection and Gland Seal System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Valves & Dampers	Fluid Pressure Boundary	Cast Iron	Steam (Int)	Loss of Material	Water Chemistry Program	VIII.A.2-b	3.4.1-02	415, 422, C
					One-Time Inspection Program	VIII.A.2-b	3.4.1-02	411, 415, 422, D
				Loss of Material - Selective Leaching	One-Time Inspection Program			412, 422, F

Table 3.4.2-6 Steam and Power Conversion System - Main Steam System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Accumulators	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	C
			Steam (Int)	Loss of Material	Flow Accelerated Corrosion Program	VIII.A.1-a	3.4.1-06	403, 423, C
					Water Chemistry Program	VIII.A.1-b	3.4.1-02	403, C
					One-Time Inspection Program	VIII.A.1-b	3.4.1-02	403, D
			Treated Water (Int)	Loss of Material	One-Time Inspection Program	VIII.F.1-b	3.4.1-02	E
Filters/Strainers	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	401, A
			Steam (Int)	Loss of Material	Flow Accelerated Corrosion Program	VIII.A.2-a	3.4.1-06	403, 423, C
					Water Chemistry Program	VIII.A.2-b	3.4.1-02	403, C
					One-Time Inspection Program	VIII.A.2-b	3.4.1-02	403, D
			Treated Water (Int)	Loss of Material	One-Time Inspection Program	VIII.E5-a	3.4.1-02	E
Indicators/ Recorders	Fluid Pressure Boundary	Stainless Steel	Steam (Int)	Cracking	Water Chemistry Program			403, 404, J
				Loss of Material	Water Chemistry Program			403, 404, J

Table 3.4.2-6 Steam and Power Conversion System - Main Steam System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Ejectors	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	401, A
			Steam (Int)	Loss of Material	Water Chemistry Program	VIII.A.1-b	3.4.1-02	403, C
					One-Time Inspection Program	VIII.A.1-b	3.4.1-02	403, D
Fasteners	Fluid Pressure Boundary	Carbon Steel	Containment Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	A
			Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VIII.H.1-a	3.4.1-13	401, A
					Bolting Integrity Program	VIII.H.2-a	3.4.1-08	406, C
				Loss of Preload	Bolting Integrity Program	VIII.H.2-a	3.4.1-08	406, C
Heat Exchangers	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	A
			Steam (Int)	Loss of Material	One-Time Inspection Program	VIII.B1.1-a	3.4.1-07	E
			Treated Water (Int)	Loss of Material	One-Time Inspection Program	VIII.F.1-b	3.4.1-02	E

Table 3.4.2-6 Steam and Power Conversion System - Main Steam System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Pipe & Fittings	Fluid Pressure Boundary	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VIII.H.1-a	3.4.1-13	A
			Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VIII.H.1-a	3.4.1-13	A
			Steam (Int)	Loss of Material	Flow Accelerated Corrosion Program	VIII.A.1-a	3.4.1-06	423, A
						VIII.B1.1-c	3.4.1-06	423, A
					Water Chemistry Program	VIII.A.1-b	3.4.1-02	424, A
						VIII.B1.1-a	3.4.1-07	424, A
					One-Time Inspection Program	VIII.A.1-b	3.4.1-02	B
			Treated Water (Int)	Loss of Material	Flow Accelerated Corrosion Program	VIII.F.1-a	3.4.1-06	423, A
					One-Time Inspection Program	VIII.F.1-b	3.4.1-02	411, B
					Water Chemistry Program	VIII.F.1-b	3.4.1-02	411, A
		Copper Alloys	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program			402, F
		Stainless Steel	Steam (Int)	Cracking	Water Chemistry Program			404, G
			Steam (Int)	Loss of Material	Water Chemistry Program			404, G
Pumps	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-13	A
			Treated Water (Int)	Loss of Material	One-Time Inspection Program	VIII.E.5-a	3.4.1-02	E

Table 3.4.2-6 Steam and Power Conversion System - Main Steam System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Traps (Steam)	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VIII.H.1-a	3.4.1-13	A
			Steam (Int)	Loss of Material	Flow Accelerated Corrosion Program	VIII.A.2-a	3.4.1-06	403, 423, C
					Water Chemistry Program	VIII.A.2-b	3.4.1-02	403, C
					One-Time Inspection Program	VIII.A.2-b	3.4.1-02	403, D
			Treated Water (Int)	Loss of Material	One-Time Inspection Program	VIII.F.1-b	3.4.1-02	E
Valves & Dampers	Fluid Pressure Boundary	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VIII.H.1-a	3.4.1-13	A
			Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VIII.H.1-a	3.4.1-13	A
			Steam (Int)	Loss of Material	Flow Accelerated Corrosion Program	VIII.A.2-a	3.4.1-06	423, A
						VIII.B1.2-b	3.4.1-06	423, A
					Water Chemistry Program	VIII.A.2-b	3.4.1-02	424, A
						VIII.B1.2-a	3.4.1-07	424, A
					One-Time Inspection Program	VIII.A.2-b	3.4.1-02	B

Table 3.4.2-6 Steam and Power Conversion System - Main Steam System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Valves & Dampers	Fluid Pressure Boundary	Carbon Steel	Treated Water (Int)	Loss of Material	Flow Accelerated Corrosion Program	VIII.F.2-a	3.4.1-06	423, A
					One-Time Inspection Program	VIII.F.2-b	3.4.1-02	411, B
					Water Chemistry Program	VIII.F.2-b	3.4.1-02	411, A
		Stainless Steel	Steam (Int)	Cracking	Water Chemistry Program			404, F
				Loss of Material	Water Chemistry Program			404, F
			Treated Water (Int)	Loss of Material	One-Time Inspection Program	VIII.E.5-b	3.4.1-02	417, 419, D
					Water Chemistry Program	VIII.E.5-b	3.4.1-02	417, 419, C

Table 3.4.2-7 Steam and Power Conversion System - Turbine Generator System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Accumulators	Fluid Pressure Boundary	Carbon Steel	Air/Gas (Int)	Loss of Material	One-Time Inspection Program	VII.H.2.3-a	3.3.1-05	E
			Oil (Int)	Loss of Material	One-Time Inspection Program	VII.G.7-a	3.3.1-06	E
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	A
		Stainless Steel	Oil (Int)	Loss of Material	One-Time Inspection Program			G
Fasteners	Fluid Pressure Boundary	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	A
				Loss of Preload	Bolting Integrity Program	VIII.H.2-a	3.4.1-08	406, C
Filters/Strainers	Fluid Pressure Boundary	Carbon Steel	Oil (Int)	Loss of Material	One-Time Inspection Program	VII.G.7-a	3.3.1-06	E
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	A
Heat Exchangers	Fluid Pressure Boundary	Carbon Steel	Air/Gas (Int)	Loss of Material	One-Time Inspection Program	VII.H.2.3-a	3.3.1-05	E
			Oil (Int)	Loss of Material	One-Time Inspection Program	VII.G.7-a	3.3.1-06	E
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	A
			Raw Water (Int)	Loss of Material	One-Time Inspection Program	VII.C1.3-a	3.3.1-17	E

Table 3.4.2-7 Steam and Power Conversion System - Turbine Generator System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Pipe & Fittings	Fluid Pressure Boundary	Carbon Steel	Oil (Int)	Loss of Material	One-Time Inspection Program	VII.G.7-b	3.3.1-06	E
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	A
			Raw Water (Int)	Loss of Material	One-Time Inspection Program	VII.C1.1-a	3.3.1-17	E
			Steam (Int)	Loss of Material	One-Time Inspection Program	VIII.B1.1-c	3.3.1-17	E
			Treated Water (Int)	Loss of Material	One-Time Inspection Program	VII.C2.1-a	3.3.1-17	E
		Stainless Steel	Oil (Int)	Loss of Material	One-Time Inspection Program			G
Pumps	Fluid Pressure Boundary	Carbon Steel	Oil (Int)	Loss of Material	One-Time Inspection Program	VII.G.7-a	3.3.1-06	E
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	A
		Cast Iron	Oil (Int)	Loss of Material	One-Time Inspection Program	VII.G.7-a	3.3.1-06	E
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	A

Table 3.4.2-7 Steam and Power Conversion System - Turbine Generator System - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Valves & Dampers	Fluid Pressure Boundary	Carbon Steel	Oil (Int)	Loss of Material	One-Time Inspection Program	VII.G.7-b	3.3.1-06	E
			Plant Indoor Air (Ext)	Loss of Material	System Monitoring Program	VIII.H.1-b	3.4.1-05	A
			Raw Water (Int)	Loss of Material	One-Time Inspection Program	VII.C1.2-a	3.3.1-17	E
			Steam (Int)	Loss of Material	One-Time Inspection Program	VIII.B1.2-b	3.3.1-17	E
			Treated Water (Int)	Loss of Material	One-Time Inspection Program	VII.C2.2-a	3.3.1-17	E

Notes for Tables 3.4.2-1 through 3.4.2-7

A Consistent with NUREG-1801 item for component, material, environment, and aging management program. AMP is consistent with NUREG-1801 AMP.

B Consistent with NUREG-1801 item for component, material, environment, and aging management program. AMP has exceptions to NUREG-1801 AMP.

C Component is different, but consistent with NUREG-1801 for material, environment, aging effect and AMP. AMP is consistent with NUREG-1801 AMP.

D Component is different, but consistent with NUREG-1801 for material, environment, aging effect and AMP. AMP has exceptions to NUREG-1801 AMP.

E Consistent with NUREG-1801 for material, environment, aging effect but a different AMP is credited.

F Material not in NUREG-1801 for this component.

G Environment not in NUREG-1801 for this component and material.

H Aging effect not in NUREG-1801 for this component, material and environment combination.

I Aging effect in NUREG-1801 for this component, material and environment combination is not applicable.

J Neither the component nor the material and environment combination is evaluated in NUREG-1801.

Plant-specific notes:

401 The external surface temperature of this component is < 212°F.

402 Copper alloys are not addressed in S&PC Supergroup of GALL.

403 These components are not addressed in MSS subgroup (VIII.A, VIII.B1, and VIII.F) of GALL.

404 The combination of Stainless Steels and environment of Steam is not evaluated in S&PC Supergroup of GALL.

405 Use GALL line of Condensate Storage Tank in VIII.E for this evaluation.

406 Loss of preload is included here in response to recent NRC RAIs on non-primary system, high temperature bolting that may experience loss of preload. The Palisades Bolting Integrity Program manages potential bolting AERMs and event driven degradation. GALL reconciliation is based on Loss of Material.

407 Buried stainless steel piping are not addressed in S&PC Supergroup of GALL.

408 Use the GALL line of FWS or AFW piping for this evaluation.

409 These components associated with AFW turbine are normally exposed to air in standby with steam supply isolated.

410 Aging effect of cracking for internal environment is not evaluated in S&PC Supergroup of GALL.

411 For carbon steels, Galvanic Corrosion and/or MIC are conservatively included as aging mechanisms for Loss of Material.

412 Selective Leaching is not addressed in S&PC Supergroup of GALL.

413 Use GALL VIII.A for evaluation of steam environment and VIII.E.2 for condensate environment.

414 The environment of Treated Water is not addressed in GALL VIII.C. Use GALL line of Condensate System for this evaluation.

415 Internal environment of components associated with condenser vacuum. Air Ejectors are not addressed in S&PC Supergroup of GALL.

416 The internal environment of Air is not addressed in S&PC Supergroup of GALL.

417 Demineralized Make-Up Water (DMW) is not addressed in GALL. Use GALL line of Condensate System or Feedwater System for this evaluation.

418 Palisades does not maintain a separate Above ground Carbon Steel Tanks Program (GALL XI.M29). It is implemented by System Monitoring Program (inspection) and One-Time Inspection Program (tank bottom thickness).

419 For stainless steels in treated water < 212°F, MIC is conservatively included as an aging mechanism for Loss of Material.

420 For carbon steels outside the plant buildings, Crevice and Pitting corrosion are considered to be applicable aging mechanisms.

421 The AFW isolation valves of backup supplies from Fire Protection Water and Service Water are normally closed. It is not covered by Fire Protection Water program or Service Water Program. The One-Time Inspection Program will determine whether further actions are required.

422 Cast Irons are considered to be equivalent to Carbon Steels in evaluation of the aging effects of Loss of Material, except Selective Leaching.

423 As described in Plant OE database, erosion had occurred to some components that also are susceptible to FAC aging effect. Thus it is conservatively assumed this aging effect is

applicable to the components susceptible to FAC. This aging effect is covered by the Flow Accelerated Corrosion Program.

424 General Corrosion is not evaluated for carbon steel main steam piping and valves in GALL VIII.B1.1-a and VIII.B1.2-a.

3.5 Aging Management of Containments, Structures, and Component Supports

3.5.1 Introduction

This section provides the results of the aging management review for those components identified in Section 2.4, Containments, Structures, and Component Supports, as being subject to aging management review. The systems, or portions of systems, which are addressed in this section, are described in the indicated sections.

- Auxiliary Building (Section 2.4.1)
- Component Supports (Section 2.4.2)
- Containment (Section 2.4.3)
- Containment Interior Structures (Section 2.4.4)
- Discharge Structure (Section 2.4.5)
- Feedwater Purity Building (Section 2.4.6)
- Intake Structure (Section 2.4.7)
- Miscellaneous Structural and Bulk Commodities (Section 2.4.8)
- Switchyard and Yard Structures (Section 2.4.9)
- Turbine Building (Section 2.4.10)

Table 3.5.1, Summary of Aging Management Evaluations in Chapters II and III of NUREG-1801 for Structures and Component Supports, provides the summary of the programs evaluated in NUREG-1801 for the Structures and Component Supports component groups that are relied on for license renewal.

This table uses the format described in Figure 3.0-1 above. Note that this table only includes those component groups that are applicable to a PWR.

3.5.2 Results

The following tables summarize the results of the aging management review for systems in the Containments, Structures, and Component Supports group:

Table 3.5.2-1, Structures and Component Supports - Auxiliary Building - Summary of Aging Management Evaluation

Table 3.5.2-2, Structures and Component Supports - Component Supports - Summary of Aging Management Evaluation

Table 3.5.2-3, Structures and Component Supports - Containment - Summary of Aging Management Evaluation

Table 3.5.2-4, Structures and Component Supports - Containment Interior Structures - Summary of Aging Management Evaluation

Table 3.5.2-5, Structures and Component Supports - Discharge Structure - Summary of Aging Management Evaluation

Table 3.5.2-6, Structures and Component Supports - Feedwater Purity Building - Summary of Aging Management Evaluation

Table 3.5.2-7, Structures and Component Supports - Intake Structure - Summary of Aging Management Evaluation

Table 3.5.2-8, Structures and Component Supports - Miscellaneous Structural and Bulk Commodities - Summary of Aging Management Evaluation

Table 3.5.2-9, Structures and Component Supports - Switchyard and Yard Structures - Summary of Aging Management Evaluation

Table 3.5.2-10, Structures and Component Supports - Turbine Building - Summary of Aging Management Evaluation

The materials that specific components are fabricated from, the environments to which components are exposed, the potential aging effects requiring management, and the aging management programs used to manage these aging effects are provided for each of the above systems in the following subsections of Section 3.5.2.1, Materials, Environment, Aging Effects Requiring Management and Aging Management Programs:

Section 3.5.2.1.1, Auxiliary Building

Section 3.5.2.1.2, Component Supports

Section 3.5.2.1.3, Containment

Section 3.5.2.1.4, Containment Interior Structures

Section 3.5.2.1.5, Discharge Structure

Section 3.5.2.1.6, Feedwater Purity Building

Section 3.5.2.1.7, Intake Structure

Section 3.5.2.1.8, Miscellaneous Structural and Bulk Commodities

Section 3.5.2.1.9, Switchyard and Yard Structures

Section 3.5.2.1.10, Turbine Building

3.5.2.1 Materials, Environment, Aging Effects Requiring Management and Aging Management Programs

3.5.2.1.1 Auxiliary Building

Materials

The materials of construction for the Auxiliary Building components are:

- Carbon Steel
- Concrete
- Galvanized
- Stainless Steel

Environment

The Auxiliary Building components are exposed to the following environments:

- Atmosphere/Weather (Ext)
- Plant Indoor Air (Ext)
- Soil (Ext)
- Treated Water (Ext)

Aging Effects Requiring Management

The following aging effects, associated with the Auxiliary Building, require management:

- Change in Material Properties
- Cracking
- Cracking and Expansion
- Cracking, Loss of Bond/Material
- Loss of Leak Tightness
- Loss of Material
- Loss of Strength
- Reduction in Concrete Anchor Capacity

Aging Management Programs

The following aging management programs manage the aging effects for the Auxiliary Building components:

- Boric Acid Corrosion Program

- Structural Monitoring Program
- Water Chemistry Program

3.5.2.1.2 Component Supports

Materials

The materials of construction for the Component Supports components are:

- Aluminum
- Boron Carbide
- Bronze
- Carbon Steel
- Cast Iron
- Concrete
- Galvanized
- Stainless Steel

Environment

The Component Supports components are exposed to the following environments:

- Atmosphere/Weather (Ext)
- Containment Air (Ext)
- Plant Indoor Air (Ext)
- Raw Water (Ext)
- Treated Water (Ext)

Aging Effects Requiring Management

The following aging effects, associated with the Component Supports, require management:

- Cracking
- Loss of Material
- Reduction in Conc Anchor Capacity

Aging Management Programs

The following aging management programs manage the aging effects for the Component Supports components:

- ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program
- Bolting Integrity Program
- Boric Acid Corrosion Program
- Structural Monitoring Program
- Water Chemistry Program

3.5.2.1.3 Containment

Materials

The materials of construction for the Containment components are:

- Carbon Steel
- Concrete
- Elastomers
- Stainless Steel

Environment

The Containment components are exposed to the following environments:

- Atmosphere/Weather (Ext)
- Containment Air (Ext)
- Soil (Ext)

Aging Effects Requiring Management

The following aging effects, associated with the Containment, require management:

- Change in Material Properties
- Cracking
- Cracking and Expansion
- Cracking, Loss of Bond/Material
- Loss of Leak Tightness
- Loss of Material

Aging Management Programs

The following aging management programs manage the aging effects for the Containment components:

- Boric Acid Corrosion Program
- Containment Inservice Inspection Program
- Containment Leakage Testing Program
- Structural Monitoring Program

3.5.2.1.4 Containment Interior Structures

Materials

The materials of construction for the Containment Interior Structures components are:

- Carbon Steel
- Concrete
- Stainless Steel
- Galvanized

Environment

The Containment Interior Structures components are exposed to the following environments:

- Containment Air (Ext)

Aging Effects Requiring Management

The following aging effects, associated with the Containment Interior Structures, require management:

- Change in Material Properties
- Cracking
- Cracking and Expansion
- Cracking, Loss of Bond/Material
- Loss of Material
- Reduction in Conc Anchor Capacity

Aging Management Programs

The following aging management programs manage the aging effects for the Containment Interior Structures components:

- Boric Acid Corrosion Program
- Structural Monitoring Program
- Water Chemistry Program

3.5.2.1.5 Discharge Structure

Materials

The materials of construction for the Discharge Structure components are:

- Carbon Steel
- Cast Iron
- Concrete

Environment

The Discharge Structure components are exposed to the following environments:

- Atmosphere/ Weather (Ext)
- Soil (Ext)
- Raw Water (Ext)

Aging Effects Requiring Management

The following aging effects, associated with the Discharge Structure, require management:

- Change in Material Properties
- Cracking
- Cracking and Expansion
- Cracking, Loss of Bond/Material
- Loss of Material
- Reduction in Concrete Anchor Capacity

Aging Management Programs

The following aging management programs manage the aging effects for the Discharge Structure components:

- Structural Monitoring Program

3.5.2.1.6 Feedwater Purity Building

Materials

The materials of construction for the Feedwater Purity Building components are:

- Carbon Steel
- Concrete

Environment

The Feedwater Purity Building components are exposed to the following environments:

- Atmosphere/Weather (Ext)
- Plant Indoor Air (Ext)
- Soil (Ext)

Aging Effects Requiring Management

The following aging effects, associated with the Feedwater Purity Building, require management:

- Change in Material Properties
- Cracking
- Cracking and Expansion
- Cracking, Loss of Bond/Material
- Loss of Material
- Loss of Strength
- Reduction in Concrete Anchor Capacity

Aging Management Programs

The following aging management programs manage the aging effects for the Feedwater Purity Building components:

- Structural Monitoring Program

3.5.2.1.7 Intake Structure

Materials

The materials of construction for the Intake Structure components are:

- Carbon Steel
- Cast Iron
- Concrete
- Galvanized

Environment

The Intake Structure components are exposed to the following environments:

- Atmosphere/Weather (Ext)
- Plant Indoor Air (Ext)
- Raw Water (Ext)
- Soil (Ext)

Aging Effects Requiring Management

The following aging effects, associated with the Intake Structure, require management:

- Change in Material Properties
- Cracking
- Cracking and Expansion
- Cracking, Loss of Bond/Material
- Loss of Material
- Loss of Strength
- Reduction in Concrete Anchor Capacity

Aging Management Programs

The following aging management programs manage the aging effects for the Intake Structure components:

- Structural Monitoring Program

3.5.2.1.8 Miscellaneous Structural and Bulk Commodities

Materials

The materials of construction for the Miscellaneous Structural and Bulk Commodities components are:

- Built-up Roofing
- Carbon Steel
- Concrete
- Elastomers
- Fire Stops [Sealant/Maranite]
- Fire Wraps [Mineral-Wool Batts]
- Galvanized
- Soil

Environment

The Miscellaneous Structural and Bulk Commodities components are exposed to the following environments:

- Containment Air (Ext)
- Atmosphere/Weather (Ext)
- Plant Indoor Air (Ext)
- Raw Water (Ext)

Aging Effects Requiring Management

The following aging effects, associated with the Miscellaneous Structural and Bulk Commodities, require management:

- Change in Material Properties
- Cracking
- Cracking and Expansion
- Cracking/Delamination
- Cracking, Loss of Bond/Material
- Loss of Form
- Loss of Leak Tightness
- Loss of Material

- Loss of Material/Form
- Separation

Aging Management Programs

The following aging management programs manage the aging effects for the Miscellaneous Structural and Bulk Commodities components:

- Fire Protection Program
- Overhead Load Handling Systems Inspection Program
- Structural Monitoring Program
- Containment Inservice Inspection Program

3.5.2.1.9 **Switchyard and Yard Structures**

Materials

The materials of construction for the Switchyard and Yard Structures components are:

- Carbon Steel
- Concrete
- Galvanized

Environment

The Switchyard and Yard Structures components are exposed to the following environments:

- Atmosphere/Weather (Ext)
- Plant Indoor Air (Ext)
- Soil (Ext)

Aging Effects Requiring Management

The following aging effects, associated with the Switchyard and Yard Structures, require management:

- Change in Material Properties
- Cracking
- Cracking and Expansion
- Cracking, Loss of Bond/Matl
- Loss of Material

- Loss of Strength
- Reduction in Concrete Anchor Capacity

Aging Management Programs

The following aging management programs manage the aging effects for the Switchyard and Yard Structures components:

- Structural Monitoring Program

3.5.2.1.10 Turbine Building

Materials

The materials of construction for the Turbine Building components are:

- Bronze
- Carbon Steel
- Concrete
- Galvanized

Environment

The Turbine Building components are exposed to the following environments:

- Atmosphere/ Weather (Ext)
- Soil (Ext)
- Plant Indoor Air (Ext)

Aging Effects Requiring Management

The following aging effects, associated with the Turbine Building, require management:

- Change in Material Properties
- Cracking
- Cracking and Expansion
- Cracking, Loss of Bond/Matl
- Loss of Leak Tightness
- Loss of Material
- Loss of Strength
- Reduction in Concrete Anchor Capacity

Aging Management Programs

The following aging management programs manage the aging effects for the Turbine Building components:

- Structural Monitoring Program

3.5.2.2 Further Evaluation of Aging Management as Recommended by NUREG-1801

NUREG-1801 Volume 1 Tables provide the basis for identifying those programs that warrant further evaluation by the reviewer in the license renewal application. For the Containments, Structures, and Component Supports, those programs are addressed in the following sections.

3.5.2.2.1 PWR Containment

3.5.2.2.1.1 Aging of Inaccessible Concrete Areas

NUREG 1800 states that cracking, spalling, and increases in porosity and permeability due to leaching of calcium hydroxide and aggressive chemical attack; and cracking, spalling, loss of bond, and loss of material due to corrosion of embedded steel; could occur in inaccessible areas of PWR concrete and steel containments. NUREG-1801 recommends further evaluation of plant-specific programs to manage the aging effects for inaccessible areas if specific criteria defined in NUREG-1801 cannot be satisfied.

Loss of Material (Spalling, Scaling) and Cracking due to Freeze - Thaw (GALL IIA1.1-a)

Accessible Areas: In accordance with NUREG-1801 as revised by ISG-03, accessible reinforced concrete structures and components will be monitored by the Containment Inservice Inspection Program to manage Loss of Material (spalling, scaling) and cracking due to Freeze - thaw.

Inaccessible Areas: Palisades is located in an area with severe weathering conditions as noted on Figure 1 of ASTM C33-99. Freeze-thaw is not considered an aging mechanism for concrete components below the frost line (depth of 42 inches, per Michigan Building Code).

Discussion: The Palisades concrete structures and concrete are designed in accordance with ACI 318-63 & 71, and constructed using ingredients conforming to ACI and ASTM standards. Palisades specifications require all concrete to contain an air-entraining agent in sufficient quantity to maintain specified percentages based on nominal maximum size aggregate. For severe weather exposures, the air content identified varies from 3 to 5 percent. Containment replacement concrete for steam generator replacement access had 3 to 7 percent air entrainment specified. Water/cement ratios for concrete mixes range from 0.44 for the 5000 psi concrete and 0.45 to 0.46 for the 4000 psi concrete mix designs used in the primary containment construction (FSAR Section 5.8.7.1), and 0.33 for 5000 psi concrete used for closing the opening in the shell for steam generator replacement (FSAR Section 5.8.9.3.1.4).

Review of Operating Experience and Maintenance Rule Structures Monitoring results confirmation that freeze-thaw degradation of Palisades' concrete has not occurred.

As described in NUREG-1557 "Summary of Technical Information and Agreements from Nuclear Management and Resources Council [NUMARC] Industry Reports Addressing License Renewal," freeze-thaw does not cause loss of material from reinforced concrete in foundations, and in above and below grade exterior concrete, for plants located in a geographic region of negligible weathering conditions (weathering index <100 day-inch/yr). Similarly, freeze-thaw damage is not significant for reinforced concrete in foundations, and in above and below grade exterior concrete, for plants located in areas in which weathering conditions are considered severe (weathering index >500 day-inch/yr) or moderate (100-500 day-inch/yr), provided that the concrete mix design meets the air content (entrained air 3-6%) and water-to-cement ratio (0.35-0.45) specified in ACI 318-63 or ACI 349-85. Palisades water-cement ratios range between 0.33 and 0.46, and ACI 201.2R-77 Section 1.4.2 recommends a not to exceed water-cement ratio of 0.50 for other than "thin" structures. In addition, at Palisades, the ground tends to freeze and stay frozen through the winter, which means that below grade concrete is only exposed, essentially, to one freeze-thaw cycle in a typical year. Since damage to concrete from freeze-thaw is related to repeated cycles of freeze-thaw as noted in EPRI Structural Tools, Rev1, Section 5.3.1.1, and no evidence of freeze-thaw damage has been identified in exposed concrete which is subjected to repeated cycles, it is concluded that below grade concrete will not experience damage or degradation.

Therefore, since these conditions for Palisades concrete are satisfied, aging management is not required for below grade concrete.

Increase in Porosity, Permeability, and Loss of Strength Due to Leaching (GALL IIA1.1-b)

Accessible Areas: In accordance with NUREG-1801 as revised by ISG-03, accessible reinforced concrete structures and components will be monitored by the Containment Inservice Inspection Program to manage increase in porosity and permeability, and loss of strength due to leaching of calcium hydroxide.

Inaccessible Areas (Containment Shell and Basemat Concrete): The Palisades concrete structures and concrete components are designed in accordance with ACI 318-63 and constructed using ingredients conforming to ACI and ASTM standards, which provide for good quality, dense, well cured, and low permeability concrete. Cracking is controlled through proper arrangement and distribution of reinforcing bars.

Concrete structures and concrete components are constructed of a dense, well-cured concrete with an amount of cement suitable for strength development, and achievement of a water-to-cement ratio (Palisades water/cement ratio 0.45), which is characteristic of concrete having low permeability. This is consistent with the recommendations and guidance provided by ACI 201.2R-77.

In addition, concrete components must be exposed to flowing water through the concrete component in order for leaching to be an issue. Ground water elevation is Elev.580', with primary containment basement floor at Elev. 590'. In addition, the containment bottom/floor has a ¼” thick, steel liner plate welded to embedments and covered with 18" concrete, further reducing the likelihood of water flowing through the floor.

Natural groundwater movement in this area is from the plant site to Lake Michigan. The rate of groundwater flow estimated during site exploration and discussed in the FSAR Section 2.2.1, is 650 feet per year (0.074 feet/hr), which is not considered an aggressive flow rate. Groundwater elevation at the plant site is the same as Lake Michigan (elevation corresponds with lake water level).

Leaching of Calcium Hydroxide is readily noticeable as white deposits that remain on the concrete surface after a solution of water-free lime from the concrete and carbon dioxide from the air is absorbed and dries. The Containment ISI Program inspects concrete surfaces for signs of leaching.

No significant signs of leaching have been documented during these inspection walkdowns.

In summary, the accessible concrete will be inspected in accordance with ASME Section Subsection IWL requirements. Inaccessible concrete is not subject to flowing water, and was constructed consistent with ACI 201.2R such that below grade aging management is not required. Therefore, the conditions identified in NUREG-1801 as revised by ISG-03 are satisfied, and aging management of increase in porosity and permeability, loss of strength due to leaching of calcium hydroxide for below grade inaccessible concrete is not required.

Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) due to Aggressive Chemical Attack (GALL IIA1.1-c)

Inaccessible Areas: ISG-3 indicates that a plant-specific aging management program is required for below-grade exterior reinforced concrete (basemat, embedded walls) if the environment is aggressive (pH < 5.5, chlorides >500 ppm, or sulfates > 1500 ppm). Examination of representative samples of below-grade concrete, when excavated for any reason, is to be included as part of a plant-specific program. Periodic monitoring of below-grade water chemistry (including consideration of potential seasonal variations) is an acceptable approach to demonstrate that the below-grade environment is aggressive or non-aggressive.

Design and construction of Palisades' reinforced concrete provides for dense, well cured, and low permeability concrete that provides an acceptable degree of protection against exposure of below-grade exterior reinforced concrete to an aggressive environment. Cracking of concrete is controlled through proper arrangement and distribution of reinforcing bars. Continued or frequent cyclic exposure to the following aggressive environments is necessary for aggressive chemicals to cause a significant increase in porosity and permeability, cracking, loss of material (spalling, scaling):

Acidic solutions with pH < 5.5

Chloride solutions >500 ppm

Sulfate solutions >1500 ppm

Since aggressive chemicals are contained at plant sites, system leakage is possible that could cause the reinforced concrete to be exposed to chemicals beyond these limits. However, leaks are not expected to continue for the extensive periods required for degradation, and repairs would be completed prior to loss of intended function. It is not likely that leaks inside the structure would get outside to cause an aggressive chemical attack on embedded concrete.

An aggressive environment may also occur when reinforced concrete is exposed to aggressive aqueous solutions such as groundwater or aggressive water flow. Palisades groundwater water sample measurements, summarized below, have confirmed that parameters are well below threshold limits that could cause concrete degradation i.e., (an aggressive environment does not exist).

Palisades Groundwater Sampling results from 1966, 1996, and 2004:

Chemistry/Year	1966 (20 locations)	1996	2004
pH	Range 6.1 - 7.7 (> 5.5)	No reading	7.0
Chlorides - ppm	Range 4.0 - 39 (<500)	23	139
Sulfates - ppm	Range 9.47 - 33.17 (<1500)	15.2	11.5

Natural groundwater movement in this area is from the plant site to Lake Michigan. The rate of groundwater flow estimated during site exploration and discussed in the FSAR Section 2.2.1 Groundwater, is 650 feet per year (0.074'/hr), which is not considered an aggressive flow rate. Groundwater elevation at the plant site is the same as Lake Michigan (elevation corresponds with lake water level).

Lake Michigan water samples, listed below also confirm that an aggressive environment does not exist.

Samples from 1962 to1966 of Lake Michigan water at or near the site and 1992 and 2004 site results:

Chemistry/Year	1962-1966	1992	2004
pH	Range 7.6 to 8.2 (>5.5)	8.2	7.9
Chlorides - ppm	Range 5.0 to 32.0 (<500)	11.5	12
Sulfates - ppm	Range 16.0 to 28.0 (<1500)	29	24.4

In addition, FSAR Chapter 2 - Table 2-12 - Analysis of Soil Samples shows that site soil pH (9 samples) ranged between 8.1 and 8.5, indicating that soil around and under site structures is also alkaline and does not present an aggressive environment for inaccessible concrete.

In summary, Palisades ground water sample measurements, taken over the course of many years and during varying seasons, confirm that parameters have remained consistent and are well below threshold limits that could cause concrete degradation (i.e., an aggressive environment does not exist). The rate of ground water flow is not considered an aggressive flow rate. Therefore, the conditions identified in NUREG-1801 as revised by ISG-03 are satisfied; and aging management of increased porosity and permeability, cracking, loss of material (spalling, scaling) due to aggressive chemical attack of below grade inaccessible concrete is not required.

In addition it is concluded that additional groundwater monitoring over the period of license extension is not necessary. Sampling to date has shown no variance in over 40 years and the chemical parameters are well below limits considered aggressive. It would require a significant environmental event to substantially affect the quality of groundwater in the vicinity of Palisades. A change in the environment due to a chemical release would be considered as an “abnormal event.” NUREG-1800, “Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants,” states that aging effects from abnormal events need not be postulated specifically for license renewal.

Expansion and Cracking due to Reaction with Aggregates (GALL IIA1.1-d)

Accessible Areas: In accordance with NUREG-1801, as revised by ISG-03, accessible reinforced concrete structures and components will be monitored by the Containment Inservice Inspection Program to manage expansion and cracking due to reaction with aggregates.

Inaccessible Areas: The aggregate used in the concrete of the Palisades components did not come from a region known to yield aggregates suspected of or known to cause aggregate reactions. Materials for concrete used in Palisades structures and components were specifically investigated, tested and examined in accordance with pertinent ASTM standards. All aggregates used at Palisades conform to the requirements of ASTM C33 "Standard Specification of Concrete Aggregates." Appendix XI of ASTM C33 identifies methods for evaluating potential reactivity of aggregates including ASTM C295, ASTM C289, ASTM C227, and ASTM C342. Palisades aggregates were tested to assure compliance with ASTM C-33 (FSAR 5.8.7.2): Petrographic Analysis - ASTM C-295, Potential Reactivity (Chemical) - ASTM C-289, and Potential Reactivity (Mortar Bar) - ASTM C-227. Palisades aggregates were not tested to ASTM C-342, presumably because this method does not provide reliable results as noted in ACI 201.2R-77 Chapter 5, Section 5.3.3. Low alkali Portland Cement (ASTM C150 Type II) was used in the concrete mixes used in all Palisades concrete structures, which mitigates harmful expansion due to alkali aggregate reaction.

Therefore, the conditions identified in NUREG-1801 as revised by ISG-03 are satisfied and aging management of expansion and cracking due to reaction with aggregates for below grade inaccessible concrete is not required.

Aggressive Groundwater Environment-Embedded Steel (GALL IIA1.1-e)

Inaccessible Areas: For the aging effects of cracking, loss of bond, loss of material (spalling, scaling) due to corrosion of embedded steel for concrete components, ISG-3 indicates that a plant-specific aging management program is required if the below-grade environment is aggressive (pH < 5.5, chlorides >500 ppm, or sulfates > 1500 ppm). Examination of representative samples of below-grade concrete, when excavated for any reason, is to be included as part of a plant-specific program. Note: Periodic monitoring of below-grade water chemistry (including consideration of potential seasonal variations) is an acceptable approach to demonstrate that the below-grade environment is aggressive or non-aggressive.

The groundwater environment that exists at Palisades has been reviewed in detail in the discussion for GALL IIA1.1-c above, and will not be repeated here.

Palisades groundwater water sample measurements, taken over the course of many years and during varying seasons, confirm that parameters have remained consistent and are well below threshold limits that could cause concrete degradation or corrosion of embedded steel (i.e., an aggressive environment does not exist). The rate of groundwater flow is not considered an aggressive flow rate. The conditions identified in NUREG-1801 as revised by ISG-03 are satisfied; therefore, aging management of cracking, loss of bond, loss of material (spalling, scaling) due to corrosion of embedded steel for below grade inaccessible concrete is not required.

In addition it is concluded that additional groundwater monitoring over the period of license extension is not necessary. Sampling to date has shown no variance in over 40 years and the chemical parameters are well below limits considered aggressive. It would require a significant environmental event to substantially affect the quality of groundwater in the vicinity of Palisades. A change in the environment due to a chemical release would be considered as an “abnormal event.” NUREG-1800, “Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants,” states that aging effects from abnormal events need not be postulated specifically for license renewal.

3.5.2.2.1.2 **Cracking, Distortion, and Increase in Component Stress Level due to Settlement; Reduction of Foundation Strength due to Erosion of Porous Concrete Subfoundations, if not Covered by Structures Monitoring Program**

NUREG 1800 states that cracking, distortion, and increase in component stress level due to settlement could occur in PWR concrete and steel containments. Also, reduction of foundation strength due to erosion of porous concrete subfoundations could occur in all types of PWR containments. Some plants may rely on a de-watering system to lower the site ground water level. If the plant’s CLB credits a de-watering system, NUREG-1801 recommends verification of the continued functionality of the de-watering system during the period of extended operation. NUREG-1801 recommends no further evaluation if this activity is included in the scope of the applicant's structures monitoring program.

Cracks; distortion; increase in component stress level Due to Settlement (GALL IIA1.1-f)

Concrete structures can be affected by differential settlement between supporting foundations, within the building, or between buildings. For buildings experiencing significant settlement, cracks on structural members may be visibly detected. Cracks, distortion, and an increase in component stress level due to settlement are not considered as aging effects requiring management for the Palisades Primary Containment Structure since it is founded on highly dense, compacted sand, that remained after removal of the sand dunes for site preparation. (FSAR Section 2.3.4, Engineering Geology). The Palisades Primary Containment Structure did not require a de-watering system to control settlement, since subsurface conditions of dense compact sand and compacted backfill do not require such a system.

For concrete structures founded on dense soil or backfill, if in the past 20 years of experience for a structure, the total differential settlement experienced are well within the permissible limits for this type of structure and no settlement has manifested itself via cracked walls or cracked foundations, then it can be concluded that cracking due to settlement is not significant, and would not be applicable for the structure during the period of extended operation. No settlement monitoring program for Palisades structures has been formally implemented. In support of NUREG 0820, Integrated Plant Safety Assessment, Systematic Evaluation Program, Palisades Plant,…Final Report October 1982, the NRC evaluation for topics II-4.D, Stability of Slopes and II-4.F, Settlement of Foundations and Buried Equipment, provided NRC staff conclusions on the site conditions. On the settlement issue, the NRC concluded that, "the settlement of foundations and buried equipment will not be a safety problem of concern."

Reduction in foundation strength, cracking, differential settlement due to erosion of porous concrete subfoundation (GALL IIA1.1-g)

The Palisades primary containment base mat rests directly on native soil (dense fine sand). There is no porous concrete sub-foundation below the base mat. Therefore, erosion of porous sub-foundation cement by ground water, a matter addressed in Section II A1 of NUREG 1801, is not an issue at the Palisades Plant. Palisades does not have a porous concrete foundation and has no subsurface drainage system, as was identified at other facilities in NRC Information Notice 97-11. In addition, natural groundwater movement in this area is from the plant site to Lake Michigan. The rate of groundwater flow estimated during site exploration and discussed in the FSAR Section 2.2.1, is 650 feet per year (0.074'/hr), which is not considered an aggressive flow rate.

3.5.2.2.1.3 Reduction of Strength and Modulus of Concrete Structures due to Elevated Temperature

NUREG 1800 states that reduction of strength and modulus of elasticity due to elevated temperatures could occur in PWR concrete and steel containments. NUREG-1801 recommends further evaluation if any portion of the concrete containment components exceeds specified temperature limits, i.e., general area temperature 66°C (150°F) and local area temperature 93°C (200°F).

Reduction of strength and modulus due to elevated temperature (>150°F general; >200°F local)-Containment Shell Operating Temperature Evaluation (GALL IIA1.1-h)

Subsection CC-3400 of ASME Section III, Division 2, specifies the concrete temperature limits for normal operation or any other long-term period. The temperatures shall not exceed 150°F except for local areas, such as around penetrations, which are not allowed to exceed 200°F. If significant equipment loads are supported by concrete at temperatures exceeding 150°F, an evaluation of the ability to withstand the postulated design loads is to be made.

Appendix A of ACI 349-85 specifies that for normal operation or any other long term period the concrete temperature shall not exceed 150 ºF except for local areas, which are allowed to have increased temperatures not to exceed 200 ºF. The Palisades containment shell reinforced concrete structures have general area temperatures less than 150ºF during normal operation. FSAR Table 9-13, shows a maximum ambient design basis temperature of 104°F.

Analyses have determined that, due to the increased ambient containment shell temperatures specified in the EEQ program, the maximum boundary temperature in the main steam penetration room near the main steam penetration increased from, 104°F to 110°F, and the reactor containment temperature near the main steam penetration increased from 104°F to 110°F. The analysis concluded the maximum temperature of the concrete at the main steam penetration increased from 143°F to <179°F, which is less than the limit imposed by ACI 349 of 200°F and provides adequate margin.

The main steam and feedwater penetrations through containment shown on FSAR Figure 5.8-2 are provided with 6" of insulation between the pipe and the concrete shell, and fills the conical steel penetration closure that is welded to the containment liner plate. FSAR Figure 5.8-18 shows plots of thermal gradients at the main steam penetration. It shows the 143°F gradient at the liner plate in the penetration opening, which, based on the above EEQ discussion, could be <179°F. Temperatures at this localized area are less than the 200°F that is allowed at localized areas.

This situation is judged to be acceptable since local temperature is less than 200°F, and reinforcing steel exists around the penetrations to distribute compressive stresses due to heating. (Reference FSAR Section 5.8.6.4.1)

It is, therefore, concluded that the conditions identified in GALL IIA1.1-h are satisfied and aging management of reduction of strength and modulus due to elevated temperature for containment shell concrete components is not required.

3.5.2.2.1.4 Loss of Material due to Corrosion in Inaccessible Areas of Steel Containment Shell or Liner Plate

NUREG 1800 states that loss of material due to corrosion could occur in inaccessible areas of the steel containment shell or the steel liner plate for all types of PWR containments. NUREG-1801 recommends further evaluation of plant-specific programs to manage this aging effect for inaccessible areas if specific criteria defined in NUREG-1801 cannot be satisfied.

NUREG-1801, Item IIA2.1-a, states that loss of material due to corrosion is not significant if four conditions are satisfied. Each condition, and a Palisades discussion for that condition, is itemized below:

1. Concrete meeting the requirements of ACI 318 or 349 and the guidance of 201.2R was used for the containment concrete in contact with the embedded containment shell or liner.

The Palisades containment structure was designed and constructed in accordance with ACI-318-63, ACI-301-72 (proposed) and the ASME Pressure Vessel Code, Sections III, VIII and IX, 1965. (Reference FSAR Section 5.1.6.2). Palisades' concrete, meeting the requirements of ACI 318 (and is consistent with the guidance of 201.2R-77), was used for the containment concrete in contact with the embedded containment shell or liner, as discussed in FSAR Section 5.8.2 and Section 5.8.7.1 These materials produced an excellent high strength, dense, sound concrete.

2. The concrete is monitored to ensure that it is free of penetrating cracks that provide a path for water seepage to the surface of the containment shell or liner.

The containment exterior concrete is monitored by the Palisades Containment Inservice Inspection Program to ensure that it is free of penetrating cracks that might provide a path for water seepage to the surface of the containment shell or liner. The containment interior 18" reinforced concrete floor placed over the containment bottom steel liner is monitored by the Palisades Structural Monitoring Program to ensure that penetrating cracks are not occurring.

3. The moisture barrier, at the junction where the shell or liner becomes embedded, is subject to aging management activities in accordance with IWE requirements.

The moisture barrier, at the junction where the shell or liner becomes embedded, is subject to aging management activities in accordance with Palisades Containment Inservice Inspection Program requirements.

4. Borated water spills and water ponding on the containment concrete floor are not common and when detected are cleaned up in a timely manner.

Borated water spills and water ponding on the containment concrete floor are not common, and when detected, are cleaned up in a timely manner, in accordance with the Palisades Boric Acid Corrosion Program.

It is, therefore, concluded that corrosion is not significant for inaccessible areas of the containment liner.

3.5.2.2.1.5 Loss of Prestress due to Relaxation, Shrinkage, Creep, and Elevated Temperature

NUREG 1800 states that loss of prestress forces due to relaxation, shrinkage, creep, and elevated temperature for PWR prestressed concrete containments is a TLAA as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c).

See Section 4.5 for a more detailed discussion of this issue.

The Containment Inservice Inspection Program includes the requirements for periodic containment tendon surveillance. The program predicts time-dependent lower limits of the lift-off force (predicted lower limits, PLL) for each tendon subgroup by regression analysis of individual tendon surveillance data. The PLL calculations are revised at each inspection, as required by the program.

The program maintains trend lines of the data for each tendon surveyed. The program inspects a sample of tendons from each group (dome, vertical, and hoop) in each inspection interval to confirm that the trend lines remain within the tolerances of the predicted lower limits, and therefore that tendon prestresses will remain above their respective minimum required values (MRV) for the succeeding inspection interval. The program provides appropriate actions if surveillance data indicate that a trend line may cross its MRV.

3.5.2.2.1.6 **Cumulative Fatigue Damage**

NUREG 1800 states that, if included in the current licensing basis, fatigue analyses of containment steel liner plates and steel containment shells (including welded joints) and penetrations (including penetration sleeves, dissimilar metal welds, and penetration bellows) for all types of PWR containments are TLAAs as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c).

See Section 4.6.1 and Section 4.6.2 for more detailed discussions of the fatigue in the containment liner and penetrations, respectively.

Liner

The Palisades containment design relies on the liner to maintain a leak-tight containment. However, There are no design conditions under which the liner plate is relied upon to assist the concrete in maintaining the integrity of the structure. Forces are transmitted between the liner plate and the concrete through the anchorage system and through direct contact (pressure). At times, forces may also be transmitted by bond and/or friction. These forces cause, or are caused by, liner plate strains. The liner plate is designed to withstand the predicted strains. The effect of concrete cracking on the liner plate has also been considered.

The 1/4 inch liner plate will yield when subjected to the effects of concrete creep and shrinkage, prestressing forces and high temperature. Because the liner plate shares deformation compatibility with the containment shell along the continuous angle anchors and because the shell is very stiff, these loads do not lead to a severe condition in the liner plate. The most significant contributor to what (if calculated) would have been the fatigue usage factor is a single end-of-design-life strain cycle due to a design basis accident.

The cyclical load events that contribute to fatigue in the containment liner during the licensed operating period result in only insignificant fatigue usage factors. No changes in operation are anticipated during the extended licensed operating period that would increase the frequency of the contributing events. The resulting fatigue usage factors in the containment liner will therefore remain insignificant for the 60-year extended licensed operating period.

Penetrations

FSAR Section 5.8.3 and Section 5.8.6.4 describe design of both the “large penetrations” (personnel air lock and equipment hatch) and the “small penetrations” (escape air lock, piping, ventilation, and electrical penetrations) to limiting strain criteria for a set of design loads, including thermal loads and design basis accident (DBA) loads.

Although the FSAR Section 5.8.6.4.1 description states (1) that “The allowable value of stress intensity Sa was determined from Figure N 415(a) of the referenced ASME Code article,” and (2) that “For all load combinations, the strains in the pipe penetrations did not exceed [those for which an equivalent Sa would exceed] the values given in the ASME B&PV Code, Figure N 415(a),” it appears that the only event for which the N-415(a) stress criterion was used was the end-of-life design basis event.

3.5.2.2.1.7 Cracking due to Cyclic Loading and SCC

NUREG 1800 states that cracking of containment penetrations (including penetration sleeves, penetration bellows, and dissimilar metal welds) due to cyclic loading or SCC could occur in all types of PWR and BWR containments. Cracking could also occur in vent line bellows, vent headers and downcomers due to SCC for BWR containments. A visual VT-3 examination would not detect such cracks. NUREG-1801 recommends further evaluation of the inspection methods implemented to detect these aging effects.

See Section 4.6.1 and Section 4.6.2 for more detailed discussions of the fatigue in the containment liner and penetrations, respectively.

No expansion bellows are required in the Palisades containment design. All piping and ventilation penetrations are of the rigid welded type and are solidly anchored to the containment shell, thus precluding any requirement for expansion bellows. (Reference FSAR Section 5.8.6.2.2) Stress concentrations around openings in the liner plate were calculated using the theory of elasticity. These stress concentrations were then reduced by thickening the liner plate around each penetration in accordance with the ASME B&PV Code, Section III, 1965.

Anchor bolts are provided as part of each penetration assembly. When the penetration assembly has no significant external loads, the anchors maintain the strain compatibility between the liner plate and the concrete. When significant loads are present, the anchors control the inward displacement of the liner plate. The stress level in the anchor bolts from external loads is in accordance with the AISC Code. (Reference FSAR Section 5.8.6.4.1)

Therefore, no further evaluation is required since these GALL Items are not applicable to Palisades' design.

3.5.2.2.2 **Class 1 Structures**

3.5.2.2.2.1 **'Aging of Structures not Covered by Structures Monitoring Program**

NUREG 1800 states that NUREG-1801 recommends further evaluation of certain structure/aging effect combinations if they are not covered by the structures monitoring program. This includes (1) scaling, cracking, and spalling due to repeated freeze-thaw for Groups 1-3, 5, 7-9 structures; (2) scaling, cracking, spalling and increase in porosity and permeability due to leaching of calcium hydroxide and aggressive chemical attack for Groups 1-5, 7-9 structures; (3) expansion and cracking due to reaction with aggregates for Groups 1-5, 7-9 structures; (4) cracking, spalling, loss of bond, and loss of material due to corrosion of embedded steel for Groups 1-5, 7-9 structures; (5) cracks, distortion, and increase in component stress level due to settlement for Groups 1-3, 5, 7-9 structures; (6) reduction of foundation strength due to erosion of porous concrete subfoundation for Groups 1-3, 5-9 structures; (7) loss of material due to corrosion of structural steel components for Groups 1-5, 7-8 structures; (8) loss of strength and modulus of concrete structures due to elevated temperatures for Groups 1-5; and (9) crack initiation and growth due to SCC and loss of material due to crevice corrosion of stainless steel liner for Groups 7 and 8 structures. Further evaluation is necessary only for structure/aging effect combinations not covered by the structures monitoring program.

Loss of Material (Spalling, Scaling) and Cracking due to Freeze - Thaw - (GALL IIIA3.1-a, III.A6.1-a, IIIA8.1-a)

Accessible Areas: In accordance with NUREG-1801 as clarified by ISG-03, accessible reinforced concrete structures and components will be monitored by the Palisades Structural Monitoring Program to manage Loss of Material (spalling, scaling) and cracking due to freeze - thaw.

Inaccessible Areas: Palisades is located in an area with severe weathering conditions as noted on Figure 1 of ASTM C33-99. Freeze-thaw is not considered an aging mechanism for concrete components below the frost line (depth of 42 inches in accordance with Michigan Building Code).

The Palisades concrete structures and concrete are designed in accordance with ACI 318-63 and -71, and constructed using ingredients conforming to ACI and ASTM standards. Palisades specifications require all concrete to contain an air-entraining agent in sufficient quantity to maintain specified percentages based on nominal maximum size aggregate. For severe weather exposures, the air content identified varies from 3 to 5 percent. Water/cement ratios, for concrete mixes, range from 0.45 to 0.46 for the 4000 psi concrete used in the primary containment internal concrete construction, and in other PAL structures. Water/cement ratio for 3000 psi concrete used in the original Auxiliary Building and other original structures as previously identified, is 0.47, slightly above the recommended range.

Review of Operating Experience and Maintenance Rule Structures Monitoring results provide confirmation that freeze-thaw degradation of Palisades concrete has not occurred.

As described in NUREG-1557 “Summary of Technical Information and Agreements from Nuclear Management and Resources Council [NUMARC] Industry Reports Addressing License Renewal,” freeze-thaw does not cause loss of material from reinforced concrete in foundations, and in above and below grade exterior concrete, for plants located in a geographic region of negligible weathering conditions (weathering index <100 day-inch/yr). Similarly, freeze-thaw damage is not significant for reinforced concrete in foundations, and in above and below grade exterior concrete, for plants located in areas in which weathering conditions are considered severe (weathering index >500 day-inch/yr) or moderate (100-500 day-inch/yr), provided that the concrete mix design meets the air content (entrained air 3-6%) and water-to-cement ratio (0.35-0.45) specified in ACI 318-63 or ACI 349-85. Palisades water-cement ratios range between 0.45 and 0.47 and ACI 201.2R-77 (Reference 7.22) Section 1.4.2 recommends a not to exceed water-cement ratio of 0.50 for other than “thin” structures. In addition, at Palisades, the ground tends to freeze and stay frozen through the winter, which means that below grade concrete only is exposed, essentially, to one freeze-thaw cycle in a typical year. Since damage to concrete from freeze-thaw is related to repeated cycles of freeze-thaw as noted in EPRI Structural Tools, Rev1, Section 5.3.1.1, and no evidence of freeze-thaw damage has been identified in exposed concrete which is subjected to repeated cycles, it is concluded that below grade concrete will not experience damage or degradation.

Therefore, since these conditions for Palisades concrete are essentially satisfied, aging management is not required for below grade concrete.

Increase in Porosity, Permeability, and Loss of Strength Due to Leaching -(GALL IIIA3.1-b, III.A6.1-b, IIIA8.1-b)

Accessible Areas: In accordance with NUREG-1801 as clarified by ISG-03, accessible reinforced concrete structures and components will be monitored by the Palisades Structural Monitoring Program to manage increase in porosity and permeability, loss of strength due to leaching of calcium hydroxide.

Inaccessible Areas: The Palisades concrete structures and concrete components are designed in accordance with ACI 318-63 and constructed using ingredients conforming to ACI and ASTM standards, which provide for a good quality, dense, well cured, and low permeability concrete. Cracking is controlled through proper arrangement and distribution of reinforcing bars.

Concrete structures and concrete components are constructed of a dense, well-cured concrete with an amount of cement suitable for strength development, and achievement of a water-to-cement ratio (Palisades water/cement ratio 0.45), which is characteristic of concrete having low permeability. This is consistent with the recommendations and guidance provided by ACI 201.2R-77.

In addition, concrete components must be exposed to flowing water through the concrete component for this to be a significant issue. Natural groundwater movement in this area is from the plant site to Lake Michigan. The rate of groundwater flow estimated during site exploration and shown in the FSAR, is 650 feet per year (0.074 feet/hr), which is not considered an aggressive flow rate. Groundwater elevation at the plant site is the same as Lake Michigan (elevation corresponds with lake water level).

Leaching of Calcium Hydroxide is readily noticeable as white deposits that remain on the concrete surface after a solution of water-free lime from the concrete and carbon dioxide from the air is absorbed and dries. The Palisades Structural Monitoring Program inspects concrete surfaces for signs of leaching. No significant signs of leaching have been documented during these inspection walkdowns.

Therefore, the conditions identified in NUREG-1801 as revised by ISG-03 are satisfied and aging management of increase in porosity and permeability, loss of strength due to leaching of calcium hydroxide for below grade inaccessible concrete is not required.

Increase in Porosity and Permeability, cracking, loss of material (spalling, scaling) due to Aggressive Chemical Attack (GALL III.A3.1-f, III.A4, III.A6.1-e, III.A8.1-e)

Accessible Areas: In accordance with NUREG-1801 as revised by ISG-03, accessible reinforced concrete structures and components will be monitored by the Structures Monitoring Program to manage increase in porosity and permeability, cracking, loss of material (spalling, scaling) due to aggressive chemical attack.

Inaccessible Areas: A plant-specific aging management program is required for below-grade exterior reinforced concrete (basemat, embedded walls) if the environment is aggressive (pH < 5.5, chlorides >500 ppm, or sulfates > 1500 ppm). Examination of representative samples of below-grade concrete, when excavated for any reason, is to be included as part of a plant-specific program. Note: Periodic monitoring of below-grade water chemistry (including consideration of potential seasonal variations) is an acceptable approach to demonstrate that the below-grade environment is aggressive or non-aggressive.

At Palisades, design and construction of reinforced concrete provides for dense, well cured, and low permeability concrete that provides acceptable degree of protection against exposure of below grade exterior reinforced concrete to an aggressive environment. Cracking of concrete is controlled through proper arrangement and distribution of reinforcing bars. Continued or frequent cyclic exposure to the following aggressive environments is necessary for aggressive chemicals to cause a significant increase in porosity and permeability, cracking, loss of material (spalling, scaling):

Acidic solutions with pH < 5.5

Chloride solutions >500 ppm

Sulfate solutions >1500 ppm

Since aggressive chemicals are contained at plant sites, system leakage is possible that could cause the embedded steel to be exposed to chemicals beyond these limits. However, leaks are not expected to continue for the extensive periods required for degradation, and repairs would be completed prior to loss of intended function. It is not likely that leaks inside the structure would get outside to cause an aggressive chemical attack on embedded concrete.

An aggressive environment may also occur where embedded steel is exposed to aggressive aqueous solutions such as groundwater or aggressive water flow. Palisades groundwater water sample measurements, summarized below, have confirmed that parameters are well below threshold limits that could cause concrete degradation or corrosion of embedded steel (i.e., an aggressive environment does not exist).

Palisades Groundwater Sampling from 1966 (18 Sampling Locations results), 1996 South Monitoring Well, and 2004 Well #14 at Palisades:

Chemistry/Year	1966 (18 locations)	1996	2004
pH	Range 6.1-7.7 (> 5.5)	No reading	7.0

Chemistry/Year	1966 (18 locations)	1996	2004
Chlorides - ppm	Range 4.0-39 (<500)	23	139
Sulfates - ppm	Range 9.47-33.17 (<1500)	15.2	11.5

Natural groundwater movement in this area is from the plant site to Lake Michigan. The rate of groundwater flow estimated during site exploration and described in the FSAR Section 2.2.1 Groundwater, is 650 feet per year (0.074 feet/hr), which is not considered an aggressive flow rate. Groundwater elevation at the plant site is the same as Lake Michigan (elevation corresponds with lake water level).

Lake Michigan water samples also confirm an aggressive environment does not exist.

Samples from 1962 to1966 of Lake Michigan water at or near the site and 1992 and 2004 site results:

Chemistry/Year	1962-1966	1992	2004
pH	Range 7.6 - 8.2 (>5.5)	8.2	7.9
Chlorides - ppm	Range 5.0 - 32.0 (<500)	11.5	12
Sulfates - ppm	Range 20.0 to 28.0 (<1500)	29	24.4

In addition, FSAR Chapter 2 - Table 2-12 - Analysis of Soil Samples shows that site soil pH of 9 samples ranged between 8.1 and 8.5, indicating that soil around and under site structures is also alkaline and does not present an aggressive environment for inaccessible concrete.

In summary, Palisades groundwater water sample measurements have confirmed that parameters are well below threshold limits that could cause concrete degradation (i.e., an aggressive environment does not exist). The rate of groundwater flow is not considered an aggressive flow rate. The conditions identified in NUREG-1801 as revised by ISG-03 are satisfied; therefore, aging management of increased porosity and permeability, cracking, and loss of material (spalling, scaling) due to aggressive chemical attack of below grade inaccessible concrete is not required.

It is also concluded that it is not necessary to monitor groundwater chemistry over the period of license extension, since it is not credible to postulate that some environmental event will occur in the future that would affect the quality of groundwater in the vicinity of Palisades. A change in the environment due to a chemical release would be considered as an “abnormal event.” NUREG-1800, “Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants,” states that aging effects from abnormal events need not be postulated specifically for license renewal.

Expansion and cracking due to reaction with aggregates - (GALL, IIIA3.1-c, IIIA4.1-b, IIIA6.1-c, IIIA8.1-c)

Accessible Areas: In accordance with NUREG-1801 as clarified by ISG-03, accessible reinforced concrete structures and components will be monitored by the Structural Monitoring Program to manage expansion and cracking due to reaction with aggregates.

Inaccessible Areas: The aggregate used in the concrete of the Palisades components did not come from a region known to yield aggregates suspected of or known to cause aggregate reactions. Materials for concrete used in Palisades structures and components were specifically investigated, tested and examined in accordance with pertinent ASTM standards. All aggregates used at Palisades conform to the requirements of ASTM C33 “Standard Specification of Concrete Aggregates.” Appendix XI of ASTM C33 identifies methods for evaluating potential reactivity of aggregates including ASTM C295, ASTM C289, ASTM C227, and ASTM C342. Low alkali Portland Cement (ASTM C150 Type II) was used in the concrete mixes used in all PAL concrete structures, which mitigates harmful expansion due to alkali aggregate reaction. (FSAR Section 5.8.2)

Therefore, the conditions identified in NUREG-1801 as revised by ISG-03 are satisfied and aging management of expansion and cracking due to reaction with aggregates for below grade inaccessible concrete is not required.

Cracking, Spalling, Loss of Bond, and Loss of Material Due to Corrosion of Embedded steel for Groups 1-5, 7-9 Structures

Accessible Areas: In accordance with NUREG-1801 as revised by ISG-03, accessible reinforced concrete structures and components will be monitored by the Structures Monitoring Program to manage cracking, loss of bond, and loss of material (spalling, scaling) due to corrosion of embedded steel.

Inaccessible Areas: A plant-specific aging management program is required if the below-grade environment is aggressive (pH < 5.5, chlorides >500 ppm, or sulfates > 1500 ppm). Examination of representative samples of below-grade concrete, when excavated for any reason, is to be included as part of a plant-specific program. Note: Periodic monitoring of below-grade water chemistry (including consideration of potential seasonal variations) is an acceptable approach to demonstrate that the below-grade environment is aggressive or non-aggressive.

At Palisades, design and construction of reinforced concrete provides for dense, well cured, and low permeability concrete that provides acceptable degree of protection against exposure of the embedded steel (including, but not limited to, reinforcing steel, unistrut, nelson studs, anchorages consisting of steel shapes/plate, tendon sheaths, conduit, and steel pipes/drains), to an aggressive environment. Cracking of concrete is controlled through proper arrangement and distribution of reinforcing bars. Continued or frequent cyclic exposure to the following aggressive environments is necessary for aggressive chemicals to cause significant corrosion of embedded steel:

Acidic solutions with pH < 5.5

Chloride solutions >500 ppm

Sulfate solutions >1500 ppm

Since aggressive chemicals are contained at plant sites, system leakage is possible that could cause the embedded steel to be exposed to chemicals beyond these limits. However, leaks are not expected to continue for the extensive periods required for degradation, and repairs would be completed prior to loss of intended function.

An aggressive environment may also occur where embedded steel is exposed to aggressive aqueous solutions such as groundwater or aggressive water flow. PAL groundwater water sample measurements have confirmed that parameters are well below threshold limits that could cause concrete degradation or corrosion of embedded steel (an aggressive environment does not exist).

Palisades Groundwater Sampling from 1966 (18 Sampling Locations results), 1996 South Monitoring Well, and 2004 Well #14 at Palisades:

Chemistry/Year	1966	1996	2004
pH	Range 6.1-7.7 (> 5.5)	No reading	7.0
Chlorides - ppm	Range 4.0-39 (<500)	23	139
Sulfates - ppm	Range 9.47-33.17 (<1500)	15.2	11.5

Natural groundwater movement in this area is from the plant site to Lake Michigan. The rate of groundwater flow estimated during site exploration and described in the FSAR Section 2.2.1 Groundwater, is 650 feet per year (0.074 feet/hr), which is not considered an aggressive flow rate. Groundwater elevation at the plant site is the same as Lake Michigan (elevation corresponds with lake water level).

Lake Michigan water samples also confirm an aggressive environment does not exist.

Samples from 1962 to1966 of Lake Michigan water at or near the site and 1992 and 2004 site results:

Chemistry/Year	1962-1966	1992	2004
pH	Range 7.6 - 8.2 (>5.5)	8.2	7.9
Chlorides - ppm	Range 5.0 - 32.0 (<500)	11.5	12
Sulfates - ppm	Range 20.0 to 28.0 (<1500)	29	24.4

In addition, FSAR Chapter 2 - Table 2-12 - Analysis of Soil Samples shows that site soil pH of 9 samples ranged between 8.1 and 8.5, indicating that soil around and under site structures is also alkaline and does not present an aggressive environment for inaccessible concrete.

In summary, Palisades groundwater water sample measurements have confirmed that parameters are well below threshold limits that could cause concrete degradation or corrosion of embedded steel (an aggressive environment does not exist). The rate of groundwater flow is not considered an aggressive flow rate. The conditions identified in NUREG-1801 as revised by ISG-03 are satisfied; therefore, aging management of cracking, loss of bond, and loss of material (spalling, scaling) due to corrosion of embedded steel for below grade inaccessible concrete is not required.

It is also concluded that it is not necessary to monitor groundwater chemistry over the period of license extension, since it is not credible to postulate that some environmental event will occur in the future that would affect the quality of groundwater in the vicinity of Palisades. A change in the environment due to a chemical release would be considered as an “abnormal event.” NUREG-1800, “Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants,” states that aging effects from abnormal events need not be postulated specifically for license renewal.

Cracks, distortion, and increase in component stress level due to settlement for Groups 1-3, 5, 7-9 structures

Cracks, distortion, and an increase in component stress level due to settlement are not considered as aging effects requiring management for Palisades structures founded highly dense, compacted sand and/or compacted engineered fill. Palisades Structures Monitoring Program inspections include monitoring of cracking associated aging effects, irrespective of aging mechanisms.

Concrete structures can be affected by differential settlement between supporting foundations, within the building, or between buildings. For buildings experiencing significant settlement, cracks on structural members may be visibly detected. Cracks, distortion, and an increase in component stress level due to settlement are not considered as aging effects requiring management for Palisades structural concrete commodities since they are founded on highly dense, compacted sand, that remained after removal of the sand dunes for site preparation or compacted backfill.

None of the Palisades concrete structures require a de-watering system to control settlement, since subsurface conditions of dense compact sand and compacted backfill do not require such a system.

In summary, for concrete structures founded on dense soil or backfill, if in the past 20 years of experience for a structure, the total differential settlement experienced are well within the permissible limits for this type of structure and no settlement has manifested itself via cracked walls or cracked foundations, then it can be concluded that cracking due to settlement is not significant, and would not be applicable for the structure during the period of extended operation. No settlement monitoring program for PAL structures was formally implemented.

NUREG 0820, Integrated Plant Safety Assessment, Systematic Evaluation Program, Palisades Plant,…Final Report October 1982, Letter (NRC) LS05-81-04-020 for Docket No. 50-255; RE: SEP TOPICS II-4.D, STABILITY OF SLOPES AND II-4.F, SETTLEMENT OF FOUNDATIONS AND BURIED EQUIPMENT, provides results of staff inspections and conclusions on the site conditions. On the settlement issue, it was concluded that, “the settlement of foundations and buried equipment will not be a safety problem of concern.”

In conclusion, monitoring of accessible concrete for cracks and distortions to settlement will be performed in accordance with the Palisades Structures Monitoring Program. Inspection of inaccessible concrete is not required.

Reduction of foundation strength due to erosion of porous concrete subfoundation for Groups 1-3, 5-9 structures

The Palisades concrete structures are on native soil (dense fine sand) and/or engineered fill. There are no porous concrete sub-foundations below the building foundations. Therefore, erosion of porous sub-foundation cement by ground water is not an issue at the Palisades Plant. Palisades does not have a porous concrete foundation and no subsurface drainage system, as was identified at other facilities in NRC Information Notice 97-11. In addition, natural groundwater movement in this area is from the plant site to Lake Michigan. The rate of groundwater flow estimated during site exploration and shown in the FSAR, is 650 feet per year (0.074 feet/hr), which is not considered an aggressive flow rate.

Loss of material due to corrosion of structural steel components for Groups 1-5, 7-8 structures

Structural Steel Commodities potentially subject to loss of material due to corrosion (crevice, pitting, general, MIC) credit the Structural Monitoring Program for aging management.

Loss of strength and modulus of concrete structures due to elevated temperatures for Groups 1-5

For any concrete elements that exceed specified temperature limits, further evaluations are warranted. Appendix A of ACI 349-85 specifies the concrete temperature limits for normal operation or any other long-term period. The temperatures shall not exceed 150°F except for local areas that are allowed to have increased temperatures not to exceed 200°F.

General Description Of Auxiliary Building Concrete And Temperatures (GALL IIIA3.1-j)

Group 3 structures include Auxiliary, Diesel Generator, Radwaste, and Turbine Buildings: Switchgear Room, AFW Pumphouse, Utility/Piping Tunnels. The Water Treatment and Water Purity Buildings are also included in this GALL Group.

The Palisades auxiliary building houses:

- Control room

- Emergency diesel generators and related auxiliaries
- New and spent fuel handling, storage and shipment facilities
- Radwaste, chemical and volume control equipment
- Safety Injection System (majority)
- Component Cooling System (majority)
- Containment Spray System (majority)

None of the areas in the auxiliary building(s) have normal operating temperatures greater than 150°F. The bounding temperature in the Component Cooling Room 338 is 120°F, with other areas at or less than 104°F. The spent fuel pool normal operating temperature is 125°F (Ref. FSAR 5.9.3.3).

Therefore, no further evaluation is required for the Auxiliary Building.

General Description Of Containment Internal Concrete And Temperatures (GALL IIIA4.1-c)

The principal interior concrete structures and their temperature exposure are:

- The primary shield wall, which forms the reactor cavity is exposed to the highest temperature, 200°F at the cavity liner (further evaluation applies)
- Two steam generator compartments/less than 150°F
- A refueling pool which is located between the steam generator compartments and above the reactor cavity/less than 150°F
- An enclosed sump under the reactor cavity/less than 150°F
- Major equipment supports including the steam generator pedestals/less than 150°F
- The containment floor slab/ less than 150°F

The primary shield wall (bioshield) is essentially a circular cylinder, lined with ¼” steel plate, with concrete ranging in thickness from 7' to 8', with the inner 10" thickness acting as a sacrificial shield. The sacrificial shield is not reinforced, except for three horizontal “bands/hoops” of reinforcing steel and is considered non-structural, non-load bearing concrete as evidenced by the existence of the plywood construction form 10" in from the liner (ref dwg C-153). The primary shield wall cooling coils are in the sacrificial concrete, 3" in from the ¼” liner plate, with cooling coils looping radially outward to provide temperature control of a maximum of 165ºF for the outer region of reinforced concrete. Openings in the shield wall for the primary coolant pipelines are lined with ¼” steel plate, with the space between the opening and the piping filled with non-structural concrete block for shielding.

The Shield Cooling System is designed to remove heat from the biological shield surrounding the reactor vessel thereby limiting the thermal stresses in the structural concrete. The system is designed to maintain structural concrete temperature below 165°F. The system is to assure that the concrete in the reactor cavity does not overheat and develop excessive thermal stress. FSAR Figures 9-5 and 9-6 show the temperature gradient through the shield wall and the temperature 10" in (sacrificial concrete) is less than 180ºF and drops to 140ºF at 80".

This is judged to be acceptable due to the fact that reductions in excess of 10% in the compressive strength, tensile strength, and the modulus of elasticity only begins to occur in the range of 180ºF to 200ºF (Reference EPRI TR-103842 - Class I Structures License Renewal Industry Report; Revision 1 - Section 4.1.6). This is further supported by the fact that 180ºF is also the value that the original Bechtel Palisades Plant design criteria states the structural concrete is not to exceed (FSAR Section 9.0). The temperature in the structural/load bearing portion of the bioshield is less than 180ºF per FSAR figures 9.5 and 9.6. In addition, concrete mixes are designed in accordance with ACI 613 with 15% more compressive strength than the required design strength. (Specification No. 5935-C-30 for Furnishing and Delivery of Concrete).

Therefore, the conditions identified in NUREG-1801 are satisfied and aging management of reduction of strength and modulus due to elevated temperature for containment interior concrete components is not required.

Crack initiation and growth due to SCC and loss of material due to crevice corrosion of stainless steel liner for Groups 7 and 8 structures.

To foster SCC, a temperature threshold of 140 degrees F or a continuous temperature environment of 200 degrees F is necessary. The maximum temperature in the Auxiliary Building or Containment Building for fuel-related components is 125 degrees F. The normal temperature for the Spent Fuel Pool is maintained below 125 degrees F, and maximum allowed temperature are not to exceed 140 degrees F. Therefore, the temperature threshold to foster stress corrosion cracking / IGA is not present, and no aging management is required.

3.5.2.2.2.2 **Aging Management of Inaccessible Areas**

NUREG 1800 states that cracking, spalling, and increases in porosity and permeability due to aggressive chemical attack and cracking, spalling, loss of bond, and loss of material due to corrosion of embedded steel could occur in below-grade inaccessible concrete areas. NUREG-1801 recommends further evaluation to manage these aging effects in inaccessible areas of Groups 1-3, 5, 7-9 structures, if specific criteria defined in NUREG-1801 cannot be satisfied.

Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling) due to Aggressive Chemical Attack (GALL IIIA3.1-f, IIIA4-N/A, III.A6.1-e, IIIA8.1-e)

Accessible Areas: In accordance with NUREG-1801 as clarified by ISG-03, accessible reinforced concrete structures and components will be monitored by the Structural Monitoring Program to manage Increase in porosity and permeability, cracking, loss of material (spalling, scaling) due to aggressive chemical attack.

Inaccessible Areas: NUREG-1801 states that a plant-specific aging management program is required for below-grade exterior reinforced concrete (basemat, embedded walls) if the environment is aggressive (pH < 5.5, chlorides >500 ppm, or sulfates > 1500 ppm). Examination of representative samples of below-grade concrete, when excavated for any reason, is to be included as part of a plant-specific program. Note: Periodic monitoring of below-grade water chemistry (including consideration of potential seasonal variations) is an acceptable approach to demonstrate that the below-grade environment is aggressive or non-aggressive.

At Palisades, the design and construction of reinforced concrete provides for dense, well cured, and low permeability concrete that provides an acceptable degree of protection against exposure of below-grade exterior reinforced concrete to an aggressive environment. Cracking of concrete is controlled through proper arrangement and distribution of reinforcing bars. Continued or frequent cyclic exposure to the following aggressive environments is necessary for aggressive chemicals to cause a significant increase in porosity and permeability, cracking, loss of material (spalling, scaling):

Acidic solutions with pH < 5.5

Chloride solutions >500 ppm

Sulfate solutions >1500 ppm

Since aggressive chemicals are contained at plant sites, system leakage is possible that could cause the reinforced concrete to be exposed to chemicals beyond these limits. However, leaks are not expected to continue for the extensive periods required for degradation, and repairs would be completed prior to loss of intended function. It is not likely that leaks inside the structure would get outside to cause an aggressive chemical attack on embedded concrete.

An aggressive environment may also occur when reinforced concrete is exposed to aggressive aqueous solutions such as groundwater or aggressive water flow. Palisades groundwater water sample measurements, summarized below, have confirmed that parameters are well below threshold limits that could cause concrete degradation or corrosion of embedded steel (i.e., an aggressive environment does not exist).

Palisades Groundwater Sampling Results from 1966, 1996, and 2004:

Chemistry/Year	1966 (18 Locations)	1996	2004
pH	Range 6.1-7.7 (> 5.5)	No reading	7.0
Chlorides - ppm	Range 4.0-39 (<500)	23	139
Sulfates - ppm	Range 9.47-33.17 (<1500)	15.2	11.5

Natural groundwater movement in this area is from the plant site to Lake Michigan. The rate of groundwater flow estimated during site exploration and described in the FSAR Section 2.2.1 Groundwater, is 650 feet per year (0.074 feet/hr), which is not considered an aggressive flow rate. Groundwater elevation at the plant site is the same as Lake Michigan (elevation corresponds with lake water level).

Lake Michigan water samples, listed below also confirm an aggressive environment does not exist.

Lake Michigan Water Sampling Results from 1962 to 1966, 1992, 2004:

Chemistry/Year	1962-1966	1992	2004
pH	Range 7.6 - 8.2 (>5.5)	8.2	7.9
Chlorides - ppm	Range 5.0 - 32.0 (<500)	11.5	12
Sulfates - ppm	Range 20.0 to 28.0 (<1500)	29	24.4

In addition, FSAR Table 2-12 - Analysis of Soil Samples, shows that site soil pH of nine samples ranged between 8.1 and 8.5, indicating that soil around and under site structures is also alkaline and does not present an aggressive environment for inaccessible concrete.

It is, therefore, concluded that Palisades groundwater water sample measurements have confirmed that parameters are well below threshold limits that could cause concrete degradation (an aggressive environment does not exist). The rate of groundwater flow is not considered an aggressive flow rate. The conditions identified in NUREG-1801 as revised by ISG-03 are satisfied; therefore, aging management of increased porosity and permeability, cracking, and loss of material (spalling, scaling) due to aggressive chemical attack of below grade inaccessible concrete is not required.

It is also concluded that formal groundwater monitoring over the period of license extension is unnecessary, since it is not credible to postulate that some environmental event will occur in the future that would substantially affect the quality of groundwater in the vicinity of Palisades. A change in the environment due to a chemical release would be considered as an “abnormal event.” NUREG-1800, “Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants,” states that aging effects from abnormal events need not be postulated specifically for license renewal.

Cracking, Loss of Bond, Loss of Material (Spalling, Scaling)/Corrosion of Embedded Steel (GALL IIIA3.1-e, IIIA4.1-d, III.A6.1-d, IIIA8.1-d)

Accessible Areas: In accordance with NUREG-1801 as clarified by ISG-03, accessible reinforced concrete structures and components will be monitored by the Structural Monitoring Program to manage cracking, loss of bond, loss of material (spalling, scaling)/corrosion of embedded steel.

Inaccessible Areas: NUREG-1801 states that a plant-specific aging management program is required if the below-grade environment is aggressive (pH < 5.5, chlorides >500 ppm, or sulfates > 1500 ppm). Examination of representative samples of below-grade concrete, when excavated for any reason, is to be included as part of a plant-specific program. Note: Periodic monitoring of below-grade water chemistry (including consideration of potential seasonal variations) is an acceptable approach to demonstrate that the below-grade environment is aggressive or non-aggressive.

The groundwater environment that exists at Palisades has been reviewed in detail in the discussion for GALL IIA1.1-c, and will not be repeated here.

It is concluded that Palisades groundwater water sample measurements confirm that chemistry parameters are well below threshold limits that could cause concrete degradation or corrosion of embedded steel (i.e., an aggressive environment does not exist). The rate of groundwater flow is not considered an aggressive flow rate. The conditions identified in NUREG-1801 as revised by ISG-03 are satisfied; therefore, aging management of cracking, loss of bond, and loss of material (spalling, scaling) due to corrosion of embedded steel for below grade inaccessible concrete is not required.

It is also concluded that formal groundwater monitoring over the period of license extension is unnecessary, since it is not credible to postulate that some environmental event will occur in the future that would affect the quality of groundwater in the vicinity of Palisades. A change in the environment due to a chemical release would be considered as an "abnormal event." NUREG-1800, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," states that aging effects from abnormal events need not be postulated specifically for license renewal.

3.5.2.2.3 Component Supports

3.5.2.2.3.1 Aging of Supports not Covered by Structures Monitoring Program

NUREG-1801 recommends further evaluation of certain component support/aging effect combinations if they are not covered by the structures monitoring program. This includes (1) reduction in concrete anchor capacity due to degradation of the surrounding concrete, for Groups B1-B5 supports; (2) loss of material due to environmental corrosion, for Groups B2-B5 supports; and (3) reduction/loss of isolation function due to degradation of vibration isolation elements, for Group B4 supports. Further evaluation is necessary only for structure/aging effect combinations not covered by the structures monitoring program.

NUREG-1801 Items III.B1.1.4-a, III.B1.2.3-a, III.B2.2-a, III.B3.2-a, III.B4.3-a, and III.B5.2-a discuss Aging Effect/Mechanism: Reduction in Anchor Bolt Capacity due to Local Concrete Degradation / Service Induced Cracking. NUREG-1801 requires further evaluation of this aging effect and mechanism if the Structures Monitoring Program does not have this aging effect and mechanism in-scope. The Palisades Structural Monitoring Program includes Local Concrete Degradation / Service Induced Cracking in-scope. Therefore no further evaluation is required.

NUREG-1801 Items III.B2.1-a, III.B3.1-a, III.B4.1-a, and III.B5.1-a discuss Aging Effect/Mechanism: Loss of Material due to Environmental Corrosion (i.e., pitting corrosion, general corrosion, etc.). NUREG-1801 requires further evaluation of this aging effect and mechanism if the Structures Monitoring Program does not have this aging effect and mechanism in-scope. The Palisades Structural Monitoring Program includes Loss of Material due to Environmental Corrosion (i.e., pitting corrosion, general corrosion, etc.) in-scope. Therefore no further evaluation is required.

NUREG-1801 Item III.B4.2-a discusses Vibration Isolation Elements and Aging Effect/Mechanism: Reduction or Loss of Isolation Function due Radiation Hardening, Temperature, Humidity, Sustained Vibratory Loading. The Palisades Structural Monitoring Program is credited with age managing vibration isolation elements for the Emergency Diesel Generator. Therefore no further evaluation is required.

3.5.2.2.3.2 Cumulative Fatigue Damage due to Cyclic Loading

NUREG-1800 states that fatigue of component support members, anchor bolts, and welds for Groups B1.1, B1.2, and B1.3 component supports is a TLAA as defined in 10 CFR 54.3 only if a CLB fatigue analysis exists. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c).

NUREG-1801 Items III.B1.1.1-c, and III.B1.2.1-c discuss Aging Effect/Mechanism: Cumulative Fatigue Damage due to Fatigue for ASME Class 1, 2, and 3 Component Supports. A TLAA is required if, as part of the current CLB, a fatigue analysis is performed for these supports. Palisades does not perform fatigue analysis for ASME Class 1, 2, or 3 Component Supports (i.e. support members, welds, bolted connections, and support anchorage to building structure). Therefore this item is not a TLAA at Palisades.

Note that certain Class 1 components have fatigue analyses that include their integral supports. These are addressed in Section 4.3 of the LRA.

3.5.2.2.4 Quality Assurance for Aging Management of Non-Safety Related Components

Quality Assurance Program applicability to non-safety-related components is addressed in Appendix B, Section 1.2.

3.5.2.3 Time-Limited Aging Analysis

The time-limited aging analyses (TLAA) identified below are associated with the Containments, Structures, and Component Supports components. The section of the LRA that contains the TLAA review results is indicated in parenthesis.

- Fatigue (Section 4.6, Containment Liner Plate and Penetrations Load Cycles)
- Fatigue (Section 4.7.1, Crane Load Cycles)
- Loss of Preload (Section 4.5, Concrete Containment Tendon Prestress Analysis)

3.5.3 Conclusion

The Containments, Structures, and Component Supports components that are subject to aging management review have been identified in accordance with the requirements of 10 CFR 54.4. The aging management programs selected to manage aging effects for the Containments, Structures, and Component Supports components are identified in the summaries in Section 3.5.2.1 above.

A description of these aging management programs is provided in Appendix B, along with the demonstration that the identified aging effects will be managed for the period of extended operation.

Therefore, based on the demonstrations provided in Appendix B, the effects of aging associated with the Containments, Structures, and Component Supports components will be adequately managed so that there is reasonable assurance that the intended function(s) will be maintained consistent with the current licensing basis during the period of extended operation.

Table 3.5.1 Summary of Aging Management Evaluations in Chapters II and III of NUREG-1801 for Structures and Component Supports

Item Number	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Common Components of All Types of PWR and BWR Containment					
3.5.1-01	Penetration sleeves, penetration bellows, and dissimilar metal welds	Cumulative fatigue damage (CLB fatigue analysis exists)	TLAA evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA (see [SRP] subsection 3.5.2.2.1.6)	Further evaluation documented in Section 3.5.2.2.1.6. The aging mechanism fatigue is potentially applicable to many component types in the Structures and Component Supports Supergroup, but only selected components or locations require explicit analysis as TLAAs and/or warrant aging management for fatigue. The Palisades evaluation of these components is discussed in Section 4.6. Therefore, cumulative fatigue damage is not identified as an aging effect in Tables 3.5.2-1 through 3.5.2-10.below.
3.5.1-02	Penetration sleeves, bellows, and dissimilar metal welds.	Cracking due to cyclic loading, or crack initiation and growth due to SCC	Containment ISI and Containment leak rate test	Yes, detection of aging effects is to be evaluated (see [SRP] subsection 3.5.2.2.1.7)	Further evaluation documented in Section 3.5.2.2.1.7.
3.5.1-03	Penetration sleeves, penetration bellows, and dissimilar metal welds	Loss of material due to corrosion	Containment ISI and Containment leak rate test	No	
3.5.1-04	Personnel airlock and equipment hatch	Loss of material due to corrosion	Containment ISI and Containment leak rate test	No	
3.5.1-05	Personnel airlock and equipment hatch	Loss of leak tightness in closed position due to mechanical wear of locks, hinges and closure mechanism	Containment leak rate test and Plant Technical Specifications	No	

Table 3.5.1 Summary of Aging Management Evaluations in Chapters II and III of NUREG-1801 for Structures and Component Supports

Item Number	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.5.1-06	Seals, gaskets, and moisture barriers	Loss of sealant and leakage through containment due to deterioration of joint seals, gaskets, and moisture barriers	Containment ISI and Containment leak rate test	No	
PWR Concrete (Reinforced and Prestressed) and Steel Containment BWR Concrete (Mark II and III) and Steel (Mark I, II, and III) Containment					
3.5.1-07	Concrete elements: foundation, walls, dome.	Aging of accessible and inaccessible concrete areas due to leaching of calcium hydroxide, aggressive chemical attack, and corrosion of embedded steel	Containment ISI	Yes, if aging mechanism is significant for inaccessible areas (see [SRP] subsection 3.5.2.2.1.1)	Further evaluation documented in Section 3.5.2.2.1.1.
3.5.1-08	Concrete elements: foundation	Cracks, distortion, and increases in component stress level due to settlement	Structures Monitoring	No, if within the scope of the applicant's structures monitoring program (see [SRP] subsection 3.5.2.2.1.2)	Further evaluation documented in Section 3.5.2.2.1.2.
3.5.1-09	Concrete elements: foundation	Reduction in foundation strength due to erosion of porous concrete subfoundation	Structures Monitoring	No, if within the scope of the applicant's structures monitoring program (see [SRP] subsection 3.5.2.2.1.2)	Further evaluation documented in Section 3.5.2.2.1.2.

Table 3.5.1 Summary of Aging Management Evaluations in Chapters II and III of NUREG-1801 for Structures and Component Supports

Item Number	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.5.1-10	Concrete elements: foundation, dome, and wall	Reduction of strength and modulus due to elevated temperature	Plant specific	Yes, for any portions of concrete containment that exceed specified temperature limits (see [SRP] subsection 3.5.2.2.1.3)	Further evaluation documented in Section 3.5.2.2.1.3.
3.5.1-11	Prestressed containment: tendons and anchorage components	Loss of prestress due to relaxation, shrinkage, creep, and elevated temperature	TLAA evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA (see [SRP] subsection 3.5.2.2.1.5)	Further evaluation documented in Section 3.5.2.2.1.5.
3.5.1-12	Steel elements: liner plate, containment shell	Loss of material due to corrosion in accessible and inaccessible areas	Containment ISI and Containment leak rate test	Yes, if corrosion is significant for inaccessible areas (see [SRP] subsection 3.5.2.2.1.4)	Further evaluation documented in Section 3.5.2.2.1.4.
3.5.1-13	BWR only				
3.5.1-14	Steel elements: protected by coating	Loss of material due to corrosion in accessible areas only	Protective coating monitoring and maintenance	No	
3.5.1-15	Prestressed containment: tendons and anchorage components	Loss of material due to corrosion of prestressing tendons and anchorage components	Containment ISI	No	

Table 3.5.1 Summary of Aging Management Evaluations in Chapters II and III of NUREG-1801 for Structures and Component Supports

Item Number	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.5.1-16	Concrete elements: foundation, dome, and wall	Scaling, cracking, and spalling due to freeze-thaw; expansion and cracking due to reaction with aggregate	Containment ISI	No	
3.5.1-17	BWR only				
3.5.1-18	BWR only				
3.5.1-19	BWR only				
Class 1 Structures					

Table 3.5.1 Summary of Aging Management Evaluations in Chapters II and III of NUREG-1801 for Structures and Component Supports

Item Number	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.5.1-20	All Groups except Group 6: accessible interior/exterior concrete & steel components	All types of aging effects	Structures Monitoring	No, if within the scope of the applicant's structures monitoring program (see [SRP] subsection 3.5.2.2.2.1) ISG-3 has clarified this wording as follows: No, if within the scope of the applicant's structures monitoring program and a plant-specific aging management program is required for inaccessible areas as stated (see subsection 3.5.2.2.2.1)	Further evaluation documented in Section 3.5.2.2.2.1.

Table 3.5.1 Summary of Aging Management Evaluations in Chapters II and III of NUREG-1801 for Structures and Component Supports

Item Number	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.5.1-21	Groups 1-3, 5, 7-9: inaccessible concrete components, such as exterior walls below grade and foundation	Aging of inaccessible concrete areas due to aggressive chemical attack, and corrosion of embedded steel	Plant-specific	Yes, if an aggressive below-grade environment exists (see [SRP] subsection 3.5.2.2.2.2) ISG-3 has clarified this wording as follows: Yes, a plant-specific aging management program is required for inaccessible areas as stated (see subsection 3.5.2.2.2.2)	Further evaluation documented in Section 3.5.2.2.2.2.
3.5.1-22	Group 6: all accessible/ inaccessible concrete, steel, and earthen components	All types of aging effects, including loss of material due to abrasion, cavitation, and corrosion	Inspection of Water-Control Structures or FERC/US Army Corps of Engineers dam inspections and maintenance	No	
3.5.1-23	Group 5: liners	Crack initiation and growth from SCC and loss of material due to crevice corrosion	Water Chemistry Program and Monitoring of spent fuel pool water level	No	

Table 3.5.1 Summary of Aging Management Evaluations in Chapters II and III of NUREG-1801 for Structures and Component Supports

Item Number	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.5.1-24	Groups 1-3, 5, 6: all masonry block walls	Cracking due to restraint, shrinkage, creep, and aggressive environment	Masonry Wall	No	
3.5.1-25	Groups 1-3, 5, 7-9: foundation	Cracks, distortion, and increases in component stress level due to settlement	Structures Monitoring	No, if within the scope of the applicant's structures monitoring program (see [SRP] subsection 3.5.2.2.1.2)	Further evaluation documented in Section 3.5.2.2.1.2.
3.5.1-26	Groups 1-3, 5-9: foundation	Reduction in foundation strength due to erosion of porous concrete subfoundation	Structures Monitoring	No, if within the scope of the applicant's structures monitoring program (see [SRP] subsection 3.5.2.2.1.2)	Further evaluation documented in Section 3.5.2.2.1.2.
3.5.1-27	Groups 1-5: concrete	Reduction of strength and modulus due to elevated temperature	Plant-specific	Yes, for any portions of concrete that exceed specified temperature limits (see [SRP] subsection 3.5.2.2.1.3)	Further evaluation documented in Section 3.5.2.2.1.3.
3.5.1-28	Groups 7, 8: liners	Crack Initiation and growth due to SCC; Loss of material due to crevice corrosion	Plant-specific	Yes	

Table 3.5.1 Summary of Aging Management Evaluations in Chapters II and III of NUREG-1801 for Structures and Component Supports

Item Number	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
Component Supports					
3.5.1-29	All Groups: support members: anchor bolts, concrete surrounding anchor bolts, welds, grout pad, bolted connections, etc.	Aging of component supports	Structures Monitoring	No, if within the scope of the applicant's structures monitoring program (see [SRP] subsection 3.5.2.2.3.1)	Further evaluation documented in Section 3.5.2.2.3.1.
3.5.1-30	Groups B1.1, B1.2, and B1.3: support members: anchor bolts, welds	Cumulative fatigue damage (CLB fatigue analysis exists)	TLAA evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA (see [SRP] subsection 3.5.2.2.3.2)	Further evaluation documented in Section 3.5.2.2.3.2.
3.5.1-31	All Groups: support members: anchor bolts, welds	Loss of material due to boric acid corrosion	Boric acid corrosion	No	
3.5.1-32	Groups B1.1, B1.2, and B1.3: support members: anchor bolts, welds, spring hangers, guides, stops, and vibration isolators	Loss of material due to environmental corrosion; loss of mechanical function due to corrosion, distortion, dirt, overload, etc.	ISI	No	
3.5.1-33	Group B1.1: high strength low-alloy bolts	Crack initiation and growth due to SCC	Bolting integrity	No	

Table 3.5.2-1 Structures and Component Supports - Auxiliary Building - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Building Framing - Carbon Steel, Protected (column, hanger, beam, truss, decking, floor grating or plate, catwalk, threaded fastener, concrete expansion bolt, column base plate, weld, etc.)	HELB Shielding Pipe Whip Restraint Shelter/ Protection Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	III.A3.2-a	3.5.1-20	580, 584, H
						III.B5.1-b	3.5.1-31	583, A
					Structural Monitoring Program	III.A3.2-a	3.5.1-20	580, A
						III.B5.1-a	3.5.1-31	A
Building Framing - Concrete, Below Grade (wall footing, foundation, slab, grout, reinforcement, duct banks, cable pits, tunnels, etc.)	Flood Protection Structural Support for Safety Related	Concrete	Soil (Ext)	Loss of Material	None Required	III.A3.1-a	3.5.1-20	571, A
						III.A3.1-g	3.5.1-21	544, A
				Change in Material Properties	None Required	III.A3.1-b	3.5.1-20	557, A
						III.A3.1-g	3.5.1-21	544, A
				Cracking	None Required	III.A3.1-h	3.5.1-25	547, A
				Cracking and Expansion	None Required	III.A3.1-c	3.5.1-20	543, A
				Cracking / Loss of Bond/Matl	None Required	III.A3.1-e	3.5.1-21	544, A

Table 3.5.2-1 Structures and Component Supports - Auxiliary Building - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Building Framing - Concrete, Exposed (foundations, concrete & masonry wall, beam, roof slab, grout, reinforcements, concrete around expansion & grouted anchors)	Flood Protection Missile Barrier Radiation Shielding Shelter/ Protection Structural Support for Safety Related	Concrete	Atmosphere/ Weather (Ext)	Change in Material Properties	Structural Monitoring Program	III.A3.1-b	3.5.1-20	A
						III.A3.1-f	3.5.1-20	A
				Cracking	Structural Monitoring Program	III.A3.3-a	3.5.1-24	A
						III.A3.1-a	3.5.1-20	A
						III.A3.1-h	3.5.1-25	A
				Cracking and Expansion	Structural Monitoring Program	III.A3.1-c	3.5.1-20	A
				Cracking, Loss of Bond/Material	Structural Monitoring Program	III.A3.1-d	3.5.1-20	A
				Loss of Material	Structural Monitoring Program	III.A3.1-a	3.5.1-20	A
						III.A3.1-f	3.5.1-20	A
				Reduction in Concrete Anchor Capacity	Structural Monitoring Program	III.B5.2-a	3.5.1-29	A

Table 3.5.2-1 Structures and Component Supports - Auxiliary Building - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Building Framing - Concrete, Protected (foundations, concrete & masonry wall, column, pedestal, beam, floor slab, grout, reinforcements, concrete around expansion & grouted anchors, cable pits, tunnels, etc.)	Direct Flow Fire Barrier HELB Shielding Missile Barrier Pipe Whip Restraint Radiation Shielding Shelter/ Protection Structural Support for Safety Related	Concrete	Plant Indoor Air (Ext)	Cracking	Structural Monitoring Program	III.A3.3-a	3.5.1-24	A
						III.A3.1-h	3.5.1-25	A
				Cracking and Expansion	Structural Monitoring Program	III.A3.1-c	3.5.1-20	A
				Cracking, Loss of Bond/Material	Structural Monitoring Program	III.A3.1-d	3.5.1-20	A
				Loss of Material	Structural Monitoring Program	III.A3.1-f	3.5.1-20	A
				Loss of Strength	Structural Monitoring Program	III.A3.1-b	3.5.1-20	A
				Reduction in Concrete Anchor Capacity	Structural Monitoring Program	III.B5.2-a	3.5.1-29	A
Flood Barrier - Carbon Steel, Exposed (water tight doors)	Flood Protection Structural Support for Safety Related	Carbon Steel	Atmosphere/ Weather (Ext)	Loss of Material	Structural Monitoring Program	III.A3.2-a	3.5.1-20	580, C
Flood Barrier - Carbon Steel, Protected (water tight doors and gate)	Flood Protection Structural Support for Safety Related	Carbon Steel	Plant Indoor Air (Ext)	Loss of Leak Tightness	Structural Monitoring Program	III.A3.2-a	3.5.1-20	531, H
				Loss of Material	Boric Acid Corrosion Program	III.B5.1-b	3.5.1-31	583, C
					Structural Monitoring Program	III.A3.2-a	3.5.1-20	580, C

Table 3.5.2-1 Structures and Component Supports - Auxiliary Building - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Fuel Related Component - Carbon Steel, Protected (anchor bolts for SFP gates, liners, transfer tube appurtenances	Structural Support for Safety Related	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Water Chemistry Program	III.A5.2-b	3.5.1-23	578, 579, F
Fuel Related Component - Stainless, Protected (anchor bolts for SFP gates, liners, transfer tube appurtenances	Structural Support for Safety Related	Stainless Steel	Plant Indoor Air (Ext)	Cracking	None Required	III.A5.2-b	3.5.1-23	535, G
Fuel Related Component - Stainless, Borated (liner plates, gates, transfer tube expansion bellows)	Expansion / Separation Fluid Pressure Boundary Structural Support for Safety Related	Stainless Steel	Treated Water (Ext)	Cracking	None Required	III.A5.2-b	3.5.1-23	535, E
				Loss of Material	Water Chemistry Program	III.A5.2-b	3.5.1-23	523, 578, A

Table 3.5.2-1 Structures and Component Supports - Auxiliary Building - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
HELB/MELB Component - Carbon Steel, Protected (doors, scuttle, blowout panels, floor drains & screens, guard pipes, louvers, whip restraints, bellows, spray shields, etc.)	Direct Flow Flood Protection HELB Shielding Pipe Whip Restraint	Carbon Steel	Plant Indoor Air (Ext)	Loss of Leak Tightness	Structural Monitoring Program	III.A3.2-a	3.5.1-20	532, H
				Loss of Material	Boric Acid Corrosion Program	III.B5.1-b	3.5.1-31	583, C
					Structural Monitoring Program	III.A3.2-a	3.5.1-20	580, C
HELB/MELB Component - Concrete, Protected (curbs & pipe whip restraint grout, concrete at locations of expansion & grouted anchors)	Flood Protection HELB Shielding Pipe Whip Restraint Direct Flow	Concrete	Plant Indoor Air (Ext)	Cracking	Structural Monitoring Program	III.A3.3-a	3.5.1-24	A
				Cracking and Expansion	Structural Monitoring Program	III.A3.1-c	3.5.1-20	A
				Cracking, Loss of Bond/Material	Structural Monitoring Program	III.A3.1-d	3.5.1-20	A
				Loss of Material	Structural Monitoring Program	III.A3.1-f	3.5.1-20	A
				Reduction in Conc Anchor Capacity	Structural Monitoring Program	III.B5.2-a	3.5.1-29	A
HVAC Component - Stainless, Protected (Control Room vestibule door)	Fluid Pressure Boundary Structural Support for Safety Related	Stainless Steel	Plant Indoor Air (Ext)	Cracking	None Required	III.A3.2-a		588, F

Table 3.5.2-1 Structures and Component Supports - Auxiliary Building - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
HVAC Component - Carbon Steel, Protected (Control Room vestibule door)	Fluid Pressure Boundary Structural Support for Safety Related	Carbon Steel	Plant Indoor Air (Ext)	Loss of Leak Tightness	Structural Monitoring Program	III.A3.2-a	3.5.1-20	533, H
				Loss of Material	Boric Acid Corrosion Program	III.B5.1-b	3.5.1-31	583, C
					Structural Monitoring Program	III.A3.2-a	3.5.1-20	580, C
HVAC Component - Concrete, Protected (Control Room vestibules, concrete & masonry walls, floors, ceilings)	Fluid Pressure Boundary Structural Support for Safety Related	Concrete	Plant Indoor Air (Ext)	Cracking	Structural Monitoring Program	III.A3.3-a	3.5.1-24	A
				Cracking and Expansion	Structural Monitoring Program	III.A3.1-c	3.5.1-20	A
				Cracking, Loss of Bond/Material	Structural Monitoring Program	III.A3.1-d	3.5.1-20	A
				Loss of Material	Structural Monitoring Program	III.A3.1-f	3.5.1-20	A
				Reduction in Concrete Anchor Capacity	Structural Monitoring Program	III.B5.2-a	3.5.1-29	A
HVAC Component - Galvanized, Protected (damper & louver frames)	Fire Barrier Structural Support for Safety Related	Galvanized	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	III.B5.1-b	3.5.1-31	581, 583, C
					Structural Monitoring Program	III.A3.2-a	3.5.1-20	581, C

Table 3.5.2-1 Structures and Component Supports - Auxiliary Building - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Missile Shield - Carbon Steel, Protected (steel doors and structural steel missile barrier)	Missile Barrier Structural Support for Safety Related	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	III.B5.1-b	3.5.1-31	583, C
					Structural Monitoring Program	III.A3.2-a	3.5.1-20	580, C
Operator Access Component - Carbon Steel, Protected (stairs, floors, platforms, welds, bolted connections, etc.)	Structural Support for Non-Safety Related Structural Support for Regulated Events	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	III.B5.1-b	3.5.1-31	583, A
					Structural Monitoring Program	III.A3.2-a	3.5.1-20	A

Table 3.5.2-1 Structures and Component Supports - Auxiliary Building - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Operator Access Component - Concrete, Protected (stairs, floors, platforms, concrete at locations of expansion & grouted anchors, etc.)	Structural Support for Non-Safety Related Structural Support for Regulated Events	Concrete	Plant Indoor Air (Ext)	Cracking	Structural Monitoring Program	III.A3.3-a	3.5.1-24	A
						III.A3.1-h	3.5.1-25	A
				Cracking and Expansion	Structural Monitoring Program	III.A3.1-c	3.5.1-20	A
				Cracking, Loss of Bond/Material	Structural Monitoring Program	III.A3.1-d	3.5.1-20	A
				Loss of Material	Structural Monitoring Program	III.A3.1-f	3.5.1-20	A
				Loss of Strength	Structural Monitoring Program	III.A3.1-b	3.5.1-20	A
				Reduction in Concrete Anchor Capacity	Structural Monitoring Program	III.B5.2-a	3.5.1-29	A
Operator Access Component - Galvanized, Protected (stairs, walkways, removable platform, welds, bolted connections)	Structural Support for Non-Safety Related Structural Support for Regulated Events	Galvanized	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	III.B5.1-b	3.5.1-31	581, 583, C
					Structural Monitoring Program	III.A3.2-a	3.5.1-20	581, A

Table 3.5.2-2 Structures and Component Supports - Component Supports - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
ASME 1 Support - Containment - Carbon Steel, Protected	Structural Support for Safety Related	Carbon Steel	Containment Air (Ext)	Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	III.B1.1.1-a	3.5.1-32	A
					Boric Acid Corrosion Program	III.B1.1.1-b	3.5.1-31	A
ASME 1 Support - Containment - Concrete, Protected	Structural Support for Safety Related	Concrete	Containment Air (Ext)	Reduction in Conc Anchor Capacity	Structural Monitoring Program	III.B1.1.4-a	3.5.1-29	A
ASME 1 Support - Containment - Sliding Material, Cont Cavity	Expansion/ Separation Structural Support for Safety Related	Bronze	Containment Air (Ext)	Loss of Material	None Required	III.A4.2-b	3.5.1-20	515, I
ASME 1 Support - Containment - Sliding Material, Protected	Expansion/ Separation Structural Support for Safety Related	Bronze	Containment Air (Ext)	Loss of Mechanical Function	None Required	III.B1.1.3-a	3.5.1-32	515, I
ASME Class 1 Tubing Support - Auxiliary Bldg, Carbon Steel, Protected	Structural Support for Safety Related	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	III.B1.1.1-a	3.5.1-32	A
					Boric Acid Corrosion Program	III.B1.1.1-b	3.5.1-31	583, A

Table 3.5.2-2 Structures and Component Supports - Component Supports - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
ASME 2 & 3 Support - Turbine (Water Treatment Area) - Carbon Steel, Protected	Structural Support for Regulated Events Structural Support for Safety Related	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	III.B1.2.1-a	3.5.1-32	A
ASME 2 & 3 Support - Turbine (Water Treatment Area) - Concrete, Protected	Structural Support for Regulated Events Structural Support for Safety Related	Concrete	Plant Indoor Air (Ext)	Reduction in Conc Anchor Capacity	Structural Monitoring Program	III.B1.2.3-a	3.5.1-29	A
ASME Class 2 & 3 Piping & Mechanical Component Support - Auxiliary Bldg, Carbon Steel, Protected	Structural Support for Regulated Events Structural Support for Safety Related	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	III.B1.2.1-a	3.5.1-32	A
					Boric Acid Corrosion Program	III.B1.2.1-b	3.5.1-31	583, A
ASME Class 2 & 3 Piping & Mechanical Component Support - Auxiliary Bldg, Cast Iron, Protected	Structural Support for Safety Related	Cast Iron	Plant Indoor Air (Ext)	Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	III.B1.2.1-a	3.5.1-32	582, F
					Boric Acid Corrosion Program	III.B1.2.1-b	3.5.1-31	582, F

Table 3.5.2-2 Structures and Component Supports - Component Supports - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
ASME Class 2 & 3 Piping & Mechanical Component Support - Auxiliary Bldg, Concrete, Protected	Structural Support for Regulated Events Structural Support for Safety Related	Concrete	Plant Indoor Air (Ext)	Reduction in Conc Anchor Capacity	Structural Monitoring Program	III.B1.2.3-a	3.5.1-29	A
ASME Class 2 & 3 Piping & Mechanical Component Support - Auxiliary Bldg, Galvanized, Protected	Structural Support for Regulated Events Structural Support for Safety Related	Galvanized	Plant Indoor Air (Ext)	Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	III.B1.2.1-a	3.5.1-32	581, A
					Boric Acid Corrosion Program	III.B1.2.1-b	3.5.1-31	581, 583, A
ASME Class 2 & 3 Piping & Mechanical Component Support - Auxiliary Bldg, Sliding Material, Protected	Expansion/ Separation Structural Support for Safety Related	Bronze	Plant Indoor Air (Ext)	Loss of Mechanical Function	None Required	III.B1.2.2-a	3.5.1-32	515, I
ASME Class 2 & 3 Piping & Mechanical Component Support - Containment Bldg, Carbon Steel, Cont Cavity	Structural Support for Safety Related	Carbon Steel	Containment Air (Ext)	Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	III.B1.2.1-a	3.5.1-32	A
					Boric Acid Corrosion Program	III.B1.2.1-b	3.5.1-31	A

Table 3.5.2-2 Structures and Component Supports - Component Supports - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
ASME Class 2 & 3 Piping & Mechanical Component Support - Containment Bldg, Carbon Steel, Protected	Structural Support for Safety Related	Carbon Steel	Containment Air (Ext)	Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	III.B1.2.1-a	3.5.1-32	A
					Boric Acid Corrosion Program	III.B1.2.1-b	3.5.1-31	A
ASME Class 2 & 3 Piping & Mechanical Component Support - Containment Bldg, Concrete, Protected	Structural Support for Safety Related	Concrete	Containment Air (Ext)	Reduction in Conc Anchor Capacity	Structural Monitoring Program	III.B1.2.3-a	3.5.1-29	A
ASME Class 2 & 3 Piping & Mechanical Component Support - Containment Bldg, Galvanized, Cont Cavity	Structural Support for Safety Related	Galvanized	Containment Air (Ext)	Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	III.B1.2.1-a	3.5.1-32	581, A
					Boric Acid Corrosion Program	III.B1.2.1-b	3.5.1-31	581, A
ASME Class 2 & 3 Piping & Mechanical Component Support - Containment Bldg, Galvanized, Protected	Structural Support for Safety Related	Galvanized	Containment Air (Ext)	Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	III.B1.2.1-a	3.5.1-32	A
					Boric Acid Corrosion Program	III.B1.2.1-b	3.5.1-31	581, A

Table 3.5.2-2 Structures and Component Supports - Component Supports - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
ASME Class 2 & 3 Piping & Mechanical Component Support - Containment Bldg, Sliding Material, Protected	Expansion/ Separation Structural Support for Safety Related	Bronze	Containment Air (Ext)	Loss of Mechanical Function	None Required	III.B1.2.2-a	3.5.1-32	515, I
ASME Class 2 & 3 Piping & Mechanical Component Support - Containment Bldg, Stainless, Borated	Structural Support for Safety Related	Stainless Steel	Treated Water (Ext)	Loss of Material	Water Chemistry Program	III.A5.2-b	3.5.1-23	567, 587, C
ASME Class 2 & 3 Piping & Mechanical Component Support - Intake Structure Bldg, Carbon Steel, Protected	Structural Support for Safety Related	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	III.B1.2.1-a	3.5.1-32	A
ASME Class 2 & 3 Piping & Mechanical Component Support - Intake Structure Bldg, Concrete, Protected	Structural Support for Safety Related	Concrete	Plant Indoor Air (Ext)	Reduction in Conc Anchor Capacity	Structural Monitoring Program	III.B1.2.3-a	3.5.1-29	A

Table 3.5.2-2 Structures and Component Supports - Component Supports - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
ASME Class 2 & 3 Piping & Mechanical Component Support - Intake Structure Bldg, Galvanized, Protected	Structural Support for Safety Related	Galvanized	Plant Indoor Air (Ext)	Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	III.B1.2.1-a	3.5.1-32	A
ASME Class 2 & 3 Piping & Mechanical Component Support - Turbine Bldg, Carbon Steel, Protected	Structural Support for Regulated Events Structural Support for Safety Related	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	III.B1.2.1-a	3.5.1-32	A
ASME Class 2 & 3 Piping & Mechanical Component Support - Turbine Bldg, Concrete, Protected	Structural Support for Regulated Events Structural Support for Safety Related	Concrete	Plant Indoor Air (Ext)	Reduction in Conc Anchor Capacity	Structural Monitoring Program	III.B1.2.3-a	3.5.1-29	A
ASME Class 2 & 3 Piping & Mechanical Component Support - Turbine Bldg, Galvanized, Protected	Structural Support for Regulated Events Structural Support for Safety Related	Galvanized	Plant Indoor Air (Ext)	Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	III.B1.2.1-a	3.5.1-32	581, A

Table 3.5.2-2 Structures and Component Supports - Component Supports - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
ASME Class 2 & 3 Piping & Mechanical Component Support - Turbine Bldg, Sliding, Protected	Expansion/ Separation Structural Support for Safety Related	Bronze	Plant Indoor Air (Ext)	Loss of Mechanical Function	None Required	III.B1.2.2-a	3.5.1-32	515, I
ASME Class 2 & 3 Piping & Mechanical Component Support - Yard, Aluminum, Exposed	Structural Support for Safety Related	Aluminum	Atmosphere/ Weather (Ext)	Loss of Material	Boric Acid Corrosion Program			501, F
ASME Class 2 & 3 Piping & Mechanical Component Support - Yard, Carbon Steel, Below Grade	Structural Support for Regulated Events Structural Support for Safety Related	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	III.B1.2.1-a	3.5.1-32	518, A
ASME Class 2 & 3 Piping & Mechanical Component Support - Yard, Carbon Steel, Exposed	Structural Support for Regulated Events Structural Support for Safety Related	Carbon Steel	Atmosphere/ Weather (Ext)	Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	III.B1.2.1-a	3.5.1-32	A

Table 3.5.2-2 Structures and Component Supports - Component Supports - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
ASME Class 2 & 3 Piping & Mechanical Component Support - Yard, Concrete, Exposed	Structural Support for Regulated Events Structural Support for Safety Related	Concrete	Atmosphere/ Weather (Ext)	Reduction in Conc Anchor Capacity	Structural Monitoring Program	III.B1.2.3-a	3.5.1-29	A
Elec Component Support - Auxiliary Bldg, Carbon Steel, Protected	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	III.B2.1-b	3.5.1-31	583, A
						III.B3.1-b	3.5.1-31	583, A
					Structural Monitoring Program	III.B2.1-a	3.5.1-29	A
						III.B3.1-a	3.5.1-29	A
Elec Component Support - Auxiliary Bldg, Carbon Steel, Raw Water	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related	Carbon Steel	Raw Water (Ext)	Loss of Material	Structural Monitoring Program	III.B2.1-a	3.5.1-29	594, A

Table 3.5.2-2 Structures and Component Supports - Component Supports - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Elec Component Support - Auxiliary Bldg, Concrete, Protected	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related	Concrete	Plant Indoor Air (Ext)	Reduction in Conc Anchor Capacity	Structural Monitoring Program	III.B2.2-a	3.5.1-29	A
						III.B3.2-a	3.5.1-29	A
Elec Component Support - Auxiliary Bldg, Concrete, Raw Water	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related	Concrete	Raw Water (Ext)	Reduction in Conc Anchor Capacity	Structural Monitoring Program	III.B2.2-a	3.5.1-29	A

Table 3.5.2-2 Structures and Component Supports - Component Supports - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Elec Component Support - Auxiliary Bldg, Galvanized, Protected	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related	Galvanized	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	III.B2.1-b	3.5.1-31	581, A
						III.B3.1-b	3.5.1-31	581, A
					Structural Monitoring Program	III.B2.1-a	3.5.1-29	581, A
						III.B3.1-a	3.5.1-29	581, A
Elec Component Support - Auxiliary Bldg, Galvanized, Raw Water	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related	Galvanized	Raw Water (Ext)	Loss of Material	Structural Monitoring Program	III.B2.1-a	3.5.1-29	581, 594, A
Elec Component Support - Containment Bldg, Carbon Steel, Containment Cavity	Structural Support for Safety Related	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	III.B2.1-b	3.5.1-31	A
						III.B3.1-b	3.5.1-31	A
					Structural Monitoring Program	III.B2.1-a	3.5.1-29	A
						III.B3.1-a	3.5.1-29	A

Table 3.5.2-2 Structures and Component Supports - Component Supports - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Elec Component Support - Containment Bldg, Carbon Steel, Protected	Structural Support for Safety Related	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	III.B2.1-b	3.5.1-31	A
						III.B3.1-b	3.5.1-31	A
					Structural Monitoring Program	III.B2.1-a	3.5.1-29	A
						III.B3.1-a	3.5.1-29	A
Elec Component Support - Containment Bldg, Concrete, Protected	Structural Support for Safety Related	Concrete	Containment Air (Ext)	Reduction in Conc Anchor Capacity	Structural Monitoring Program	III.B2.2-a	3.5.1-29	A
						III.B3.2-a	3.5.1-29	A
Elec Component Support - Containment Bldg, Galvanized, Containment Cavity	Structural Support for Safety Related	Galvanized	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	III.B2.1-b	3.5.1-31	581, A
						III.B3.1-b	3.5.1-31	581, A
					Structural Monitoring Program	III.B2.1-a	3.5.1-29	581, A
						III.B3.1-a	3.5.1-29	581, A
Elec Component Support - Containment Bldg, Galvanized, Protected	Structural Support for Safety Related	Galvanized	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	III.B2.1-b	3.5.1-31	581, A
						III.B3.1-b	3.5.1-31	581, A
					Structural Monitoring Program	III.B2.1-a	3.5.1-29	581, A
						III.B3.1-a	3.5.1-29	581, A
Elec Component Support - Discharge Structure, Carbon Steel, Protected	Structural Support for Non-Safety Related	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Structural Monitoring Program	III.B2.1-a	3.5.1-29	A
						III.B3.1-a	3.5.1-29	A

Table 3.5.2-2 Structures and Component Supports - Component Supports - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Elec Component Support - Discharge Structure, Galvanized, Protected	Structural Support for Non-Safety Related	Galvanized	Plant Indoor Air (Ext)	Loss of Material	Structural Monitoring Program	III.B2.1-a	3.5.1-29	581, A
						III.B3.1-a	3.5.1-29	581, A
Elec Component Support - Intake Structure Bldg, Carbon Steel, Protected	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Structural Monitoring Program	III.B2.1-a	3.5.1-29	A
						III.B3.1-a	3.5.1-29	A
Elec Component Support - Intake Structure Bldg, Concrete, Protected	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related	Concrete	Plant Indoor Air (Ext)	Reduction in Conc Anchor Capacity	Structural Monitoring Program	III.B2.2-a	3.5.1-29	A
						III.B3.2-a	3.5.1-29	A

Table 3.5.2-2 Structures and Component Supports - Component Supports - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Elec Component Support - Intake Structure Bldg, Galvanized, Protected	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related	Galvanized	Plant Indoor Air (Ext)	Loss of Material	Structural Monitoring Program	III.B2.1-a	3.5.1-29	581, A
						III.B3.1-a	3.5.1-29	581, A
Elec Component Support - Switch Yard Relay House Group Bldg, Carbon Steel, Protected	Structural Support for Regulated Events	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Structural Monitoring Program	III.B2.1-a	3.5.1-29	A
						III.B3.1-a	3.5.1-29	A
Elec Component Support - Switch Yard Relay House Group Bldg, Concrete, Protected	Structural Support for Regulated Events	Concrete	Plant Indoor Air (Ext)	Reduction in Conc Anchor Capacity	Structural Monitoring Program	III.B2.2-a	3.5.1-29	A
						III.B3.2-a	3.5.1-29	A
Elec Component Support - Switch Yard Relay House Group Bldg, Galvanized, Protected	Structural Support for Regulated Events	Galvanized	Plant Indoor Air (Ext)	Loss of Material	Structural Monitoring Program	III.B2.1-a	3.5.1-29	581, A
						III.B3.1-a	3.5.1-29	581, A

Table 3.5.2-2 Structures and Component Supports - Component Supports - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Elec Component Support - Turbine Bldg, Carbon Steel, Protected	Structural Support for Regulated Events	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Structural Monitoring Program	III.B2.1-a	3.5.1-29	A
						III.B3.1-a	3.5.1-29	A
Elec Component Support - Turbine Bldg, Concrete, Protected	Structural Support for Regulated Events	Concrete	Plant Indoor Air (Ext)	Reduction in Conc Anchor Capacity	Structural Monitoring Program	III.B2.2-a	3.5.1-29	A
						III.B3.2-a	3.5.1-29	A
Elec Component Support - Turbine Bldg, Galvanized, Protected	Structural Support for Regulated Events	Galvanized	Plant Indoor Air (Ext)	Loss of Material	Structural Monitoring Program	III.B2.1-a	3.5.1-29	581, A
						III.B3.1-a	3.5.1-29	581, A
Elec Component Support - Yard, Carbon Steel, Exposed	Structural Support for Regulated Events	Carbon Steel	Atmosphere/ Weather (Ext)	Loss of Material	Structural Monitoring Program	III.B2.1-a	3.5.1-29	A
						III.B3.1-a	3.5.1-29	A
Elec Component Support - Yard, Concrete, Exposed	Structural Support for Regulated Events	Concrete	Atmosphere/ Weather (Ext)	Reduction in Conc Anchor Capacity	Structural Monitoring Program	III.B2.2-a	3.5.1-29	A
						III.B3.2-a	3.5.1-29	A
Elec Component Support - Yard, Galvanized, Exposed	Structural Support for Regulated Events	Galvanized	Atmosphere/ Weather (Ext)	Loss of Material	Structural Monitoring Program	III.B2.1-a	3.5.1-29	581, A
						III.B3.1-a	3.5.1-29	581, A
Elec Component Support - Yard, Galvanized, Raw Water	Structural Support for Regulated Events	Galvanized	Raw Water (Ext)	Loss of Material	Structural Monitoring Program	III.B3.1-a	3.5.1-29	581, 594, A

Table 3.5.2-2 Structures and Component Supports - Component Supports - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
High Strength Bolting - Containment Building, Carbon Steel, Protected	Structural Support for Safety Related	Carbon Steel	Containment Air (Ext)	Cracking	Bolting Integrity Program	III.B1.1.2-a	3.5.1-33	A
				Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	III.B1.1.1-a	3.5.1-32	A
						III.B1.2.1-a	3.5.1-32	A
					Boric Acid Corrosion Program	III.B1.1.1-b	3.5.1-31	A
						III.B1.2.1-b	3.5.1-31	A
Non-ASME Component Support - Auxiliary Bldg, Aluminum, Protected	Structural Support for Safety Related	Aluminum	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	VII.A1.1-a	3.3.1-11	503, F
Non-ASME Piping & Mechanical Component Support - Auxiliary Bldg, Carbon Steel, Protected	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	III.B2.1-b	3.5.1-31	583, A
						III.B4.1-b	3.5.1-31	583, A
					Structural Monitoring Program	III.B2.1-a	3.5.1-29	A
						III.B4.1-a	3.5.1-29	A

Table 3.5.2-2 Structures and Component Supports - Component Supports - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Non-ASME Piping & Mechanical Component Support - Auxiliary Bldg, Concrete, Protected	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related	Concrete	Plant Indoor Air (Ext)	Reduction in Conc Anchor Capacity	Structural Monitoring Program	III.B2.2-a	3.5.1-29	A
						III.B4.3-a	3.5.1-29	A
Non-ASME Piping & Mechanical Component Support - Auxiliary Bldg, Galvanized, Protected	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related	Galvanized	Plant Indoor Air (Ext)	Loss of Material	Boric Acid Corrosion Program	III.B2.1-b	3.5.1-31	581, 583, A
						III.B4.1-b	3.5.1-31	581, 583, A
					Structural Monitoring Program	III.B2.1-a	3.5.1-29	581, A
						III.B4.1-a	3.5.1-29	581, A
Non-ASME Piping & Mechanical Component Support - Auxiliary Bldg, Stainless, Borated	Structural Support for Safety Related	Stainless Steel	Treated Water (Ext)	Loss of Material	Water Chemistry Program	III.A5.2-b	3.5.1-23	578, C

Table 3.5.2-2 Structures and Component Supports - Component Supports - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Non-ASME Piping & Mechanical Component Support - Boiler Building, Carbon Steel, Protected	Structural Support for Non-Safety Related	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Structural Monitoring Program	III.B2.1-a	3.5.1-29	A
						III.B4.1-a	3.5.1-29	A
Non-ASME Piping & Mechanical Component Support - Boiler Building, Concrete, Protected	Structural Support for Non-Safety Related	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Structural Monitoring Program	III.B2.2-a	3.5.1-29	A
						III.B4.3-a	3.5.1-29	A
Non-ASME Piping & Mechanical Component Support - Containment Bldg, Aluminum, Protected	Structural Support for Non-Safety Related	Aluminum	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program			502, F
Non-ASME Piping & Mechanical Component Support - Containment Bldg, Carbon Steel, Containment Cavity	Structural Support for Non-Safety Related Structural Support for Safety Related	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	III.B2.1-b	3.5.1-31	A
						III.B4.1-b	3.5.1-31	A
					Structural Monitoring Program	III.B2.1-a	3.5.1-29	A
						III.B4.1-a	3.5.1-29	A

Table 3.5.2-2 Structures and Component Supports - Component Supports - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Non-ASME Piping & Mechanical Component Support - Containment Bldg, Carbon Steel, Protected	Structural Support for Non-Safety Related Structural Support for Safety Related	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	III.B2.1-b	3.5.1-31	A
						III.B4.1-b	3.5.1-31	A
					Structural Monitoring Program	III.B2.1-a	3.5.1-29	A
						III.B4.1-a	3.5.1-29	A
Non-ASME Piping & Mechanical Component Support - Containment Bldg, Concrete, Protected	Structural Support for Non-Safety Related Structural Support for Safety Related	Concrete	Containment Air (Ext)	Reduction in Conc Anchor Capacity	Structural Monitoring Program	III.B2.2-a	3.5.1-29	A
						III.B4.3-a	3.5.1-29	A
Non-ASME Piping & Mechanical Component Support - Containment Bldg, Galvanized, Containment Cavity	Structural Support for Non-Safety Related Structural Support for Safety Related	Galvanized	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	III.B2.1-b	3.5.1-31	581, A
						III.B4.1-b	3.5.1-31	581, A
					Structural Monitoring Program	III.B2.1-a	3.5.1-29	581, A
						III.B4.1-a	3.5.1-29	581, A

Table 3.5.2-2 Structures and Component Supports - Component Supports - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Non-ASME Piping & Mechanical Component Support - Containment Bldg, Galvanized, Protected	Structural Support for Non-Safety Related Structural Support for Safety Related	Galvanized	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	III.B2.1-b	3.5.1-31	581, A
						III.B4.1-b	3.5.1-31	581, A
					Structural Monitoring Program	III.B2.1-a	3.5.1-29	581, A
						III.B4.1-a	3.5.1-29	581, A
Non-ASME Piping & Mechanical Component Support - Containment Bldg, Stainless, Borated	Structural Support for Non-Safety Related Structural Support for Safety Related	Stainless Steel	Treated Water (Ext)	Loss of Material	Water Chemistry Program	III.A5.2-b	3.5.1-23	567, 587, C
Non-ASME Piping & Mechanical Component Support - Discharge Structure, Carbon Steel, Protected	Structural Support for Non-Safety Related	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Structural Monitoring Program	III.B2.1-a	3.5.1-29	A
						III.B4.1-a	3.5.1-29	A

Table 3.5.2-2 Structures and Component Supports - Component Supports - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Non-ASME Piping & Mechanical Component Support - Discharge Structure, Cast Iron, Protected	Structural Support for Non-Safety Related	Cast Iron	Plant Indoor Air (Ext)	Loss of Material	Structural Monitoring Program	III.B2.1-a	3.5.1-29	582, F
Non-ASME Piping & Mechanical Component Support - Discharge Structure, Concrete, Protected	Structural Support for Non-Safety Related	Concrete	Plant Indoor Air (Ext)	Reduction in Conc Anchor Capacity	Structural Monitoring Program	III.B2.2-a	3.5.1-29	A
						III.B4.3-a	3.5.1-29	A
Non-ASME Piping & Mechanical Component Support - Discharge Structure, Galvanized, Protected	Structural Support for Non-Safety Related	Galvanized	Plant Indoor Air (Ext)	Loss of Material	Structural Monitoring Program	III.B2.1-a	3.5.1-29	581, A
						III.B4.1-a	3.5.1-29	581, A
Non-ASME Piping & Mechanical Component Support - Feedwater Purity Bldg, Carbon Steel, Protected	Structural Support for Regulated Events	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Structural Monitoring Program	III.B2.1-a	3.5.1-29	A
						III.B4.1-a	3.5.1-29	A

Table 3.5.2-2 Structures and Component Supports - Component Supports - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Non-ASME Piping & Mechanical Component Support - Feedwater Purity Bldg, Concrete, Protected	Structural Support for Regulated Events	Concrete	Plant Indoor Air (Ext)	Reduction in Conc Anchor Capacity	Structural Monitoring Program	III.B2.2-a	3.5.1-29	A
						III.B4.3-a	3.5.1-29	A
Non-ASME Piping & Mechanical Component Support - Intake Structure Bldg, Carbon Steel, Protected	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Structural Monitoring Program	III.B2.1-a	3.5.1-29	A
						III.B4.1-a	3.5.1-29	A
Non-ASME Piping & Mechanical Component Support - Intake Structure Bldg, Concrete, Protected	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related	Concrete	Plant Indoor Air (Ext)	Reduction in Conc Anchor Capacity	Structural Monitoring Program	III.B2.2-a	3.5.1-29	A
						III.B4.3-a	3.5.1-29	A

Table 3.5.2-2 Structures and Component Supports - Component Supports - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Non-ASME Piping & Mechanical Component Support - Intake Structure Bldg, Galvanized, Protected	Structural Support for Non-Safety Related Structural Support for Safety Related	Galvanized	Plant Indoor Air (Ext)	Loss of Material	Structural Monitoring Program	III.B2.1-a	3.5.1-29	581, A
						III.B4.1-a	3.5.1-29	581, A
Non-ASME Piping & Mechanical Component Support - Turbine Bldg, Carbon Steel, Protected	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Structural Monitoring Program	III.B2.1-a	3.5.1-29	A
						III.B4.1-a	3.5.1-29	A
Non-ASME Piping & Mechanical Component Support - Turbine Bldg, Concrete, Protected	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related	Concrete	Plant Indoor Air (Ext)	Reduction in Conc Anchor Capacity	Structural Monitoring Program	III.B2.2-a	3.5.1-29	A
						III.B4.3-a	3.5.1-29	A

Table 3.5.2-2 Structures and Component Supports - Component Supports - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Non-ASME Piping & Mechanical Component Support - Turbine Bldg, Galvanized, Protected	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related	Galvanized	Plant Indoor Air (Ext)	Loss of Material	Structural Monitoring Program	III.B2.1-a	3.5.1-29	581, A
						III.B4.1-a	3.5.1-29	581, A
Non-ASME Piping & Mechanical Component Support - Water Treatment Bldg, Carbon Steel, Protected	Structural Support for Non-Safety Related Structural Support for Regulated Events	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Structural Monitoring Program	III.B2.1-a	3.5.1-29	A
						III.B4.1-a	3.5.1-29	A
Non-ASME Piping & Mechanical Component Support - Water Treatment Bldg, Concrete, Protected	Structural Support for Non-Safety Related Structural Support for Regulated Events	Concrete	Plant Indoor Air (Ext)	Reduction in Conc Anchor Capacity	Structural Monitoring Program	III.B2.2-a	3.5.1-29	A
						III.B4.3-a	3.5.1-29	A

Table 3.5.2-2 Structures and Component Supports - Component Supports - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Non-ASME Piping & Mechanical Component Support - Yard, Carbon Steel, Exposed	Structural Support for Regulated Events Structural Support for Safety Related	Carbon Steel	Atmosphere/ Weather (Ext)	Loss of Material	Structural Monitoring Program	III.B2.1-a	3.5.1-29	A
						III.B4.1-a	3.5.1-29	A
Non-ASME Piping & Mechanical Component Support - Yard, Concrete, Exposed	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related	Concrete	Atmosphere/ Weather (Ext)	Reduction in Conc Anchor Capacity	Structural Monitoring Program	III.B2.2-a	3.5.1-29	A
						III.B4.3-a	3.5.1-29	A
Non-ASME Piping & Mechanical Component Support - Yard, Galvanized, Exposed	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related	Galvanized	Atmosphere/ Weather (Ext)	Loss of Material	Structural Monitoring Program	III.B2.1-a	3.5.1-29	581, A
						III.B4.1-a	3.5.1-29	581, A

Table 3.5.2-2 Structures and Component Supports - Component Supports - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Spent Fuel Storage Rack - Auxiliary Building, Boron Carbide, Borated Water	Radiation Shielding	Boron Carbide	Treated Water (Ext)	Loss of Material	None Required	VII.A2.1-b	3.3.1-10	595, F
Spent Fuel Storage Rack - Auxiliary Building, Stainless Steel, Borated Water	Structural Support for Safety Related	Stainless Steel	Treated Water (Ext)	Loss of Material	Water Chemistry Program	VII.A2.1-c	3.3.1-13	H

Table 3.5.2-3 Structures and Component Supports - Containment - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Containment Shell & Base Slab - Containment Bldg, Carbon Steel, Protected (air locks, equipment hatch, liner plate, penetrations)	Heat Sink Pressure Boundary/ Fission Product Retention Structural Support for Safety Related Structural Support for Regulated Events	Carbon Steel	Containment Air (Ext)	Loss of Leak Tightness	Containment Leakage Testing Program	II.A3.2-b	3.5.1-05	538, A
				Loss of Material	Boric Acid Corrosion Program	II.A1.2-a	3.5.1-12	577, A
					Containment Inservice Inspection Program	II.A1.2-a	3.5.1-12	A
						II.A3.1-a	3.5.1-03	A
						II.A3.2-a	3.5.1-04	A
					Containment Leakage Testing Program	II.A1.2-a	3.5.1-12	A
						II.A3.1-a	3.5.1-03	A
						II.A3.2-a	3.5.1-04	A
Containment Shell & Base Slab - Containment Bldg, Concrete, Below Grade (base mat, foundation, wall, embedded steel, etc.)	Pressure Boundary/ Fission Product Retention Radiation Shielding Structural Support for Safety Related Shelter/ Protection	Concrete	Soil (Ext)	Change in Material Properties	None Required	II.A1.1-b	3.5.1-07	572, A
						II.A1.1-c	3.5.1-07	573, A
				Cracking	None Required	II.A1.1-f	3.5.1-08	547, A
				Cracking and Expansion	None Required	II.A1.1-d	3.5.1-16	574, A
				Cracking, Loss of Bond/Matl	None Required	II.A1.1-e	3.5.1-07	573, A
				Loss of Material	None Required	II.A1.1-a	3.5.1-16	575, A
					None Required	II.A1.1-c	3.5.1-07	573, A

Table 3.5.2-3 Structures and Component Supports - Containment - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Containment Shell & Base Slab - Containment Bldg, Concrete, Exposed (dome, wall, base mat, embedded steel, etc.)	Heat Sink Missile Barrier Pressure Boundary/ Fission Product Retention Radiation Shielding Shelter / Protection Structural Support for Regulated Events Structural Support for Safety Related	Concrete	Atmosphere/ Weather (Ext)	Change in Material Properties	Containment Inservice Inspection Program	II.A1.1-b	3.5.1-07	A
						II.A1.1-c	3.5.1-07	A
				Cracking	Containment Inservice Inspection Program	II.A1.1-a	3.5.1-16	A
					Structural Monitoring Program	II.A1.1-f	3.5.1-08	547, A
				Cracking and Expansion	Containment Inservice Inspection Program	II.A1.1-d	3.5.1-16	A
				Cracking, Loss of Bond/Matl	Containment Inservice Inspection Program	II.A1.1-e	3.5.1-07	A
				Loss of Material	Containment Inservice Inspection Program	II.A1.1-a	3.5.1-16	A
						II.A1.1-c	3.5.1-07	A
Containment Shell & Base Slab - Containment Bldg, Elastomer, Protected (seals, gaskets, moisture barriers)	Pressure Boundary/ Fission Product Retention Shelter/ Protection	Elastomers	Containment Air (Ext)	Change in Material Properties	Containment Inservice Inspection Program	II.A3.3-a	3.5.1-06	A
					Containment Leakage Testing Program	II.A3.3-a	3.5.1-06	A
				Cracking	Containment Inservice Inspection Program	II.A3.3-a	3.5.1-06	A
					Containment Leakage Testing Program	II.A3.3-a	3.5.1-06	A

Table 3.5.2-3 Structures and Component Supports - Containment - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Containment Shell & Base Slab - Containment Bldg, Stainless Steel, Protected (fuel transfer tube and closure flange)	Heat Sink Pressure Boundary/ Fission Product Retention	Stainless Steel	Containment Air (Ext)	Loss of Material	Containment Inservice Inspection Program	II.A3.1-a	3.5.1-03	570, A
					Containment Leakage Testing Program	II.A3.1-a	3.5.1-03	570, A
Containment Shell Prestressing System - Containment Bldg, Carbon Steel, Exposed (includes tendons and anchorage components)	Pressure Boundary/ Fission Product Retention Structural Support for Safety Related	Carbon Steel	Atmosphere/ Weather (Ext)	Loss of Material	Containment Inservice Inspection Program	II.A1.3-a	3.5.1-15	A

Table 3.5.2-4 Structures and Component Supports - Containment Interior Structures - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Building Framing - Carbon Steel, Protected (column, beam, truss, platform, floor grating or plate, catwalk, bracing, threaded fastener, concrete expansion bolt, column base plate, welds, etc.)	Heat Sink HELB Shielding Pipe Whip Restraint Structural Support for Regulated Events Structural Support for Safety Related Support for Non-Safety Related	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	III.A4.2-a	3.5.1-20	580, 585, H
						III.B5.1-b	3.5.1-29	A
					Structural Monitoring Program	III.A4.2-a	3.5.1-20	580, A
Building Framing - Concrete, Containment Cavity (reactor shield walls including reinforcements, inserts, grouted anchors)	Heat Sink HELB Shielding Missile Barrier Radiation Shielding Structural Support for Safety Related	Concrete	Containment Air (Ext)	Cracking and Expansion	Structural Monitoring Program	III.A4.1-b	3.5.1-20	A
				Cracking, Loss of Bond/Material	Structural Monitoring Program	III.A4.1-d	3.5.1-20	A
				Loss of Material	Structural Monitoring Program	III.A4.1-a	3.5.1-20	A
				Change in Material Properties	None Required	III.A4.1-c	3.5.1-27	576, A

Table 3.5.2-4 Structures and Component Supports - Containment Interior Structures - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Building Framing - Concrete, Protected (concrete & masonry wall, column, pedestal, beam, slab, grout, reinforcements, concrete around expansion & grouted anchors, etc.)	Direct Flow Heat Sink HELB Shielding Pipe Whip Restraint Radiation Shielding Structural Support for Regulated Events Structural Support for Non-Safety Related Structural Support for Safety Related	Concrete	Containment Air (Ext)	Cracking	Structural Monitoring Program			559, H
				Cracking and Expansion	Structural Monitoring Program	III.A4.1-b	3.5.1-20	A
				Cracking, Loss of Bond/Material	Structural Monitoring Program	III.A4.1-d	3.5.1-20	A
				Loss of Material	Structural Monitoring Program	III.A4.1-a	3.5.1-20	A
				Reduction in Concrete Anchor Capacity	Structural Monitoring Program	III.B5.2-a	3.5.1-29	A
Fuel Related Component - Stainless, Protected (refueling cavity sump, containment sump liner and screen, transfer tube)	Structural Support for Safety Related	Stainless Steel	Containment Air (Ext)	Cracking	None Required	III.A5.2-b	3.5.1-23	535, E
				Loss of Material	Water Chemistry Program	III.A5.2-b	3.5.1-23	578, 587, A
					Structural Monitoring Program	III.A5.2-b	3.5.1-23	596, E

Table 3.5.2-4 Structures and Component Supports - Containment Interior Structures - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Fuel Related Component - Carbon Steel, Protected (anchor bolts for refueling cavity liner and transfer tube appurtenances)	Structural Support for Safety Related	Carbon Steel	Containment Air (Ext)	Loss of Material	Water Chemistry Program	III.A5.2-b	3.5.1-23	578, 579, F
HELB/MELB Component - Carbon Steel, Protected (steel curbs, pipe whip restraints, spray shields, etc.)	HELB Shielding Pipe Whip Restraint Structural Support for Safety Related	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	III.B5.1-b	3.5.1-31	C
					Structural Monitoring Program	III.A4.2-a	3.5.1-20	580, C
HELB/MELB Component - Concrete, Protected (curbs, sump, pipe whip restraint grout, concrete at locations of expansion & grouted anchors, etc.)	Direct Flow Flood Protection Heat Sink HELB Shielding Pipe Whip Restraint	Concrete	Containment Air (Ext)	Cracking	Structural Monitoring Program			H
				Cracking and Expansion	Structural Monitoring Program	III.A4.1-b	3.5.1-20	A
				Cracking, Loss of Bond/Material	Structural Monitoring Program	III.A4.1-d	3.5.1-20	A
				Loss of Material	Structural Monitoring Program	III.A4.1-a	3.5.1-20	A
				Reduction in Conc Anchor Capacity	Structural Monitoring Program	III.B5.2-a	3.5.1-29	A

Table 3.5.2-4 Structures and Component Supports - Containment Interior Structures - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
HVAC Component - Carbon Steel, Protected (damper & louver mounting frames)	Structural Support for Safety Related	Carbon Steel	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	III.B5.1-b	3.5.1-31	C
					Structural Monitoring Program	III.A4.2-a	3.5.1-20	580, C
HVAC Component - Galvanized, Protected (damper & louver mounting frames)	Structural Support for Safety Related	Galvanized	Containment Air (Ext)	Loss of Material	Boric Acid Corrosion Program	III.B5.1-b	3.5.1-31	C
					Structural Monitoring Program	III.A4.2-a	3.5.1-20	580, C
Missile Shield - Concrete, Containment Cavity (removable missile shield above reactor vessel)	Heat Sink Missile Barrier	Concrete	Containment Air (Ext)	Cracking and Expansion	Structural Monitoring Program	III.A4.1-b	3.5.1-20	A
				Cracking, Loss of Bond/Material	Structural Monitoring Program	III.A4.1-d	3.5.1-20	A
				Loss of Material	Structural Monitoring Program	III.A4.1-a	3.5.1-20	A
				Reduction in Conc Anchor Capacity	Structural Monitoring Program	III.B5.2-a	3.5.1-29	A

Table 3.5.2-5 Structures and Component Supports - Discharge Structure - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Building Framing - Cast Iron, Raw Water (sluice gates)	Structural Support for Non-Safety Related	Cast Iron	Raw Water (Ext)	Loss of Material	Structural Monitoring Program	III.A6.2-a	3.5.1-22	580, 582, A
Building Framing - Discharge/Intake Crib - Carbon Steel, Raw Water (sluice gate, column, bracket, beam, bracing, threaded fastener, connector, weld, etc.)	Structural Support for Non-Safety Related	Carbon Steel	Raw Water (Ext)	Loss of Material	Structural Monitoring Program	III.A6.2-a	3.5.1-22	580, A
Building Framing - Concrete, Below Grade (wall, footing, foundation, slab, grout, reinforcement, trenches, cable pits, tunnels, etc.)	Structural Support for Non-Safety Related	Concrete	Soil (Ext)	Loss of Material	None Required	III.A6.1-a	3.5.1-22	571, A
						III.A6.1-e	3.5.1-22	544, A
				Change in Material Properties	None Required	III.A6.1-b	3.5.1-22	557, A
						III.A6.1-e	3.5.1-22	544, A
				Cracking and Expansion	None Required	III.A6.1-c	3.5.1-22	543, A
				Cracking, Loss of Bond/Matl	None Required	III.A6.1-d	3.5.1-22	544, A
				Cracking	None Required	III.A6.1-f	3.5.1-22	547, A

Table 3.5.2-5 Structures and Component Supports - Discharge Structure - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Building Framing - Concrete, Exposed (foundations, masonry & concrete wall, floor/ roof slab, grout, reinforcements, steel shapes, concrete around expansion & grouted anchors)	Structural Support for Non-Safety Related	Concrete	Atmosphere/ Weather (Ext)	Change in Material Properties	Structural Monitoring Program	III.A6.1-b	3.5.1-22	A
						III.A6.1-e	3.5.1-22	A
				Cracking	Structural Monitoring Program	III.A6.3-a	3.5.1-24	A
						III.A6.1-a	3.5.1-22	A
						III.A6.1-f	3.5.1-22	A
				Cracking and Expansion	Structural Monitoring Program	III.A6.1-c	3.5.1-22	A
				Cracking, Loss of Bond/Material	Structural Monitoring Program	III.A6.1-d	3.5.1-22	A
				Loss of Material	Structural Monitoring Program	III.A6.1-a	3.5.1-22	A
						III.A6.1-e	3.5.1-22	A
				Reduction in Concrete Anchor Capacity	Structural Monitoring Program	III.B5.2-a	3.5.1-29	A

Table 3.5.2-5 Structures and Component Supports - Discharge Structure - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Building Framing - Concrete, Raw Water (wall, footing, foundation, slab, grout, reinforcements)	Structural Support for Non-Safety Related	Concrete	Raw Water (Ext)	Change in Material Properties	Structural Monitoring Program	III.A6.1-b	3.5.1-22	557, A
						III.A6.1-e	3.5.1-22	A
				Cracking	Structural Monitoring Program	III.A6.1-a	3.5.1-22	A
						III.A6.1-f	3.5.1-22	A
				Cracking and Expansion	Structural Monitoring Program	III.A6.1-c	3.5.1-22	A
				Loss of Material	Structural Monitoring Program	III.A6.1-a	3.5.1-22	A
						III.A6.1-e	3.5.1-22	A
						III.A6.1-h	3.5.1-22	A

Table 3.5.2-6 Structures and Component Supports - Feedwater Purity Building - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Building Framing - Carbon Steel, Protected (column, beam, bracing, threaded fastener, concrete expansion bolt, column base plate, welds, etc.)	Structural Support for Regulated Events	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Structural Monitoring Program	III.A3.2-a	3.5.1-20	580, A
Building Framing - Concrete, Below Grade (pedestal, grade beam, footing, foundation, slab, grout, reinforcement, trenches, cable pits, etc.)	Structural Support for Regulated Events	Concrete	Soil (Ext)	Loss of Material	None Required	III.A3.1-a	3.5.1-20	571, A
						III.A3.1-g	3.5.1-21	544, A
				Change in Material Properties	None Required	III.A3.1-b	3.5.1-20	557, A
						III.A3.1-g	3.5.1-21	544, A
				Cracking and Expansion	None Required	III.A3.1-c	3.5.1-20	543, A
				Cracking, Loss of Bond/Matl	None Required	III.A3.1-e	3.5.1-21	544, A
				Cracking	None Required	III.A3.1-h	3.5.1-25	547, A

Table 3.5.2-6 Structures and Component Supports - Feedwater Purity Building - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Building Framing - Concrete, Exposed (foundations, masonry/ concrete wall, grout, reinforcement, concrete around expansion & grouted anchors, etc.)	Structural Support for Regulated Events	Concrete	Atmosphere/ Weather (Ext)	Change in Material Properties	Structural Monitoring Program	III.A3.1-b	3.5.1-20	A
						III.A3.1-f	3.5.1-20	A
				Cracking	Structural Monitoring Program	III.A3.3-a	3.5.1-24	A
						III.A3.1-a	3.5.1-20	A
						III.A3.1-h	3.5.1-25	A
				Cracking and Expansion	Structural Monitoring Program	III.A3.1-c	3.5.1-20	A
				Cracking, Loss of Bond/Material	Structural Monitoring Program	III.A3.1-d	3.5.1-20	A
				Loss of Material	Structural Monitoring Program	III.A3.1-a	3.5.1-20	A
						III.A3.1-f	3.5.1-20	A
				Reduction in Concrete Anchor Capacity	Structural Monitoring Program	III.B5.2-a	3.5.1-29	A

Table 3.5.2-6 Structures and Component Supports - Feedwater Purity Building - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Building Framing - Concrete, Protected (foundations, masonry/ concrete wall, column, pedestal, beam, slab, grout, reinforcement, concrete around expansion & grouted anchors, etc.)	Structural Support for Regulated Events	Concrete	Plant Indoor Air (Ext)	Cracking	Structural Monitoring Program	III.A3.3-a	3.5.1-24	A
						III.A3.1-h	3.5.1-25	A
				Cracking and Expansion	Structural Monitoring Program	III.A3.1-c	3.5.1-20	A
				Cracking, Loss of Bond/Material	Structural Monitoring Program	III.A3.1-d	3.5.1-20	A
				Loss of Material	Structural Monitoring Program	III.A3.1-f	3.5.1-20	A
				Loss of Strength	Structural Monitoring Program	III.A3.1-b	3.5.1-20	A
				Reduction in Concrete Anchor Capacity	Structural Monitoring Program	III.B5.2-a	3.5.1-29	A

Table 3.5.2-7 Structures and Component Supports - Intake Structure - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Building Framing - Carbon Steel, Raw Water (gates, guides, and trash racks)	Structural Support for Regulated Events Structural Support for Safety Related	Carbon Steel	Raw Water (Ext)	Loss of Material	Structural Monitoring Program	III.A6.2-a	3.5.1-22	580, A
Building Framing - Cast Iron, Raw Water (Sluice Gates)	Structural Support for Regulated Events Structural Support for Safety Related	Cast Iron	Raw Water (Ext)	Loss of Material	Structural Monitoring Program	III.A6.2-a	3.5.1-22	580, 582, A
Building Framing - Galvanized, Raw Water (fasteners and anchor bolts)	Structural Support for Regulated Events Structural Support for Safety Related	Galvanized	Raw Water (Ext)	Loss of Material	Structural Monitoring Program	III.A6.2-a	3.5.1-22	581, A

Table 3.5.2-7 Structures and Component Supports - Intake Structure - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Building Framing - Concrete, Below Grade (wall, foundation, slab, grout, reinforcement, trenches, cable pits, tunnels, etc.)	Shelter / Protection Structural Support for Regulated Events Structural Support for Safety Related	Concrete	Soil (Ext)	Loss of Material	None Required	III.A6.1-a	3.5.1-22	571, A
						III.A6.1-e	3.5.1-22	544, A
				Change in Material Properties	None Required	III.A6.1-b	3.5.1-22	557, A
						III.A6.1-e	3.5.1-22	544, A
				Cracking and Expansion	None Required	III.A6.1-c	3.5.1-22	543, A
				Cracking, Loss of Bond/Matl	None Required	III.A6.1-d	3.5.1-22	544, A
				Cracking	None Required	III.A6.1-f	3.5.1-22	547, A

Table 3.5.2-7 Structures and Component Supports - Intake Structure - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Building Framing - Concrete, Exposed (wall, beam, floor slab, roof slab, grout, reinforcements, grouted anchors, etc.)	Fire Barrier Flood Protection Missile Barrier Shelter/ Protection Structural Support for Regulated Events Structural Support for Safety Related	Concrete	Atmosphere/ Weather (Ext)	Change in Material Properties	Structural Monitoring Program	III.A6.1-b	3.5.1-22	A
						III.A6.1-e	3.5.1-22	A
				Cracking	Structural Monitoring Program	III.A6.3-a	3.5.1-24	A
						III.A6.1-a	3.5.1-22	A
						III.A6.1-f	3.5.1-22	A
				Cracking and Expansion	Structural Monitoring Program	III.A6.1-c	3.5.1-22	A
				Cracking, Loss of Bond/Material	Structural Monitoring Program	III.A6.1-d	3.5.1-22	A
				Loss of Material	Structural Monitoring Program	III.A6.1-a	3.5.1-22	A
						III.A6.1-e	3.5.1-22	A
				Reduction in concrete Anchor Capacity	Structural Monitoring Program	III.B5.2-a	3.5.1-29	A

Table 3.5.2-7 Structures and Component Supports - Intake Structure - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Building Framing - Concrete, Protected (wall, beam, floor slab, grout, reinforcements, grouted anchors, etc.)	Direct Flow Flood Protection Shelter/ Protection Structural Support for Regulated Events Structural Support for Safety Related	Concrete	Plant Indoor Air (Ext)	Cracking	Structural Monitoring Program	III.A6.3-a	3.5.1-24	A
						III.A6.1-f	3.5.1-22	A
				Cracking and Expansion	Structural Monitoring Program	III.A6.1-c	3.5.1-22	A
				Cracking, Loss of Bond/Material	Structural Monitoring Program	III.A6.1-d	3.5.1-22	A
				Loss of Material	Structural Monitoring Program	III.A6.1-a	3.5.1-22	A
						III.A6.1-e	3.5.1-22	A
				Loss of Strength	Structural Monitoring Program	III.A6.1-b	3.5.1-22	A
				Reduction in Concrete Anchor Capacity	Structural Monitoring Program	III.B5.2-a	3.5.1-29	A
Building Framing - Concrete, Raw Water (wall, column, beam, footing, foundation slab, floor slab, grout, reinforcements, etc.)	Structural Support for Regulated Events Structural Support for Safety Related	Concrete	Raw Water (Ext)	Change in Material Properties	Structural Monitoring Program	III.A6.1-b	3.5.1-22	A
						III.A6.1-e	3.5.1-22	A
				Cracking	Structural Monitoring Program	III.A6.1-a	3.5.1-22	A
						III.A6.1-f	3.5.1-22	A
				Cracking and Expansion	Structural Monitoring Program	III.A6.1-c	3.5.1-22	A
				Loss of Material	Structural Monitoring Program	III.A6.1-a	3.5.1-22	A
						III.A6.1-e	3.5.1-22	A
						III.A6.1-h	3.5.1-22	A

Table 3.5.2-7 Structures and Component Supports - Intake Structure - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Flood Barrier - Concrete, Protected (concrete interior wall in southeast corner)	Flood Protection	Concrete	Plant Indoor Air (Ext)	Cracking and Expansion	Structural Monitoring Program	III.A6.1-c	3.5.1-22	A
				Cracking, Loss of Bond/Material	Structural Monitoring Program	III.A6.1-d	3.5.1-22	A
				Loss of Material	Structural Monitoring Program	III.A6.1-e	3.5.1-22	A
				Loss of Strength	Structural Monitoring Program	III.A6.1-b	3.5.1-22	A
HELB/MELB Component - Carbon Steel, Protected (steel curbs, floor drains, shields, etc.)	Direct Flow Flood Protection Structural Support for Safety Related	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Structural Monitoring Program	III.A6.2-a	3.5.1-22	580, C
HELB/MELB Component - Concrete, Protected (concrete / masonry wall, curbs, etc.)	Direct Flow Flood Protection	Concrete	Plant Indoor Air (Ext)	Cracking	Structural Monitoring Program	III.A6.3-a	3.5.1-24	A
				Cracking and Expansion	Structural Monitoring Program (XI.S6)	III.A6.1-c	3.5.1-22	A
				Cracking, Loss of Bond/Material	Structural Monitoring Program (XI.S6)	III.A6.1-d	3.5.1-22	A
				Loss of Material	Structural Monitoring Program (XI.S6)	III.A6.1-e	3.5.1-22	A
				Reduction in Conc Anchor Capacity	Structural Monitoring Program (XI.S6)	III.B5.2-a	3.5.1-29	A

Table 3.5.2-8 Structures and Component Supports - Miscellaneous Structural and Bulk Commodities - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Built-Up Roofing - Auxiliary Bldg - Tarred, Exposed	Shelter/ Protection	Built-up Roofing	Atmosphere/ Weather (Ext)	Change in Material Properties	Structural Monitoring Program			J
				Loss of Form	Structural Monitoring Program			J
				Loss of Material	Structural Monitoring Program			J
Built-Up Roofing - Discharge Structure - Tarred, Exposed	Structural Support for Non-Safety Related	Built-up Roofing	Atmosphere/ Weather (Ext)	Change in Material Properties	Structural Monitoring Program			J
				Loss of Form	Structural Monitoring Program			J
				Loss of Material	Structural Monitoring Program			J
Built-Up Roofing - Intake Structure Bldg - Tarred, Exposed	Shelter/ Protection	Built-up Roofing	Atmosphere/ Weather (Ext)	Change in Material Properties	Structural Monitoring Program			J
				Loss of Form	Structural Monitoring Program			J
				Loss of Material	Structural Monitoring Program			J
Built-Up Roofing - Switch Yard Relay House Bldg - Tarred, Exposed	Structural Support for Regulated Events	Built-up Roofing	Atmosphere/ Weather (Ext)	Change in Material Properties	Structural Monitoring Program			J
				Loss of Form	Structural Monitoring Program			J
				Loss of Material	Structural Monitoring Program			J

Table 3.5.2-8 Structures and Component Supports - Miscellaneous Structural and Bulk Commodities - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Built-Up Roofing - Water Treatment Bldg - Tarred, Exposed	Structural Support for Non-Safety Related	Built-up Roofing	Atmosphere/ Weather (Ext)	Change in Material Properties	Structural Monitoring Program			J
				Loss of Form	Structural Monitoring Program			J
				Loss of Material	Structural Monitoring Program			J
Crane - Auxiliary Bldg - Carbon Steel, Protected	Structural Support for Safety Related	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Overhead Load Handling Systems Inspection Program	VII.B.1-b	3.3.1-16	B
						VII.B.2-a	3.3.1-16	B
Crane - Containment Bldg - Carbon Steel, Protected	Structural Support for Safety Related	Carbon Steel	Containment Air (Ext)	Loss of Material	Overhead Load Handling Systems Inspection Program	VII.B.1-b	3.3.1-16	B
						VII.B.2-a	3.3.1-16	B
Crane Lift Device - Containment Bldg - Carbon Steel, Protected	Structural Support for Non-Safety Related	Carbon Steel	Containment Air (Ext)	Loss of Material	Overhead Load Handling Systems Inspection Program	VII.B.1-b	3.3.1-16	D
Crane Support - Auxiliary Bldg - Carbon Steel, Protected	Structural Support for Non-Safety Related Structural Support for Safety Related	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Overhead Load Handling Systems Inspection Program	VII.B.1-b	3.3.1-16	B

Table 3.5.2-8 Structures and Component Supports - Miscellaneous Structural and Bulk Commodities - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Crane Support - Containment Bldg - Carbon Steel, Protected	Structural Support for Non-Safety Related Structural Support for Safety Related	Carbon Steel	Containment Air (Ext)	Loss of Material	Overhead Load Handling Systems Inspection Program	VII.B.1-b	3.3.1-16	B
Fire Barrier - Auxiliary Bldg - Carbon Steel, Protected	Fire Barrier	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Fire Protection Program	VII.G.3-d	3.3.1-20	591, B

Table 3.5.2-8 Structures and Component Supports - Miscellaneous Structural and Bulk Commodities - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Fire Barrier - Auxiliary Bldg - Concrete, Exposed	Fire Barrier	Concrete	Atmosphere/ Weather (Ext)	Change in Material Properties	Fire Protection Program	VII.G.3-b	3.3.1-30	B
					Structural Monitoring Program	VII.G.3-b	3.3.1-30	A
				Cracking	Fire Protection Program	VII.G.3-b	3.3.1-30	B
					Structural Monitoring Program	VII.G.3-b	3.3.1-30	A
				Cracking and Expansion	Fire Protection Program	VII.G.3-b	3.3.1-30	B
					Structural Monitoring Program	VII.G.3-b	3.3.1-30	A
				Cracking, Loss of Bond/Material	Fire Protection Program	VII.G.3-c	3.3.1-30	B
					Structural Monitoring Program	VII.G.3-c	3.3.1-30	A
				Loss of Material	Fire Protection Program	VII.G.3-b	3.3.1-30	B
					Structural Monitoring Program	VII.G.3-b	3.3.1-30	A

Table 3.5.2-8 Structures and Component Supports - Miscellaneous Structural and Bulk Commodities - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Fire Barrier - Auxiliary Bldg - Concrete, Protected	Fire Barrier	Concrete	Plant Indoor Air (Ext)	Cracking	Fire Protection Program	VII.G.3-b	3.3.1-30	B
					Structural Monitoring Program	VII.G.3-b	3.3.1-30	A
				Cracking and Expansion	Fire Protection Program	VII.G.3-b	3.3.1-30	B
					Structural Monitoring Program	VII.G.3-b	3.3.1-30	A
				Cracking, Loss of Bond/Material	Fire Protection Program	VII.G.3-c	3.3.1-30	B
					Structural Monitoring Program	VII.G.3-c	3.3.1-30	A
				Loss of Material	Fire Protection Program	VII.G.3-b	3.3.1-30	H
					Structural Monitoring Program	VII.G.3-b	3.3.1-30	H
Fire Barrier - Auxiliary Bldg - Fire Stop, Protected	Fire Barrier	Fire Stops [Sealant / Maranite]	Plant Indoor Air (Ext)	Cracking / Delamination	Fire Protection Program	VII.G.3-a	3.3.1-20	592, B
						VII.G.4-a	3.3.1-20	592, B
				Loss of Material	Fire Protection Program	VII.G.3-a	3.3.1-20	H
						VII.G.4-a	3.3.1-20	H
				Separation	Fire Protection Program	VII.G.3-a	3.3.1-20	592, B
						VII.G.4-a	3.3.1-20	592, B

Table 3.5.2-8 Structures and Component Supports - Miscellaneous Structural and Bulk Commodities - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Fire Barrier - Auxiliary Bldg - Fire Wrap, Protected	Fire Barrier	Fire Wraps [Mineral-Wool Batts]	Plant Indoor Air (Ext)	Cracking	Fire Protection Program			J
				Loss of Material	Fire Protection Program			J
Fire Barrier - Containment Bldg - Carbon Steel, Protected	Fire Barrier	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Fire Protection Program	VII.G.5-c	3.3.1-20	B
Fire Barrier - Containment Bldg - Concrete, Exposed	Fire Barrier	Concrete	Atmosphere/ Weather (Ext)	Change in Material Properties	Containment Inservice Inspection Program	II.A1.1-b	3.5.1-07	A
						II.A1.1-c	3.5.1-07	A
				Cracking	Containment Inservice Inspection Program	II.A1.1-a	3.5.1-16	A
					Structural Monitoring Program	II.A1.1-f	3.5.1-08	A
				Cracking and Expansion	Containment Inservice Inspection Program	II.A1.1-d	3.5.1-16	A
				Cracking, Loss of Bond/Material	Containment Inservice Inspection Program	II.A1.1-e	3.5.1-07	A
				Loss of Material	Containment Inservice Inspection Program	II.A1.1-a	3.5.1-16	A
						II.A1.1-c	3.5.1-07	A
Fire Barrier - Intake Structure Bldg - Carbon Steel, Protected	Fire Barrier	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Fire Protection Program	VII.G.1-d	3.3.1-20	591, B

Table 3.5.2-8 Structures and Component Supports - Miscellaneous Structural and Bulk Commodities - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Fire Barrier - Intake Structure Bldg - Concrete, Exposed	Fire Barrier	Concrete	Atmosphere/ Weather (Ext)	Change in Material Properties	Fire Protection Program	VII.G.1-b	3.3.1-30	B
					Structural Monitoring Program	VII.G.1-b	3.3.1-30	A
				Cracking	Fire Protection Program	VII.G.1-b	3.3.1-30	B
					Structural Monitoring Program	VII.G.1-b	3.3.1-30	A
				Cracking and Expansion	Fire Protection Program	VII.G.1-b	3.3.1-30	B
					Structural Monitoring Program	VII.G.1-b	3.3.1-30	A
				Cracking, Loss of Bond/Material	Fire Protection Program	VII.G.1-c	3.3.1-30	B
					Structural Monitoring Program	VII.G.1-c	3.3.1-30	A
				Loss of Material	Fire Protection Program	VII.G.1-b	3.3.1-30	B
					Structural Monitoring Program	VII.G.1-b	3.3.1-30	A
Fire Barrier - Intake Structure Bldg - Fire Stop, Protected	Fire Barrier	Fire Stops [Sealant / Maranite]	Plant Indoor Air (Ext)	Cracking / Delamination	Fire Protection Program	VII.G.1-a	3.3.1-20	592, B
				Loss of Material	Fire Protection Program	VII.G.1-a	3.3.1-20	H
				Separation	Fire Protection Program	VII.G.1-a	3.3.1-20	592, B

Table 3.5.2-8 Structures and Component Supports - Miscellaneous Structural and Bulk Commodities - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Fire Barrier - Intake Structure Bldg - Fire Wrap, Protected	Fire Barrier	Fire Wraps [Mineral-Wool Batts]	Plant Indoor Air (Ext)	Cracking	Fire Protection Program			J
				Loss of Material	Fire Protection Program			J
Fire Barrier - Turbine Bldg - Carbon Steel, Protected	Fire Barrier	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Fire Protection Program	VII.G.2-d	3.3.1-20	B
Fire Barrier - Turbine Bldg - Concrete, Exposed	Fire Barrier	Concrete	Atmosphere/ Weather (Ext)	Change in Material Properties	Fire Protection Program	VII.G.2-b	3.3.1-30	B
					Structural Monitoring Program	VII.G.2-b	3.3.1-30	A
				Cracking	Fire Protection Program	VII.G.2-b	3.3.1-30	B
					Structural Monitoring Program	VII.G.2-b	3.3.1-30	A
				Cracking and Expansion	Fire Protection Program	VII.G.2-b	3.3.1-30	B
					Structural Monitoring Program	VII.G.2-b	3.3.1-30	A
				Cracking, Loss of Bond/Material	Fire Protection Program	VII.G.2-c	3.3.1-30	B
					Structural Monitoring Program	VII.G.2-c	3.3.1-30	A
				Loss of Material	Fire Protection Program	VII.G.2-b	3.3.1-30	B
					Structural Monitoring Program	VII.G.2-b	3.3.1-30	A

Table 3.5.2-8 Structures and Component Supports - Miscellaneous Structural and Bulk Commodities - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Fire Barrier - Turbine Bldg - Concrete, Protected	Fire Barrier	Concrete	Plant Indoor Air (Ext)	Cracking	Fire Protection Program	VII.G.2-b	3.3.1-30	B
					Structural Monitoring Program	VII.G.2-b	3.3.1-30	A
				Cracking and Expansion	Fire Protection Program	VII.G.2-b	3.3.1-30	B
					Structural Monitoring Program	VII.G.2-b	3.3.1-30	A
				Cracking, Loss of Bond/Material	Fire Protection Program	VII.G.2-c	3.3.1-30	B
					Structural Monitoring Program	VII.G.2-c	3.3.1-30	A
				Loss of Material	Fire Protection Program	VII.G.2-b	3.3.1-30	B
					Structural Monitoring Program	VII.G.2-b	3.3.1-30	A
Fire Barrier - Turbine Bldg - Fire Stop, Protected	Fire Barrier	Fire Stops [Sealant / Maranite]	Plant Indoor Air (Ext)	Cracking / Delamination	Fire Protection Program	VII.G.2-a	3.3.1-20	592, B
				Loss of Material	Fire Protection Program	VII.G.2-a	3.3.1-20	H
				Separation	Fire Protection Program	VII.G.2-a	3.3.1-20	592, B
Fire Barrier - Turbine Bldg - Fire Wrap, Protected	Fire Barrier	Fire Wraps [Mineral-Wool Batts]	Plant Indoor Air (Ext)	Cracking	Fire Protection Program			J
				Loss of Material	Fire Protection Program			J

Table 3.5.2-8 Structures and Component Supports - Miscellaneous Structural and Bulk Commodities - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Fire Barrier - Water Treatment Bldg - Concrete, Exposed	Fire Barrier	Concrete	Atmosphere/ Weather (Ext)	Change in Material Properties	Fire Protection Program	VII.G.2-b	3.3.1-30	B
					Structural Monitoring Program	VII.G.2-b	3.3.1-30	A
				Cracking	Fire Protection Program	VII.G.2-b	3.3.1-30	B
					Structural Monitoring Program	VII.G.2-b	3.3.1-30	A
				Cracking and Expansion	Fire Protection Program	VII.G.2-b	3.3.1-30	B
					Structural Monitoring Program	VII.G.2-b	3.3.1-30	A
				Cracking, Loss of Bond/Material	Fire Protection Program	VII.G.2-c	3.3.1-30	B
					Structural Monitoring Program	VII.G.2-c	3.3.1-30	A
				Loss of Material	Fire Protection Program	VII.G.2-b	3.3.1-30	B
					Structural Monitoring Program	VII.G.2-b	3.3.1-30	A

Table 3.5.2-8 Structures and Component Supports - Miscellaneous Structural and Bulk Commodities - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Fire Barrier - Water Treatment Bldg - Concrete, Protected	Fire Barrier	Concrete	Plant Indoor Air (Ext)	Cracking	Fire Protection Program	VII.G.2-b	3.3.1-30	B
					Structural Monitoring Program	VII.G.2-b	3.3.1-30	A
				Cracking and Expansion	Fire Protection Program	VII.G.2-b	3.3.1-30	B
					Structural Monitoring Program	VII.G.2-b	3.3.1-30	A
				Cracking, Loss of Bond/Material	Fire Protection Program	VII.G.2-c	3.3.1-30	B
					Structural Monitoring Program	VII.G.2-c	3.3.1-30	A
				Loss of Material	Fire Protection Program	VII.G.2-b	3.3.1-30	B
					Structural Monitoring Program	VII.G.2-b	3.3.1-30	A
Fire Barrier - Water Treatment Bldg - Fire Stop, Protected	Fire Barrier	Fire Stops [Sealant / Maranite]	Plant Indoor Air (Ext)	Cracking / Delamination	Fire Protection Program	VII.G.2-a	3.3.1-20	592, B
				Loss of Material	Fire Protection Program	VII.G.2-a	3.3.1-20	H
				Separation	Fire Protection Program	VII.G.2-a	3.3.1-20	592, B
Fire Barrier - Water Treatment Bldg - Fire Wrap, Protected	Fire Barrier	Fire Wraps [Mineral-Wool Batts]	Plant Indoor Air (Ext)	Cracking	Fire Protection Program			J
				Loss of Material	Fire Protection Program			J

Table 3.5.2-8 Structures and Component Supports - Miscellaneous Structural and Bulk Commodities - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Flood Barrier - Auxiliary Bldg - Elastomer, Protected	Flood Protection	Elastomers	Plant Indoor Air (Ext)	Change in Material Properties	Structural Monitoring Program			J
				Cracking	Structural Monitoring Program			J
Flood Barrier - Turbine Building - Elastomer, Protected	Flood Protection	Elastomers	Plant Indoor Air (Ext)	Change in Material Properties	Structural Monitoring Program			J
				Cracking	Structural Monitoring Program			J
HELB/MELB & EQ Civil/ Structural Component - Auxiliary Bldg - Elastomer, Protected	Direct Flow Flood Protection	Elastomers	Plant Indoor Air (Ext)	Change in Material Properties	Structural Monitoring Program			J
				Cracking	Structural Monitoring Program			J
HELB/MELB & EQ Civil/ Structural Component - Intake Structure Bldg - Elastomer, Protected	Direct Flow Flood Protection	Elastomers	Plant Indoor Air (Ext)	Change in Material Properties	Structural Monitoring Program			J
				Cracking	Structural Monitoring Program			J

Table 3.5.2-8 Structures and Component Supports - Miscellaneous Structural and Bulk Commodities - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Mech General Component Support - Containment Bldg - Elastomer, Protected	Structural Support for Safety Related	Elastomers	Containment Air (Ext)	Change in Material Properties	Structural Monitoring Program			J
				Cracking	Structural Monitoring Program			J
Riprap - Yard - Soil, Submerged	Structural Support for Non-Safety Related	Soil	Raw Water (Ext)	Loss of Material/Form	Structural Monitoring Program	III.A6.4-a	3.5.1-22	A
Roof Flashing - Auxiliary Bldg - Galvanized, Exposed	Shelter/ Protection	Galvanized	Atmosphere/ Weather (Ext)	Loss of Material	Structural Monitoring Program	III.A3.2-a	3.5.1-20	A
Roof Flashing - Intake Structure Bldg - Galvanized, Exposed	Shelter/ Protection	Galvanized	Atmosphere/ Weather (Ext)	Loss of Material	Structural Monitoring Program	III.A6.2-a	3.5.1-22	A
Roof Flashing - Switchyard Relay House - Galvanized, Exposed	Structural Support for Regulated Events	Galvanized	Atmosphere/ Weather (Ext)	Loss of Material	Structural Monitoring Program	III.A3.2-a	3.5.1-20	A
Seal, Gasket or Filler - Auxiliary Bldg - Elastomer, Exposed	Shelter/ Protection	Elastomers	Atmosphere/ Weather (Ext)	Cracking	Structural Monitoring Program			J

Table 3.5.2-8 Structures and Component Supports - Miscellaneous Structural and Bulk Commodities - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Seal, Gasket or Filler - Auxiliary Bldg - Elastomer, Protected	Direct Flow Expansion / Separation Flood Protection Shelter/ Protection	Elastomers	Plant Indoor Air (Ext)	Change in Material Properties	Structural Monitoring Program	III.B4.2-a	3.5.1-29	593, A
				Cracking	Structural Monitoring Program	III.B4.2-a	3.5.1-29	593, A
Seal, Gasket or Filler - Containment Bldg - Elastomer, Protected	Expansion / Separation Flood Protection Shelter/ Protection	Elastomers	Containment Air (Ext)	Change in Material Properties	Structural Monitoring Program			J
				Cracking	Structural Monitoring Program			J
Seal, Gasket or Filler - Discharge Structure - Elastomer, Exposed	Structural Support for Non-Safety Related	Elastomers	Atmosphere/ Weather (Ext)	Change in Material Properties	Structural Monitoring Program			J
				Cracking	Structural Monitoring Program			J
Seal, Gasket or Filler - Discharge Structure - Elastomer, Protected	Structural Support for Non-Safety Related	Elastomers	Plant Indoor Air (Ext)	Change in Material Properties	Structural Monitoring Program			J
				Cracking	Structural Monitoring Program			J
Seal, Gasket or Filler - Intake Structure Bldg - Elastomer, Exposed	Shelter/ Protection	Elastomers	Atmosphere/ Weather (Ext)	Change in Material Properties	Structural Monitoring Program			J
				Cracking	Structural Monitoring Program			J

Table 3.5.2-8 Structures and Component Supports - Miscellaneous Structural and Bulk Commodities - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Seal, Gasket or Filler - Switchyard Relay House/ Switchgear/ Safeguard Bldg - Elastomer, Exposed	Shelter / Protection Structural Support for Regulated Events	Elastomers	Atmosphere/ Weather (Ext)	Change in Material Properties	Structural Monitoring Program			J
				Cracking	Structural Monitoring Program			J
Seal, Gasket or Filler - Turbine Bldg - Elastomer, Exposed	Structural Support for Non-Safety Related	Elastomers	Atmosphere/ Weather (Ext)	Change in Material Properties	Structural Monitoring Program			J
				Cracking	Structural Monitoring Program			J
Seal, Gasket or Filler - Turbine Bldg - Elastomer, Protected	Flood Protection Shelter/ Protection	Elastomers	Plant Indoor Air (Ext)	Change in Material Properties	Structural Monitoring Program			J
				Cracking	Structural Monitoring Program			J

Table 3.5.2-9 Structures and Component Supports - Switchyard and Yard Structures - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Building Framing - Safeguard Bus/Switchgear - Carbon Steel, Protected (floor beam, panels, welds, threaded fasteners, concrete expansion bolt, etc.)	Structural Support for Regulated Events	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Structural Monitoring Program	III.A3.2-a	3.5.1-20	580, A
Building Framing - Switchyard - Concrete, Below Grade (grade beam, footing, foundation, slab, grout, reinforcement, trenches, cable pits, tunnels, etc.)	Structural Support for Regulated Events	Concrete	Soil (Ext)	Loss of Material	None Required	III.A3.1-a	3.5.1-20	571, A
						III.A3.1-g	3.5.1-21	544, A
				Change in Material Properties	None Required	III.A3.1-b	3.5.1-20	557, A
						III.A3.1-g	3.5.1-21	544, A
				Cracking and Expansion	None Required	III.A3.1-c	3.5.1-20	543, A
				Cracking, Loss of Bond/Matl	None Required	III.A3.1-e	3.5.1-21	544, A
				Cracking	None Required	III.A3.1-h	3.5.1-25	547, A

Table 3.5.2-9 Structures and Component Supports - Switchyard and Yard Structures - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Building Framing - Switchyard - Concrete, Exposed (masonry/ concrete wall, grout, reinforcements, foundations, concrete around expansion & grouted anchors, bus supports, etc.)	Structural Support for Regulated Events	Concrete	Atmosphere/ Weather (Ext)	Change in Material Properties	Structural Monitoring Program	III.A3.1-b	3.5.1-20	A
						III.A3.1-f	3.5.1-20	544, A
				Cracking	Structural Monitoring Program	III.A3.1-a	3.5.1-20	A
						III.A3.3-a	3.5.1-24	A
						III.A3.1-h	3.5.1-25	A
				Cracking and Expansion	Structural Monitoring Program	III.A3.1-c	3.5.1-20	A
				Cracking, Loss of Bond/Matl	Structural Monitoring Program	III.A3.1-d	3.5.1-20	A
				Loss of Material	Structural Monitoring Program	III.A3.1-a	3.5.1-20	A
						III.A3.1-f	3.5.1-20	A
				Reduction in Concrete Anchor Capacity	Structural Monitoring Program	III.B5.2-a	3.5.1-29	A

Table 3.5.2-9 Structures and Component Supports - Switchyard and Yard Structures - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Building Framing - Switchyard - Concrete, Protected (masonry roof bearing walls, reinforcements, concrete around expansion & grouted anchors, etc.)	Structural Support for Regulated Events	Concrete	Plant Indoor Air (Ext)	Cracking	Structural Monitoring Program	III.A3.1-h	3.5.1-25	A
						III.A3.3-a	3.5.1-24	A
				Cracking and Expansion	Structural Monitoring Program	III.A3.1-c	3.5.1-20	A
				Cracking, Loss of Bond/Matl	Structural Monitoring Program	III.A3.1-d	3.5.1-20	A
				Loss of Material	Structural Monitoring Program	III.A3.1-f	3.5.1-20	A
				Loss of Strength	Structural Monitoring Program	III.A3.1-b	3.5.1-20	A
				Reduction in Concrete Anchor Capacity	Structural Monitoring Program	III.B5.2-a	3.5.1-29	A
Metal Roofing & Siding - Switchgear & Safeguard Bus - Carbon Steel, Exposed (flashing)	Structural Support for Regulated Events	Galvanized	Atmosphere/ Weather (Ext)	Loss of Material	Structural Monitoring Program	III.A3.2-a	3.5.1-20	581, C

Table 3.5.2-9 Structures and Component Supports - Switchyard and Yard Structures - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Missile Shield - Yard - Concrete, Exposed (Tank vault roof, pavement over buried piping)	Missile Barrier	Concrete	Atmosphere/ Weather (Ext)	Change in Material Properties	Structural Monitoring Program	III.A3.1-b	3.5.1-20	A
						III.A3.1-f	3.5.1-20	A
				Cracking	Structural Monitoring Program	III.A3.1-a	3.5.1-20	A
						III.A3.1-h	3.5.1-25	A
				Cracking and Expansion	Structural Monitoring Program	III.A3.1-c	3.5.1-20	A
				Cracking, Loss of Bond/Matl	Structural Monitoring Program	III.A3.1-d	3.5.1-20	A
				Loss of Material	Structural Monitoring Program	III.A3.1-a	3.5.1-20	A
					Structural Monitoring Program	III.A3.1-f	3.5.1-20	A
				Reduction in Concrete Anchor Capacity	Structural Monitoring Program	III.B5.2-a	3.5.1-29	A

Table 3.5.2-9 Structures and Component Supports - Switchyard and Yard Structures - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Tank Foundations - Building & Yard - Concrete, Below Grade (concrete foundation)	Structural Support for Regulated Events Structure Functional Support	Concrete	Soil (Ext)	Loss of Material	None Required	III.A8.1-a	3.5.1-20	571, A
						III.A8.1-e	3.5.1-21	544, A
				Change in Material Properties	None Required	III.A8.1-b	3.5.1-20	557, A
						III.A8.1-e	3.5.1-21	544, A
				Cracking and Expansion	None Required	III.A8.1-c	3.5.1-20	543, A
				Cracking, Loss of Bond/Matl	None Required	III.A8.1-d	3.5.1-21	544, A
				Cracking	None Required	III.A8.1-f	3.5.1-25	547, A
Tank Foundations - Building & Yard - Concrete, Exposed (concrete foundation)	Structural Support for Regulated Events Structural Support for Safety Related	Concrete	Atmosphere / Weather	Cracking	Structural Monitoring Program	III.A8.1-a	3.5.1-20	A
				Loss of Material	Structural Monitoring Program	III.A8.1-a	3.5.1-20	A
				Change in Material Properties	Structural Monitoring Program	III.A8.1-b	3.5.1-20	A
				Cracking and Expansion	Structural Monitoring Program	III.A8.1-c	3.5.1-20	A
				Cracking, Loss of Bond/Matl	Structural Monitoring Program	III.A8.1-d	3.5.1-21	A

Table 3.5.2-10 Structures and Component Supports - Turbine Building - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Building Framing - Boiler Buildings Area - Carbon Steel, Protected (column, hanger, beam, truss, decking, platform, floor grating or plate, catwalk, threaded fastener, concrete expansion bolt, column base plate, welds, etc.)	Structural Support for Non-Safety Related	Carbon Steel	Plant Indoor air (Ext)	Loss of Material	Structural Monitoring Program	III.A3.2-a	3.5.1-20	A
Building Framing - Boiler Buildings Area - Concrete, Below Grade (wall, pedestal, grade beam, footing, foundation, slab, grout, reinforcement, cable pits, tunnels, etc.)	Structural Support for Non-Safety Related	Concrete	Soil (Ext)	Loss of Material	None Required	III.A3.1-a	3.5.1-20	571, A
						III.A3.1-g	3.5.1-21	544, A
				Change in Material Properties	None Required	III.A3.1-b	3.5.1-20	557, A
						III.A3.1-g	3.5.1-21	544, A
				Cracking and Expansion	None Required	III.A3.1-c	3.5.1-20	543, A
				Cracking, Loss of Bond/Matl	None Required	III.A3.1-e	3.5.1-21	544, A
				Cracking	None Required	III.A3.1-h	3.5.1-25	547, A

Table 3.5.2-10 Structures and Component Supports - Turbine Building - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Building Framing - Boiler Buildings Area - Concrete, Exposed (foundations, masonry/ concrete wall, grout, reinforcements, concrete around expansion & grouted anchors, etc.)	Structural Support for Non-Safety Related	Concrete	Atmosphere / Weather (Ext)	Cracking	Structural Monitoring Program	III.A3.1-a	3.5.1-20	A
						III.A3.1-h	3.5.1-25	A
						III.A3.3-a	3.5.1-24	A
				Loss of Material	Structural Monitoring Program	III.A3.1-a	3.5.1-20	A
						III.A3.1-f	3.5.1-20	A
				Change in Material Properties	Structural Monitoring Program	III.A3.1-b	3.5.1-20	A
						III.A3.1-f	3.5.1-20	A
				Cracking and Expansion	Structural Monitoring Program	III.A3.1-c	3.5.1-20	A
				Cracking, Loss of Bond/Matl	Structural Monitoring Program	III.A3.1-d	3.5.1-20	A
				Reduction in Concrete Anchor Capacity	Structural Monitoring Program	III.B5.2-a	3.5.1-29	A

Table 3.5.2-10 Structures and Component Supports - Turbine Building - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Building Framing - Boiler Buildings Area - Concrete, Protected (foundations, concrete & masonry wall, column, beam, floor/roof slab, grout, reinforcements, concrete around expansion & grouted anchors, etc.)	Structural Support for Non-Safety Related	Concrete	Plant Indoor Air (Ext)	Loss of Strength	Structural Monitoring Program	III.A3.1-b	3.5.1-20	A
				Cracking and Expansion	Structural Monitoring Program	III.A3.1-c	3.5.1-20	A
				Cracking, Loss of Bond/Matl	Structural Monitoring Program	III.A3.1-d	3.5.1-20	A
				Loss of Material	Structural Monitoring Program	III.A3.1-f	3.5.1-20	A
				Cracking	Structural Monitoring Program	III.A3.1-h	3.5.1-25	A
						III.A3.3-a	3.5.1-24	A
				Reduction in Concrete Anchor Capacity	Structural Monitoring Program	III.B5.2-a	3.5.1-29	A

Table 3.5.2-10 Structures and Component Supports - Turbine Building - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Building Framing - Carbon Steel, Protected (column, beam, truss, decking, platform, floor grating or plate, catwalk, bracing, threaded fastener, concrete expansion bolt, column base plate, welds, etc.)	Flood Protection HELB Shielding Pipe Whip Restraint Shelter/ Protection Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Structural Monitoring Program	III.A3.2-a	3.5.1-20	580, A

Table 3.5.2-10 Structures and Component Supports - Turbine Building - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Building Framing - Concrete, Below Grade (wall, pedestal, grade beam, footing, foundation, slab, grout, reinforcement, cable pits, tunnels, etc.)	Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related	Concrete	Soil (Ext)	Loss of Material	None Required	III.A3.1-a	3.5.1-20	571, A
						III.A3.1-g	3.5.1-21	544, A
				Change in Material Properties	None Required	III.A3.1-b	3.5.1-20	557, A
						III.A3.1-g	3.5.1-21	544, A
				Cracking and Expansion	None Required	III.A3.1-c	3.5.1-20	543, A
				Cracking, Loss of Bond/Matl	None Required	III.A3.1-e	3.5.1-21	544, A
				Cracking	None Required	III.A3.1-h	3.5.1-25	547, A

Table 3.5.2-10 Structures and Component Supports - Turbine Building - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Building Framing - Concrete, Exposed (foundations, masonry/ concrete wall, grout, reinforcements, concrete around expansion & grouted anchors, etc.)	Flood Protection Shelter/ Protection Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related	Concrete	Atmosphere/ Weather (Ext)	Change in Material Properties	Structural Monitoring Program	III.A3.1-b	3.5.1-20	A
						III.A3.1-f	3.5.1-20	A
				Cracking	Structural Monitoring Program	III.A3.3-a	3.5.1-24	A
						III.A3.1-a	3.5.1-20	A
						III.A3.1-h	3.5.1-25	A
				Cracking and Expansion	Structural Monitoring Program	III.A3.1-c	3.5.1-20	A
				Cracking, Loss of Bond/Matl	Structural Monitoring Program	III.A3.1-d	3.5.1-20	A
				Loss of Material	Structural Monitoring Program	III.A3.1-a	3.5.1-20	A
						III.A3.1-f	3.5.1-20	A
				Reduction in Concrete Anchor Capacity	Structural Monitoring Program	III.B5.2-a	3.5.1-29	A

Table 3.5.2-10 Structures and Component Supports - Turbine Building - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Building Framing - Concrete, Protected (foundations, concrete & masonry wall, column, beam, floor/roof slab, grout, reinforcements, concrete around expansion & grouted anchors, etc.)	Flood Protection HELB Shielding Missile Barrier Shelter/ Protection Structural Support for Non-Safety Related Structural Support for Regulated Events Structural Support for Safety Related	Concrete	Plant Indoor Air (Ext)	Cracking	Structural Monitoring Program	III.A3.3-a	3.5.1-24	A
						III.A3.1-h	3.5.1-25	A
				Cracking and Expansion	Structural Monitoring Program	III.A3.1-c	3.5.1-20	A
				Cracking, Loss of Bond/Matl	Structural Monitoring Program	III.A3.1-d	3.5.1-20	A
				Loss of Material	Structural Monitoring Program	III.A3.1-f	3.5.1-20	A
				Loss of Strength	Structural Monitoring Program	III.A3.1-b	3.5.1-20	A
				Reduction in Concrete Anchor Capacity	Structural Monitoring Program	III.B5.2-a	3.5.1-29	A
Building Framing - Sliding Material, Protected (vertical supports with sliding surfaces)	Expansion / Separation	Bronze	Plant Indoor Air (Ext)	Loss of Material	None Required	III.A4.2-b	3.5.1-20	586, I

Table 3.5.2-10 Structures and Component Supports - Turbine Building - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Building Framing - Water Treatment Area - Carbon Steel, Protected (column, beam, bracing, threaded fastener, weld, etc.)	Structural Support for Non-Safety Related Structural Support for Regulated Events	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Structural Monitoring Program	III.A3.2-a	3.5.1-20	580, A
Building Framing - Water Treatment Area - Concrete, Below Grade (wall, pedestal, grade beam, footing, foundation, slab, grout, reinforcement, trenches, cable pits, tunnels, etc.)	Structural Support for Non-Safety Related Structural Support for Regulated Events	Concrete	Soil (Ext)	Loss of Material	None Required	III.A3.1-a	3.5.1-20	571, A
						III.A3.1-g	3.5.1-21	544, A
				Change in Material Properties	None Required	III.A3.1-b	3.5.1-20	557, A
						III.A3.1-g	3.5.1-21	544, A
				Cracking and Expansion	None Required	III.A3.1-c	3.5.1-20	543, A
				Cracking, Loss of Bond/Matl	None Required	III.A3.1-e	3.5.1-21	544, A
				Cracking	None Required	III.A3.1-h	3.5.1-25	547, A

Table 3.5.2-10 Structures and Component Supports - Turbine Building - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Building Framing - Water Treatment Area - Concrete, Exposed (foundations, wall, grout, reinforcements, concrete around expansion & grouted anchors, etc.)	Structural Support for Non-Safety Related Structural Support for Regulated Events	Concrete	Atmosphere/ Weather (Ext)	Change in Material Properties	Structural Monitoring Program	III.A3.1-b	3.5.1-20	A
						III.A3.1-f	3.5.1-20	A
				Cracking	Structural Monitoring Program	III.A3.3-a	3.5.1-24	A
						III.A3.1-a	3.5.1-20	A
						III.A3.1-h	3.5.1-25	A
				Cracking and Expansion	Structural Monitoring Program	III.A3.1-c	3.5.1-20	A
				Cracking, Loss of Bond/Matl	Structural Monitoring Program	III.A3.1-d	3.5.1-20	A
				Loss of Material	Structural Monitoring Program	III.A3.1-a	3.5.1-20	A
						III.A3.1-f	3.5.1-20	A
				Reduction in Concrete Anchor Capacity	Structural Monitoring Program	III.B5.2-a	3.5.1-29	A

Table 3.5.2-10 Structures and Component Supports - Turbine Building - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Building Framing - Water Treatment Area - Concrete, Protected (foundations, concrete/ masonry wall, beam, floor slab, grout, reinforcements, concrete around expansion & grouted anchors, etc.)	Structural Support for Regulated Events Structural Support for Non-Safety Related	Concrete	Plant Indoor Air (Ext)	Cracking	Structural Monitoring Program	III.A3.3-a	3.5.1-24	A
						III.A3.1-h	3.5.1-25	A
				Cracking and Expansion	Structural Monitoring Program	III.A3.1-c	3.5.1-20	A
				Cracking, Loss of Bond/Matl	Structural Monitoring Program	III.A3.1-d	3.5.1-20	A
				Loss of Material	Structural Monitoring Program	III.A3.1-f	3.5.1-20	A
				Loss of Strength	Structural Monitoring Program	III.A3.1-b	3.5.1-20	A
				Reduction in Concrete Anchor Capacity	Structural Monitoring Program	III.B5.2-a	3.5.1-29	A
Flood Barrier - Carbon Steel, Protected (flood doors, hatch, standpipe)	Flood Protection Structural Support for Safety Related	Carbon Steel	Plant Indoor Air (Ext)	Loss of Leak Tightness	Structural Monitoring Program	III.A3.2-a	3.5.1-20	531, H
				Loss of Material	Structural Monitoring Program	III.A3.2-a	3.5.1-20	580, C

Table 3.5.2-10 Structures and Component Supports - Turbine Building - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
HELB/MELB Component - Carbon Steel, Protected (curbs, floor drains, pipe whip restraints, spray shields, etc.)	Flood Protection HELB Shielding Pipe Whip Restraint Structural Support for Safety Related	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Structural Monitoring Program	III.A3.2-a	3.5.1-20	580, C
HELB/MELB Component - Concrete, Protected (concrete/ masonry wall, whip restraint grout, concrete around expansion & grouted anchors, etc.)	Flood Protection HELB Shielding Pipe Whip Restraint	Concrete	Plant Indoor Air (Ext)	Cracking	Structural Monitoring Program	III.A3.3-a	3.5.1-24	A
				Cracking and Expansion	Structural Monitoring Program	III.A3.1-c	3.5.1-20	A
				Cracking, Loss of Bond/Matl	Structural Monitoring Program	III.A3.1-d	3.5.1-20	A
				Loss of Material	Structural Monitoring Program	III.A3.1-f	3.5.1-20	A
				Reduction in Conc Anchor Capacity	Structural Monitoring Program	III.B5.2-a	3.5.1-29	A
HVAC Component - Carbon Steel, Protected (Control Room vestibule door)	Fluid Pressure Boundary Structural Support for Non-Safety Related	Carbon Steel	Plant Indoor Air (Ext)	Loss of Leak Tightness	Structural Monitoring Program	III.A3.2-a	3.5.1-20	533, H
				Loss of Material	Structural Monitoring Program	III.A3.2-a	3.5.1-20	580, C

Table 3.5.2-10 Structures and Component Supports - Turbine Building - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
HVAC Component - Concrete, Protected (Control Room vestibules, concrete & masonry walls, floors, ceilings)	Fluid Pressure Boundary Structural Support for Non-Safety Related	Concrete	Plant Indoor Air (Ext)	Cracking	Structural Monitoring Program	III.A3.3-a	3.5.1-24	A
						III.A3.1-h	3.5.1-25	A
				Cracking and Expansion	Structural Monitoring Program	III.A3.1-c	3.5.1-20	A
				Cracking, Loss of Bond/Matl	Structural Monitoring Program	III.A3.1-d	3.5.1-20	A
				Loss of Material	Structural Monitoring Program	III.A3.1-f	3.5.1-20	A
				Reduction in Conc Anchor Capacity	Structural Monitoring Program	III.B5.2-a	3.5.1-29	A

Table 3.5.2-10 Structures and Component Supports - Turbine Building - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Missile Shield - concrete, Exposed (concrete and/or masonry walls protecting CCW room door and containment escape hatch	Missile Barrier	Concrete	Atmosphere/ Weather (Ext)	Change in Material Properties	Structural Monitoring Program	III.A3.1-b	3.5.1-20	A
						III.A3.1-f	3.5.1-20	A
				Cracking	Structural Monitoring Program	III.A3.3-a	3.5.1-24	A
						III.A3.1-a	3.5.1-20	A
						III.A3.1-h	3.5.1-25	A
				Cracking and Expansion	Structural Monitoring Program	III.A3.1-c	3.5.1-20	A
				Cracking, Loss of Bond/Matl	Structural Monitoring Program	III.A3.1-d	3.5.1-20	A
				Loss of Material	Structural Monitoring Program	III.A3.1-a	3.5.1-20	A
						III.A3.1-f	3.5.1-20	A
				Reduction in Conc Anchor Capacity	Structural Monitoring Program	III.B5.2-a	3.5.1-29	A
Missile Shield - Concrete, Protected (concrete/ masonry walls, floor, & roof protecting Control Room Door)	Missile Barrier	Concrete	Plant Indoor Air (Ext)	Cracking	Structural Monitoring Program	III.A3.1-h	3.5.1-25	A
				Cracking and Expansion	Structural Monitoring Program	III.A3.1-c	3.5.1-20	A
				Cracking, Loss of Bond/Matl	Structural Monitoring Program	III.A3.1-d	3.5.1-20	A
				Loss of Material	Structural Monitoring Program	III.A3.1-f	3.5.1-20	A
				Reduction in Conc Anchor Capacity	Structural Monitoring Program	III.B5.2-a	3.5.1-29	A

Table 3.5.2-10 Structures and Component Supports - Turbine Building - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Operator Access Component - Carbon Steel, Protected (stairs, floors, platforms)	Structural Support for Non-Safety Related Structural Support for Regulated Events	Carbon Steel	Plant Indoor Air (Ext)	Loss of Material	Structural Monitoring Program	III.A3.2-a	3.5.1-20	A
Operator Access Component - Concrete, Protected (stairs, floors, platforms, concrete at locations of expansion & grouted anchors, etc.)	Structural Support for Non-Safety Related Structural Support for Regulated Events	Concrete	Plant Indoor Air (Ext)	Cracking	Structural Monitoring Program	III.A3.3-a	3.5.1-24	A
						III.A3.1-h	3.5.1-25	A
				Cracking and Expansion	Structural Monitoring Program	III.A3.1-c	3.5.1-20	A
				Cracking, Loss of Bond/Matl	Structural Monitoring Program	III.A3.1-d	3.5.1-20	A
				Loss of Material	Structural Monitoring Program	III.A3.1-f	3.5.1-20	A
				Loss of Strength	Structural Monitoring Program	III.A3.1-b	3.5.1-20	A
				Reduction in Conc Anchor Capacity	Structural Monitoring Program	III.B5.2-a	3.5.1-29	A
Operator Access Component - Galvanized, Protected (stairs, floors, platforms)	Structural Support for Non-Safety Related Structural Support for Regulated Events	Galvanized	Plant Indoor Air (Ext)	Loss of Material	Structural Monitoring Program	III.A3.2-a	3.5.1-20	581, A

Notes for Tables 3.5.2-1 through 3.5.2-10

A Consistent with NUREG-1801 item for component, material, environment, and aging management program. AMP is consistent with NUREG-1801 AMP.

B Consistent with NUREG-1801 item for component, material, environment, and aging management program. AMP has exceptions to NUREG-1801 AMP.

C Component is different, but consistent with NUREG-1801 for material, environment, aging effect and AMP. AMP is consistent with NUREG-1801 AMP.

D Component is different, but consistent with NUREG-1801 for material, environment, aging effect and AMP. AMP has exceptions to NUREG-1801 AMP.

E Consistent with NUREG-1801 for material, environment, aging effect but a different AMP is credited.

F Material not in NUREG-1801 for this component.

G Environment not in NUREG-1801 for this component and material.

H Aging effect not in NUREG-1801 for this component, material and environment combination.

I Aging effect in NUREG-1801 for this component, material and environment combination is not applicable.

J Neither the component nor the material and environment combination is evaluated in NUREG-1801.

Plant-specific notes:

501 Aging effect/mechanism is applicable for SIRW Tank supports on roof of Auxiliary Building.

502 Aging effect/mechanism is applicable for permanent radiation shield support beam inside Containment Building pressurizer shed.

503 Aging effect/mechanism is applicable for new fuel racks adjacent to Spent Fuel Pool.

504 Not Used.

505 Not Used.

506 Not Used.

507 Not Used.

508 Not Used.

509 Not Used.

510 Not Used.

511 Not Used.

512 Not Used.

513 Not Used.

514 Not Used.

515 Lubrite does not require Aging Management.

516 Not Used.

517 Not Used.

518 Aging effect/mechanism is applicable to the condensate storage tank valve pit supports.

519 Not Used.

520 Not Used.

521 Not Used.

522 Not Used.

523 Spent Fuel Pool Liner is susceptible to Crevice and Pitting Corrosion. GALL Item III.A5.2-b mandates Water Chemistry Program to manage contaminants.

524 Not Used.

525 Not Used.

526 Not Used.

527 Not Used.

528 Not Used.

529 Not Used.

530 Not Used.

531 Aging Effect is applicable to Floor Doors.

532 Aging Effect is applicable to HELB Doors.

533 Aging Effect is applicable to Control Room Vestibule Doors.

534 Not Used.

535 Aging Effect is not applicable for Fuel Pool Liner since temperature threshold to initiate SCC is not present.

536 Not Used.

537 Not Used.

538 “Loss of Leak Tightness/Mechanical Wear” AERM is only applicable for Pressure Boundary Intended Function.

539 Not Used.

540 Not Used.

541 Not Used.

542 Not Used.

543 See further evaluation. Cracking and Expansion due to aggregate reaction is not applicable. Palisades concrete aggregates were tested for reactivity and accepted. Aggregate came from source known not to be reactive. Therefore, inspection of inaccessible (below grade) concrete is not required. The Structures Monitoring Program will monitor this aging effect/mechanism for accessible (exposed to weather or protected) concrete.

544 Aging effect is not applicable. Palisades does not have an aggressive environment. Reinforcing and embedment steel are in sound, dense concrete.

545 Not Used.

546 Not Used.

547 See Further Evaluation discussion. Cracking due to settlement does not apply. No de-watering system is relied upon. There is no history of settlement and NRC Systematic Evaluation Program Topic II-4.F evaluation supports this conclusion. (See FSAR Table 1-3). Therefore, inspection of inaccessible (below grade) concrete is not required. The Structures Monitoring Program will monitor this aging effect/mechanism for accessible (exposed to weather or protected) concrete.

548 Not Used.

549 Not Used.

550 Not Used.

551 Not Used.

552 Not Used.

553 Not Used.

554 Not Used.

555 Not Used.

556 Not Used.

557 PAL concrete design and construction is consistent with ACI 201.2R-77 recommendations. Concrete is dense, well cured, low permeability and adequately reinforced to control cracking. Leaching aging effect does not apply.

558 Not Used.

559 GALL Group 4-Containment internal Structures does not have entry for Masonry Walls. PAL has masonry wall enclosure inside containment.

560 Not Used.

561 Not Used.

562 Not Used.

563 Not Used.

564 Not Used.

565 Not Used.

566 Not Used.

567 Aging Effect and Mechanism is applicable to supports immersed in Borated Water in the Refueling Cavity and Water Chemistry Program controls contaminants to minimize possibility for Aging Effect and Mechanism to occur.

568 Not Used.

569 Not Used

570 Stainless Steel transfer tube with dissimilar metal weld to carbon steel penetration.

571 See further evaluation. Cracking of inaccessible (below grade) concrete due to freeze/thaw is not an AERM. Palisades water-to-cement ratio (3 -5%) and air content in the concrete mix design are per applicable ACI standards and no freeze-thaw degradation has been identified. Therefore, inspection of inaccessible (below grade) concrete is not required. The Structures Monitoring Program will monitor this aging effect/mechanism for accessible concrete exposed to weather.

572 See Further Evaluation discussion. Palisades concrete design and construction is consistent with ACI 201.2R-77 recommendations. Concrete is dense, well cured, low permeability and adequately reinforced to control cracking. Leaching aging effect does not apply. Therefore, inspection of inaccessible (below grade) concrete is not required. ASME Section XI Subsection IWL will monitor this aging effect for accessible (exposed to weather or protected) concrete.

573 See Further Evaluation discussion. This aging effect/mechanism is not applicable. Palisades does not have an aggressive ground water environment based on a long history

of ground water sampling results. Reinforcing and embedment steel are in sound, dense concrete. Therefore, inspection of inaccessible (below grade) concrete is not required. ASME Section XI Subsection IWL will monitor this aging effect/mechanism for accessible (exposed to weather or protected) concrete.

574 See Further Evaluation discussion. Cracking and Expansion due to aggregate reaction is not applicable. Palisades concrete aggregates were tested for reactivity and accepted. Aggregate came from a source known not to be reactive. Therefore, inspection of inaccessible (below grade) concrete is not required. ASME Section XI Subsection IWL will monitor this aging effect/mechanism for accessible (exposed to weather or protected) concrete.

575 See Further Evaluation discussion. Cracking of inaccessible (below grade) concrete due to freeze/thaw is not an AERM. Palisades water-to-cement ratio (3 - 5%) and air content in the concrete mix design are per applicable ACI standards, and no freeze-thaw degradation has been identified. Therefore, inspection of inaccessible (below grade) concrete is not required. ASME Section XI Subsection IWL will monitor this aging effect/mechanism for accessible (exposed to weather or protected) concrete.

576 Temperatures for reactor cavity concrete are designed to be below 165 F which is above 150 F. Further evaluation has been performed to justify acceptability.

577 See Further Evaluation discussion. For embedded containment liner, the four conditions are satisfied with regards to concrete quality, concrete inspections for cracks are performed via IWE (external), and borated water spills are infrequent and identified and cleaned up in accordance with the Boric Acid Corrosion Program.

578 Technical Specification 3.7.14.1 includes a surveillance requirement to verify minimum spent fuel pool level.

579 Carbon steel anchor bolts supporting the SFP and reactor cavity liner plates are protected from corrosive environments by the liner plates. However, since leakage of the liners could expose the anchor bolts to pool environment, the programs credited to age manage the liner plates are also credited for the anchor bolts.

580 Protective coatings not credited in managing effects of aging.

581 Galvanized material is treated the same as carbon steel. No credit is taken for the galvanized coating.

582 Cast iron is considered consistent with carbon steel and is evaluated the same, but with the additional aging effect/mechanism of loss of material due to selective leaching also evaluated.

583 GALL III.B5.1-b environment is inside PWR containment. Palisades will also manage boric acid wastage in the Auxiliary Building since it contains boric acid systems as well.

584 GALL III.A3.2-a does not address the mechanism of Boric Acid Wastage. Boric acid wastage of structural steel in the Auxiliary Building will be managed by the Boric Acid Corrosion Program.

585 GALL III.A4.2-a does not address the mechanism of Boric Acid Wastage. Boric acid wastage of structural steel in the Containment Building will be managed by the Boric Acid Corrosion Program.

586 Lubrite sliding surface for structural steel in the turbine building is a material and environment combination not addressed in the GALL. However, it is similar to GALL line item IIIA4.2-b (lubrite sliding surface for RPV support shoes in the containment building) so it is being aligned to that GALL line item. Per EPRI Structural Tools Section 9.3.1, Lubrite has no aging effects requiring management (AERM). This is contrary to GALL IIIA4.2-b that identifies lock-up due to wear as an AERM. This specific application is as a sliding seismic joint that does not see wear.

587 The primary environment for refueling cavity liner and transfer tube is containment air. However, since the refueling cavity is submerged in borated water during refueling, it is evaluated using a more conservative treated water environment category. The water chemistry program and SFP level monitoring (during refueling) well age manage the liner for Loss of Material due to crevice and pitting corrosion, consistent with GALL.

588 Stainless steel control room vestibule doors do not have any aging effects requiring management.

589 Not Used.

590 Not Used.

591 This component group also includes steel fire barriers as well as fire doors.

592 Fire stop materials consist of various sealants, including silicone, maranite and ceramic fiber insulation. Aging effect terminology used in GALL is slightly different but overall deterioration is the same (e.g., cracking due to shrinkage, separation due to shrinkage, etc.).

593 Aging effect terminology used in GALL for the Emergency Diesel Generators vibration isolation elements is slightly different, but overall deterioration is the same (e.g., cracking and change in material properties due to thermal exposure, etc.). Other elements included in this component (thermal expansion / seismic separation joint filler, gap or crack seal, etc.) are not addressed in the GALL.

594 Normal Environment is plant indoor air for manholes and valve pits. A raw water environment was conservatively evaluated to address potential ground water intrusion.

595 Palisades neutron absorbing material consists of boron carbide sheathed in stainless steel, unlike Boral which consists of boron carbide and aluminum alloy.

596 The primary environment for the Containment sump liner and screen is Containment air. However, since portions of the sump are periodically wetted, it is evaluated using a more conservative environment of treated water. The Structural Monitoring Program will age manage the Containment sump liner and screen for loss of material due to crevice and pitting corrosion.

3.6 Aging Management of Electrical and Instrumentation and Controls

3.6.1 Introduction

This section provides the results of the aging management review for those components identified in Section 2.5, Electrical and Instrumentation and Controls, as being subject to aging management review. The commodities containing those components, which are addressed in this section, are described above in Commodity Group Descriptions (Section 2.5.1)

Table 3.6.1, Summary of Aging Management Evaluations in Chapter VI of NUREG-1801 for Electrical Components, provides the summary of the programs evaluated in NUREG-1801 for the Electrical Component groups that are relied on for license renewal. This table uses the format described in Figure 3.0-1. Note that this table only includes those component groups that are applicable to a PWR.

3.6.2 Results

The following table summarizes the results of the aging management review for electrical commodities in the Electrical Components group:

Table 3.6.2-1, Electrical Components - Electrical Commodity Groups - Summary of Aging Management Evaluation

The materials that specific components are fabricated from, the environments to which components are exposed, the potential aging effects requiring management, and the aging management programs used to manage these aging effects are provided for each of the above commodities in the following subsections of Section 3.6.2.1, Materials, Environment, Aging Effects Requiring Management and Aging Management Programs:

Section 3.6.2.1.1, Commodity: Electrical cables and connections not subject to 10 CFR 50.49 EQ requirements

Section 3.6.2.1.2, Commodity: Electrical cables and connections used in instrumentation circuits not subject to 10 CFR 50.49 EQ requirements that are sensitive to reduction in conductor IR (ISG-15) (Nuclear Instrumentation and Radiation Monitoring Systems)

Section 3.6.2.1.3, Commodity: Electrical Portion of the Non-EQ Electrical and I&C Penetration Assemblies (Cables and Connections)

Section 3.6.2.1.4, Commodity: Fuse Holders (ISG-5)

Section 3.6.2.1.5, Commodity: Non-Segregated Phase Bus and Connections (ISG-17)

Section 3.6.2.1.6, Commodity: High-Voltage Transmission Conductors (ISG-2)

Section 3.6.2.1.7, Commodity: High-Voltage Switchyard Bus and Connections (ISG-2)

Section 3.6.2.1.8, Commodity: Inaccessible medium-voltage (2kV to 15kV) cables and connections not subject to 10 CFR 50.49 EQ requirements (ISG-18)

Section 3.6.2.1.9, Commodity: High-Voltage Insulators (ISG-2)

3.6.2.1 Materials, Environment, Aging Effects Requiring Management and Aging Management Programs

3.6.2.1.1 Commodity: Electrical cables and connections not subject to 10 CFR 50.49 EQ requirements

Materials

The materials of construction for this electrical cables and connections commodity are:

- Insulation materials - various organic polymers
- Connector Pins - Various Materials

Environment

The cables and connections of this commodity are exposed to the following environments:

- Adverse localized environment caused by heat, radiation, or moisture in the presence of oxygen
- Borated Water Leakage

Aging Effects Requiring Management

The following aging effects, associated with the cables and connections of this commodity, require management:

- Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced IR
- Electrical failure caused by thermal/ thermoxidative degradation of organics
- Radiation-induced oxidation
- Moisture intrusion
- Corrosion of connector contact surfaces caused by intrusion of borated water

Aging Management Programs

The following aging management programs manage the aging effects for the cables and connections of this commodity:

- Non-EQ Electrical Commodities Condition Monitoring Program
- Boric Acid Corrosion Program

3.6.2.1.2 Commodity: Electrical cables and connections used in instrumentation circuits not subject to 10 CFR 50.49 EQ requirements that are sensitive to reduction in conductor IR (ISG-15) (Nuclear Instrumentation and Radiation Monitoring Systems)

Materials

The materials of construction for this sensitive instrument cables and connections commodity are:

- Insulation materials - various organic polymers

Environment

The cables and connections of this commodity are exposed to the following environments:

- Adverse localized environment caused by heat, radiation, or moisture in the presence of oxygen

Aging Effects Requiring Management

The following aging effects, associated with this sensitive instrument cables and connections of this commodity, require management:

- Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced IR
- Electrical failure caused by thermal/ thermoxidative degradation of organics
- Radiation-induced oxidation
- Moisture intrusion

Aging Management Programs

The following aging management programs manage the aging effects for the sensitive instrument cables and connections of this commodity:

- Non-EQ Electrical Commodities Condition Monitoring Program

3.6.2.1.3 Commodity: Electrical Portion of the Non-EQ Electrical and I&C Penetration Assemblies (Cables and Connections)

Materials

The materials of construction for the electrical portion of this non-EQ penetration assembly commodity are:

- Insulation materials - various organic polymers

Environment

The electrical portions of this non-EQ penetration assembly commodity are exposed to the following environments:

- Adverse localized environment caused by heat, radiation, or moisture in the presence of oxygen

Aging Effects Requiring Management

The following aging effects, associated with the electrical portions of this non-EQ penetration assembly commodity, require management:

- Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced IR
- Electrical failure caused by thermal/ thermoxidative degradation of organics
- Radiation-induced oxidation
- Moisture intrusion

Aging Management Programs

The following aging management programs manage the aging effects for the electrical portions of this non-EQ penetration assembly commodity:

- Non-EQ Electrical Commodities Condition Monitoring Program

3.6.2.1.4 Commodity: Fuse Holders (ISG-5)

Materials

The materials of construction for this fuse holder commodity are:

- Phenolic
- Copper
- Aluminum

Environment

The components of this fuse holder commodity are exposed to the following environments:

- Adverse localized environment caused by heat, radiation, or moisture in the presence of oxygen.

Aging Effects Requiring Management

The following aging effects, associated with this fuse holder commodity, require management:

- Embrittlement
- Cracking
- Melting
- Discoloration
- Moisture intrusion
- Corrosion

Aging Management Programs

The following aging management programs manage the aging effects for this fuse holder commodity:

- None required

3.6.2.1.5 **Commodity: Non-Segregated Phase Bus and Connections (ISG-17)**

Materials

The materials of construction for this non-segregated phase bus commodity are:

- Various Metals
- Porcelain

Environment

The non-segregated phase bus in this commodity is exposed to the following environments:

- Adverse localized environment caused by heat, radiation, or moisture in the presence of oxygen.

Aging Effects Requiring Management

The following aging effects, associated with this non-segregated phase bus commodity, require management:

- Oxidation
- Loosening of bolted connections due to thermal cycling
- Corrosion due to moisture

Aging Management Programs

The following aging management programs manage the aging effects for this non-segregated phase bus commodity:

- Non-EQ Electrical Commodities Condition Monitoring Program

3.6.2.1.6 Commodity: High-Voltage Transmission Conductors (ISG-2)

Materials

The materials of construction for this high voltage transmission conductor commodity are:

- Aluminum
- Steel

Environment

The conductors of this high voltage transmission conductor commodity are exposed to the following environments:

- Atmosphere/Weather

Aging Effects Requiring Management

The following aging effects, associated with this high voltage transmission conductor commodity, require management:

- Loss of conductor strength
- Vibration

Aging Management Programs

The following aging management programs manage the aging effects for this high voltage transmission conductor commodity:

- None Required

3.6.2.1.7 Commodity: High-Voltage Switchyard Bus and Connections (ISG-2)

Materials

The materials of construction for this high voltage switchyard bus and connections commodity are:

- Aluminum
- Stainless Steel (Bolting)
- Copper

Environment

The busses and connections of this high voltage switchyard bus and connections commodity are exposed to the following environments:

- Atmosphere/Weather

Aging Effects Requiring Management

The following aging effects, associated with this high voltage switchyard bus and connections commodity, require management:

- Connection Surface Oxidation
- Vibration

Aging Management Programs

The following aging management programs manage the aging effects for the high voltage switchyard bus and connections commodity:

- None Required

3.6.2.1.8 Commodity: Inaccessible medium-voltage (2kV to 15kV) cables and connections not subject to 10 CFR 50.49 EQ requirements (ISG-18)

Materials

The materials of construction for this inaccessible medium voltage cables and connections commodity are:

- Insulation materials - various organic polymers

Environment

The cables and connections of this inaccessible medium voltage cables and connections commodity are exposed to the following environments:

- Adverse localized environment caused by exposure to moisture and voltage stress.

Aging Effects Requiring Management

The following aging effects, associated with this inaccessible medium voltage cables and connections commodity, require management:

- Formulation of water trees, localized damage leading to electrical failure (breakdown of insulation) caused by moisture intrusion and voltage stress.

Aging Management Programs

The following aging management programs manage the aging effects for this inaccessible medium voltage cables and connections commodity:

- Non-EQ Electrical Commodities Condition Monitoring Program

3.6.2.1.9 Commodity: High-Voltage Insulators (ISG-2)

Materials

The materials of construction for this high voltage insulator commodity are:

- Porcelain
- Cement
- Metal

Environment

The cables and connections of this high voltage insulator commodity are exposed to the following environments:

- Atmosphere/Weather

Aging Effects Requiring Management

The following aging effects, associated with this high voltage insulator commodity, require management:

- Surface Contamination
- Cracking

Aging Management Programs

The following aging management programs manage the aging effects for this high voltage insulator commodity:

- None Required

3.6.2.2 Further Evaluation of Aging Management as Recommended by NUREG-1801

NUREG-1801 Volume 1 Tables provide the basis for identifying those programs that warrant further evaluation by the reviewer in the license renewal application. For the Electrical and Instrumentation and Controls, those programs are addressed in the following sections.

3.6.2.2.1 Electrical Equipment Subject to Environmental Qualification

Environmental qualification of electrical equipment is a TLAA as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c)(1). The evaluation of this TLAA is addressed separately in Section 4.4.

10 CFR 50.49(e)(5) contains provisions for aging that require, in part, consideration of all significant types of aging degradation that can affect component functional capability. 10 CFR 50.49(e)(5) also requires component replacement or maintenance prior to the end of designated life, unless additional life is established through ongoing qualification. 10 CFR 50.49(k) and (l) permit different qualification criteria to apply based on plant vintage. Supplemental Environmental Qualification (EQ) regulatory guidance for compliance with these different qualification criteria is provided in the Regulatory Guide 1.89, Rev. 1, “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants”, the Division of Operating Reactors (DOR) Guidelines, and NUREG-0588.

The Palisades EEQ Program was established to demonstrate that certain electrical components located in harsh plant environment (that is, those areas of the plant that could be subject to the harsh environmental effects of loss of coolant accident [LOCA], high energy line breaks [HELB] or post-LOCA radiation) are qualified to perform their safety function operation in those harsh environments after the effects of in service aging. The EEQ Program manages applicable component thermal, radiation, and cyclic aging effects for the current operating license period using the qualification methods established by 10 CFR 50.49(f).

The Palisades Electrical Equipment Qualification Program allows re-analysis for maintaining qualification using the methods described above. In addition, the EEQ Program has procedural requirements in place to monitor and track aging effects of EQ equipment including cables. The requirements are listed below:

- Monitoring equipment condition and equipment performance
- Monitoring environmental conditions of plant areas, and
- Incorporating the results of testing and analysis into the plant maintenance and surveillance program.

The EEQ Program will continue to be implemented for the extended operating period in accordance with 10 CFR 50.49. Continuing the existing EEQ Program provides reasonable assurance that the aging effects will be managed and that the EQ components will continue to perform their intended functions for the period of extended operation.

3.6.2.2.2 Quality Assurance for Aging Management of Non-Safety Related Components

Quality Assurance Program applicability to non-safety-related components is addressed in Appendix B, Section 1.2.

3.6.2.3 Time-Limited Aging Analysis

The time-limited aging analyses (TLAA) identified below are associated with the Electrical and Instrumentation and Controls components. The section of the LRA that contains the TLAA review results is indicated in parenthesis.

- Section 4.4, Environmental Qualification of Electrical Equipment

3.6.3 Conclusion

The Electrical and Instrumentation and Controls components that are subject to aging management review have been identified in accordance with the requirements of 10 CFR 54.4. The aging management programs selected to manage aging effects for the Electrical and Instrumentation and Controls components are identified in the summaries in Section 3.6.2.1 above.

A description of these aging management programs is provided in Appendix B, along with the demonstration that the identified aging effects will be managed for the period of extended operation.

Therefore, based on the demonstrations provided in Appendix B, the effects of aging associated with the Electrical and Instrumentation and Controls components will be adequately managed so that there is reasonable assurance that the intended function(s) will be maintained consistent with the current licensing basis during the period of extended operation.

Table 3.6.1 Summary of Aging Management Evaluations in Chapter VI of NUREG-1801 for Electrical Components

Item Number	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.6.1-01	Electrical equipment subject to 10 CFR 50.49 environmental qualification (EQ) requirements	Degradation due to various aging mechanisms	Environmental qualification of electric components	Yes, TLAA (see [SRP] subsection 3.6.2.2.1)	Further evaluation documented in Section 3.6.2.2.1.
3.6.1-02	Electrical cables and connections not subject to 10 CFR 50.49 EQ requirements	Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced insulation resistance (IR); electrical failure caused by thermal/ thermoxidative degradation of organics; radiolysis and photolysis (ultraviolet [UV] sensitive materials only) of organics; radiation-induced oxidation; moisture intrusion	Aging management program for electrical cables and connections not subject to 10 CFR 50.49 EQ requirements	No	This is being managed by the Non-EQ Electrical Commodities Condition Monitoring Program.

Table 3.6.1 Summary of Aging Management Evaluations in Chapter VI of NUREG-1801 for Electrical Components

Item Number	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.6.1-03	Electrical cables used in instrumentation circuits not subject to 10 CFR 50.49 EQ requirements that are sensitive to reduction in conductor insulation resistance (IR)	Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced IR; electrical failure caused by thermal/ thermoxidative degradation of organics; radiation-induced oxidation; moisture intrusion	Aging management program for electrical cables used in instrumentation circuits not subject to 10 CFR 50.49 EQ requirements	No	This is being managed by the Non-EQ Electrical Commodities Condition Monitoring Program.
3.6.1-04	Inaccessible medium-voltage (2K VAC to 15K VAC) cables (e.g., installed in conduit or direct buried) not subject to 10 CFR 50.49 EQ requirements	Formation of water trees, localized damage leading to electrical failure (breakdown of insulation); water trees caused by moisture intrusion	Aging management program for inaccessible medium-voltage cables not subject to 10 CFR 50.49 EQ requirements	No	This is being managed by the Non-EQ Electrical Commodities Condition Monitoring Program.
3.6.1-05	Electrical connectors not subject to 10 CFR 50.49 EQ requirements that are exposed to borated water leakage	Corrosion of connector contact surfaces caused by intrusion of borated water	Boric acid corrosion	No	This is being managed by the Boric Acid Corrosion Program.

Table 3.6.2-1 Electrical Components - Electrical Commodity Groups - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Electrical cables and connections not subject to 10 CFR 50.49 EQ Requirements	Provide electrical connections to specified sections of an electrical circuit to deliver voltage, current, or signals	Insulation materials - various organic polymers	Adverse localized environment caused by heat, radiation, or moisture in the presence of oxygen	Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced IR; electrical failure caused by thermal/ thermoxidative degradation of organics; radiation-induced oxidation; moisture intrusion	Non-EQ Electrical Commodities Condition Monitoring Program	VI.A.1-a	3.6.1-02	A
		Connector Pins -Various Metals	Borated Water Leakage	Corrosion of connector contact surfaces caused by intrusion of borated water	Boric Acid Corrosion Program	VI.A.2-a	3.6.1-05	A

Table 3.6.2-1 Electrical Components - Electrical Commodity Groups - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Electrical cables and connections used in instrumentation circuits not subject to 10 CFR 50.49 EQ Requirements that are sensitive to reduction in conductor IR (ISG-15) (Nuclear Instrumentation and Radiation Monitoring Systems)	Provide electrical connections to specified sections of an electrical circuit to deliver voltage, current, or signals	Insulation materials - various organic polymers	Adverse localized environment caused by heat, radiation, or moisture in the presence of oxygen	Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced IR; electrical failure caused by thermal/ thermoxidative degradation of organics; radiation-induced oxidation; moisture intrusion	Non-EQ Electrical Commodities Condition Monitoring Program	VI.A.1-b	3.6.1-03	A, 602
Electrical Portion of the Non-EQ Electrical and I&C Penetration Assemblies (Cables & Connections)	Provide electrical connections to specified sections of an electrical circuit to deliver voltage, current, or signals	Insulation materials - various organic polymers	Adverse localized environment caused by heat, radiation, or moisture in the presence of oxygen	Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced IR; electrical failure caused by thermal/ thermoxidative degradation of organics; radiation-induced oxidation; moisture intrusion	Non-EQ Electrical Commodities Condition Monitoring Program	VI.A.1-a	3.6.1-02	A, 604

Table 3.6.2-1 Electrical Components - Electrical Commodity Groups - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
Fuse Holders (ISG-5)	Provide electrical connections to specified sections of an electrical circuit to deliver voltage, current, or signals.	Phenolic, Copper, Aluminum	Adverse localized environment caused by heat, radiation, or moisture in the presence of oxygen.	Embrittlement, cracking, melting, discoloration, moisture intrusion, corrosion	None required	None	None	J, 601
Non-Segregated Phase Bus and Connections (ISG-17)	Provide electrical connections to specified sections of an electrical circuit to deliver voltage, current, or signals	Various Metals, Porcelain	Adverse localized environment caused by heat, radiation, or moisture in the presence of oxygen.	Oxidation, Loosening of bolted connections due to thermal cycling, Corrosion due to moisture	Non-EQ Electrical Commodities Condition Monitoring Program	None	None	J
High-Voltage Transmission Conductors (ISG-2)	Provide electrical connections to specified sections of an electrical circuit to deliver voltage, current, or signals	Aluminum, Steel	Atmosphere/ Weather	Loss of conductor strength, Vibration	None Required	None	None	J, 603

Table 3.6.2-1 Electrical Components - Electrical Commodity Groups - Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801 Volume 2 Line Item	Table 1 Item	Notes
High-Voltage Switchyard Bus and Connections (ISG-2)	Provide electrical connections to specified sections of an electrical circuit to deliver voltage, current, or signals	Aluminum Stainless Steel (Bolting) Copper	Atmosphere/ Weather	Connection Surface Oxidation, Vibration	None Required	None	None	J, 606
Inaccessible medium-voltage (2kV to 15kV) cables and connections not subject to 10 CFR 50.49 EQ Requirements (ISG-18)	Provide electrical connections to specified sections of an electrical circuit to deliver voltage, current, or signals	Insulation materials - various organic polymers	Adverse localized environment caused by exposure to moisture and voltage stress	Formulation of water trees, localized damage leading to electrical failure (breakdown of insulation) caused by moisture intrusion and voltage stress.	Non-EQ Electrical Commodities Condition Monitoring Program	VI.A.1-c	3.6.1-04	A
High-Voltage Insulators (ISG-2)	Insulate and support an electrical conductor	Porcelain, Cement, Metal	Atmosphere/ Weather	Surface Contamination, Cracking	None Required	None	None	J, 605

Notes for Table 3.6.2-1

A Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP.

B Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP takes some exceptions to NUREG-1801 AMP.

C Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP.

D Component is different, but consistent with NUREG-1801 for material, environment, and aging effect. AMP takes some exceptions to NUREG-1801 AMP.

E Consistent with NUREG-1801 for material, environment, aging effect but a different aging management program is credited.

F Material not in NUREG-1801 for this component.

G Environment not in NUREG-1801 for this component and material.

H Aging effect not in NUREG-1801 for this component, material and environment combination.

I Aging effect in NUREG-1801 for this component, material and environment combination is not applicable.

J Neither the component nor the material and environment combination is evaluated in NUREG-1801.

Plant-specific Notes:

601 The aging management review determined that there was no aging mechanism for the in scope fuse holders. No aging management program is required for this commodity group.

602 The scope of the program is limited to select non-EQ cables associated with process and area radiation monitoring instrumentation circuits. Non-EQ Radiation Monitoring & Neutron Instrumentation systems instrumentation circuit cables require a different program other than the one presented in NUREG-1801, Section 2, Item XI.E2. This program will be consistent with ISG-15.

603 Loss of conductor strength due to corrosion of ACSR transmission conductors is not considered a significant contributor to the aging of transmission conductors. Transmission conductor vibration would be caused by wind loading. Wind loading is accounted for in the initial design and field installation of transmission conductors and high-voltage insulators. No aging management activities are required for this commodity group.

604 The Palisades analysis for the electrical penetration assembly electrical insulations contain cable and connection insulation materials similar to those of the cables, and are bounded by the cable insulation review and aging management criteria.

605 Surface contamination is not an applicable aging mechanism. The buildup of surface contamination is not prevalent at Palisades to be an aging mechanism. Cracking is not an applicable aging mechanism. Cracking or breaking of porcelain insulators is typically caused by physical damage, which is event driven rather than an age-related mechanism. No aging management activities are required for this commodity group.

606 Connection surface oxidation is not an applicable aging effect. Vibration is not an applicable aging mechanism since switchyard bus has no connections to moving or vibrating equipment. No aging management is required for this commodity group.

4.0 Time-Limited Aging Analyses

10 CFR 54.21(c)(1) requires that each application for renewal of a plant operating license include a list of time-limited aging analyses, and a demonstration for each that, either:

"(i) The analyses remain valid for the period of extended operation;

(ii) The analyses have been projected to the end of the period of extended operation; or

(iii) The effects of aging on the intended function(s) will be adequately managed for the period of extended operation."

10 CFR 54.3 defines Time Limited Aging Analyses as follows:

"Time-limited aging analyses, for the purposes of this part, are those licensee calculations and analyses that:

(1) Involve systems, structures, and components within the scope of license renewal, as delineated in §54.4(a);

(2) Consider the effects of aging;

(3) Involve time-limited assumptions defined by the current operating term, for example, 40 years;

(4) Were determined to be relevant by the licensee in making a safety determination;

(5) Involve conclusions or provide the basis for conclusions related to the operability of the system, structure, and component to perform its intended functions, as delineated in §54.4(b); and

(6) Are contained or incorporated by reference in the CLB."

This section identifies those issues that would be defined as Time-Limited Aging Analyses (TLAAs) in accordance with 10 CFR 54.21(c)(1), and, therefore, require explicit evaluation and discussion in the License Renewal Application (LRA) for Palisades. It also documents the conclusions of technical evaluations performed for each TLAA.

4.1 Identification of Time-Limited Aging Analyses

This section discusses the process used to identify and evaluate TLAAs and exemptions applicable to Palisades.

4.1.1 Process for Identification of Time-Limited Aging Analyses

The process for identifying TLAAs to be addressed by the Palisades License Renewal Application utilized extensive searches of the plant CLB, various NRC and industry documents, and LRAs from other applicants.

4.1.1.1 CLB Search for TLAAs

10 CFR 54.3 defines 6 criteria, all of which must be satisfied in order for a calculation or analysis to be classified as a TLAA. Criterion 6 of the TLAA definition indicates that a calculation or analysis must be contained or incorporated by reference in the CLB. By this definition, any calculation or analysis that is not incorporated into the CLB can not be a TLAA, even if it includes time-limited assumptions. It is logical, then, to begin the TLAA identification process with a search of the CLB.

The CLB was searched to identify discussions of calculations, analyses, and other subject matter that potentially satisfy the criteria of a TLAA. These searches were performed from a broad perspective to identify specific words that suggested time-sensitivity, as well as specific technical subjects that a knowledgeable individual might expect to be based on time-sensitive information for their conclusions. The following Palisades-specific information was searched for potential TLAAs:

- FSAR revision 24
- Technical Specifications
- NRC Safety Evaluation Reports and Safety Evaluations
- NRC Commitments
- Selected other docketed correspondence

4.1.1.2 Industry Document Search for TLAAs

TLAAs are, by definition, plant-specific issues addressed in the plant-specific CLB. TLAAs addressed by other plants, or in generic industry documents, will not necessarily be applicable to Palisades' design, but they do provide comparative data to perform a completeness check. After the initial list of potential TLAA candidates was generated directly from the Palisades CLB, various other sources of information were compared with that list to confirm its completeness. These other sources included:

- Standard Review Plan for License Renewal, NUREG-1800
- Nuclear Energy Institute “Industry Guidelines for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule”, NEI 95-10, revisions 3 and 4
- 10 CFR 54 Statements of Consideration (60FR22461)
- LRAs from other applicants

- Westinghouse (formerly Combustion Engineering) Owners Group Topical Reports (as referenced in CLB)

TLAA candidates mentioned in these sources that might have some applicability to Palisades were assessed for applicability in the same manner as potential TLAAs identified through the CLB searches.

4.1.2 Identification of Exemptions

10 CFR 54.21(c)(2) requires that the License Renewal Application (LRA) contain a list of plant-specific exemptions granted pursuant to 10 CFR 54.12, and in effect, that are based on time-limited aging analyses. To identify exemptions which might fall within this requirement, a search was performed of NRC SERs and the FSAR. This search identified a number of exemptions that have been granted to Palisades. The NRC letters granting the exemptions, and any associated SEs, were then reviewed. These reviews revealed that none of the exemptions in effect were based on analyses that would meet the criteria of a TLAA. Accordingly, it is concluded that there are no plant-specific exemptions required to be addressed in the LRA in accordance with 10 CFR 54.21(c)(2).

4.1.3 Evaluation Process of Potential Time Limited Aging Analyses

Each of the potential TLAAs, that resulted from the searches described in Sections 4.1.1 and 4.1.2, was then reviewed (screened) against the six 10 CFR 54.3(a) criteria. Those which met the six criteria were evaluated in accordance with 10 CFR 54.21(c)(1) to demonstrate that at least one of the following criteria was applicable:

i. The analyses remain valid for the period of extended operation.

ii. The analyses have been projected to the end of the period of extended operation.

iii. The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

The results of these evaluations are provided in Table 4.1-1 and are discussed in Sections 4.2 through 4.7. For NRC reviewer convenience, the information is presented in a manner consistent with the organization of the Standard Review Plan.

Table 4.1-1 Time Limited Aging Analyses

TLAA Category from SRP	TLAA Subject	LRA Section	Disposition per 10 CFR 54.21(c)(1)	Comments
Reactor Vessel Neutron Embrittlement	Upper Shelf Energy	4.2.1	(iii) effects of aging on the intended function will be adequately managed for the period of extended operation	
	Pressurized Thermal Shock	4.2.2	(iii) effects of aging on the intended function will be adequately managed for the period of extended operation	
	Pressure - Temperature (P-T) Limits	4.2.3	(iii) effects of aging on the intended function will be adequately managed for the period of extended operation	Includes LTOP setpoint calculation(s)
	Low Temperature Overpressure Protection (LTOP) PORV Setpoints	4.2.4	(iii) effects of aging on the intended function will be adequately managed for the period of extended operation	Included in Section 4.2.3
Metal Fatigue	Reactor Vessel Fatigue Analyses	4.3.2	(i) remains valid for the period of extended operation & (iii) effects of aging on the intended function will be adequately managed for the period of extended operation	
	Reactor Vessel Head Closure Stud Fatigue Analysis	4.3.3	(i) remains valid for the period of extended operation & (iii) effects of aging on the intended function will be adequately managed for the period of extended operation	

Table 4.1-1 Time Limited Aging Analyses

TLAA Category from SRP	TLAA Subject	LRA Section	Disposition per 10 CFR 54.21(c)(1)	Comments
Metal Fatigue	Control Rod Drive Mechanism (CRDM) Housing Fatigue Analyses	4.3.4	(i) remains valid for the period of extended operation	
	Steam Generator Fatigue Analyses	4.3.5	(i) remains valid for the period of extended operation	
	Pressurizer Fatigue Analyses	4.3.6	(i) remains valid for the period of extended operation & (iii) effects of aging on the intended function will be adequately managed for the period of extended operation	
	Regenerative Heat Exchanger Fatigue Analyses	4.3.7	(ii) projected to the end of the period of extended operation & (iii) effects of aging on the intended function will be adequately managed for the period of extended operation	
	ASME III Class A Primary Coolant Piping Fatigue Analyses	4.3.8	(i) remains valid for the period of extended operation & (iii) effects of aging on the intended function will be adequately managed for the period of extended operation	

Table 4.1-1 Time Limited Aging Analyses

TLAA Category from SRP	TLAA Subject	LRA Section	Disposition per 10 CFR 54.21(c)(1)	Comments
Metal Fatigue	Revised Bulletin 88-11 Fatigue Analysis of the Hot-Leg-to-Pressurizer-Surge-Line Nozzle, Surge Line, and Pressurizer Surge Nozzle	4.3.9	(i) remains valid for the period of extended operation, (ii) projected to the end of the period of extended operation, & (iii) effects of aging on the intended function will be adequately managed for the period of extended operation	
	Revised Fatigue Analysis of Nozzles from PCS Cold Legs 1B and 2A to Pressurizer Spray and of the Pressurizer Spray Nozzle	4.3.10	(i) remains valid for the period of extended operation & (iii) effects of aging on the intended function will be adequately managed for the period of extended operation	
	Pressurizer Auxiliary Spray Line Tee Fatigue Analysis in Response to NRC Bulletin 88-08	4.3.11	(i) remains valid for the period of extended operation	
	Absence of a TLAA for ASME III Class 1 HELB Locations Based on Fatigue Usage Factor	4.3.12	Not a TLAA	

Table 4.1-1 Time Limited Aging Analyses

TLAA Category from SRP	TLAA Subject	LRA Section	Disposition per 10 CFR 54.21(c)(1)	Comments
Metal Fatigue	Assumed Thermal Cycle Count for Allowable Secondary Stress Range Reduction Factor in Piping and Components	4.3.13	(i) remains valid for the period of extended operation, (ii) projected to the end of the period of extended operation, & (iii) effects of aging on the intended function will be adequately managed for the period of extended operation	
	Effects of Reactor Coolant System Environment on Fatigue Life of Piping and Components (Generic Safety Issue 190)	4.3.14	(ii) projected to the end of the period of extended operation & (iii) effects of aging on the intended function will be adequately managed for the period of extended operation	
Environmental Qualification of Electrical Equipment	Environmental Qualification of Electrical Equipment	4.4	(iii) effects of aging on the intended function will be adequately managed for the period of extended operation	
Containment Liner Plate, Metal Containments and Penetrations Fatigue Analysis	Concrete Containment Tendon Prestress Analysis	4.5	(iii) effects of aging on the intended function will be adequately managed for the period of extended operation	

Table 4.1-1 Time Limited Aging Analyses

TLAA Category from SRP	TLAA Subject	LRA Section	Disposition per 10 CFR 54.21(c)(1)	Comments
Containment Liner Plate, Metal Containments and Penetrations Fatigue Analysis	Containment Liner Plate Load Cycles	4.6.1	(i) remains valid for the period of extended operation	
	Containment Penetration Load Cycles	4.6.2	(i) remains valid for the period of extended operation	
Other Plant-Specific Time-Limited Aging Analyses	Fuel Handling Crane Load Cycles	4.7.1	(i) remains valid for the period of extended operation	
	Alloy 600 Nozzle and Safe End Life Assessment Analyses	4.7.2	(i) remains valid for the period of extended operation & (iii) effects of aging on the intended function will be adequately managed for the period of extended operation	
	ASME Code Case N-481 Relaxation of The Primary Coolant Pump Weld Category B-L-1 Inspection Interval from 10 Years to 40 Years	4.7.3	(i) remains valid for the period of extended operation & (iii) effects of aging on the intended function will be adequately managed for the period of extended operation	.

Table 4.1-1 Time Limited Aging Analyses

TLAA Category from SRP	TLAA Subject	LRA Section	Disposition per 10 CFR 54.21(c)(1)	Comments
Other Plant-Specific Time-Limited Aging Analyses	Risk-Informed Inservice Inspection Program Calculations	4.7.4	(ii) projected to the end of the period of extended operation	
	Absence of a TLAA for Reactor Coolant Pump Flywheel Fatigue or Crack Growth Analysis	4.7.5	Not a TLAA	

4.2 Reactor Vessel Neutron Embrittlement

The reactor vessel is subject to neutron irradiation from the core. This irradiation results in the embrittlement of the reactor vessel materials. The NRC approved methodology used to perform neutron fluence calculations is consistent with Regulatory Guide 1.190 and is described in WCAP-15353, "Palisades Reactor Pressure Vessel Fluence Evaluation" (Reference 2). Analyses have been performed that address the following:

- Upper shelf energy
- Pressurized thermal shock (PTS)
- PCS pressure-temperature operating limits
- Low Temperature Overpressure Protection (LTOP) PORV Setpoints

Palisades began use of low leakage core designs when Regulatory Guide 1.99, Revision 2 was issued. Gradual improvements to the low leakage core design and in the analytical methods used to estimate the amount of fast fluence accumulated at the reactor vessel beltline, resulted in the current ultra low leakage core design approved by NRC on November 14, 2000. This core design consists of third and fourth cycle assemblies loaded in peripheral locations with specially designed shield assemblies placed to reduce flux at the limiting reactor vessel beltline axial welds. The limiting welds are estimated to reach the PTS screening criterion in 2014.

The flux to the reactor vessel would have to be reduced by an additional factor of 3 in order to reach March 24, 2031. Some additional flux reduction could conceivably be achieved by installation of additional shield assemblies and/or flux suppression devices (e.g., hafnium inserts). Flux reduction of the magnitude required at Palisades would require far more extraordinary measures, however, such as installation of neutron shields on the exterior of the core support barrel. It is unlikely that a plant modification of this magnitude would be cost-effective. It is highly likely that NMC would pursue alternative solutions rather than rely on flux reduction to extend the reactor vessel life. Other alternatives that would be considered would include completion of the safety analysis as specified in 10 CFR 50.61(b)(4), and the thermal annealing treatment as specified in 10 CFR 50.61(b)(7). Any alternative that NMC may propose in the future to extend the life of the Palisades reactor vessel would, of necessity, be discussed thoroughly with the NRC, and would be subject to formal NRC review and approval before it could be implemented. The ultimate method used to manage PTS for extended plant operation would be governed by NRC regulations independently from the license renewal process.

Estimated fluence accumulated by each reactor vessel beltline material at the clad-base metal interface (CBMI) and the ¼-thickness (¼t) location at the end of the license renewal period is shown in Table 4.2-1.

Table 4.2-1 Estimated Palisades Fluence on March 24, 2031

RPV Material	Material Heat #	Fluence (10^{19} n/cm^2)	¼t Fluence (10^{19} n/cm^2)
Intermediate Shell Axial Welds 2-112A/C	W5214	2.084	1.251
Lower Shell Axial Welds 3-112A/C	W5214	2.084	1.251
	34B009	2.084	1.251
Intermediate shell to Lower Shell Circumferential Weld 9-112	27204	2.998	1.800
Intermediate Shell Plate D-3803-1	C-1279	2.998	1.800
Intermediate Shell Plate D-3803-2	A-0313	2.998	1.800
Intermediate Shell Plate D-3803-3	C-1279	2.998	1.800
Lower Shell Plate D-3804-1	C-1308A	2.998	1.800
Lower Shell Plate D-3804-2	C-1308B	2.998	1.800
Lower Shell Plate D-3804-3	B-5294	2.998	1.800

4.2.1 Upper Shelf Energy

Summary Description

10 CFR Part 50 Appendix G Paragraph IV.A.1 requires that the reactor vessel beltline materials must have Charpy upper shelf energy of no less than 68 J (50 ft-lb) throughout the life of the reactor vessel, unless otherwise approved by the NRC. In the event that the 50 ft-lb requirement cannot be satisfied as stated in 10 CFR 50 Appendix G, or by alternative procedures acceptable to the NRC, plant operation may continue provided requirement 1 of Appendix G is satisfied. This requirement states that an analysis must conservatively demonstrate the existence of equivalent margins of safety for continued operation.

Analysis

As shown in Table 4.2.1-1, the upper shelf energy for reactor vessel lower shell plate D-3804-1 is expected to decrease to less than 50 ft-lbs based on predictions using Regulatory Guide 1.99.

Table 4.2.1-1 Estimated USE on March 24, 2031

RPV Material	Material Heat #	Cu (%)	Initial USE (ft-lbs)	¼t Neutron Fluence (10^{19} n/cm^2)	USE (ft-lbs)
2-112A/C	W5214	0.213	118	1.251	73.86
3-112A/C	34B009	0.192	111	1.251	72.01
9-112	27204	0.203	84	1.800	50.83
D-3803-1	C-1279	0.24	102	1.800	63.29
D-3803-2	A-0313	0.24	87	1.800	53.98
D-3803-3	C-1279	0.24	91	1.800	56.46
D-3804-1	C-1308A	0.19	72	1.800	48.97
D-3804-2	C-1308B	0.19	76	1.800	51.69
D-3804-3	B-5294	0.12	73	1.800	55.51

The upper shelf energy for reactor vessel lower shell plate D-3804-1 is estimated to decrease below 50 ft-lbs during the year 2014.

Disposition: 10 CFR 54.21(c)(1)(iii)

10 CFR 50, Appendix G requires licensees to submit an analysis at least 3 years prior to the time that the upper-shelf energy of any of the reactor vessel material is predicted to drop below 50 ft-lb., as measured by Charpy V-notch specimen testing. NMC will comply with this requirement.

NMC will submit an equivalent margins analysis, completed in accordance with 10 CFR 50 Appendix G Section IV.A.1, for NRC approval, at least three years before any reactor vessel beltline material upper shelf energy decreases to less than 50 ft-lb.

This issue will be dispositioned using the method of 10 CFR 54.21(c)(1)(iii), the effects of aging on the intended function will be adequately managed for the period of extended operation. The relevant activities will be managed under the Reactor Vessel Integrity Surveillance Program.

4.2.2 Pressurized Thermal Shock

Summary Description

The pressurized thermal shock (PTS) rule, 10 CFR 50.61, established screening criteria that are a measure of a limiting level of reactor vessel material embrittlement beyond which operation cannot continue without further plant-specific evaluation. The screening criteria are given in terms of reference temperature RT_{PTS}. The screening criteria are 270° F for plates and axial welds, and 300° F for circumferential welds. The RT_{PTS} is defined as:

$RT_{PTS} = I + \Delta RT_{NDT} + M$

Where,

I = Initial reference temperature

ΔRT_{NDT} = Mean value of adjustment in reference temperature

M = Margin

The initial reference temperature is the measured unirradiated value as defined in the ASME Code, Paragraph NB 2331. If measured values are unavailable for the heat of the material of interest, generic values may be used. The generic values are based on the data for materials of all heats that were made by the same vendor using similar processes. The generic values of initial reference temperature for welds are defined in the PTS rule and used in this analysis. Measured values are used for the beltline plate materials.

The ΔRT_{NDT} depends upon the amount of neutron irradiation and the amounts of copper and nickel in the material. It is calculated as the product of a fluence factor and a chemistry factor. The fluence factor is calculated from the best-estimate neutron fluence at the interface of cladding, weld, and metal on the inside surface of the vessel at a location where the material receives the highest fluence at the end of the period of evaluation. The fluence value used in this analysis is based on the results of calculations performed following the guidance in Regulatory Guide 1.190. The chemistry factor may be determined using credible surveillance data or from the chemistry factor tables in the PTS rule; these tables are used in this analysis.

The margin term is intended to account for variability in initial reference temperature and the adjustment in reference temperature caused by irradiation. The value of the margin term is dependent on whether the initial reference temperature was a measured or generic value and whether the adjustment in reference temperature was determined from credible surveillance data or from the chemistry factor tables in the PTS rule.

Analysis

The results of the revised PTS analysis for the limiting material have been reviewed for compliance with 10 CFR 50.61. The methodology used in the PTS analysis is based on the projected neutron fluence at the end of the period of extended operation. The RT_{PTS} values for the intermediate and lower shell plates remain below the NRC screening criterion of 270ºF. The RT_{PTS} values for the axial and circumferential welds are projected to exceed the NRC screening criteria of 270ºF and 300ºF, respectively.

Table 4.2.2-1 Estimated RT_{PTS} on March 24, 2031

RPV Material	Material Heat #	Cu (%)	Ni (%)	$RT_{NDT(U)}$ (°F)	Margin (°F)	Fluence (10^{19} n/cm^2)	RT_{PTS} (°F)
Welds 2-112A/C	W5214	0.213	1.01	-56	65.5	2.084	287
Welds 3-112A/C	W5214	0.213	1.01	-56	65.5	2.084	287
	34B009	0.192	0.980	-56	65.5	2.084	271
Weld 9-112	27204	0.203	1.018	-56	65.5	2.998	302
Plate D-3803-1	C-1279	0.24	0.50	-5	17	2.998	209
Plate D-3803-2	A-0313	0.24	0.52	-30	34	2.998	210
Plate D-3803-3	C-1279	0.24	0.50	-5	17	2.998	209
Plate D-3804-1	C-1308A	0.19	0.48	0	34	2.998	200
Plate D-3804-2	C-1308B	0.19	0.50	-30	34	2.998	173
Plate D-3804-3	B-5294	0.12	0.55	-25	34	2.998	115

Disposition: 10 CFR 54.21(c)(1)(iii)

The Palisades reactor vessel is projected to reach the PTS screening criterion of 270º F on the beltline axial welds fabricated with weld wire heat W5214 in 2014.

10 CFR 50.61 requires the licensee to implement a flux reduction program that is reasonably practicable to avoid exceeding the screening criteria. Palisades began use of low leakage core designs in the late 1980’s when Regulatory Guide 1.99, Revision 2 was issued. Gradual improvements to the low leakage core design and in the analytical methods used to estimate the amount of fast fluence accumulated at the reactor vessel beltline, resulted in the current ultra low leakage core design.

If a flux reduction program does not prevent the reactor vessel from exceeding a PTS screening criterion at the end of a plant’s licensed life, 10 CFR 50.61 allows two options.

The licensee can submit a safety analysis pursuant to §50.61(b)(4) to determine what, if any, modifications to equipment, systems and plant operation are necessary to prevent failure of the reactor vessel from a postulated PTS event. The other option is to perform a thermal-annealing treatment of the reactor vessel pursuant to §50.61(b)(7) to recover fracture toughness. §50.61 requires that the details of the selected alternative be provided to the NRC three years prior to when the reactor vessel is projected to exceed the PTS screening criteria.

At the appropriate time, prior to exceeding the PTS screening criteria, Palisades will select the optimum alternative to manage PTS in accordance with NRC regulations, and will make the applicable submittals to obtain NRC review and approval.

This issue will be dispositioned using the method of 10 CFR 54.21(c)(1)(iii), the effects of aging on the intended function will be adequately managed for the period of extended operation. The relevant activities will be managed under the Reactor Vessel Integrity Surveillance Program.

4.2.3 Pressure-Temperature (P-T) Limits

Summary Description

10 CFR Part 50 Appendix G requires that the reactor pressure vessel be maintained within established pressure-temperature limits including during heatup and cooldown. These limits specify the maximum allowable pressure as a function of reactor coolant temperature. As the reactor pressure vessel becomes embrittled and its fracture toughness is reduced, the allowable pressure (given the required minimum temperature) is reduced.

Heatup and cooldown limit curves are calculated using the adjusted RT_{NDT} corresponding to the limiting beltline region material of the reactor vessel. The adjusted RT_{NDT} of the limiting material in the core region of the reactor vessel is determined by using the unirradiated reactor vessel material fracture toughness properties, estimating the radiation-induced RT_{NDT} and adding a margin.

Low temperature overpressure protection limits and setpoints are determined as part of the calculation of pressure/temperature operating limit curves.

Analysis

The current pressure/temperature analyses are valid beyond the current operating license period, but not to the end of the period of extended operation. These analyses are estimated to expire in 2014.

Disposition: 10 CFR 54.21(c)(1)(iii)

This issue will be dispositioned using the method of 10 CFR 54.21(c)(1)(iii), the effects of aging on the intended function will be adequately managed for the period of extended operation. 10 CFR 50, Appendices G and H, require the Pressure-Temperature operating limits contained in Technical Specifications to be updated as needed to remain valid. The relevant activities will be managed under the Reactor Vessel Integrity Surveillance Program.

4.2.4 Low Temperature Overpressure Protection (LTOP) PORV Setpoints

Low temperature overpressure protection limits and setpoints are determined as part of the calculation of pressure/temperature operating limit curves. See Section 4.2.3.

4.3 Metal Fatigue

4.3.1 Evaluation of Fatigue in Vessels, Piping, and Components

Design analyses of vessels and piping predict fatigue effects by constructing a set of normal operating and abnormal cyclic load events, each of which is assumed to occur no more than an assumed number of times during the design life or licensed operating period. Palisades maintains a record of the occurrence, and cumulative number of occurrences, of the subset of normal and abnormal cyclic events which contribute significantly to the predicted cumulative usage factor of vessels, piping, and components. The design basis number of cycles, the estimated number of cycles experienced to date, and the estimated number of cycles expected to be experienced by the end of the extended operating period, are shown in Table 4.3.1-1.

At Palisades the cumulative effect of alternating stresses produced by these events (the cumulative usage factor, CUF) was originally calculated under ASME Section III Class A rules, 1965 edition and some addenda. ASME Section III Class 1 rules were used for later replacements (e.g., the steam generators), and for certain piping modifications. Reanalyses, to incorporate events and conditions not considered in the original design, typically used the original design code for the component. The code of record for each component is listed in Table 4.3.1-2.

At almost all Palisades locations for which a fatigue analysis was performed, the cumulative usage factor (CUF) predicted by the original analysis of record, based on a 40 year life, is not expected to approach the allowable limit of 1.0 in a 60 year life. There are a number of locations in the Palisades vessels and piping for which the original fatigue analysis predicts a cumulative usage factor of greater than 0.666 for the original 40-year design life; and, therefore, for which a simple proportional extension of the original analysis to 60 years would predict a CUF exceeding the allowable 1.0. However, the plant operating history demonstrates that the number of event cycles assumed by the original analyses is conservative in all but a few cases, and that at the historical rate of occurrence, the analysis basis event cycle assumptions remain more than adequate for the extended licensed operating period. To confirm that this remains so, NMC plans to augment the existing cycle counting activities, where necessary, and continue them through the extended licensed operating period. See Appendix B for a discussion of the Fatigue Monitoring Program

However, for a few cases at some locations, addenda to the original fatigue analyses have addressed effects not contemplated in the original analyses, and plant modifications have required some new analyses. Some of these have used revised cycle estimates, or have addressed load events not contemplated in the original fatigue analyses. With these changes, at some locations, a simple projection to a 60-year design life at the expected rate of load cycle accumulation predicts a usage factor greater than 1.0. These locations, therefore, required reanalysis or other appropriate measures, as described below.

The reactor coolant environment may also reduce fatigue life. These effects are evaluated as a generic issue. At most locations the predicted cumulative usage factors are low, so that even with these effects the fatigue cycle count program will ensure that these effects do not violate the basis for the safety determination of the fatigue analysis. Alternative methods are needed at a few locations with both high predicted usage factors and with more severe effects of the reactor coolant environment, to ensure that these effects do not violate the basis for the safety determination of the fatigue analysis.

The component and piping analyses of record were reviewed to identify existing fatigue analyses of vessels and piping to ASME Section III Class A rules, and later analyses to ASME Section III Class 1 rules. The Palisades cycle count records were reviewed to determine the number of accumulated operating cycles to date for a critical set of operating and abnormal events.

Comparing the allowed and 60-year expected cycles shows that only a few of the events could reasonably be expected to occur more than 50 percent of the times assumed by the analyses in a 60 year design life. Therefore the CUF should not approach 0.5 in 60 years, even if the CUF predicted by the original 40 year analysis were at the analytic limit of 1.0, except in those locations affected by (1) modifications or additional effects not contemplated in the original analyses, or (2) the reactor coolant environmental effects described in Section 4.3.14.

Fatigue Monitoring Aging Management Program for Vessels, Piping, and Components

A number of fatigue issues at Palisades will be managed through cycle counting. The program will record plant transients for a bounding subset of reactor pressure vessel (RPV), pressurizer, steam generator, other reactor coolant system components, and Consumers Design Class I piping to ensure that fatigue effects are adequately managed for Consumers Design Class I systems and components. Cycle counting will assure that either (1) the analysis and licensing basis design cycle count assumptions are not exceeded during the period of extended operation, or (2) corrective action is taken to ensure that the basis for the safety determination remains valid, upon reaching the assumed cycle count for a given event.

This cycle counting program is described in Appendix B, Fatigue Monitoring Program.

Table 4.3.1-1 Primary Coolant system Design Transients

No	Transient	Cycles Allowed	To 9 January 2005[i]	Most Expected in 60 Years[i]
1	Plant Heatup at 100°F/hr Plant Cooldown at 100°F/hr Pressurizer Cooldown at 200°F/hr	500 500 500	134 119 119	240 240 240
2	Plant Loading at 5% of Full Power / Minute Plant Unloading at 5% of Full Power / Minute	2,000 2,000	NC NC	NC NC
3	Plant Loading at 15% of Full Power / Minute Plant Unloading at 15% of Full Power / Minute	2,000 2,000	NC NC	NC NC
4	Step Load Increase of 10% of Full Power Step Load Decrease of 10% of Full Power	2,000 2,000	NC NC	NC NC
5	Loss of Turbine Load from 100% Power	100	50	90
6	Loss of Primary Coolant Flow	100	11	20
7	Reactor Trip from 100% Power	500	126	240
8	Safety Valve Operation	200	NC	NC
9	Steady State Pressure Fluctuation (± 50 psi)	350,000	NC	NC
10	Regenerative Heat Exchanger Isolation	400	NC	NC

Table 4.3.1-1 Primary Coolant system Design Transients

No	Transient	Cycles Allowed	To 9 January 2005[i]	Most Expected in 60 Years[i]
11	Hydrostatic Test Conditions: ·Primary Side Hydrostatic Test Before Initial Startup @ 3110 psig ·Secondary Side Hydrostatic Test @ 1,250 psia with Primary Side Pressure = >600 psia	 10 10	 1 2	 1 2
12	Leak Test Conditions: ·Primary Side Leak Test @ 2485 psi ·Secondary Side Leak Test @ 1,000 psia with Primary Side Pressure >350ºF	 320 320	 0 0	 10 10
13	Steam Generator Primary-to-Secondary Pressure Test (Secondary @ 900 psia/532ºF and Primary @ 0 psig	1	0	1
14	Steam Generator Primary Head Divider Plate Pressure Differential Transients of 85 psi (from RCP starts)	2,500	NC	NC
15	Feedwater Addition (425 gpm @ 32ºF with plant in Hot Standby)	15.000	NC	NC
16	Feedwater Addition (300 gpm maximum @ 32ºF with Steam Generator Secondary Side dry @ 600ºF)	8	0	4
17	Steam Line Rupture	1	0	1
18	Design Basis Earthquake	200	NC	NC

i. NC = Not counted (will never approach)

Table 4.3.1-2 Applicable Codes for Palisades Components

Component	Code of Construction	Edition and Addenda
Reactor Vessel	ASME Section III, Class A	1965 Edition with Addenda through Winter 1965
Pressurizer	ASME Section III, Class A	1965 Edition with Addenda through Winter 1966
Reactor Coolant Pumps	ASME Section III, Class A	1965 Edition with Addenda through Winter 1965
Pressurizer Safety and Power Operated Relief Valves	ASME Section III	1965 Edition with Addenda through Winter 1965
PCS Piping and Nozzles	ANSI B31.1 Power Piping Code, Class 1[i]	1955 Edition
Containment Vessel Liner	ASME Section III	1965 Edition with no Addenda
Regenerative Heat Exchanger	ASME Section III, Class C	1965 Edition with Addenda through Winter 1965
Replacement Steam Generators	ASME Section III, Section NB/NC	1977 Edition with no Addenda
Quench Tank	ASME Section III, Class C	1965 Edition with Addenda through Winter 1965

i. PCS piping was subsequently reanalyzed to ASME Section III, Class A.

4.3.2 Reactor Vessel Fatigue Analyses

Summary Description

The Palisades Reactor Vessel was designed, constructed, and analyzed to the ASME Boiler and Pressure Vessel Code, Section III, Subsection 4 for Class A vessels, 1965, with addenda through Winter, 1965. ASME III -1965 Class A vessels require a Subsection N-415.2 fatigue analysis for parts subject to stress ranges which exceed the criteria of Subsection N-415.1. The original analyses have been amended to address issues that have arisen since fabrication. The current highest calculated fatigue usage factors, based on the number of design basis load cycles assumed by the vessel analyses, have been determined. The number of design basis load cycles for each event was selected to be adequate for the originally-licensed 40-year design life.

Of the high usage factor vessel locations, the vessel studs have the highest usage factor but are replaceable, and the control rod drive mechanism (CRDM) housings and appurtenances have been replaced. For these reasons fatigue in these components is addressed separately; the vessel studs in Section 4.3.3, and the replaced CRDM housings and appurtenances in Section 4.3.4.

Analysis

This section addresses the calculated fatigue usage factors for the reactor vessel

- Shell and bottom head
- Inlet and outlet nozzles
- Internal welded attachments
- Instrument nozzle shroud tube
- Vessel head CRDM nozzles, and
- Instrument flange bolts (on the instrument nozzle on the vessel head).

The worst-case calculated usage factor for the set of design basis load events in these locations is 0.4516 on the outlet nozzle, well within the code limit of 1.0. The number of design basis transient events is not expected to approach the number assumed by the analysis during the extended licensed operating period.

Disposition: 10 CFR 54.21(c)(1)(i) and 10 CFR 54.21(c)(1)(iii)

The number of each of the design basis events that affect the reactor pressure vessel is not expected to approach its design basis limit during the extended licensed operating period, and, therefore, the actual fatigue usage factors are not expected to approach their maximum calculated values during the extended licensed operating period. The calculated maximum usage factor for these locations on the reactor vessel is 0.4516, well within the analytical limit of 1.0. Therefore, this item is dispositioned under 10 CFR 54.21(c)(1)(i), the analysis remains valid for the period of extended operation.

The Fatigue Monitoring Program, described in Appendix B, will ensure a reanalysis or other appropriate corrective action in the unlikely event that a design basis cycle count limit is reached at any time during the extended licensed operating period.

4.3.3 Reactor Vessel Head Closure Stud Fatigue Analysis

Summary Description

The highest fatigue usage factor calculated by the reactor vessel fatigue analysis is in the vessel head studs. See also Section 4.3.2.

Analysis

The calculated lifetime usage factor in the head closure studs for the set of design basis load events is 0.8346, within the code limit of 1.0. The number of design basis transient events is not expected to approach the number assumed by the analysis during the extended licensed operating period.

Disposition: 10 CFR 54.21(c)(1)(i) and 10 CFR 54.21(c)(1)(iii)

The number of each of the design basis events that affect the reactor pressure vessel and the head closure studs is not expected to approach its design basis limit during the extended licensed operating period, and, therefore, the actual fatigue usage factors are not expected to approach their calculated values during the extended licensed operating period. The calculated maximum usage factor for the studs is 0.8346, within the analytical limit of 1.0. Therefore, this item is dispositioned under 10 CFR 54.21(c)(1)(i), the analysis remains valid for the period of extended operation.

The Fatigue Monitoring Program, described in Appendix B, will ensure a reanalysis or other appropriate corrective action in the event that a design basis cycle count limit is reached at any time during the extended licensed operating period.

4.3.4 Control Rod Drive Mechanism (CRDM) Housing Fatigue Analyses

Summary Description

The reactor control rod drive mechanisms (CRDM) are enclosed in pressure housings bolted and seal-welded to the reactor pressure vessel CRDM nozzle flanges. The CRDM housings, their seal housings, their instrument and vent tube nozzles, the flange bolts, and the Omega seal welds between the CRDM housing flanges and the reactor vessel CRDM nozzle flanges were all replaced in 2001. Extension of the operating license to March 24, 2031 therefore requires a 31 year design life. The replacements are ASME III (1989) - Class 1, NPT stamped, with a reconciliation to the 1965 code.

Analysis

CRDM Housings: The revised fatigue evaluation found that the criteria of ASME III - 1989, Paragraph NB-3222.4(d) are met, and therefore that no fatigue analysis is required.

CRDM Housing Bolts: The fatigue evaluation of record calculated a design lifetime cumulative usage factor of 0.173. This evaluation has not been revised, but will remain valid so long as the assumed number of lifetime design basis transient cycles remains valid. This analysis is very conservative since the bolts were replaced when the CRDM housings were replaced.

CRDM Flange - Reactor Vessel Nozzle Flange Bolts: The fatigue evaluation of record calculated a design lifetime cumulative usage factor of 0.624. This evaluation has not been revised, but will remain valid so long as the assumed number of lifetime design basis transient cycles remains valid. This analysis is very conservative since the bolts were replaced when the CRDM housings were replaced.

CRDM Flange - Reactor Vessel Nozzle Flange Omega Seal Welds: The fatigue evaluation calculated a design lifetime cumulative usage factor of 0.5621. This analysis used the set of design transients from the original design specification, which were based on an assumed 40-year design life.

Disposition: 10 CFR 54.21(c)(1)(i)

Since all of these components were replaced in 2001, their expected installed lifetime, including the extended licensed operating period (to March 24, 2031), will be only about 31 years, compared to the 40 years upon which the estimate of design basis event cycles was based. The two highest calculated maximum usage factors for a 40 year life in any of these components is 0.624 for the flange bolts and 0.5621 for the flange Omega seal welds, well within the analytical limit of 1.0.

In addition, the number of each of the design basis events that affect the reactor pressure vessel and the CRDM housings and appurtenances is not expected to approach its design basis limit during the extended licensed operating period. The actual fatigue usage factors are therefore not expected to approach their calculated values during the extended licensed operating period.

Therefore, this item is dispositioned under 10 CFR 54.21(c)(1)(i), the analysis remains valid for the period of extended operation.

4.3.5 Steam Generator Fatigue Analyses

Summary Description

The Palisades steam generators were replaced in 1990-1991. Extension of the operating license to March 24, 2031 therefore requires a 40 year design life.

The replacement steam generators were designed to the ASME Boiler and Pressure Vessel Code, Section III, 1977. The primary coolant pressure boundary (tube side) of the steam generators is designed to Section III Class 1 rules. Critical components of the Class 2-design secondary side (e.g., the feedwater nozzles) were also analyzed using Class 1 methods.

Analysis

Vessel and Components, Except Manway Studs: The ASME III Class 1 fatigue analyses of the replacement steam generators used the number of event cycles assumed for the original plant design for a 40-year licensed operating period. Except for the manway studs (below), the maximum usage factor at any location is 0.9158, on the main feedwater nozzle.

Manway Studs: The original ASME III Class 1 fatigue analyses calculated a worst-case usage factor for the manway studs of 0.10. However the vendor (Westinghouse) later issued a Nuclear Safety Advisory Letter identifying a significant bending load on the studs due to differential thermal expansion during the heatup and cooldown transients, which resulted in predicted lifetime usage factors greater than 1.0. The revised analysis found that the fatigue limit for the studs would be reached in about 200 reactor heatup and cooldown cycles.

Plant procedures require a periodic evaluation of the number of heatup and cooldown cycles experienced by the Primary Coolant System to assure that the manway studs are replaced before they can experience 200 heatup and cooldown cycles. Since the studs were installed at Plant Heatup Number 106, they should be replaced before Heatup Number 306. The manway stud fatigue analysis determines the evaluation interval. It does not qualify the studs for the licensed operating life, and is therefore not a TLAA.

Disposition: 10 CFR 54.21(c)(1)(i)

The replacement steam generator fatigue analyses qualify them for a 40 year design life, except for the manway studs. The analysis for the manway studs is not a TLAA, and are managed separately. The qualified 40 year design life is sufficient for the extended licensed operating period ending in 2031. Therefore, this item is dispositioned under 10 CFR 54.21(c)(1)(i), the analysis remains valid for the period of extended operation.

4.3.6 Pressurizer Fatigue Analyses

Summary Description

The pressurizer is a vertical, hemispherical-head vessel, supported from its lower head by a flanged skirt, connected to the primary coolant system by a 12 inch surge nozzle in the center of the lower head. The lower head also has 120 electric heater nozzles and 4 level instrument nozzles. There is a liquid-temperature element nozzle in the lower shell. The upper head has a manway, a spray nozzle for pressure control, 4 level instrument nozzles, a power-operated relief valve (PORV) and 3 code relief valve nozzles, and a vapor-space-temperature-element nozzle. The surge and spray nozzles have thermal sleeves.

The vessel shell, heads, skirt, surge nozzle, and manway are carbon steel. The shell and upper head are clad with 304 stainless steel. The lower head and surge nozzle are clad with Ni-Cr-Fe alloy (Alloy 600). The heater sleeves and the remainder of the nozzles are Alloy 600 or have Alloy 600 safe ends (or flanges, for the code safety valves), except the PORV nozzle safe end, which was modified as described below.

The pressurizer was designed to the ASME Boiler and Pressure Vessel Code, Section III, Class A, 1965, Winter 1966 addenda. The code design calculation includes a fatigue analysis for those nozzles or other parts which do not meet the fatigue analysis exemption criteria of Section III Paragraph N-415.1. These include all nozzles attached by J-welds and other nozzles and parts subject to more-severe thermal transients, as follows:

- The surge, spray, and temperature element nozzles and nozzle-to-shell or nozzle-to-head junctions
- Heater sleeve-to-head junctions
- Upper level nozzles
- Relief (PORV) and safety valve nozzles
- The liquid-vapor boundary region of the shell
- Manway, head, and studs, and
- Bottom head support skirt.

A revised set of external load cycles required reevaluation of fatigue in the three safety relief valve nozzles. The recalculation found that the stress intensities produced by the revised external loads are less than those calculated by the original analysis, and, therefore, that the original simplified fatigue evaluation remains valid.

The Alloy 600 safe end to-pipe weld at the power-operated relief valve nozzle was found cracked and leaking in 1993. It was repaired by two welds and a short section of stainless pipe, and the nozzle was reanalyzed. The highest usage factor calculated for the modified safe end and its connections is 0.7572 at the inside of the nozzle-head juncture. This recalculation assumed load cycles for a 40 year design life.

Analysis of the PORV nozzle safe end material removed in 1993 indicated primary water stress corrosion cracking (PWSCC), and prompted replacement of the remainder of the safe end with 316 stainless material with Alloy 690 welds, in 1995. The fatigue analysis for the currently-installed safe end and attachments calculated a worst-case usage factor of 0.084, at the inside wall of the safe end transition. See Section 4.7.2.

Other TLAAs of the Pressurizer: Thermal stratification phenomena in the surge line have required reanalysis of the surge nozzle, and concerns for high differential temperatures with auxiliary spray have required reanalysis of the spray nozzle. See Section 4.3.9 for a discussion of the surge and spray nozzles.

Primary water stress corrosion cracking (PWSCC) of the of the Alloy 600 temperature nozzles required repair, a revised fatigue analysis, and analyses of the PWSCC effects. These failures, failure of the PORV nozzle safe end, and industry-wide cracking of Alloy 600 components have required evaluation of PWSCC effects in all Alloy 600 components. See Section 4.7.2 for additional discussion.

Analysis

This section addresses the remaining applicable portions of the original code analysis and its addenda and revisions, including those for the currently-installed 316 stainless PORV nozzle safe end. This section specifically addresses the calculated fatigue usage factors for the following:

- Bottom head support skirt
- Heater sleeve-head junctions
- Liquid-vapor boundary region of the shell
- Upper level nozzles
- Power-operated relief valve (PORV) nozzle
- Code safety valve nozzles, and
- Manway, head, and studs

For these locations the worst-case calculated usage factor is 0.7572, at the inner PORV nozzle-head juncture.

Disposition: 10 CFR 54.21(c)(1)(i) and 10 CFR 54.21(c)(1)(iii)

The fatigue analysis of the replaced PORV nozzle safe end was based on a nominal 20 year life beyond its 1995 installation. However, the low worst-case usage factor of 0.084 in this component permits a simple projection to the end of the extended licensed operating period, when the service life of this component would be about 36 years. A projected usage factor based on this 36 year life would be only about 0.15, compared to the allowable 1.0.

For those portions of the original pressurizer analysis which have not been superseded, the number of each of the design basis events is not expected to approach its design basis limit during the extended licensed operating period, and, therefore, the actual fatigue usage factors are not expected to approach their calculated values during the extended licensed operating period. The calculated maximum usage factor for these locations on the pressurizer is 0.7572, within the analytical limit of 1.0, at the inner PORV nozzle-head juncture.

These items, therefore, are dispositioned under 10 CFR 54.21(c)(1)(i), the analysis remains valid for the period of extended operation.

The Fatigue Monitoring Program, described in Appendix B, will ensure that the bases for the modified PORV nozzle reanalysis, and for those portions of the original pressurizer analysis which have not been superseded, remain valid.

4.3.7 Regenerative Heat Exchanger Fatigue Analyses

Summary Description

The regenerative heat exchanger recovers energy from the letdown line to heat chemical and volume control system charging water for reactor system makeup (charging) and auxiliary pressurizer spray. The heat exchanger letdown or tube side is separated from the primary coolant system by a normally-open manual valve and a control valve, and is, therefore, a Consumers Design Class 1 component. The charging or shell side is separated from the primary coolant system by a check valve and control valve on each of the charging and auxiliary spray lines.

The charging system has three positive-displacement pumps. The pumps can not be throttled without lifting discharge safety valves, but one of them is variable speed. The charging system was designed to operate continuously, with flow from the fluid-drive variable-speed pump controlled by a primary coolant volume control signal. If the variable-speed pump is out of service, the system controls Primary Coolant System makeup by cycling a constant-speed pump on and off. This cycling of cold makeup water against the (approximately constant) hot letdown flow produces significant thermal transients in the regenerative heat exchanger.

Isolation of letdown flow introduces a similar differential thermal load and has a similar effect.

Analysis

The original design included a fatigue evaluation to ASME III -1965, Paragraph N-415. The final addendum to the original analysis was based on a revised definition of the load transient set which eliminated the 15 percent per minute load-following transient for this component. The worst-location cumulative usage factors (CUF) for a 40-year licensed operating life were 0.871 on the tubesheet, and 0.624 on the shell.

By 1993, operating experience had shown that the availability of the variable-speed pump was less than anticipated in the original design, and therefore that the number of thermal cycles from cycling a constant-speed pump was greater than anticipated. The expected number of lifetime thermal cycles and the fatigue usage factor at the most limiting location (the tube sheet) were therefore reevaluated. The transient evaluation increased the lifetime number of thermal stress cycles due to this event from 5,520 to 17,822. The fatigue evaluation found that, with the increased number of thermal cycles described above plus the remainder of the design basis event set, the CUF at the limiting tube sheet location would be 1.002, slightly above the analytic limit of 1.0. However, the design basis event set included 60,500 load-following and reactor trip events (Transient I), of which the unit had experienced only about 180 by that date (1993). The fatigue evaluation therefore reduced the design basis Transient I cycles slightly (to 60,282), and demonstrated that the analytic limit of 1.0 was then met.

A revision in 1995 to permit more frequent letdown isolation re-evaluated both thermal and pressure transient effects. The evaluation of stress pairings for Transient I load-following and reactor trip events found that those with the maximum stress range could be further reduced to 32,500 from the 60,282 assumed in the 1993 analysis, even considering the increased frequency of letdown isolations to 17,822 from 5,520 cycles. The 1995 revision shows that the highest design basis CUF is 0.880 at the inner ligament of the shell side of the tubesheet.

Based on the plant events in the first 20 years of operation, a recent revision to the calculation estimates that the number of cycles for cycling the constant-speed charging pump can be increased to 27,062, and Transient I can be reduced to 6,240 for 60-year plant life. With the new estimated numbers of transient events at the end of extended operating period, the maximum fatigue usage factors at the two most critical locations in the regenerative heat exchanger (tubesheet and tubesheet to shell junction) is 0.439.

Disposition: 10 CFR 54.21(c)(1)(ii); and Aging Management, 10 CFR 54.21(c)(1)(iii)

Re-analysis has been performed to include additional thermal cycles from cycling the constant-speed charging pump and reduced number of transients that were over estimated in the original design analysis. The projected CUF at the end of the extended operating period remains less than 1. Therefore, the regenerative heat exchanger meets the criteria of 10 CFR 54.21(c)(1)(ii).

The fatigue management cycle count program will include the letdown isolation and variable-speed charging pump out-of-service events.

4.3.8 ASME III Class A Primary Coolant Piping Fatigue Analyses

Summary Description

A piping fatigue analysis was originally applied only to the main loops of the primary coolant system, the two 42 inch hot legs and the four 30 inch cold legs, and to the connecting nozzles for smaller piping. The hot and cold legs are seam-welded from ASTM A516 Grade 70 plate, clad by roll-bonding with 1/4 inch (nominal) 304L stainless. The hot and cold legs are supported only by the reactor and steam generator nozzles and by the primary coolant pumps. The original analyses calculated fatigue usage factors for the:

- Hot legs
- Cold legs
- Safety injection-shutdown cooling nozzles
- Hot leg to surge line nozzle
- Charging Inlet nozzles
- Hot leg temperature nozzles
- Shutdown cooling outlet nozzle
- Cold leg temperature nozzles

The hot leg to surge line nozzle has been reanalyzed to address transients not contemplated in the original analysis. See Section 4.3.9. for additional discussion

The cold-leg-to-pressurizer-spray nozzles and others, not listed above, were exempt from a fatigue calculation in the original analysis, because all then-known cyclic stress ranges were below the endurance limit. However the cold-leg-to-pressurizer-spray nozzles have since been evaluated for additional transients not contemplated in the original analysis. See Section 4.3.9 for additional discussion.

Analysis

This section addresses the remaining original analyses for the hot and cold legs, and for the remaining original nozzles.

The fatigue analysis of record for the hot and cold legs is not location-specific. This analysis uses worst-case stresses at all locations in each of a typical hot leg and cold leg to determine the worst-possible-case stress ranges, and from them calculates a worst-possible usage factor for a typical hot leg and for a typical cold leg. The calculated CUFs were 0.07551 and 0.7531, respectively. These usage factors are, therefore, considerably higher than would be calculated by a location-specific analysis. The 0.7531 cold leg usage factor is higher than any for the nozzles in these original calculations.

The safety injection-shutdown cooling and charging inlet nozzles are also sample locations for evaluation of effects of the reactor coolant environment on fatigue behavior. See Section 4.3.14 below for additional discussion.

Disposition: 10 CFR 54.21(c)(1)(i) and 10 CFR 54.21(c)(1)(iii)

The number of each of the design basis events that affect the hot and cold legs is not expected to approach its design basis limit during the extended licensed operating period, and therefore that the actual fatigue usage factors are not expected to approach their calculated values during the extended licensed operating period. The calculated maximum usage factor for the hot and cold legs and their nozzles in these original calculations is 0.7531, well within the analytical limit of 1.0. In addition, the predicted hot leg and cold leg usage factors are calculated on a very conservative basis. The hot leg usage factor is quite low, and the cold leg usage factor, though appearing significant at 0.7531, would be much less if calculated by a location-specific analysis. Therefore, this item is dispositioned under 10 CFR 54.21(c)(1)(i), the analysis remains valid for the period of extended operation.

The Fatigue Monitoring Program, described in Appendix B, will ensure a reanalysis or other appropriate corrective action if a design basis cycle count limit is reached at any time during the extended licensed operating period.

4.3.9 Revised NRC Bulletin 88-11 Fatigue Analysis of the Hot Leg to Pressurizer Surge Line Nozzle, Surge Line, and Pressurizer Surge Nozzle

Summary Description

NRC Bulletin 88-11, dated December 1988, was issued to address pressurizer surge line temperature stratification concerns. The effects of thermal stratification were evaluated by the Combustion Engineering Owners Group. The Combustion Engineering Owners Group Report concluded the structural integrity of the pressurizer surge line is acceptable for the forty year life of the Plant. The NRC issued an SER on September 13, 1993 concluding that the CEOG analysis adequately demonstrates that the bounding surge line and nozzles meet ASME Code stress and fatigue requirements for the 40-year design. Consumers provided additional information detailing completion of the required actions of Bulletin 88-11, including the requirement to update the pressurizer surge line stress and fatigue analyses. See FSAR Section 4.3.7.

Analysis of Surge Line

For both nozzles and for the worst-case location in the surge line (an elbow) the calculated usage factor for the revised set of design basis load events is within the code limit of 1.0.

Pressurizer Surge Line Elbow: The fatigue evaluation for a typical Combustion Engineering plant with intermittent pressurizer spray and for the revised set of design basis load events, including the IEB 88-11 thermal stratification transients, calculated a maximum CUF of 0.937 at one of the surge line elbows.

A recent fatigue analysis, using thermal stratification conditions under the Palisades continuous pressurizer spray operation, shows that the CUF is reduced significantly to 0.0135 for the expected number of cycles at the end of the 60-year operating period. If the number of cycles at the end of the extended operating period were based on 1.5 times the 40-year design basis cycles, the CUF at the surge elbow would be 0.0447.

This location is also a NUREG/CR 6260 sample location for evaluation of environmental effects of the reactor coolant on fatigue effects, which is discussed in Section 4.3.14.

Hot Leg to Surge Line Nozzle: The fatigue evaluation for a typical Combustion Engineering plant with intermittent pressurizer spray and for the revised set of design basis load events, including the IEB 88-11 thermal stratification transients calculated a maximum CUF of 0.3818 for Palisades hot leg to surge line nozzle. With Palisades' continuous pressurizer spray operation, the CUF is reduced significantly, similar to the above surge line elbow, because piping loads due to thermal stratification are the major contributor to nozzle fatigue stress.

Pressurizer Surge Nozzle: The fatigue evaluation for a typical Combustion Engineering plant with intermittent pressurizer spray and for the revised set of design basis load events, including the IEB 88-11 thermal stratification transients, calculated a maximum CUF of 0.9611 for the Palisades pressurizer surge line nozzle. With Palisades' continuous pressurizer spray operation, the CUF is reduced significantly, similar to the above surge line elbow, because piping loads due to thermal stratification are the major contributor to the nozzle fatigue stress.

Design Basis Thermal Transients and Expected Thermal Transients: The additional thermal stratification transients are in the pressurizer surge line and nozzles during plant heatup and cooldown at differential temperatures of 320, 250, 200, and 150º F ΔT, and hot standby at 90º F ΔT. These design transients were developed by the CE owners group for a typical plant with severe thermal transients due to intermittent pressurizer spray.

The use of modulated, continuous spray for pressure control, and control of pressurizer to primary loop ΔT, significantly mitigates this problem at Palisades. An assessment of the thermal stratification event mechanisms for Palisades' operating conditions found that for almost all such events the metal ΔT would not exceed 210º F, instead of the 320º F ΔT assumed by the standard plant analysis. This is further supported by the log of pressurizer spray events at ΔT above 200º F, which has recorded only 47 events through 9 January 2005. These transients are auxiliary spray events, which have little or no effect on surge line stratification.

This moderation of the transients reduces the piping differential expansion loads and support and nozzle reactions. A recent fatigue analysis using thermal stratification conditions under the Palisades continuous pressurizer spray operation show that the CUF at the surge line elbow is reduced to less than 0.1 at the end of the extended operating period. Since piping load is the major contributor to the nozzle fatigue stress, a similar reduction is expected for the surge line nozzles.

Design Basis Cycle Count and Expected Cycle Count: The number of design cycles developed by the CE Owners Group for each of the above thermal stratification transients correspond to 500 cycles of plant heatup and cooldown. The number of transient events which might be expected to initiate these thermal stratification events will not exceed their design basis limits for the extended licensed operating period; the same is, therefore, also true for these thermal stratification events.

Disposition for Surge Line Elbow: 10 CFR 54.21(c)(1)(ii); and 10 CFR 54.21(c)(1)(iii)

A plant-unique calculation for the surge line shows that the fatigue usage factor of the surge line elbow remains less than 1 at the end of the extended operating period. The calculation includes the revised set of design basis load events; including the plant-unique IEB 88-11 thermal stratification transients. Therefore, the surge line elbow meets the revision criteria per 10 CFR 54.21(c)(1)(ii).

The Fatigue Monitoring Program, described in Appendix B, will ensure a reanalysis or other appropriate corrective action if a design basis primary coolant system cycle count limit is reached at any time during the extended licensed operating period.

Disposition for Hot Leg and Pressurizer Surge Nozzles: 10 CFR 54.21(c)(1)(i); and 10 CFR 54.21(c)(1)(iii)

The design basis analysis calculated maximum usage factor at the hot leg surge nozzle of 0.3818 is well below the analytical limit of 1.0. The usage factor remains less than 1 at the end of the extended operating period. Therefore, the surge line elbow and the pressurizer surge nozzle meet the validation criteria per 10 CFR 54.21(c)(1)(i).

The calculated maximum usage factor for the pressurizer surge nozzle of 0.9611 is below the analytical limit of 1.0. This value is the result of a generic plant analysis, which assumed worst-case stratification through the entire surge line, and which calculated transients based on intermittent pressurizer spray. Similar to the CUF of the surge line elbow, the CUF of the pressurizer is expected to reduce considerably, if the moderation of the transients of the Palisades continuous pressurizer spray is used.

As mentioned above, the number of design cycles for surge line stratification flow developed by the CE Owners Group for each of the above thermal stratification transients correspond to 500 cycles of plant heatup and cooldown.

The Fatigue Monitoring Program, described in Appendix B, will ensure a reanalysis or other appropriate corrective action if a design basis primary coolant system cycle count limit is reached at any time during the extended licensed operating period.

4.3.10 Revised Fatigue Analysis of Nozzles from PCS Cold Legs 1B and 2A to Pressurizer Spray and of the Pressurizer Spray Nozzle

Summary Description

Pressurizer spray is normally supplied by reactor coolant pump head through 3 inch nozzles on two of the four 30 inch primary coolant system (PCS) cold legs. Normal spray flow in each of these 3 inch lines is continuous, through a normally throttled 3/4 inch main spray bypass valve, and through a 3 inch main spray control valve. The charging line downstream of the regenerative heat exchanger supplies auxiliary spray from the chemical and volume control system through a 2 inch control valve. All three of these sources supply a single pressurizer spray nozzle.

The original design of the pressurizer included a fatigue analysis of the pressurizer spray nozzle. However, the normal spray piping and the auxiliary spray piping were designed to the B31.1 Code. Revised operating conditions and pressurizer cooldown rate prompted addition of a fatigue analysis for the two cold leg nozzles and for the auxiliary spray piping (see Section 4.3.11).

Analysis of Other Nozzles

Cold Leg Nozzles to Pressurizer Spray: The analysis of the auxiliary spray-reverse flow events is based on the design basis number of thermal cycles assumed for 40 years, and the calculated cumulative usage factor is 0.66.

Pressurizer Spray Nozzle: The revised pressurizer spray nozzle analysis determined that the calculated cumulative usage factor (CUF) is 0.8214 for the design basis number of high-differential-temperature spray events and 200º F per hour cooldowns assumed for 40 years, and for all other applicable transients.

The revised analysis also estimated that the maximum CUF in the spray nozzle to that date (October, 1991) was 0.353, and that accumulation at then-current trends indicated a 40-year lifetime CUF of about 0.435. On that basis a projection to the end of a 60-year extended licensed operating period would indicate a CUF of about 0.517.

Disposition Cold Leg to Pressurizer Spray Nozzles: 10 CFR 54.21(c)(1)(i)

The number of thermal cycles for each of the design basis temperature differential ranges will not exceed the design basis limit during the extended licensed operating period. The calculated 40-year plant life usage factor of 0.66 is well below the analytical limit of 1.0. Thus the usage factor remains less than 1 at the end of the extended licensed operating period. Therefore, the cold leg to pressurizer spray nozzles meets the validation criteria per 10 CFR 54.21(c)(1)(i).

Disposition Pressurizer Spray Nozzles: 10 CFR 54.21(c)(1)(iii)

The design basis cumulative usage factor (CUF) of the nozzle is 0.8214 for the original 40-year licensed operating period. The projected number of cycles of the design basis events does not exceed the design basis limit during the extended licensed operating period. Therefore, the actual fatigue usage factor is not expected to approach the calculated value at the end of the extended licensed operating period.

The fatigue cycle count program, evaluates the severity of each event conservatively. The cycle count program will ensure a reanalysis or other appropriate corrective action in the unlikely event that a design basis cycle count limit is reached at any time during the extended licensed operating period.

4.3.11 Pressurizer Auxiliary Spray Line Tee Fatigue Analysis in Response to NRC Bulletin 88-08

Summary Description

NRC Bulletin 88-08 and supplements describe observed effects of thermal cycling and thermal stratification in reactor coolant system pressure boundary components due to thermally-driven cyclic inleakage at isolation valves and similar phenomena. In 1989 a conservative, bounding analysis of the section of the Palisades auxiliary spray line from check valve CK 2118 to the pressurizer spray line tee demonstrated that fatigue due to these effects would be acceptable for the then-remaining 30 year licensed operating life.

The piping material is A 376 Type 316. The auxiliary spray line connects to the normal spray line vertically and below, so that the cooler auxiliary spray water does not cause a thermal stratification effect. The analysis assumed 500 lifetime operating basis earthquake (OBE) cycles and 500 lifetime full-range thermal expansion (heatup and cooldown) cycles. It modeled the Bulletin 88-08 phenomena (due to inleakage through the check valve) as a thermal cycle every two minutes, between 536 and 400 ºF, or 1.84×10^5 per year at a 70 percent availability factor.

The 500 lifetime OBE cycles are assumed to occur only once in a design lifetime, independent of its length. The analysis also conservatively attributed the entire 500 full-range thermal expansion to the remaining 30 year life. These OBE and full-range thermal cycles contributed a negligible 0.0063 usage factor. The assumed Bulletin 88-08 cycles contributed 0.4245 in 30 years, for an end-of-life cumulative usage factor (CUF) of 0.43

Analysis

License renewal will add an additional 20 years to the design life assumed by the analysis. Increasing the lifetime contribution of the OBE cycles and full-range thermal expansion cycles by 50% (a conservative assumption for OBE, and given the plant history, also for the full-range thermal cycles) results in a contribution of only 0.0095. Increasing the assumed remaining design life to 50 years results in a contribution from the Bulletin 88-08 cycles of 0.7075, for a 60-year CUF of 0.717, well within the allowable of 1.0.

The analysis assumed 70 percent plant availability. Recent experience has been about 90 percent, which would increase the contribution of the Bulletin 88-08 cycles to about 0.91, or a CUF of about 0.92.

Disposition: 10 CFR 54.21(c)(1)(i), the analysis remains valid for the extended operating period

The assumptions and methods of the analysis are otherwise very conservative, and this result is still within the allowable of 1.0.

4.3.12 Absence of a TLAA for ASME III Class 1 HELB Locations and Leak-Before-Break Analyses Based on Fatigue Usage Factor

Review of the Palisades licensing basis and the associated HELB reports revealed that selected break locations, either inside or outside containment, were not dependent on aging factors. Therefore, HELB analyses at Palisades are not TLAAs.

4.3.13 Assumed Thermal Cycle Count for Allowable Secondary Stress Range Reduction Factor in Piping and Components

This section addresses the issue of assumed thermal cycle counts which determine an allowable secondary stress range reduction factor for some Consumers Power Design Class 1 piping and components, and for non-Consumers Power Design Class 1 piping and components.

Summary Description

Only the primary coolant system piping and components have an ASME Class 1 fatigue analysis (see Section 4.3.8). Other Consumers Power Design Class 1 piping and components were designed to the ANSI B31.1 Power Piping Code, ASME Section VIII, or ASME III, Class 2 and 3, which requires a stress range reduction factor to the allowable stress range for secondary (expansion and displacement) stresses to account for thermal cycling. For ANSI B31.1 the allowable secondary stress range is 1.0 S_A for 7,000 equivalent full-range thermal cycles or less. The allowable secondary stress range is reduced to 0.5 S_A for thermal cycles greater than 100,000. Components designed to other codes, such as ASME VIII, have identical or very similar provisions. An increase in design life could increase the number full-range thermal cycles, therefore, design analyses under these codes are TLAAs.

Some piping within the scope of license renewal was originally designed and built to the American Standard (ASA) Code for Pressure Piping, Section 1, “Power Piping Systems,” 1955 edition.[1] However, during the implementation of IE Bulletin 79-14 work, CP Co Design Class 1 piping, except the main primary coolant piping, was designed to the USAS B31.1.0 (1967) Power Piping Code. In 1992, as a result of discussions between CP Co and the NRC, for new and existing CP Co Design Class 1 piping (except the main primary coolant piping), the code of record was changed to ANSI B31.1 (1973) Power Piping Code with the Summer (1973) Addenda (FSAR 5.10.1.1).

With regard to the stress range reduction factors and corresponding thermal cycle count assumptions, the Consumers Power Class 2 and 3 piping systems and components are designed to the same requirements as CP Co Design Class 1 piping.

The review of possible TLAAs found no Palisades piping and components design analyses to the B31.1 rules, which invoke lower stress range reduction factors for an increase in the equivalent full-range thermal and displacement cycles.

Analysis

The number of lifetime (full-range and equivalent) thermal and other displacement cycles applicable to most of the Palisades B31.1 piping and components are expected to be similar to the plant events defined in Table 4.3.1-1. Therefore, so long as the assumed number of the plant design basis event cycles is not exceeded, the secondary stress range reduction factors assumed for these B31.1 piping and components, and similar code designs remain valid.

Results of the TLAA fatigue review for B31.1 piping and similar code designs for mechanical systems within the scope of license renewal and with operating temperature in excess of 220º F for carbon steel or 270º F for austenitic stainless steel, revealed only two piping systems that have additional cycles that exceed the 7000-cycle limit of the B31.1 Code. These systems include the charging lines inboard of the regenerative heat exchanger, which experience an increase in partial-range thermal cycles due to cycling of the fixed-speed pumps. The original 6,000 events increased to 18,000 for 40-year and 27,000 for 60-year life. The effects of additional cycles have been evaluated for the Regenerative Heat Exchanger fatigue and for the Charging Inlet nozzle (see Section 4.3.7 and Section 4.3.8). The calculation will be revised to include the effects of the additional cycles on charging lines. The other system is the PCS hot leg sampling piping, which may exceed 7,000 cycles during the period of extended operation. A calculation will be performed to justify PCS

1.Predecessor to the ANSI B31.1 Power Piping Code, and sometimes identified as “ANSI B31.1 1955." The forward to USA Standard USAS B31.1.0 1967, the first edition of USAS-ANSI B31.1, similarly mentions “B31.1 1955." However the 1955 Power Piping Code preceded both the USAS and ANSI designations, and it was not numbered.

sampling to occur at any reasonable frequency for 60 years of operation without exceeding the allowable number of cycles.

Disposition for the Charging Lines Inboard of the Regenerative Heat Exchanger: Revision, 10 CFR 54.21(c)(1)(ii)

NMC will evaluate the effect the increase in variable speed charging pump out-of-service events may have on these lines, and will take actions necessary to ensure these lines meet licensing basis design criteria for the extended operating period. NMC will complete this evaluation and will advise the NRC of the results, and of any necessary corrective actions, before the end of the current licensed operating period.

Disposition for Other Piping and Components: Validation, 10 CFR 54.21(c)(1)(i), and Aging Management, 10 CFR 54.21(c)(1)(iii)

The number of lifetime (full-range and equivalent) thermal and other displacement cycles applicable to most of the Palisades B31.1 piping and components are expected to be similar to the plant events defined in Table 4.3.1-1. The number of each of these events is not expected to exceed the existing 40-year design basis for the 60-year extended licensed operating period.

The Fatigue Monitoring Program will ensure reanalysis or other appropriate corrective action in the unlikely event that a design basis cycle count limit is reached at any time during the extended licensed operating period.

4.3.14 Effects of Reactor Coolant System Environment on Fatigue Life of Piping and Components (Generic Safety Issue 190)

Summary Description

The effects of the reactor coolant environment may need to be included in the calculated fatigue life of components. GSI-190 addressed this issue. Although the parent GSI-190 safety issue has been resolved, NUREG-1800, Section 4.3.1.2, states that “The applicant's consideration of the effects of coolant environment on component fatigue life for license renewal is an area of review.” The GSI-190 review requirements are therefore imposed by the Standard Review Plan and do not depend on the individual plant licensing basis.

Analysis

NUREG/CR 6260, Table 5-43, identifies seven sample locations for older Combustion Engineering plants:

- Reactor Vessel (Lower Head to Shell Transition)
- Primary Coolant Inlet Nozzle
- Primary Coolant Outlet Nozzle

- Surge Line Elbow
- Charging Nozzles
- Safety Injection Nozzles
- Shutdown Cooling Line Inlet Transition.

Of these:

- Palisades has no "shutdown cooling line inlet transition." The safety injection and shutdown cooling functions share a common nozzle.
- NUREG/CR-6260 evaluated a long-radius elbow in the surge line because, in the sample plant, this was the highest usage factor location in the surge line and nozzles subject to NRC Bulletin 88-11 reanalysis for thermal stratification cycles. In the NUREG/CR-6260 sample plant this component is SA-376 Type 316; at Palisades the surge line material is a similar Type 316. The only Palisades location on this line with a plant-specific fatigue analysis is the hot leg nozzle to the surge line. The analysis for the surge line elbow is for a typical C-E PWR with intermittent pressurizer spray, with thermal stratification transients; and the pressurizer surge nozzle analysis uses pipe loads from the same source. The results are therefore more conservative than would be expected for Palisades, which has a continuously-modulated pressurizer spray and less-severe thermal transients.
- The Palisades charging nozzles are SB 166 Ni-Cr-Fe Alloy 600 instead of the austenitic stainless of the NUREG/CR-6260 sample plant. However, Alloy 600 is evaluated the same as austenitic stainless for these purposes.

Of the seven NUREG/CR-6260, Section 5.2, sample locations for an older Combustion Engineering plant, six are therefore applicable to Palisades. See Table 4.3.14-1, below.

Environmental effects on cracking in the charging and other Alloy 600 nozzles are also addressed in Section 4.7.2, fatigue in the charging nozzles in Section 4.3.8, and fatigue in the surge nozzles in Section 4.3.9.

All of the primary coolant system at Palisades is stainless steel, Alloy 600, or carbon steel with stainless or Alloy 600 clad. Fatigue in clad components is evaluated using base material properties only; that is, as if the coolant is in contact with the base material, consistent with NUREG/CR-6260.

Table 4.3.14-1 Summary of Fatigue Usage Factors at NUREG/CR-6260 Sample Locations Applicable to Palisades

Location	Material	CUF for Design Cycles	F_{en}	NUREG/ CR-5999 CUF for Design Cycles
Reactor Vessel (Lower Head to Shell Transition)	SA-302 Grade B	0.00364	2.53	0.009
Primary Coolant Inlet Nozzle	SA-302 Grade B	0.01702	2.53	0.043
Primary Coolant Outlet Nozzle	SA-302 Grade B	0.115	2.53	0.29
Charging Inlet Nozzles (with Thermal Sleeves)	SB-166 Alloy 600	0.288	15.35	4.428
Surge Line Elbow	SA-376 Type 316	0.0343	15.35	0.526
Safety Injection-Shutdown Cooling Nozzles	SA-516 Grade 70	0.048	1.79	0.085

Disposition: Revision, 10 CFR 54.21(c)(1)(ii); and Aging Management, 10 CFR 54.21(c)(1)(iii)

A plant-specific calculation was performed for the six sample locations applicable to Palisades, adapted from the seven identified in NUREG/CR-6260 for older-vintage Combustion Engineering plants. Detailed environmental fatigue calculation use the appropriate F_{en} relationships from NUREG/CR-6583 for carbon and low-alloy steels and from NUREG/CR-5704 for stainless steels, as appropriate for the material at each of these seven locations.[1]

1.Neither NUREG/CR-5704 nor NUREG/CR-5999 provide interim fatigue curves for Alloy 600 materials. The NUREG/CR-5704/5999 values for stainless steel will therefore be used, consistent with the practice of NUREG/CR-6260.

The calculation determines an appropriate $F_{en(i)}$ for each individual load pair in the governing fatigue calculation, so that an overall F_{en} multiplier on cumulative usage factor (CUF) for environmental effects can be determined for each location. The analysis shows that the fatigue usage factors at all NUREG/CR-6260 sample locations, but the charging nozzle, including the effects of the reactor coolant environment, will remain less than 1.0 for the extended operation period.

The Charging nozzle has an Alloy 600 safe end, and is one of the components in the Palisades Alloy 600 Inspection Program, which specifies inspection methods and inspection frequency. The Fatigue Monitoring Program will also ensure that the original design basis number of load cycles for each loading event is not exceeded. If this occurs, an evaluation will be made and appropriate actions will be taken to confirm the basis of the safety determination.

4.4 Environmental Qualification of Electrical Equipment

10 CFR 50.49, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants,” requires that certain electrical and instrument and control (I&C) equipment be qualified to perform their safety related functions in their harsh accident environments after the effects of in-service aging. The Electrical Equipment Qualification (EEQ) Program was created to establish and maintain the qualification of the applicable plant equipment.

All operating plants must meet the requirements of 10 CFR 50.49 for certain electrical and I&C components important to safety. 10 CFR 50.49 defines the scope of components to be included, requires the preparation and maintenance of a qualification binder that includes component performance specifications, electrical characteristics, and environmental conditions. Compliance with 10 CFR 50.49 provides evidence that the component will perform its intended functions during and after a design basis accident after experiencing the effects of in-service aging.

Summary Description

10 CFR 50.49(e)(5) contains provisions for aging that require, in part, consideration of all significant types of aging degradation that can affect component functional capability. 10 CFR 50.49(e)(5) also requires component replacement or maintenance prior to the end of designated life, unless additional life is established through ongoing qualification. 10 CFR 50.49(k) and (l) permit different qualification criteria to apply based on plant vintage. Supplemental Environmental Qualification (EQ) regulatory guidance for compliance with these different qualification criteria is provided in the Regulatory Guide 1.89, Rev. 1, “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants”, the Division of Operating Reactors (DOR) Guidelines, and NUREG-0588.

The Palisades EEQ Program was established to demonstrate that certain electrical components located in harsh plant environment (that is, those areas of the plant that could be subject to the harsh environmental effects of loss of coolant accident [LOCA], high energy line breaks [HELB] or post-LOCA radiation) are qualified to perform their safety function operation in those harsh environments after the effects of in service aging. The EEQ Program manages applicable component thermal, radiation, and cyclic aging effects for the current operating license period using the qualification methods established by 10 CFR 50.49(f). Maintaining qualification through the extended license renewal period requires that existing EQ evaluations be reanalyzed. A summary of Palisades' application of these 10 CFR 50.49(f) methodologies to the EQ evaluations for the period of extended operations follows:

Analysis

Under 10 CFR 54.21(c)(1)(iii), a Plant EEQ Programs which implements the requirements of 10 CFR 50.49 (as further defined and clarified by the DOR Guidelines, NUREG-0588, and RG 1.89, Rev. 1), is viewed as an Aging Management Program (AMP) for license renewal. Reanalysis of an aging evaluation to extend the qualification of components under 10 CFR 50.49(f) is performed on a routine basis as part of the EEQ Program. Important attributes of the reanalysis of an aging evaluation include analytical methods, data collection and reduction methods, the underlying assumptions, the acceptance criteria, and corrective actions (if acceptance criteria are not met).

Analytical Methods: The analytical models used in the reanalysis of an aging evaluation are the same as those previously applied during the prior evaluation. The Arrhenius methodology is an acceptable thermal model for performing a thermal aging evaluation. The analytical method used for a radiation aging evaluation is to demonstrate qualification for the total integrated dose (that is, normal radiation dose for the projected installed life plus accident radiation dose). For license renewal, one acceptable method of establishing the 60-year normal radiation dose is to multiply the 40 year normal radiation dose by 1.5 (that is, 60 years/40 years). The result is added to the accident radiation dose to obtain the total integrated dose for the component. For cyclical aging, a similar approach may be used. Other models may be justified on a case-by-case basis.

Data Collection and Reduction Methods: Reducing excess conservatism in the component service conditions (for example, temperature, radiation, and cycles) used in the prior aging evaluation is frequently employed for a reanalysis. Temperature data used in an aging evaluation is to be conservative and based on plant design temperatures or on actual plant temperature data. When used, actual plant temperature data can be obtained in several ways, including monitors used for technical specification compliance, other installed monitors, measurements made by plant operators during rounds, and temperature sensors on large motors (while the motor is not running). When used, a representative number of temperature measurements are conservatively evaluated to establish the temperatures used in an aging evaluation. Plant temperature data may be used in an aging evaluation in different ways, such as (a) directly applying the plant temperature data in the evaluation, or (b) using the plant temperature data to demonstrate conservatism when using plant design temperatures for an evaluation. Any changes to material activation energy values as part of a reanalysis are justified on a case-specific basis. Similar methods of reducing excess conservatism in the component service conditions used in prior aging evaluations may be used for radiation and cyclical aging.

Underlying Assumptions: EQ component aging evaluations contain sufficient conservatism to account for most environmental changes occurring due to plant modifications and events. When unexpected adverse conditions are identified during operational or maintenance activities that affect the normal operating environment of a qualified component, the affected EQ component is evaluated and appropriate corrective actions are taken, which may include changes to the qualification bases and conclusions.

Acceptance Criteria and Corrective Actions: The reanalysis of an aging evaluation could extend the qualification of the component. If the qualification cannot be extended by reanalysis, the component is maintained, replaced, or re-qualified prior to exceeding the period for which the current qualification remains valid.

Regulatory Issue Summary (RIS) 2003-09

On May 2, 2003, the staff issued NRC Regulatory Issue Summary (RIS) 2003-09, “Environmental Qualification of Low-Voltage Instrumentation and Control Cables”, providing the results of the staff's technical assessment of GSI-168, following completion of the NRC-sponsored cable test research. The staff's technical assessment of GSI-168 in RIS-2003-09 is stated as follows:

> For license renewal, a re-analysis (based on the Arrhenius methodology) to extend the life of the cables by using the available margin based on a knowledge of the actual operating environment compared to the qualification environment, coupled with observations of the condition of the cables during walk-downs, was found to be an acceptable approach. Monitoring I&C cable condition could provide the basis for extending cable life.

The Palisades Electrical Equipment Qualification Program allows re-analysis for maintaining qualification using the methods described above. In addition, the EEQ Program has procedural requirements in place to monitor and track aging effects of EQ equipment including cables. The requirements are listed below:

- Monitoring equipment condition and equipment performance

- Monitoring environmental conditions of plant areas, and
- Incorporating the results of testing and analysis into the plant maintenance and surveillance program.

Disposition: 10 CFR 54.21(c)(1)(iii)

The EEQ Program will continue to be implemented for the extended operating period in accordance with 10 CFR 50.49. Continuing the existing EEQ Program provides reasonable assurance that the aging effects will be managed and that the EQ components will continue to perform their intended functions for the period of extended operation. Therefore, the disposition of the TLAA is the performance of the existing EEQ Program which meets the requirement of 10 CFR 54.21(c)(1)(iii). See Appendix B, Electrical Equipment Qualification Program, for additional information.

4.5 Concrete Containment Tendon Prestress Analysis

Summary Description

The Palisades containment is a Consumers Design Class 1 structure. It consists of a post-tensioned, reinforced concrete cylinder and dome connected to and supported by a massive, reinforced concrete foundation slab. It is designed to ACI 318-63, with some adaptations of the design rules and equations, as described in FSAR Section 5.8.

The post-tensioning system consists of three groups of tendons:

- 3 sets of 55 dome tendons (165 total), spaced 120 degrees apart and anchored at the vertical face of the dome ring girder.
- 178 vertical tendons anchored at the top surface of the ring girder and at the bottom of the base slab.
- 6 sets of hoop tendons (502 total), spaced 60 degrees apart, each spanning 120 degrees of arc, anchored at 6 vertical buttresses. (The midpoint of each hoop tendon therefore passes under the anchor buttress for two other sets of hoop tendons.)

Palisades tendons are ungrouted. The tendon sheaths (glands, conduits) are filled with a corrosion preventive medium (CPM). Some sheaths were blocked. Tendons were not installed in 2 of the 180 vertical sheaths, nor in 20 of the 522 hoop sheaths. Evaluation of the original containment design determined that these tendons were not required. See FSAR Section 5.8.2, Section 5.8.5.1, and Section 5.8.5.2 for additional information.

Analysis

The original design included a calculation of expected loss of prestress for the plant design life In accordance with ACI 318-63. The calculation evaluated loss of prestress due to friction and initial seating loss, tendon relaxation, concrete elasticity, concrete shrinkage, and concrete creep. FSAR Section 5.8.5.3.1 lists the predicted values for remaining prestress at the end of the 40 year design life. This original analysis was conservative, as demonstrated by a regression analysis of tendon surveillance data from the twentieth and twenty-fifth-year tendon surveillances. This regression analysis indicated that the effective dome, hoop, and vertical tendon forces would remain significantly higher than values predicted by the original relaxation estimates beyond the 40-year licensed operating period.

Periodic surveillances of containment tendons for degradation are required by 10 CFR 50.55a and Palisades Technical Specification 5.5.5. See FSAR Section 5.8.8 for additional information on the existing surveillance program requirements and results. The program is also described in Appendix B, Containment Inservice Inspection Program.

Disposition: 10 CFR 54.21(c)(1)(iii)

Periodic tendon surveillance program activities are implemented as part of the Containment Inservice Inspection Program. The program follows requirements of ASME Section XI Subsection IWL (1998), NRC-approved Relief Requests, 10 CFR 50.55a, including the 61 FR 154 revision, and the guidance of Regulatory Guide 1.35[1]. The program predicts time-dependent lower limits of the lift-off force (predicted lower limits, PLL) for each tendon subgroup by regression analysis of individual tendon surveillance data, and maintains trend lines of the data for each tendon surveyed. The program inspects a sample of tendons from each group (dome, vertical, and hoop) in each inspection interval to confirm that the trend lines remain within the tolerances of the predicted lower limits, and therefore that tendon prestresses will remain above their respective minimum required values (MRV) for the succeeding inspection interval. The program provides for appropriate actions if surveillance data indicate that a trend line may cross its MRV. See Appendix B, Containment Inservice Inspection Program, for a more detailed description of the aging management program.

4.6 Containment Liner Plate and Penetrations Load Cycles

The containment structure is a Consumers Design Class 1 structure. The entire interior surface of the containment structure is lined with 1/4 inch thick steel plate to ensure a high degree of leak tightness. Numerous mechanical and electrical systems penetrate the containment wall through steel penetrations which are welded to the containment liner plate. See FSAR Section 5.8.2 for additional information.

The containment liner plate and penetrations were conservatively designed, in part, to the rules of the ASME Boiler and Pressure Vessel Code, Section III-1965. This code edition

1.The one- and three-year tendon inspections used Regulatory Guide 1.35 Revision 1. Subsequent inspections use Revision 3. Data from the one and three-year inspections cannot be used for predicted lower limit (PLL) calculations because the Regulatory Guide 1.35 Revision 1 methods required detensioning and retensioning the tendons classifies containment as a Class B vessel. A fatigue analysis is required under this code edition only for Class A vessels (reactor coolant pressure boundary, etc.). However the Palisades containment liner and penetration designs use some of the methods and data from Section III, Article 4, for design of Class A vessels for fatigue loads.

4.6.1 Containment Liner Plate Load Cycles

Summary Description

The containment liner plate is stitch welded to a gridwork of structural steel angles embedded in the concrete. The anchoring system is designed to prevent significant distortion of the liner plate during accident conditions and to ensure that the liner maintains its leak-tight integrity. The liner plate has been coated on the inside for corrosion protection. There is no paint on the side in contact with the concrete. See FSAR Section 5.8.2 for additional information.

The Palisades containment design relies on the liner only to maintain a leak-tight containment. There are no design conditions under which the liner plate is relied upon to assist the concrete in maintaining the integrity of the structure. At times, provide assistance in order to maintain deformation compatibility.

Stress concentrations around penetration openings in the liner plate are calculated using the theory of elasticity. These stress concentrations were then reduced, by thickening the liner plate around each penetration in accordance with the ASME B&PV Code, Section III, 1965.

FSAR Section 5.8.3.2.1 states that the following fatigue loads were considered in the design of the liner plate:

1. Thermal cycling due to annual outdoor temperature variations. The number of cycles for this loading was 40 cycles for the Plant life of 40 years. However, the impact of outdoor temperature variations upon liner plate stresses, insulated by the 3'6" concrete containment wall, is negligible in comparison with the stresses caused by the design basis accident temperature. The annual outdoor temperature variations are not the controlling design consideration because the design loads related to accidents result in higher stress conditions.
2. Thermal cycling due to containment interior temperature varying during the start-up and shutdown of the reactor system. The number of cycles for this loading was assumed to be 500 cycles.
3. Thermal cycling due to the DBA was assumed to be one cycle.

The liner plate was analyzed for these fatigue loadings using Figure N-415(a) of the ASME Boiler and Pressure Vessel Code, Section III, Article 4, 1965. Since this figure does not extend below ten cycles, ten cycles were used conservatively for the DBA instead of one cycle as indicated above. The resulting stresses were insignificant.

Analysis

The Palisades containment design relies on the liner to maintain a leak-tight containment. However, there are no design conditions under which the liner plate is relied upon to assist the concrete in maintaining the integrity of the structure. Forces are transmitted between the liner plate and the concrete through the anchorage system and through direct contact (pressure). At times, forces may also be transmitted by bond and/or friction. These forces cause, or are caused by, liner plate strains. The liner plate is designed to withstand the predicted strains. The effect of concrete cracking on the liner plate has also been considered.

The allowable liner plate strains/stress was conservatively based on the ASME B&PV Code, Section III, Article 4, 1965. Specifically, the following sections were adopted as guides in establishing allowable strain limits:

1. Paragraph N-412(m) - Thermal Stress, Subparagraph 2
2. Paragraph N-412(n) - Operational Cycle
3. Paragraph N-414.5, Table N-413, Figures N-414 and N-415(a) - Peak Stress Intensity
4. Paragraph N-415.1 - Vessels Not Requiring Analysis for Cyclic Operation

The liner strains/stresses due the (non-DBA) loads are relatively small such that the number of environmental and operational load cycles is insignificant compared to the allowable number of cycles on the fatigue curve in code Fig. N-415 (a), or compared to 3 times the S_m value of code of Table N-421 at the operational temperature. The results of the analysis confirm that the design of the containment liner complies with the provisions of code paragraph N-415.1 for not requiring a fatigue analysis for design load cycles.

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

Of the design basis fatigue load cycles of the containment liner, only the number of environmental and operational load cycles would increase due to the 60-year extended licensed operating period.

Of these two events, the effect of the assumed summer-winter annual cycles is negligible, and will remain negligible on increase from 40 to 60 cycles for the 60-year extended licensed operating period.

The assumed 500 containment interior operational heatup and cooldown cycles is very conservative, since it corresponds to an average of 8 1/3 cycles per year, or a PCS cooldown and heatup every 6 weeks. This is more than adequate to accommodate the 60-year extended licensed operating period.

Therefore, there will be negligible change in the fatigue resistance of the containment liner for the 60-year extended licensed operating period.

4.6.2 Containment Penetration Load Cycles

Summary Description

FSAR Section 5.8.3 and Section 5.8.6.4 describe design of both the “large penetrations” (personnel air lock and equipment hatch) and the “small penetrations” (escape air lock, piping, ventilation, and electrical penetrations) to limiting strain criteria for a set of design loads, including thermal loads and design basis accident (DBA) loads.

FSAR Section 5.8.6.4.1 describes the design of the pipe penetrations to acceptance criteria for combined primary and secondary stresses under the ASME Boiler and Pressure Vessel Code, Section III-1965, Article 4. Figure N 415(a) of that article is a fatigue S-N diagram (stress versus number of allowed cycles, or vice-versa) for alternating stress intensity Sa versus N, for carbon steel.

Although the FSAR § 5.8.6.4.1 description states (1) that “The allowable value of stress intensity Sa was determined from Figure N 415(a) of the referenced ASME Code article,” and (2) that “For all load combinations, the strains in the pipe penetrations did not exceed [those for which an equivalent Sa would exceed] the values given in the ASME B&PV Code, Figure N 415(a),” it appears that the only event for which the N-415(a) stress criterion was used was the end-of-life design basis event.

Numerous mechanical and electrical subsystems penetrate the containment wall through steel penetrations, which are welded to the containment liner plate. Personnel and equipment access to the containment is provided by a personnel air lock with 2 - 3 foot 6 inch x 6 foot 8 inch doors, an escape air lock with 2 - 30 inch diameter doors, and an equipment hatch with a single door that provides total access to the 12 foot diameter [equipment] passageway. The air locks and hatch were fabricated from ASTM A-516, Grade 70, firebox quality steel, made to the requirements of SA-300, Charpy V notch tested at a temperature of 0º F, and conforming to the requirements of the ASME Boiler and Pressure Vessel Code, Section III.

Containment penetrations are designed to maintain the leak tightness of the containment structure under normal and accident conditions. They are designated as CPCo Design Class 1 components of the containment structure.

Piping, HVAC, and electrical penetrations are designed, fabricated, inspected, and installed in accordance with the ASME B&PV Code, Section III, Subsection B.

The design and fabrication of the personnel airlock, the escape airlock, and all portions of the equipment hatch extending beyond the concrete shell conform to the requirements of the ASME B&PV Code, Section III.

The limiting steel strain/stress in the penetrations is conservatively based on the ASME B&PV Code, Section III, Article 4, 1965.

The inservice functionality of each individual penetration is monitored by means of local leak-rate testing (LLRT, also referred to as Type B Test in 10CFR50, Appendix J) in accordance with Plant Technical Specifications at a frequency of at least every refueling, but not exceeding a two year interval.

Analysis

The allowable liner plate strains/stress was conservatively based on the ASME B&PV Code, Section III, Article 4, 1965. Specifically, the following sections were adopted as guides in establishing allowable strain limits:

1. Paragraph N-412(m) - Thermal Stress, Subparagraph 2
2. Paragraph N-412(n) - Operational Cycle
3. Paragraph N-414.5, Table N-413, Figures N-414 and N-415(a) - Peak Stress Intensity
4. Paragraph N-415.1 - Vessels Not Requiring Analysis for Cyclic Operation

The liner strains/stresses due the (non-DBA) loads are relatively small such that the number of environmental and operational load cycles is insignificant compared to the allowable number of cycles on the fatigue curve in code Fig. N-415 (a), or compared to 3 times the S_m value of code of Table N-421 at the operational temperature. The results of the analysis confirm that the design of the containment liner complies with the provisions of code paragraph N-415.1 for not requiring a fatigue analysis for design load cycles.

<u>Inservice Local Leak Rate Tests (LLRT)</u>

Plant Technical Specifications requires that the inservice functionality of each individual penetration shall be established by means of local leak-rate testing (LLRT, also referred to as Type B Test in 10CFR50, Appendix J) at a frequency of at least every refueling, not exceeding a two year interval, except as specified below:

1. The equipment hatch and the fuel transfer tube shall be tested at each refueling shutdown or after each time used, if that is sooner.
2. A full personnel airlock penetration test shall be performed at six month intervals. During the period between the six month tests, when containment integrity is required, a reduced pressure test for the door seals shall be performed within 72 hours either after each

airlock door opening or after the first of a series of openings. Airlocks opened during periods when containment integrity is not required shall be tested at the end of such periods at full pressure.

Local leak rate tests are to be performed at a pressure of not less than 55 psig.

Local leak rate tests for checking Personnel Airlock door seals shall be performed at a pressure of not less than 10 psig. The Personnel Airlock door seals are tested at a lower pressure due to the design of the doors. This pressure is sufficient to determine Personnel Airlock door seal operability.

Local leak rate tests for the Emergency Escape Airlock shall be performed at 55 psig. A seal contact check shall be performed on the Emergency Escape Airlock following each full pressure test. Emergency Escape Airlock door opening, solely for the purpose of strongback removal and the performance of the seal contact check, does not necessitate additional pressure testing.

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

Of the design basis fatigue load cycles of the containment penetrations, only the number of environmental and operational load cycles would increase due to the 60-year extended licensed operating period.

Of these two events, the effect of the assumed summer-winter annual cycles is negligible, and will remain negligible on increase from 40 to 60 cycles for the 60-year extended licensed operating period.

The assumed 500 containment interior operational heatup and cooldown cycles is very conservative, since it corresponds to an average of 8 1/3 cycles per year, or a PCS cooldown and heatup every 6 weeks. This is more than adequate to accommodate the 60-year extended licensed operating period.

Therefore, there will be negligible change in the fatigue resistance of the containment penetrations for the 60-year extended licensed operating period.

In addition, periodic inservice LLRTs required by Plant Technical Specifications monitor the continued inservice leaktight functionality of each individual penetration through the extended licensed operating period.

4.7 Other Plant-Specific Time-Limited Aging Analyses

4.7.1 Crane Load Cycles

Summary Description

A crane evaluation to the Crane Manufacturers Association of America Standard CMAA-70 assumes a number of rated lifts in the design lifetime in order to establish the design Service Level, and hence the allowable stresses. At Palisades, two cranes have been reanalyzed to CMAA-70 design criteria. The NUREG-0612 heavy loads evaluation of the reactor building polar crane was performed to CMAA 70. A redesign of the Spent Fuel Pool Crane for dry fuel storage also included a NUREG-0612 evaluation to CMAA-70 design criteria. The limiting components of the containment polar crane (135 tons) and the redesigned spent fuel pool crane (110 tons) are now rated for CMAA-70 “Service Level A - Standby or Infrequent Service.”

Therefore, the analyses of the Polar Crane and the Spent Fuel Pool Crane are TLAAs for license renewal purposes.

Analysis

Containment Polar Crane: The polar crane was originally designed to Electric Overhead Crane Institute Specification 61. The subsequent NUREG-0612 heavy loads evaluation of the polar crane was performed to CMAA 70 (1975). The minimally-rated components are CMAA 70 Service Level A. Since the minimally-rated components are Service Level A, the effective crane design life for fatigue or allowed number of rated lifts depends on this classification, which assumes 20,000 to 100,000 rated lifts in a design lifetime.

Separate evaluations have been performed of polar crane planned engineered lifts (over the rated capacity). The evaluations were done to ANSI/ASME Standard B30.2 (1996). Lifts have been evaluated and approved up to 140 T, less than 4 percent over the 135 T rating.

Spent Fuel Pool Crane: The redesign to 110 T for dry cask storage included a NUREG-0612 evaluation to CMAA-70 Service Level A design criteria, and other considerations; and replacement of the trolley with a 110 T single-failure-proof trolley meeting NUREG 0554 guidelines. The structural redesign and evaluation considered the load combinations of ASME NOG 1, Section NOG 4140, “Load Combinations.”

Disposition: 10 CFR 54.21(c)(1)(i)

Polar Crane (L-1): Polar crane rated or near-rated lifts are limited to the reactor head plus CRDMs and insulation, and reactor internals. Only a few rated lifts are performed each refueling outage, and none during operation. Therefore this machine cannot realistically approach the 20,000 to 100,000 rated lifts, assumed for components evaluated to CMAA 70 (1975) Service Level A, during a 60 year licensed operating period.

Spent Fuel Pool Crane: Approximately 11 dry cask storage campaigns are expected between rerating and the end of the 60 year extended license. This will require loading about 64 casks. Each will require about two lifts of 100 T or more per cask, and some additional lifts of between 50 and 100 T. The total for 64 casks and 11 campaigns is about 140 lifts of 100 T or more, and about 162 lifts between 50 and 100 T. Other lifts, and lifts prior to rerating the crane, were determined to be inconsequential. Therefore, this machine can not realistically approach the 20,000 to 100,000 rated lifts assumed for its design evaluation during the 60 year extended licensed operating period.

4.7.2 Alloy 600 Nozzle and Safe End Life Assessment Analyses

Summary Description

Alloy 600 (Ni-Cr-Fe alloy) was used to clad the pressurizer lower head and surge nozzle, for the pressurizer heater sleeves, and for smaller nozzles and safe ends and flanges on larger nozzles of the Palisades reactor vessel head, primary coolant system loop piping, and pressurizer. There were originally 251 Alloy 600 heater sleeves, nozzles, safe ends, and flanges in the Palisades primary coolant system.[1]

By the 1990's Alloy 600 had become recognized as being susceptible to a cracking phenomenon identified as Primary Water Stress Corrosion Cracking (PWSCC). A significant contributing factor to the initiation and propagation of PWSCC was determined to be temperature. Thus, since the pressurizer has the highest operating temperature of any location in the Primary Coolant System (PCS), a reasonable first assumption is that the Alloy 600 in its nozzles would develop PWSCC first.

Evidence of PWSCC was first noted at Palisades in September 1993 when a through-wall crack was found in the pressurizer Power Operated Relief Valve (PORV) nozzle safe end to pipe weld heat affected zone. The crack was found after the PCS leak rate increased while the plant was in hot shutdown. The plant was returned to cold shutdown and the leak was repaired. The cracked section of field weld was removed, the safe end modified for a new weld prep, and a short new section of stainless steel pipe installed to replace the material that was removed. The failed section was saved for thorough analysis of the failure mechanism. The failure mechanism was attributed to PWSCC.

Second and third failures of Alloy 600, apparently by PWSCC, occurred in October 1993 immediately following the repair of the PORV nozzle safe end. Eddy Current inspection identified several axially oriented cracks in the vicinity of the “J groove” weld that attaches the nozzle to the pressurizer head. This type of cracking in this style of weld has been seen

1.Now 250. The PORV nozzle safe end has been has been replaced with 316 stainless plus an Alloy 690 weld.

in multiple other nuclear plants and is caused by PWSCC. An inspection of the lower shell temperature element also identified leakage from around the nozzle. Both temperature element nozzles were repaired by welding a new attachment pad on the exterior of the pressurizer. See FSAR Section 4.3.7 for additional information.

As a result of the PORV nozzle safe end failure and the metallurgical exams performed on the failed material, it was decided that replacement of the safe end with a less susceptible material was prudent. During the 1995 refueling outage, the Alloy 600 PORV nozzle safe end was replaced with a new Type 316 stainless steel safe end/spool piece using Alloy 690 for the attachment weld to eliminate PCS contact with Alloy 600 from the pressurizer PORV nozzle. The NRC reviewed the use of forged (SB-564) Alloy 690 material, and issued an SER.

Due to the previous PWSCC failures, a heightened awareness to the potential for future PWSCC exists for the Alloy 600 penetrations in the entire Primary Coolant system. During the 1995 refueling outage, inspections of many of the Alloy 600 penetrations were performed to establish a reference for future PWSCC inspections. No indication of PWSCC was detected during those inspections. Also during the 1995 refueling outage, the Mechanical Stress Improvement Process (MSIP) was applied to the PCS nozzles on both ends of the pressurizer surge line and to the shutdown cooling outlet nozzle to mitigate any possible PWSCC by removing tensile stresses on the inside diameter of the piping. It consisted of using a hydraulically powered clamp ring assembly to squeeze the piping in the vicinity of the weldment, leaving a permanently deformed ring that is deformed approximately 1% in the radial direction. MSIP is accepted by the NRC for mitigating inter-granular stress corrosion cracking in BWRs. See FSAR Section 4.3.7 for additional information.

The 1995 inspections included eight in-core instrumentation (ICI) nozzles. The Combustion Engineering Owner's Group timing model described in CEOG-97-244 indicated that these nozzles were highly susceptible to PWSCC, due to high hillside angles (54.4º), high yield strength (60 ksi), and long service hours (approximately 100,000 effective full-power hours). Eddy current inspection of the ICI nozzles detected no evidence of cracking.

Analysis

The inspection methods and intervals of the Palisades Alloy 600 aging management program were determined from evaluations of the susceptibility of all 250 remaining heater sleeves, nozzles, safe ends, and flanges to primary water stress corrosion cracking (PWSCC).

Alloy 600 was also used for the primary-side steam generator head drain plugs, steam generator tubes, and reactor flange leak detector taps. None of these were evaluated for

PWSCC, nor included in the Alloy 600 inspection program. The steam generator drain plugs are solid tapered plugs inserted from the inside and therefore not subject to ejection. The steam generator tubes are subject to the tube inspection and aging management program, whose supporting analyses do not depend on licensed design life. The reactor flange leak detector taps are not primary system pressure boundaries.

Assessment of Reactor Vessel Head Vent, In-Core Instrumentation, and CRDM Nozzles; and Pressurizer Relief Valve Flanges, Heater Sleeves, and Level Nozzle Safe Ends: The assessments of the susceptibility of these components to PWSCC are based on industry experience and industry analyses, but are qualitative and do not depend on the licensed operating period.

Fracture Mechanics Assessment of the Most-Susceptible Pressurizer Locations; and of All Primary Coolant System Alloy 600 Loop Penetrations and Nozzle Safe Ends: An ASME XI, IWB-3612 fracture mechanics analysis was performed on 13 types of Alloy 600 nozzles and safe ends on the pressurizer and on the primary coolant hot and cold legs. This analysis omits those locations for which only a qualitative assessment was performed, as described above, and the pressurizer PORV nozzle safe end, which was replaced with other material.

The analysis calculated allowable axial and circumferential flaw depths at three aspect ratios (l/a = 2, 4, and 6) for one 18-month operating cycle in each of these nozzles or safe ends, to confirm that a limiting allowable flaw is detectable; and calculated time to failure (assuming a uniform frequency of design basis cycles) for an initial 0.010-inch-deep axial and circumferential flaw, at each of these three aspect ratios.

Table 4.7.2-1 Summary of Fracture Mechanics Assessment of Susceptible Pressurizer and Primary Coolant System Alloy 600 Locations

Nozzle or Safe End (SE)		Limiting Initial and Allowable Flaw Type (all are l/a = 6)	Limiting Allowable Flaw Depth for an 18-Month Cycle, in.	Life for 0.010" Initial Flaw, yr.	TLAA ?
1	Pressurizer Temperature Element TE-0101 and TE-0102 Nozzles	Axial	0.020	7.68	No
2	Pressurizer Surge SE	Circumferential	0.320	40	Yes

Table 4.7.2-1 Summary of Fracture Mechanics Assessment of Susceptible Pressurizer and Primary Coolant System Alloy 600 Locations

Nozzle or Safe End (SE)		Limiting Initial and Allowable Flaw Type (all are l/a = 6)	Limiting Allowable Flaw Depth for an 18-Month Cycle, in.	Life for 0.010" Initial Flaw, yr.	TLAA ?
3a	Pressurizer Spray SE, 540º F	Circumferential	0.360	40	No
3b	Pressurizer Spray SE, 640º F	Circumferential	0.130	5.36	No
4	Hot Leg Surge SE	Axial	0.320	40	Yes
5	Hot Leg -Shutdown Cooling SE	Axial	0.300	40	Yes
6	Hot Leg Loop Drain Nozzle	Circumferential	0.140	40	Yes
7	Hot Leg Pressure and Sampling Nozzle	Circumferential	0.125	40	Yes
8	Hot Leg RTD J-Weld Nozzle	Axial	0.130	40	Yes
9	Cold Leg Safety Injection-Shutdown Cooling SE	Axial	0.520	40	Yes
10	Cold Leg Spray Nozzle	Circumferential	0.225	40	Yes
11a	Cold Leg Charging Nozzles	Circumferential	0.195	40	Yes
11b	Cold Leg Loop Drain Nozzles	Circumferential	0.195	40	Yes
12	Cold Leg Pressure and Sampling Nozzle	Circumferential	1.180	40	Yes
13	Cold Leg RTD J-Weld Nozzle	Circumferential	0.240	40	Yes

The analysis found that fatigue alone accounted for only about one percent of crack growth. Primary water stress corrosion cracking accounted for the remainder.The analysis also found that a 0.010-inch flaw might grow beyond critical size in a 40-year design life in only the pressurizer temperature nozzles and the spray nozzle safe end, and that in all but the temperature nozzles the limiting allowable flaw depth was greater than that detectable by approved methods.

Values for all but the pressurizer surge and spray nozzle safe end (Locations 2 and 3 above) were calculated with a simplifying assumption of uniform through-wall stress equal to room temperature yield stress. With this assumption, an initial 0.010-inch deep, l/a = 6 axial flaw in the pressurizer temperature nozzles would fail in 7.68 years.

The final calculation for the surge and spray nozzle safe ends assumed the more realistic NUREG-0313 stress distribution model, and found that an initial 0.010-inch deep, l/a = 6 circumferential flaw in the surge nozzle safe end would only fail after 40 years. A similar flaw in the spray nozzle would also endure 40 years at 540º F, but would fail in only 5.36 years at 640º F.

Additional time-dependent analyses have been performed to support safety determinations for these nozzles:

- A fatigue analysis of the replaced PORV safe end. See Section 4.3.6.
- Revised fatigue analyses of the hot-leg-to-surge-line-nozzle and the pressurizer surge nozzle for thermal stratification events. See Section 4.3.9.
- Revised fatigue analyses of the cold-leg-to-spray-line-nozzles and the pressurizer spray nozzle for high-thermal-differential auxiliary spray actuations. See Section 4.3.9.
- Fatigue analyses of the weld pad repair of the pressurizer temperature element nozzles, an ASME IX Appendix A fracture mechanics crack growth analysis of the vessel wall, and additional fracture mechanics and corrosion life assessments of the repaired nozzles, weld repair, and vessel wall inside the penetration, are discussed below.
- A service life assessment of the pressurizer spray and surge nozzle safe ends are discussed below.

Repaired Pressurizer Temperature Element Nozzles: The inner sections of these two nozzles are pieces of Alloy 600 pipe inserted through the pressurizer vessel wall and attached by welding the inner end of the nozzle to the inner wall of the vessel, using a “J”-shaped weld preparation machined in the vessel inner wall. Leaks through the J welds of these two nozzles were repaired by adding a weld-deposited pad on the vessel exterior. In similar situations at other units these nozzles have been repaired by the “half nozzle” method, in which the nozzle is cut within the vessel wall thickness to detach it from the J weld at the inner wall. The outer portion of the nozzle is then welded to the outer surface of the vessel, with a gap left between the nozzle parts to eliminate differential expansion loads. At Palisades these two nozzles were repaired with similar welds without the complete cut, but with a partial electrodischarge machining (EDM) cut of the TE-0101 upper head nozzle, the one subject to the highest differential thermal expansion loads and fatigue usage factor.

The upper head nozzle is subject to the highest fatigue usage, highest temperature, and worst environment, and therefore this nozzle has the most limiting fatigue and fracture mechanics design life of these two nozzles. Fatigue analyses of the weld pad repairs installed in 1993 for the temperature element nozzles were based on half the specified event cycles for the pressurizer, that is, for a nominal remaining life of 20 years. The analysis assumed a very conservative stress concentration factor of 5.0, for weld discontinuities. The maximum calculated fatigue usage factor is 0.823, for the upper head nozzle repair pad.

An ASME XI Appendix A crack growth evaluation of postulated flaws in the pressurizer base metal at the original J-welds of the temperature element nozzles found that flaws left in place would not propagate sufficiently to permit a brittle fracture during the remaining portion of the original 40-year licensed operating period. For crack growth the only significant events are the cool-down transient and system leak tests. The calculation assumed the design basis 500 cool-down events and 320 leak tests. However this analysis was later superseded by a bounding ASME XI Paragraph IWB 3132.4 analysis of the hot leg, piping RTD and sampling nozzles, pressurizer instrument nozzles, and pressurizer heater sleeves, described below.

The fatigue analysis of the weld pad repairs and the ASME XI Appendix A crack growth evaluation are therefore TLAAs. The following additional fracture mechanics and corrosion evaluations of the two pressurizer temperature nozzles support safety determinations, but either the evaluations were not used for the final safety determination, or did not depend on design life, and are therefore not TLAAs.

The fracture mechanics crack growth assessment of the Alloy 600 pressurizer temperature nozzles found that a crack below the detectable size might propagate through the nozzle wall in as few as 7.68 years, for which fatigue contributed only about one percent. A subsequent service life assessment of these nozzles also considered general corrosion in the pressurizer wall, and found that the limiting concern was the previous 7.68-year crack growth assessment. This limit was however not used as the basis for the safety determination of the final justification for continued operation.

The final justification for continued operation (final JCO) of these nozzles, welds, and the adjacent pressurizer wall evaluated corrosion of the carbon steel pressurizer wall, and primary water stress corrosion cracking (PWSCC) and fatigue in the nozzle material and in the repair weld pad. The final JCO found that the 7.68-year service life estimate from the fracture mechanics crack growth assessment is acceptable, because other effects (corrosion) are not limiting, and because the effects of the predicted cracks can be detected before a full circumferential failure can occur.

Corrosion: The carbon steel vessel wall is subject to corrosion because it is exposed to reactor coolant through the observed cracks in both nozzles, and because of the EDM cut through the upper head temperature nozzle. The final JCO cited the above service life assessment, a separate corrosion evaluation report, and examination results of a similar situation at another plant to demonstrate that the corrosion rate is “extremely low,” and to conclude that the nozzles are acceptable, considering this effect, “for more than 20 years.” This statement of the safety determination is not, per se, time-limited, but reflects the intent of the analysis to show that effects are acceptable for at least the original licensed operating period.

The supporting corrosion evaluation in the service life assessment was quantitative, concluding that corrosion might increase the diameter of the bore through the pressurizer base metal surrounding the upper temperature nozzle by about 0.28 inch, and about 0.14 inch at the lower nozzle, for the extended 60 year licensed operating period. This corrosion evaluation did not complete a safety determination on a quantitative basis. The safety determination of the final JCO was made by a qualitative rather than a quantitative assessment of these facts.

PWSCC at the temperature element nozzle Inner Surface: Experience and the expected state of stress indicate that axial cracks may occur but that circumferential cracks are extremely unlikely. Axial cracks result in leaks which are readily detected on inspection, but which do not permit nozzle ejection.

PWSCC at the Weld Pad Root: The weld pad root is exposed to primary coolant only via existing or new cracks in the nozzles, or through the EDM cut in the upper temperature nozzle. The weld material should be less susceptible to PWSCC than the nozzle material, and is thicker than the nozzle material.

This final JCO concluded that corrosion of the base material should be minimal and acceptable for the remaining life of the plant, that PWSCC would produce axial cracks in the nozzle material before any full circumferential failures of either the nozzle or nozzle-to-pad weld region, and which would therefore be detected as minor leakage, and repaired, before a full circumferential break might occur. This determination that a detectable leak will occur before a full circumferential break, plus limited expected corrosion, are the bases for the safety determination. The corrosion assessment was performed for the extended 60-year licensed operating period. The limited, 7.68-year predicted life before a new through-wall axial crack might occur is one basis for the inspection interval for these pressurizer nozzles, but does not qualify the nozzles for a stated design life, and its calculation is therefore not a TLAA.

Pressurizer Spray and Surge Nozzle Service Life Assessment

The Service life Assessment of the Pressurizer Spray and Surge Nozzle Safe Ends summarized the results of the fracture mechanics survey of most-susceptible locations. The assessment noted that with a 40-year predicted life the surge nozzle is of less concern than the spray nozzle, since the spray nozzle safe end predicted life is somewhere between 5.36 years at 640º F and 40 years at 540º F. The Palisades continuous spray operation maintains safe end mean temperature nearer the lower value, with greater life. The assessment made recommendations for material procurement against possible replacement, and for inspections. The inservice inspection plan requires volumetric and surface (penetrant) inspection of the spray nozzle safe end every other refueling outage. For the surge nozzle and others for which the fracture mechanics survey of most susceptible locations predicted a 40-year life, a safety determination is implied in that predicted life. However for the spray nozzle, with a more limited or ambiguous predicted life, no such inference can be drawn. This assessment of the likely range of the remaining life of the spray nozzle supports the inspection plan but supports no explicit safety determination, does not depend on the licensed operating period, and is therefore not a TLAA.

Final NRC Safety Evaluation: The final JCO for the temperature element nozzle repairs, the fracture mechanics and fatigue evaluations of other Alloy 600 components, and the proposed Alloy 600 inspection and aging management program were evaluated by the NRC staff and found acceptable in a letter from the NRC to Consumers Power, “Cracking of Inconel 600 Components”, dated June 27, 1995.

Bounding Fracture Mechanics Analysis of the Hot Leg, Piping RTD and Sampling Nozzles, Pressurizer Instrument Nozzles, and Pressurizer Heater Sleeves: A subsequent, bounding ASME XI Paragraph IWB 3132.4 analysis of these components addressed repairs of small-bore nozzles attached at the inner pressure boundary wall by J-welds whose pressure boundaries have been breached by PWSCC attack in the J-weld penetration areas, and provided a justification for leaving a flaw in the nozzle remnant. The justification includes the evaluations of the effects of corrosion, stress corrosion, stress corrosion cracking, fatigue crack growth and environmental factors.

The bounding fracture mechanics portion of the analysis employs elastic-plastic methods with IWB 3600 and Regulatory Guide 1.161 acceptance criteria. The Palisades analysis demonstrates the validity of the bounding fracture mechanics analysis by demonstrating that the plant-specific load and thermal events are within those assumed by the bounding analysis. The basis for the safety determination of the fracture mechanics evaluation calculation will therefore remain valid so long as the numbers of these events do not exceed the design basis values. This analysis confirms the conclusions of the earlier final JCO.

Disposition: 10 CFR 54.21(c)(1)(i) and 10 CFR 54.21(c)(1)(iii)

Disposition for Fatigue Analyses of the Weld Pad Repairs Installed in 1993 for the Pressurizer Temperature Element Nozzles: 10 CFR 54.21(c)(1)(iii)

NMC will monitor the cumulative number of pressurizer temperature element nozzle fatigue cycles within the Fatigue Monitoring Program, and maintain a special action level to ensure that appropriate actions are taken if at any time the cycle count for any design basis event since 1993 reaches the number assumed by these analyses.

For this purpose the fatigue management program will compare cycle counts since the repair in 1993 to appropriate action levels. Since the fatigue analyses were based on half of the 40-year pressurizer design basis event cycles, the action levels for cycles since then will be about half of the 40-year pressurizer design basis event cycles for each event.

Disposition for Corrosion Life Assessment of the TE 0101 Temperature Element Nozzle Bore in the Carbon Steel Pressurizer Wall: 10 CFR 54.21(c)(1)(i)

The evaluation estimated a repair lifetime of 52.3 years for this effect following initiation of leakage. Leakage was first detected in 1993, after 22 years of operation, which indicates that the pressurizer wall can withstand this effect for a total plant life of over 70 years. Therefore, the current analyses remain valid for the period of extended operation.

Disposition for Cycle-Dependent Aspects of the Bounding Fracture Mechanics Analysis of the Hot Leg, Piping RTD and Sampling Nozzles, Pressurizer Instrument Nozzles, and Pressurizer Heater Sleeves; and Disposition for Fatigue Portions of All Other Alloy 600 Fracture Mechanics Analyses for a 40-Year Design Life: 10 CFR 54.21(c)(1)(i) and 10 CFR 54.21(c)(1)(iii)

The Palisades plant-specific bounding fracture mechanics analysis demonstrates the validity of the cycle-dependent aspects of the generic bounding fracture mechanics analysis (WCAP-15973-P) by demonstrating that the plant-specific load and thermal events are within those assumed by the generic bounding analysis. The basis for the safety determination of the fracture mechanics evaluation calculation will therefore remain valid so long as the numbers of these events do not exceed the design basis values.

The fatigue cycle count program described in Appendix B, Fatigue Monitoring Program, will ensure a reanalysis or other appropriate corrective action if a design basis primary coolant system cycle count limit is reached at any time during the extended licensed operating period.

Disposition for All Alloy 600 Heater Sleeves, Nozzles, Safe Ends, and Flanges: 10 CFR 54.21(c)(1)(iii)

The Palisades Alloy 600 Program identifies the Alloy 600 components in the primary coolant system, ranks them according to PWSCC susceptibility, and establishes a program for inspection, repairs, and mitigation. All 250 remaining Alloy 600 heater sleeves, nozzles, safe ends, and flanges are subject to the inspection program. At all 250 locations the program requires at least an insulated VT-2 visual inspection for leakage every refueling outage. Locations which are more susceptible to PWSCC, or whose failure could result in a more-significant safety hazard, are also subject to initial or periodic bare-metal VT-2, volumetric, or penetrant inspections.

NMC will re-evaluate effects of primary water stress corrosion cracking for all Alloy 600 components for which the current analyses found acceptable crack sizes at 40 years to identify those for which the analysis would predict unacceptable crack sizes at 60 years, and to identify appropriate additional inspections for them. NMC will complete these re-evaluations before the period of extended operation.

4.7.3 ASME Code Case N-481 Relaxation of The Primary Coolant Pump Weld Category B-L-1 Inspection Interval from 10 Years to 40 Years

Summary Description

The Byron-Jackson DFSS (Diffuser Single Suction) pumps at Palisades are vertical, single stage, centrifugal, with top-mounted motors, bottom suction, and single discharge. These pumps are classified as Type E under recent ASME code editions (volute, radially-split casing, with diffuser vanes that also provide a structural function). Each casing hub and inner volute was cast separately from the outer volute shell, and the two were then welded together. The castings are Grade CF8M austenitic stainless.

The casings were designed and fabricated to the ASME Boiler and Pressure Vessel Code, Section III, 1968. The casings met all requirements of Paragraph N-415.1 for exemption from a fatigue analysis. However, additional crack growth and thermal embrittlement analyses were performed to support relaxation of the ASME Section XI 10-year Category B-L-1 volumetric inspection interval for casing welds, under ASME Code Case N-481.

Analysis

The analysis includes support for its assumption of an initial flaw depth of 8 percent of the wall thickness (8% t), instead of the 25% t specified by Code Case N-481; based on the

ASME code radiographic inspection standard used for these pumps which requires detection of a 2% t flaw.

Crack Growth: The crack growth analysis demonstrated that:

- The initial flaw would not exceed 25% t due to effects of the alternating stress for the design basis primary system cyclic events assumed for 40 year design life,
- The resulting 25% t flaw is stable for design, emergency, and faulted loads, and
- The resulting 25% t flaw would not grow to an unstable end-point size for another 5 years, assuming a uniform rate of design cyclic events equal to their assumed total number divided by 40 years.

Thermal Embrittlement: The evaluation of the reduction of fracture toughness due to thermal embrittlement assumed that weld material behaved like the CF8M castings, which is conservative, since weld materials are not as subject to these effects. The evaluation used the J-R curve methodology of NUREG/CR-4513, which yields an aged material toughness K_{Jc}.

The evaluation of aged material toughness K_{Jc} used the most-conservative infinite time-at-temperature or “saturation” J-R curve for each material. The results of this evaluation of thermal embrittlement effects therefore do not depend on design life, and this aspect of the analysis is therefore not a TLAA.

End-Point Crack Size Determination: A crack will become unacceptable if any of the following would occur under design, emergency, or faulted loads:

- The crack is unstable against non-ductile propagation (“brittle fracture”).
- The crack is unstable against ductile tearing.
- The remaining ligament cannot carry its design loads, based on its flow stress.

Stability Against Brittle Fracture: Stability against brittle fracture is, in part, determined by the material fracture toughness, and, therefore, depends, in part, on any reduction of fracture toughness due to thermal embrittlement. Stability is indicated if the applied stress intensity factor K_I (ASME Section XI, Equation A-3300 (1)), for all load conditions at all locations of concern, remains less than the aged material toughness K_{Jc}.

At Palisades the maximum end-point values (at the volute crotch) is K_I = 130, minimum K_{Jc} = 152.3, and median K_{Jc} = 223.5 (all in ksi-in$^{1/2}$). Therefore substantial margins would remain at the end of a 40 year life, even assuming K_I is the maximum that would result from a complete design basis complement of load cycles. Of these two parameters, K_I depends on the loading and the flaw sizes at the end of evaluation period, and the aged material toughness K_{Jc} in this analysis is the saturation J-R curve for each material. Thus the expected additional thermal aging effect will not diminish the K_{Jc} assumed by the analysis, the analysis will remain acceptable so long as the design basis complement of load cycles is not exceeded, which will ensure that the worst-location K_I remains below 130 ksi-in$^{1/2}$.

Stability Against Ductile Tearing: Stability against ductile tearing is ensured if the applied crack extension J never reaches the J-integral required to initiate ductile tearing. This condition is also satisfied so long as K_I does not exceed K_{Jc}, as above.

Flow Stress Ligament Capacity Limit: The flow stress ligament capacity limit is defined as the crack size for which a ligament flow stress evaluation no longer demonstrates that the ligament will support the required loads. At Palisades the flow stress ligament capacity limit is the most time limiting of these three criteria. The critical crack size based on this criterion would be exceeded in a little over 45 years, assuming the 40 year design basis cycle count limit is reached at 40 years for all load events, and that the load cycles continue to accumulate at a constant rate.

Since the margin to the K_{Jc} vs. K_I brittle fracture stability criterion will not change, the cycle-dependent ligament flow stress criterion will remain limiting for the extended licensed operating period.

Disposition: 10 CFR 54.21(c)(1)(i) and 10 CFR 54.21(c)(1)(iii)

The projected number of each of the design basis events of the primary coolant pumps will approach its design basis limit during the extended licensed operating period. Therefore the maximum end-of- 60-year life crack size is not expected to approach the critical crack size during the 60-year extended licensed operating period.

The fatigue cycle count program described in Section 4.3 and Appendix B, Fatigue Monitoring Program, will ensure a reanalysis or other appropriate corrective action if a design basis cycle count limit is reached at any time during the extended licensed operating period.

4.7.4 Risk-Informed Inservice Inspection Program Calculations

The scope and inspection intervals of the Palisades Risk-Informed Inservice Inspection (RI-ISI) Program depended on probabilities of failure of each pipe segment calculated at the time of program implementation (approximately year 29) and the time of initial license expiration (year 40).

Analysis

The scope of the RI-ISI program may increase somewhat when the extended operating period is considered.

Disposition: 10 CFR 54.21(c)(1)(ii)

The supporting calculations for the Palisades RI-ISI program will be reviewed, and updated as needed, to reflect a 60-year operating period; and the program inspection scope will be updated accordingly, before the period of extended operation.

Therefore, these items are dispositioned under 10 CFR 54.21(c)(1)(ii), the analyses will be projected to the end of the period of extended operation.

4.7.5 Absence of a TLAA for Reactor Coolant Pump Flywheel Fatigue or Crack Growth Analysis

The four original Palisades reactor coolant pump motors were built by Allis-Chalmers. Westinghouse built an additional motor. One of the five is maintained as a spare, and any combination of four may be installed. All have flywheels at the top of the motor to provide additional rotational inertia for gradual coastdown and continued circulation, in case of a power supply loss or inadvertent trip.

A reactor coolant pump flywheel could theoretically burst because of centrifugal stresses, which could produce missiles inside containment and could also damage pump seals or other pressure boundary components. This concern is the subject of Regulatory Guide 1.14. The flywheels may therefore be subject to crack growth or fatigue analyses, inservice inspection, or both, to assess and reduce the probability of a failure.

Early Technical specifications required periodic, relatively frequent, inspections of reactor coolant pump motor flywheels. To justify a longer inspection frequency, the Combustion Engineering Owners Group prepared report SIR-94-080 (Reference 3). This report used a crack growth analysis of Palisades' reactor coolant pump flywheels to establish acceptable limits for the flywheel inspection interval. The analysis does not depend on the licensed operating period (10 CFR 54.3 Criterion 3) and is not a TLAA. The evaluation determined that the primary coolant pump would be subject to approximately 500 startup/shutdown cycles, and the crack growth fatigue analysis assumed 4000 cycles. It was concluded that a ten year inspection interval was acceptable since an assumed preservice flaw would not grow to a critical flaw size during the period between inspections (Reference 4). (In fact, the report justified that the assumed preservice flaw would not grow to critical flaw size during the entire licensed operating period, or extended licensed operating period) The NRC approved the change to the inservice inspection requirements to extend the flywheel examination frequency to once each ten years ((Reference 5), (Reference 6), and (Reference 7)).

Section 4.7 References

1. RG 1.190. US NRC Regulatory Guide 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” March 2001.

2. WCAP 15353, Westinghouse Report WCAP-15353, Revision 0, “Palisades Reactor Pressure Vessel Neutron Fluence Evaluation,” January 2000.

3. Combustion Engineering Owners Group (CEOG) Report SIR 94-080. “Relaxation of Reactor Coolant Pump Flywheel Inspection Requirements.”

4. Letter Bordine (Consumers) to USNRC, “Docket No. 50-255 - License DPR 20 - Palisades Plant - Technical Specifications Change Request - Primary Coolant Pump Flywheel Inspections.” 1 October 1997

5. US NRC Letter, R. G. Schaff, Project Manager, Division of Reactor Projects, Office of Nuclear Reactor Regulation; to N. L. Haskell, Director of Licensing, Palisades Plant. “Palisades Plant: Issuance of Amendment Re: Primary Coolant Pump Flywheel Inspection Technical Specifications (TAC No. M94567).” 15 May 1998.

6. “Consumers Energy Company, Docket No. 50-255, Palisades Plant, Amendment to Facility Operating License. Amendment No. 182, License No. DPR 20." Washington: US NRC, 15 May 1998.

7. “Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 182 to Facility Operating License No. DPR 20, Consumers Energy Company, Palisades Plant, Docket No. 50-255.” Washington: US NRC, 15 May 1998.

APPENDIX A

FSAR SUPPLEMENT

APPENDIX A
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A1.0 Appendix A Introduction

10 CFR 54.21(d) requires an application for a renewed operating license to include an FSAR Supplement. This supplement must contain summary descriptions of the programs and activities for managing the effects of aging, and of the evaluation of Time-Limited Aging Analyses (TLAAs) for the period of extended operation. This appendix provides that supplement for the Palisades Plant FSAR.

Section A2.0 of this appendix contains summary descriptions of the programs used to manage the effects of aging during the period of extended operation. Section A3.0 contains descriptions of programs used for management of TLAAs during the period of extended operation. Section A4.0 contains summaries of TLAAs applicable to the period of extended operation.

In a periodic FSAR update following NRC issuance of the renewed operating license, in accordance with 10 CFR 50.71(e), the summary descriptions of Aging Management Programs and Time Limited Aging Analyses, provided in Appendix A, will be incorporated into appropriate sections of the FSAR.

A2.0 Summary Descriptions of Aging Management Programs

This section provides summaries of programs credited in the License Renewal Application for managing the effects of aging during the period of extended operation.

The activities implemented to manage aging at the Palisades Plant may be performed under discrete programs as defined herein, or they may be incorporated into other plant programs. The program summaries provided should be interpreted as summaries of activities to be performed to manage aging, and not as specific commitments to maintain unique programs with the specific titles and content listed.

It should also be noted that these summaries do not specifically invoke or reference the Generic Aging Lessons Learned Report, NUREG-1801. The activities credited for managing aging at Palisades were developed, to a large extent, to be responsive to the revision of GALL that existed at the time that the license renewal application was developed. It is expected that changes will be made to these programs in the future as a result of advances in the state of knowledge in the industry, plant modifications, and operating experience. However, no commitment is made to update any aging management program in response to changes in the GALL. Future changes that may occur to aging management programs or activities will be managed under 10 CFR 50.59 and/or other regulatory and administrative requirements appropriate to the changes being made.

A2.1 Alloy 600 Inspection Program

The Alloy 600 Program manages aging due to PWSCC of the Primary Coolant System (PCS) pressure boundary Alloy 600 components, including Inconel 82/182 weld joints, reactor vessel head penetrations, etc. The program includes (a) PWSCC susceptibility assessment using industry models to identify susceptible components, (b) monitoring and control of primary coolant chemistry to mitigate PWSCC, (c) in-service inspections (ISI) of pressurizer penetrations, reactor vessel head penetrations and Alloy 82/182 PCS pressure boundary welds in accordance with ASME Section XI, Subsection IWB, Table IWB-2500-1, and (d) augmented inspections or preemptive repair/replacement of susceptible components or welds.

A2.2 ASME Section XI, Subsections IWB, IWC, IWD, IWF Inservice Inspection Program

ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program facilitates inspections to identify and correct degradation in Class 1, 2, and 3 piping, components, their supports and integral attachments. The program includes periodic visual, surface and/or volumetric examinations and leakage tests of all Class 1, 2 and 3 pressure-retaining components, their supports and integral attachments, including welds, pump casings, valve bodies, pressure-retaining bolting, piping/component supports, and reactor head closure studs. These are identified in ASME Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," or commitments requiring augmented inservice inspections, and are within the scope of license renewal. This program is in accordance with 10 CFR 50.55a.

A2.3 Bolting Integrity Program

The Bolting Integrity Program manages the aging effects associated with bolting through the performance of periodic inspections. The program also includes repair/replacement controls for ASME Section XI related bolting and generic guidance regarding material selection, thread lubrication and assembly of bolted joints. The program considers the guidelines delineated in NUREG-1339 for a bolting integrity program, EPRI NP-5769 (with the exceptions noted in NUREG-1339) for safety related bolting, and EPRI TR-104213 for non-safety related bolting.

A2.4 Boric Acid Corrosion Program

The Palisades Boric Acid Corrosion Program monitors component degradation due to boric acid leakage through the performance of periodic inspections. It implements the recommendations of NRC Generic Letter 88-05. The program requires periodic visual inspection of all systems within the scope of license renewal that contain borated water for evidence of leakage, accumulations of dried boric acid, or boric acid wastage. The program also provides for visual inspections and early discovery of borated water leaks such that structures, electrical and mechanical components that may be contacted by leaking borated water will not be adversely affected such that their intended functions are impaired.

A2.5 Buried Services Corrosion Monitoring Program

The Buried Services Corrosion Monitoring Program manages aging effects on the external surfaces of carbon steel, low-alloy steel, and stainless steel components that are buried in soil or sand. This program includes (a) visual inspections of external surfaces of buried components for evidence of coating damage and substrate degradation to manage the effects of aging, (b) visual inspection of the external surfaces of buried stainless steel components for evidence of crevice corrosion, pitting, and MIC. The periodicity of these inspections for carbon, low-alloy, and stainless steel will be based on opportunities for inspection such as scheduled maintenance work.

A2.6 Closed Cycle Cooling Water Program

The Closed Cycle Cooling Water Program manages aging effects in closed cycle cooling water systems that are not subject to significant sources of contamination, in which water chemistry is controlled and heat is not directly rejected to the ultimate heat sink. The program includes (a) maintenance of system corrosion inhibitor concentrations to minimize degradation, and (b) periodic or one-time testing and inspections to assess component aging. This program is based on the guidelines in EPRI TR-107396.

The program scope includes activities to manage aging in the Component Cooling Water (CCS) System, Emergency Diesel Generator (EDG) Jacket Cooling Water (Emergency Power System), and Shield Cooling System (SCS).

A2.7 Containment Inservice Inspection Program

The Containment Inservice Inspection (ISI) Program is designed to ensure that containment shell concrete, the post-tensioning system and steel pressure retaining elements continue to provide an acceptable level of structural integrity. In addition, it is designed to ensure that the liner (with associated moisture barriers), other leakage limiting steel barriers and pressure retaining bolted connections have not degraded.

A2.8 Containment Leakage Testing Program

The Containment Leakage Testing Program ensures that containment leakage is maintained below the upper acceptance limit of La = 0.1% / day. This testing program, in conjunction with the Containment Inservice Inspection Program, provides assurance that age related (and other) deterioration of the containment leakage limiting boundary is appropriately managed to ensure that postulated post-accident releases are limited to an acceptable level. The program is implemented through the following testing and examination activities:

- Overall containment leakage (integrated leakage rate or Type A) test to assess the leak tight integrity of the entire pressure boundary.
- Visual examinations of the containment exterior and interior.
- Local (Type B & C) tests to assess the leak tight integrity of individual penetrations.

A2.9 Diesel Fuel Monitoring and Storage Program

The Diesel Fuel Monitoring and Storage Program assures the continued availability and quality of fuel oil to be used in diesel generators and diesel fire pumps. The program includes (a) monitoring and trending of fuel oil chemistry to maintain fuel oil quality and mitigate corrosion, (b) periodic draining, cleaning, and internal inspection of fuel oil storage tanks, and (c) verification of program effectiveness by a one-time measurement of fuel oil storage tank bottom thickness confirming the absence of an aging effect. Fuel oil quality is maintained by monitoring and controlling fuel oil contamination in accordance with the guidelines of the American Society for Testing Materials (ASTM) Standards D 1796, D 2276, D 2709, and D 4057.

A2.10 Fire Protection Program

The Fire Protection Program includes (a) fire barrier inspections, (b) electric and diesel-driven fire pump tests, and (c) periodic maintenance, testing, and inspection of water-based fire protection systems. Periodic visual inspections of fire barrier penetration seals, fire dampers, fire barrier walls, ceilings and floors, and periodic visual inspections and functional tests of fire-rated doors are performed to ensure that functionality and operability is maintained. Periodic testing of the fire pumps ensures that an adequate flow of firewater is supplied and that there is no degradation of diesel fuel supply lines. Periodic maintenance, testing and inspection activities of water-based fire protection systems provides reasonable assurance that fire water systems are capable of performing their intended function. Inspection and testing include periodic hydrant inspections, fire main flushing, sprinkler inspections, pipe wall thickness testing and flow tests.

A2.11 Flow Accelerated Corrosion Program

The Flow Accelerated Corrosion Program manages aging effects due to flow-accelerated corrosion (FAC) on the internal surfaces of carbon or low alloy steel piping, elbows, reducers, expanders, and valve bodies which contain high energy fluids (both single phase and two phase). The program implements the EPRI guidelines in NSAC-202L-R2 for an effective FAC program and includes (a) an analysis using a predictive code such as CHECWORKS™ to determine critical locations, (b) baseline inspections to determine the extent of thinning at these locations, (c) follow-up inspections to confirm the predictions, and (d) repairing or replacing components, as necessary.

A2.12 Non-EQ Electrical Commodities Condition Monitoring Program

The Non-EQ Electrical Commodities Condition Monitoring Program manages aging in selected non-EQ commodity groups within the scope of 10 CFR 54. Features of the program periodic inspection of insulated cables and connectors, testing of sensitive instrumentation circuits, testing of medium voltage cables, and inspection of manholes for the presence of water.

A2.13 One-Time Inspection Program

The One-Time Inspection Program addresses potentially long incubation periods for certain aging effects, including various corrosion mechanisms, cracking, and selective leaching, and provides a means of verifying that an aging effect is either not occurring or progressing so slowly as to have negligible effect on the intended function of the structure or component. Hence, the One-Time Inspection Program provides measures for verifying an aging management program is not needed, verifying the effectiveness of an existing program, or determining that degradation is occurring which will require evaluation and corrective action.

The program includes (a) determination of appropriate inspection sample size, (b) identification of inspection locations, (c) selection of examination technique, with acceptance criteria, and (d) evaluation of results to determine the need for additional inspections or other corrective actions. The inspection sample includes locations where the most severe aging effect(s) would be expected to occur. Inspection methods may include visual (or remote visual), surface or volumetric examinations, or other established NDE techniques.

This program is used for a variety of purposes, including the following:

- To verify the effectiveness of water chemistry control for managing the effects of aging in stagnant or low-flow portions of piping or components, exposed to a treated water environment.
- To manage the aging effects of loss of material due to aging mechanisms such as general, crevice, pitting, and galvanic corrosion; selective leaching; and MIC.
- To verify that cracking due to stress corrosion cracking or cyclic loading, in small bore (< 4" NPS) ASME class 1 piping, is not occurring.
- To verify, for components in the Compressed Air System, that there are no aging effects requiring management in the dry air environment.

A2.14 Open Cycle Cooling Water Program

The Open Cycle Cooling Water Program manages aging effects such as loss of material due to general, pitting, and crevice corrosion, erosion, MIC, and loss of heat transfer due to biological/corrosion product fouling (e.g., sedimentation, silting) caused by exposure of internal surfaces of metallic components to raw, untreated (e.g., service) water. The program scope includes activities to manage aging in the Service Water System (SWS) and Circulating Water system (CWS). The aging effects are managed through (a) monitoring and control of biofouling, (b) flow balancing and flushing, (c) heat exchanger testing (d) routine inspection and maintenance program activities to ensure that aging effects do not impair component intended function. Inspection methods include visual (VT), ultrasonic (UT), radiographic (RT), and eddy current (ECT). This program is responsive to NRC GL 89-13.

A2.15 Overhead Load Handling Systems Inspection Program

The Overhead Load Handling Systems Inspection Program provides for inspections of the structural components and rails of cranes and fuel handling machines associated with heavy load handling that are subject to the requirements of NUREG-0612 and are within the scope of license renewal requiring aging management. For Palisades these are the Containment Building Polar Crane, the Spent Fuel Pool Overhead Crane, the Containment Building jib and boom cranes, and the reactor and spent fuel pool fuel handling machines. These cranes comply with the Maintenance Rule requirements provided in 10 CFR 50.65. The Overhead Load Handling Systems Inspections Program is primarily focused on structural components that make up the bridge and trolley of the overhead cranes that are within the scope of NUREG-0612.

A2.16 Reactor Vessel Integrity Surveillance Program

The Reactor Vessel Integrity Surveillance Program manages the aging effect reduction of fracture toughness due to neutron embrittlement of the low alloy steel reactor vessel. Monitoring methods will be in accordance with 10 CFR 50, Appendix H. This program includes (a) capsule insertion, withdrawal and materials testing/evaluation, (including upper shelf energy and RT_{NDT} determinations), (b) fluence and uncertainty calculations, (c) monitoring of Effective Full Power Years (EFPY), (d) development of pressure temperature limitations, and (e) determination of low temperature overpressure protection (LTOP) set points. The program ensures the reactor vessel materials (a) meet the fracture toughness requirements of 10 CFR 50, Appendix G, and (b) have adequate margins against brittle fracture caused by Pressurized Thermal Shock (PTS) in accordance with 10 CFR 50.61.

A2.17 Reactor Vessel Internals Inspection Program

The Reactor Vessel Internals Inspection Program manages the aging effects for reactor vessel internals. The program provides for (a) Inservice Inspection (ISI) in accordance with ASME Section XI requirements, including examinations performed during the 10-year ISI examination; (b) Participation in industry initiatives to evaluate the significance of void swelling; (c) Monitoring and control of reactor coolant water chemistry in accordance with the EPRI guidelines in TR-105714 (See Water Chemistry Program) to mitigate SCC or IASCC; and (d) Participation in industry initiatives that will generate additional data on aging mechanisms relevant to RVI and develop appropriate inspection techniques to permit detection and characterization of features of interest.

A2.18 Steam Generator Tube Integrity Program

The Steam Generator Tube Integrity Program manages the aging effects of steam generator tubes and tube repairs. The Program also manages the aging effects of accessible steam generator secondary side internal components and incorporates the guidance of NEI 97-06. The program manages aging effects through a balance of mitigation, inspection, evaluation, repair, and leakage monitoring measures. Component degradation is mitigated by controlling primary and secondary water chemistry. Eddy current testing is used to detect steam generator tube flaws and degradation. Visual examinations are performed to identify degradation of accessible steam generator secondary side internal components. Primary to secondary leakage is monitored during plant operation.

A2.19 Structural Monitoring Program

The Structural Monitoring Program is designed to ensure that age related (as well as other) deterioration of plant structures (including masonry walls) and components within its scope is appropriately managed to ensure that each such structure or component retains the ability to perform its intended function. The program is implemented through visual examination of these structures, components and other specified items. Damage or degradation found during visual examination may be further evaluated by measurements and testing techniques as appropriate.

This program also implements provisions of the Maintenance Rule, 10 CFR 50.65, that relate to masonry walls and water-control structures. It conforms to the guidance contained in RG 1.160 and NUMARC 93-01 as well as Nuclear Energy Institute publication NEI 96-03. This NEI document, which supplements NUMARC 93-01, contains additional guidance specific to the monitoring of structures.

A2.20 System Monitoring Program

The System Monitoring Program manages aging effects for normally accessible, external surfaces of piping, tanks, and other components and equipment within the scope of License Renewal. These aging effects are managed through visual inspection and monitoring of external surfaces for leakage and evidence of material degradation. The program relies upon periodic system walkdowns to monitor degradation of the protective paint or coating, and/or the exterior steel surface area (if no paint or coatings exist, or if the existing protective paint and coatings are degraded to a point whereby the exterior steel surface is exposed). Palisades does not take credit for any above ground coating or paint for mitigating corrosion even though the tanks may be painted or coated. However, inspections of the above ground coating or paint will provide an indication of the condition of the material underneath the coating or paint.

A2.21 Water Chemistry Program

The Water Chemistry Program manages aging effects such as loss-of-material due to general, pitting and crevice corrosion; cracking due to SCC; and steam generator tube degradation caused by denting, intergranular attack (IGA) and outer diameter stress corrosion cracking (ODSCC), by controlling the environment to which internal surfaces of systems and components are exposed. The aging effects are minimized by controlling the chemical species that cause the underlying mechanisms that result in these aging effects. The program provides assurance that an elevated level of contaminants and, where applicable, oxygen does not exist in the systems and components covered by the program, thus minimizing the occurrences of aging effects, and maintaining each component's ability to perform the intended functions. The program is based on the guidelines in EPRI TR-105714, and TR-102134.

A3.0 Summary Descriptions of Time Limited Aging Analysis Management Programs

A3.1 Electrical Equipment Qualification Program

The Electrical Equipment Qualification Program is an existing program that implements the requirements of 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," at Palisades. 10 CFR 50.49 defines the scope of components to be included, requires the preparation and maintenance of a list of in-scope components, and requires the preparation and maintenance of a qualification file that includes component performance specifications, electrical characteristics, the environmental conditions to which the components could be subjected, and the basis for qualification. 10 CFR 50.49(e)(5) contains provisions for aging that require, in part, consideration of all significant types of aging degradation that can affect component functional capability. 10 CFR 50.49(e)(5) also requires replacement or refurbishment of qualified components prior to the end of its designated life, unless additional life is established through ongoing qualification. EQ programs manage component thermal, radiation, and cyclical aging through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods.

A3.2 Fatigue Monitoring Program

The Fatigue Monitoring Program is a new program that ensures that limits on fatigue usage are not exceeded during the renewal term. The program monitors and tracks selected cyclic loading transients (cycle counting) and their effects on susceptible components. Palisades has selected this option under 10 CFR 54.21 to manage cracking due to metal fatigue of the reactor coolant pressure boundary during the extended period of operation.

The Fatigue Monitoring Program provides cycle counting activities for confirming analytically derived cumulative usage values for applicable locations. Specific locations that may be subject to cyclic loading that could cause fatigue cracking are monitored using a computer-based monitoring program provided by EPRI, called FatiguePro. If warranted, other monitoring methods in addition to cycle counting may also be employed under this program to monitor specific locations.

A4.0 Evaluation Summaries of Time-Limited Aging Analyses

As part of a License Renewal Application, 10 CFR 54.21(c) requires that an evaluation of time-limited aging analyses (TLAAs) for the period of extended operation be provided. The following TLAAs have been identified and evaluated to meet this requirement. These discussions will be inserted into the FSAR sections appropriate to the subject matter.

A4.1 Reactor Vessel Neutron Embrittlement

A4.1.1 Upper Shelf Energy

The Charpy upper shelf energy is associated with the determination of acceptable reactor vessel toughness during the license renewal period. 10 CFR Part 50 Appendix G Paragraph IV.A.1 requires that the reactor vessel beltline materials must have Charpy upper shelf energy of no less than 68 J (50 ft-lb) throughout the life of the reactor vessel, unless otherwise approved by the NRC. In the event that the 50 ft-lb requirement cannot be satisfied as stated in 10 CFR 50 Appendix G, or by alternative procedures acceptable to the NRC, the reactor vessel may continue to operate provided requirement 1 of Appendix G is satisfied. This requirement states that an analysis must conservatively demonstrate, the existence of equivalent margins of safety for continued operation.

Analysis

As shown in Table A4.1.1-1, the upper shelf energy for reactor vessel beltline materials at the end of the extended period of operation is expected to decrease to less than 50 ft-lbs based on predictions using Regulatory Guide 1.99. A low upper-shelf fracture mechanics analysis has been performed to evaluate the plate and weld material for ASME Levels A, B, C, and D Service Loadings, based on the acceptance criteria of the ASME Code, Section XI, Appendix K. Combustion Engineering Report NPSD-993 has determined that Combustion Engineering reactor vessels maintain equivalent margins of safety, if plate material and circumferential weld material maintains at least 30 ft-lb upper shelf energy, and if longitudinal weld material maintains at least 34 ft-lb upper shelf energy. This analysis has not been submitted to the NRC.

Table A4.1.1-1Estimated USE on March 24, 2031

RPV Material	Material Heat #	Cu (%)	Initial USE (ft-lbs)	¼t Neutron Fluence (10^{19} n/cm^2)	USE (ft-lbs)
2-112A/C	W5214	0.213	118	1.251	73.86

Table A4.1.1-1Estimated USE on March 24, 2031

RPV Material	Material Heat #	Cu (%)	Initial USE (ft-lbs)	¼t Neutron Fluence (10^{19} n/cm^2)	USE (ft-lbs)
3-112A/C	34B009	0.192	111	1.251	72.01
9-112	27204	0.203	84	1.800	50.83
D-3803-1	C-1279	0.24	102	1.800	63.29
D-3803-2	A-0313	0.24	87	1.800	53.98
D-3803-3	C-1279	0.24	91	1.800	56.46
D-3804-1	C-1308A	0.19	72	1.800	48.97
D-3804-2	C-1308B	0.19	76	1.800	51.69
D-3804-3	B-5294	0.12	73	1.800	55.51

Disposition: Aging Management, 10 CFR 54.21(c)(1)(iii)

10 CFR 50, Appendix G requires licensees to submit an analysis at least 3 years prior to the time that the upper-shelf energy of any of the reactor vessel material is predicted to drop below 50 ft-lb., as measured by Charpy V-notch specimen testing. NMC will comply with this requirement.

NMC will submit an equivalent margins analysis, completed in accordance with 10 CFR 50 Appendix G Section IV.A.1, for NRC approval, at least three years before any reactor vessel beltline material upper shelf energy decreases to less than 50 ft-lb.

This issue will be dispositioned using the method of 10 CFR 54.21(c)(1)(iii), the effects of aging on the intended function will be adequately managed for the period of extended operation

A4.1.2 Pressurized Thermal Shock

The pressurized thermal shock (PTS) rule, 10 CFR 50.61, established screening criteria that are a measure of a limiting level of reactor vessel material embrittlement beyond which operation cannot continue without further plant-specific evaluation. The screening criteria are given in terms of reference temperature RT_{PTS}. The screening criteria are 270° F for plates and axial welds, and 300° F for circumferential welds.

Analysis

The results of the PTS analysis for the limiting material have been reviewed for compliance with 10 CFR 50.61. The methodology used in the PTS analysis is based on the projected neutron fluence at the end of the period of extended operation. As shown in Table A4.1.2-1, the RT_{PTS} values for the intermediate and lower shell plates remain below the NRC screening criterion of 270ºF. The RT_{PTS} values for the axial and circumferential welds are projected to exceed the NRC screening criteria of 270ºF and 300ºF, respectively. The vessel is projected to reach the PTS screening criterion of 270º F on the beltline axial welds fabricated with weld wire heat W5214 in 2014.

Table A4.1.2-1Estimated RT_{PTS} on March 24, 2031

RPV Material	Material Heat #	Cu (%)	Ni (%)	$RT_{NDT(U)}$ (°F)	Margin (°F)	Fluence (10^{19} n/cm^2)	RT_{PTS} (°F)
Axial Welds 2-112A/C	W5214	0.213	1.01	-56	65.5	2.084	287
Axial Welds 3-112A/C	W5214	0.213	1.01	-56	65.5	2.084	287
	34B009	0.192	0.980	-56	65.5	2.084	271
Circumferential Weld 9-112	27204	0.203	1.018	-56	65.5	2.998	302
Plate D-3803-1	C-1279	0.24	0.50	-5	17	2.998	209
Plate D-3803-2	A-0313	0.24	0.52	-30	34	2.998	210
Plate D-3803-3	C-1279	0.24	0.50	-5	17	2.998	209
Plate D-3804-1	C-1308A	0.19	0.48	0	34	2.998	200
Plate D-3804-2	C-1308B	0.19	0.50	-30	34	2.998	173
Plate D-3804-3	B-5294	0.12	0.55	-25	34	2.998	115

Disposition: Aging Management, 10 CFR 54.21(c)(1)(iii)

10 CFR 50.61 requires the licensee to implement a flux reduction program that is reasonably practicable to avoid exceeding the screening criteria. If the flux reduction program does not prevent the reactor vessel from exceeding the PTS screening criterion at the end of life, 10 CFR 50.61 allows two options. The licensee can submit a safety analysis pursuant to 10 CFR 50.61(b)(4) to determine what, if any, modifications to equipment, systems and plant operation are necessary to prevent failure of the reactor vessel from a postulated PTS event. The other option is to perform a thermal-annealing treatment of the reactor vessel pursuant to 10 CFR 50.61(b)(7) to recover fracture toughness. 10 CFR 50.61 requires the details of the selected alternative be provided to the NRC three years prior to when the reactor vessel is projected to exceed the PTS screening criteria. At the appropriate time, prior to exceeding the PTS screening criteria, Palisades will select the

optimum alternative to manage PTS in accordance with NRC regulations and make relevant submittals to obtain NRC review and approval.

This issue will be dispositioned using the method of 10 CFR 54.21(c)(1)(iii), the effects of aging on the intended function will be adequately managed for the period of extended operation.

A4.1.3 Pressure-Temperature (P-T) Limits

10 CFR Part 50 Appendix G requires that the reactor pressure vessel be maintained within established pressure-temperature limits including during heatup and cooldown. These limits specify the maximum allowable pressure as a function of reactor coolant temperature, and are contained in Technical Specifications. As the reactor pressure vessel becomes embrittled and its fracture toughness is reduced, the allowable pressure (given the required minimum temperature) is reduced.

Low temperature overpressure protection limits and setpoints are determined as part of the calculation of pressure/temperature operating limit curves.

Analysis

The current pressure/temperature analyses are valid beyond the current operating license period, but not to the end of the period of extended operation.

Disposition

This issue will be dispositioned using the method of 10 CFR 54.21(c)(1)(iii), the effects of aging on the intended function will be adequately managed for the period of extended operation. The Pressure-Temperature operating limits contained in Technical Specifications will be updated as required by either Appendices G or H of 10 CFR 50, or as operational needs dictate.

A4.1.4 Low Temperature Overpressure Protection (LTOP) PORV Setpoints

Low temperature overpressure protection limits and setpoints are determined as part of the calculation of pressure/temperature operating limit curves. See Section A4.1.3.

A4.2 Metal Fatigue

A4.2.1 Reactor Vessel Fatigue Analyses

The Palisades Reactor Vessel was designed, constructed, and analyzed to the ASME Boiler and Pressure Vessel Code, Section III, Subsection 4 for Class A vessels, 1965, with addenda through Winter, 1965. The original analyses have been corrected and amended to address issues that have arisen since fabrication. The current design basis highest calculated fatigue usage factors, based on the number of design basis load cycles assumed by the vessel analyses, have been determined. The number of design basis load cycles for each event was selected to be adequate for the originally-licensed 40-year design life.

Analysis

This section addresses the calculated fatigue usage factors for the reactor vessel

- Shell and bottom head
- Inlet and outlet nozzles
- Internal welded attachments
- Instrument nozzle shroud tube
- Vessel head CRDM nozzles, and
- Instrument flange bolts (on the instrument nozzle on the vessel head).

The worst-case calculated usage factor for the set of design basis load events in these locations is 0.4516 on the outlet nozzle, well within the code limit of 1.0.

The number of each of the design basis events that affect the reactor pressure vessel is not expected to approach its design basis limit during the extended licensed operating period. Therefore, the actual fatigue usage factors are not expected to approach their calculated values during the extended licensed operating period. The calculated maximum usage factor for these locations on the reactor vessel is 0.4516, well within the analytical limit of 1.0.

Disposition

This issue is dispositioned using the method of 10 CFR 54.21(c)(1)(i), the analysis remains valid for the period of extended operation.

In addition, the Fatigue Monitoring Program will ensure that the effects of aging will be adequately managed for the period of extended operation under 10 CFR 54.21(c)(1)(iii) by assuring that a reanalysis or other appropriate corrective action is taken if a design basis cycle count limit is reached at any time during the extended licensed operating period.

A4.2.2 Reactor Vessel Head Closure Stud Fatigue Analysis

The highest fatigue usage factor calculated by the reactor vessel fatigue analysis is in the vessel head studs.

Analysis

The calculated lifetime usage factor in the head closure studs for the set of design basis load events is 0.8346, within the code limit of 1.0. The number of design basis transient events is not expected to approach the number assumed by the analysis during the extended licensed operating period.

Disposition

This issue is dispositioned using the method of 10 CFR 54.21(c)(1)(i), the analysis remains valid for the period of extended operation.

In addition, the Fatigue Monitoring Program will ensure that the effects of aging will be adequately managed for the period of extended operation under 10 CFR 54.21(c)(1)(iii) by assuring that a reanalysis or other appropriate corrective action is taken if a design basis cycle count limit is reached at any time during the extended licensed operating period.

A4.2.3 Control Rod Drive Mechanism (CRDM) Housing Fatigue Analyses

The reactor control rod drive mechanisms (CRDM) are enclosed in pressure housings, bolted and seal-welded to the reactor pressure vessel CRDM nozzle flanges. The CRDM housings, their seal housings, their instrument and vent tube nozzles, the flange bolts, and the Omega seal welds between the CRDM housing flanges and the reactor vessel CRDM nozzle flanges were all replaced in 2001. Extension of the operating license to March 24, 2031, therefore, only requires a 30 year design life. The replacements are ASME III (1989) - Class 1, NPT stamped, with reconciliation to the 1965 code.

Analysis

CRDM Housings: The revised fatigue evaluation found that the criteria of ASME III - 1989, Paragraph NB-3222.4(d) are met, and, therefore, that no fatigue analysis is required.

CRDM Seal Housings and Tool Access Tube Assemblies: The fatigue evaluation finds the criteria of N-415.2(d)(6) of ASME III are met. A revised fatigue analysis was not performed. The replacement material is stronger than the analyzed material.

CRDM Housing Bolts: The fatigue evaluation of record calculated a design lifetime cumulative usage factor of 0.173. This evaluation has not been revised, but will remain valid so long as the assumed number of lifetime design basis transient cycles remains valid. This analysis is very conservative since the bolts were replaced when the CRDM housings were replaced.

CRDM Flange - Reactor Vessel Nozzle Flange Bolts: The fatigue evaluation of record calculated a design lifetime cumulative usage factor of 0.624. This evaluation has not been revised, but will remain valid so long as the assumed number of lifetime design basis transient cycles remains valid. This analysis is very conservative since the bolts were replaced when the CRDM housings were replaced.

CRDM Flange - Reactor Vessel Nozzle Flange Omega Seal Welds: The fatigue evaluation calculated a design lifetime cumulative usage factor of 0.5621. This analysis used the set of design transients from the original design specification, which were based on an assumed 40-year design life.

Since all of these components were replaced in 2001, their expected installed lifetime, including the extended licensed operating period (to March 24, 2031), will be only about 30 years, compared to the 40 years upon which the estimate of design basis event cycles was based. The two highest calculated maximum usage factors for a 40 year life in any of these components is 0.624 for the flange bolts and 0.5621 for the flange Omega seal welds, well within the analytical limit of 1.0.

In addition, the number of each of the design basis events that affect the reactor pressure vessel and the CRDM housings and appurtenances is not expected to approach its design basis limit during the extended licensed operating period. The actual fatigue usage factors are, therefore, not expected to approach their calculated values during the extended licensed operating period.

Disposition

This issue is dispositioned using the method of 10 CFR 54.21(c)(1)(i), the analysis remains valid for the period of extended operation.

A4.2.4 Steam Generator Fatigue Analyses

The Palisades steam generators were replaced in 1990-1991. Extension of the operating license to March 24, 2031 therefore requires a 40 year design life. The replacement steam generators were designed to the ASME Boiler and Pressure Vessel Code, Section III, 1977. The primary coolant pressure boundary (tube side) of the steam generators is designed to Section III Class 1 rules. Critical components of the Class 2-design secondary side (e.g., the feedwater nozzles) were also analyzed using Class 1 methods.

Analysis

Vessel and Components, Except Manway Studs: The ASME III Class 1 fatigue analyses of the replacement steam generators used the number of event cycles assumed for the original plant design for a 40-year licensed operating period.

Except for the manway studs (below), the maximum usage factor at any location is 0.9158, on the main feedwater nozzle.

Manway Studs: The original ASME III Class 1 fatigue analyses calculated a worst-case usage factor for the manway studs of 0.10. However the vendor (Westinghouse) later issued a Nuclear Safety Advisory Letter identifying a significant bending load on the studs due to differential thermal expansion during the heatup and cooldown transients, which resulted in predicted lifetime usage factors greater than 1.0. The revised analysis found that the fatigue limit for the studs would be reached in about 200 reactor heatup and cooldown cycles.

This problem has been addressed by a requirement in plant procedures to evaluate the number of heatup and cooldown cycles every 5 operating years, and to replace the manway studs before they can experience 200 heatup and cooldown cycles. Since the studs were installed at Plant Heatup Number 106, they should be replaced before Heatup Number 306. This five-year evaluation interval was based on a very conservative assumption that no more than 12.5 startup and shutdown cycles would occur per operating year, and, therefore, less than 63 cycles should accumulate between evaluations.

However, since the fatigue life analysis that supports the current design and licensing basis safety determination for the manway studs does not depend on the licensed operating period, the analysis is not a TLAA.

The replacement steam generator fatigue analyses qualify them for a 40 year design life, except for the manway studs. The replacement steam generators were installed in 1991. The qualified 40 year design life is therefore sufficient for the extended licensed operating period ending in 2031.

Disposition

This issue is dispositioned using the method of 10 CFR 54.21(c)(1)(i), the analysis remains valid for the period of extended operation.

A4.2.5 Pressurizer Fatigue Analyses

The pressurizer was designed to the ASME Boiler and Pressure Vessel Code, Section III, Class A, 1965, Winter 1966 addenda. The code design calculation includes a fatigue analysis for those nozzles or other parts which do not meet the fatigue analysis exemption criteria of Section III Paragraph N-415.1. These include all nozzles attached by J-welds and other nozzles and parts subject to more-severe thermal transients:

- The surge, spray, and temperature element nozzles and nozzle-to-shell or nozzle-to-head junctions
- Heater sleeve-to-head junctions
- Upper level nozzles
- Relief (PORV) and safety valve nozzles
- The liquid-vapor boundary region of the shell
- Manway, head, and studs, and
- Bottom head support skirt.

A revised set of external load cycles required reevaluation of fatigue in the three safety relief valve nozzles. The recalculation found that the stress intensities produced by the revised external loads are less than those calculated by the original analysis, and, therefore, that the original simplified fatigue evaluation remains valid.

The Alloy 600 safe end to-pipe weld at the power-operated relief valve nozzle was found cracked and leaking in 1993. The Alloy 600 safe end was repaired with a short stainless steel pipe, and the nozzle was reanalyzed. The highest usage factor calculated for the modified safe end and its connections is 0.7572 at the inside of the nozzle-head juncture. This calculation assumed load cycles for a 40 year design life.

Analysis of the PORV nozzle safe end material removed in 1993 indicated primary water stress corrosion cracking (PWSCC), and prompted replacement of the remainder of the safe end with 316 stainless material with Alloy 690 welds, in 1995. The fatigue analysis for the currently-installed safe end and attachments calculated a worst-case usage factor of 0.084, at the inside wall of the safe end transition.

Other TLAAs of the Pressurizer: Thermal stratification phenomena in the surge line have required reanalysis of the surge nozzle, and concerns for high differential temperatures with auxiliary spray have required reanalysis of the spray nozzle.

Primary water stress corrosion cracking (PWSCC) of the Alloy 600 temperature nozzles required repair, a revised fatigue analysis, and analyses of the PWSCC effects. These failures, failure of the PORV nozzle safe end, and industry-wide cracking of Alloy 600 components have required evaluation of PWSCC effects in all Alloy 600 components.

Analysis and Disposition

Fatigue usage factors have been calculated for the pressurizer

- Bottom head support skirt
- Heater sleeve-head junctions
- Liquid-vapor boundary region of the shell
- Upper level nozzles
- Power-operated relief valve (PORV) nozzle
- Code safety valve nozzles, and
- Manway, head, and studs

For these locations the worst-case calculated usage factor is 0.7572, at the inner PORV nozzle-head juncture. This issue is dispositioned using the method of 10 CFR 54.21(c)(1)(i), the analysis remains valid for the period of extended operation.

The fatigue analysis of the replaced PORV nozzle safe end was based on a nominal 20 year life beyond its 1995 installation. However, the low worst-case usage factor of 0.084 in this component permits a simple projection to the end of the extended licensed operating period, when the service life of this component would be about 36 years. A projected usage factor based on this 36 year life would be only about 0.15, compared to the allowable 1.0. This issue is dispositioned using the method of 10 CFR 54.21(c)(1)(i), the analysis remains valid for the period of extended operation.

For those portions of the original pressurizer analysis which have not been superseded, the number of each of the design basis events is not expected to approach its design basis limit during the extended licensed operating period, and, therefore, the actual fatigue usage factors are not expected to approach their calculated values during the extended licensed operating period. The calculated maximum usage factor for these locations on the pressurizer is 0.7572, within the analytical limit of 1.0, at the inner PORV nozzle-head juncture. This issue is dispositioned using the method of 10 CFR 54.21(c)(1)(i), the analysis remains valid for the period of extended operation.

In addition, the Fatigue Monitoring Program will ensure that the effects of aging will be adequately managed for the period of extended operation under 10 CFR 54.21(c)(1)(iii) by assuring that a reanalysis or other appropriate corrective action is taken if a design basis cycle count limit is reached at any time during the extended licensed operating period.

A4.2.6 Regenerative Heat Exchanger Fatigue Analyses

The regenerative heat exchanger recovers energy from the letdown line to heat chemical and volume control system charging water for reactor system makeup (charging) and auxiliary pressurizer spray. The charging system has three positive-displacement pumps. The pumps cannot be throttled without lifting discharge safety valves, but one of them is variable speed. The charging system was designed to operate continuously, with flow from the fluid-drive variable-speed pump controlled by a primary coolant volume control signal. If the variable-speed pump is out of service, the system controls Primary Coolant System makeup by cycling a constant-speed pump on and off. This cycling of cold makeup water against the (approximately constant) hot letdown flow produces significant thermal transients in the regenerative heat exchanger.

Isolation of letdown flow introduces a similar differential thermal load and has a similar effect.

Analysis

The original design included a fatigue evaluation to ASME III -1965, Paragraph N-415. The final addendum to the original analysis was based on a revised definition of the load transient set which eliminated the 15 percent per minute load-following transient for this component. The worst-location cumulative usage factors (CUF) for a 40-year licensed operating life were 0.871 on the tubesheet, and 0.624 on the shell.

By 1993, operating experience had shown that the availability of the variable-speed pump was less than anticipated in the original design, and therefore that the number of thermal cycles from cycling a constant-speed pump was greater than anticipated. The expected number of lifetime thermal cycles and the fatigue usage factor at the most limiting location (the tube sheet) were therefore reevaluated. The transient evaluation increased the lifetime number of thermal stress cycles due to this event from 5,520 to 17,822. The fatigue evaluation found that, with the increased number of thermal cycles described above plus the remainder of the design basis event set, the CUF at the limiting tube sheet location would be 1.002, slightly above the analytic limit of 1.0. However, the design basis event set included 60,500 load-following and reactor trip events (Transient I), of which the unit had experienced only about 180 by that date (1993). The fatigue evaluation therefore reduced the design basis Transient I cycles slightly (to 60,282), and demonstrated that the analytic limit of 1.0 was then met. The 1993 reanalysis retained the simplifying assumption from the original analysis, that the thermal effect of cycling a constant-speed charging pump produced the maximum stress range for all load pairings.

A revision in 1995 to permit more frequent letdown isolation re-evaluated both thermal and pressure transient effects. The evaluation of stress pairings for Transient I load-following and reactor trip events found that those with the maximum stress range could be further reduced to 32,500 from the 60,282 assumed in the 1993 analysis, even considering the increased frequency of letdown isolations. This 1995 revision again evaluated only the limiting inner ligament of the shell side of the tubesheet, for which the calculated lifetime CUF is 0.880.

Based on the plant events in the first 20 years of operation, a recent revision to the calculation estimates that the number of cycles for cycling the constant-speed charging pump can be increased to 27,062, and Transient I can be reduced to 6,240 for 60-year plant life. With the new estimated numbers of transient events at the end of extended operating period, the maximum fatigue usage factors at the two most critical locations in the regenerative heat exchanger (tubesheet and tubesheet to shell junction) is 0.439.

Disposition: Revision, 10 CFR 54.21(c)(1)(ii); and Aging Management, 10 CFR 54.21(c)(1)(iii)

Re-analysis has been performed to include additional thermal cycles from cycling the constant-speed charging pump and reduced number of transients that were over estimated in the original design analysis. The projected CUF at the end of the extended operating period remains less than 1. Therefore, the regenerative heat exchanger meets the criteria of 10 CFR 54.21(c)(1)(ii).

The fatigue management cycle count program will include the letdown isolation and variable-speed charging pump out-of-service events.

A4.2.7 ASME III Class A Primary Coolant Piping Fatigue Analyses

A piping fatigue analysis was originally applied only to the main loops of the primary coolant system, the two 42 inch hot legs and the four 30 inch cold legs, and to the connecting nozzles for smaller piping. The original analyses calculated fatigue usage factors for the:

- Hot legs
- Cold legs
- Safety injection-shutdown cooling nozzles
- Hot leg to surge line nozzle
- Charging Inlet nozzles
- Hot leg temperature nozzles
- Shutdown cooling outlet nozzle
- Cold leg temperature nozzles

The hot leg to surge line nozzle has been reanalyzed to address transients not contemplated in the original analysis.

The design stress ranges of the cold-leg-to-pressurizer-spray nozzles were below the endurance limit in the original design basis analysis. However the cold-leg-to-pressurizer-spray nozzles have since been evaluated for additional transients not contemplated in the original analysis. The fatigue issue of the cold-leg-to-pressurizer-spray nozzles is addressed in Section A4.2.8.

The following section addresses the remaining original analyses for the hot and cold legs, and for the remaining original nozzles.

Analysis

The fatigue analysis of record for the hot and cold legs uses the bounding stresses at all locations in each of hot leg and cold leg to determine the worst possible stress ranges. These usage factors are therefore considerably higher than would be calculated by a location-specific analysis. The maximum CUF is 0.07551 for the hot leg, and 0.7531 for the cold leg.

The cold leg charging nozzle was reanalyzed to account for the additional cycles due to the constant-speed charging pump recycling. CUF at the end of the extended operation period is 0.526.

The cold leg safety injection-shutdown cooling and charging inlet nozzles are also the sample locations, which were selected for evaluation of effects of the reactor coolant environment on fatigue behavior.

Disposition: Validation, 10 CFR 54.21(c)(1)(i), and Aging Management, 10 CFR 54.21(c)(1)(iii)

The number of each of the design basis events that affect the hot and cold legs is not expected to approach its design basis limit during the extended licensed operating period, and therefore that the actual fatigue usage factors are not expected to approach their calculated values during the extended licensed operating period. The calculated maximum usage factor for the hot and cold legs and their nozzles in these original calculations is 0.7531, well within the analytical limit of 1.0.

The predicted hot leg and cold leg usage factors are calculated on a very conservative basis. The hot leg usage factor is quite low, and the cold leg usage factor, though appearing significant at 0.7531, would be much less if calculated by a location-specific analysis.

The Fatigue Monitoring Program will ensure a reanalysis or other appropriate corrective action in the unlikely event that a design basis cycle count limit is reached at any time during the extended licensed operating period.

A4.2.8 Revised Bulletin 88-11 Fatigue Analysis of the Hot-Leg-to-Pressurizer-Surge-Line Nozzle, Surge Line, and Pressurizer Surge Nozzle

NRC Bulletin 88-11, dated December 1988, was issued to address pressurizer surge line temperature stratification concerns. The effects of thermal stratification were evaluated by the Combustion Engineering Owners Group. The Combustion Engineering Owners Group Report concluded the structural integrity of the pressurizer surge line is acceptable for the forty year life of the Plant. The NRC issued an SER on September 13, 1993 concluding that the CEOG analysis adequately demonstrates that the bounding surge line and nozzles meet ASME Code stress and fatigue requirements for the 40-year design. CPCo provided additional information detailing completion of the required actions of Bulletin 88-11, including the requirement to update the pressurizer surge line stress and fatigue analyses. See FSAR Section 4.3.7.

Analysis

For both surge line nozzles and the surge line elbow, the calculated usage factors for the revised set of design basis load events are within the code limit of 1.0.

Pressurizer Surge Line Elbow: The fatigue evaluation for a typical Combustion Engineering plant with intermittent pressurizer spray and for the revised set of design basis load events, including the IEB 88-11 thermal stratification transients, calculated a maximum CUF of 0.937 at one of the surge line elbows.

A recent fatigue analysis using thermal stratification conditions under the Palisade continuous pressurizer spray operation show that the CUF is reduced significantly to 0.0135 for the expected number of cycles at the end of the 60-year operating period. If the number of cycles at the end of the extended operating period were based on 1.5 times the 40-year design basis cycles, the CUF at the surge elbow would be 0.0447.

This location is also a NUREG/CR 6260 sample location for evaluation of environmental effects of the reactor coolant on fatigue effects.

Hot Leg to Surge Line Nozzle: The fatigue evaluation for a typical Combustion Engineering plant with intermittent pressurizer spray and for the revised set of design basis load events, including the IEB 88-11 thermal stratification transients calculated a maximum CUF of 0.3818 for Palisades hot leg to surge line nozzle. With Palisades continuous pressurizer spray operation, the CUF is reduced significantly, similar to the above surge line elbow, because piping loads due to thermal stratification are the major contributor to nozzle fatigue stress.

Pressurizer Surge Nozzle: The fatigue evaluation for a typical Combustion Engineering plant with intermittent pressurizer spray and for the revised set of design basis load events, including the IEB 88-11 thermal stratification transients calculated a maximum CUF of 0.9611 for the Palisades pressurizer surge line nozzle. With Palisades continuous pressurizer spray operation, the CUF is reduced significantly, similar to the above surge line elbow, because piping loads due to thermal stratification are the major contributor to the nozzle fatigue stress.

Design Basis Thermal Transients and Expected Thermal Transients: The additional thermal stratification transients are in the pressurizer surge line and nozzles during plant heatup and cooldown at differential temperatures of 320, 250, 200, and 150º F ΔT, and hot standby at 90º F ΔT. These design transients were developed by the CE owners group for a typical plant with severe thermal transients due to intermittent pressurizer spray.

The use of modulated, continuous spray for pressure control, and control of pressurizer to primary loop ΔT, significantly mitigates this problem at Palisades. An assessment of the thermal stratification event mechanisms for Palisades operating conditions found that for almost all such events the metal ΔT would not exceed 210º F, instead of the 320º F ΔT assumed by the standard plant analysis. This is further supported by the log of pressurizer spray events at ΔT above 200º F, which has recorded only 47 through 9 January 2005.

These transients are auxiliary spray events, which have little or no effect on surge line stratification.

This moderation of the transients reduces the piping differential expansion loads and support and nozzle reactions. A recent fatigue analysis using thermal stratification conditions under the Palisade continuous pressurizer spray operation show that the CUF at the surge line elbow is reduced to less than 0.1 at the end of the extended operation period. Since piping load is the major contributor to the nozzle fatigue stress, a similar reduction is expected for the surge line nozzles.

Design Basis Cycle Count and Expected Cycle Count: The number of design cycles developed by the CE Owners Group for each of the above thermal stratification transients correspond to 500 cycles of plant heatup and cooldown. The number of transient events which might be expected to initiate these thermal stratification events will not exceed their design basis limits for the extended licensed operating period, the same is therefore also true for these thermal stratification events.

Disposition for Surge Line Elbow: Revision, 10 CFR 54.21(c)(1)(ii); and Aging Management, 10 CFR 54.21(c)(1)(iii)

A plant-unique calculation for the surge line shows that the fatigue usage factor of the surge line elbow remains less than 1 at the end of the extended operating period. The calculation includes the revised set of design basis load events; including the plant-unique IEB 88-11 thermal stratification transients. Therefore, the surge line elbow meets the revision criteria per 10 CFR 54.21(c)(1)(ii).

The Fatigue Monitoring Program will ensure that the effects of aging will be adequately managed for the period of extended operation by assuring that a reanalysis or other appropriate corrective action is taken if a design basis primary coolant system cycle count limit is reached at any time during the extended licensed operating period.

Disposition for Hot Leg and Pressurizer Surge Nozzles: Validation, 10 CFR 54.21(c)(1)(i); and Aging Management, 10 CFR 54.21(c)(1)(iii)

The design basis analysis calculated maximum usage factor at the hot leg surge nozzle of 0.3818 is well below the analytical limit of 1.0. The usage factor remains less than 1 at the end of the extended operating period. Therefore, the surge line elbow and the pressurizer surge nozzle meet the validation criteria per 10 CFR 54.21(c)(1)(i).

The calculated maximum usage factor for the pressurizer surge nozzle of 0.9611 is below the analytical limit of 1.0. This value is the result of a generic plant analysis, which assumed worst-case stratification through the entire surge line, and which calculated transients based on intermittent pressurizer spray. Similar to the CUF of the surge line elbow, the CUF of the pressurizer is expected to reduce considerably, if the moderation of the transients of the Palisades continuous pressurizer spray is used.

As mentioned above, the number of design cycles for surge line stratification flow developed by the CE Owners Group for each of the above thermal stratification transients correspond to 500 cycles of plant heatup and cooldown.

The Fatigue Monitoring Program will ensure that the effects of aging will be adequately managed for the period of extended operation by assuring that a reanalysis or other appropriate corrective action is taken if a design basis primary coolant system cycle count limit is reached at any time during the extended licensed operating period.

A4.2.9 Revised Fatigue Analysis of Nozzles from PCS Cold Legs 1B and 2A to Pressurizer Spray and of the Pressurizer Spray Nozzle

Summary Description

Pressurizer spray is normally supplied by reactor coolant pump head through 3 inch nozzles on two of the four 30 inch primary coolant system (PCS) cold legs. Normal spray flow in each of these 3 inch lines is continuous, through a normally throttled 3/4 inch main spray bypass valve, and through a 3 inch main spray control valve. The charging line downstream of the regenerative heat exchanger supplies auxiliary spray from the chemical and volume control system through a 2 inch control valve. All three of these sources supply a single pressurizer spray nozzle.

The original design of the pressurizer included a fatigue analysis of the pressurizer spray nozzle. However, the normal spray piping and the auxiliary spray piping were designed to the B31.1 Code. Revised operating conditions and pressurizer cooldown rate prompted addition of a fatigue analysis for the two cold leg nozzles and for the auxiliary spray piping.

Analysis of Other Nozzles

Cold Leg Nozzles to Pressurizer Spray: The analysis of the auxiliary spray-reverse flow events is based on the design basis number of thermal cycles assumed for 40 years, and the calculated cumulative usage factor is 0.66.

Pressurizer Spray Nozzle: The revised pressurizer spray nozzle analysis determined that the calculated cumulative usage factor (CUF) is 0.8214 for the design basis number of high-differential-temperature spray events and 200º F per hour cooldowns assumed for 40 years, and for all other applicable transients.

The revised analysis also estimated that the maximum CUF in the spray nozzle to that date (October, 1991) was 0.353, and that accumulation at then-current trends indicated a 40-year lifetime CUF of about 0.435. On that basis a projection to the end of a 60-year extended licensed operating period would indicate a CUF of about 0.517.

Disposition Cold Leg to Pressurizer Spray Nozzles: Validation, 10 CFR 54.21(c)(1)(i)

The number of thermal cycles for each of the design basis temperature differential ranges will not exceed the design basis limit during the extended licensed operating period. The calculated 40-year plant life usage factor of 0.66 is well below the analytical limit of 1.0. Thus the usage factor remains less than 1 at the end of the extended licensed operating period.

Disposition Pressurizer Spray Nozzles: Aging Management, 10 CFR 54.21(c)(1)(iii)

The design basis cumulative usage factor (CUF) of the nozzle is 0.8214 for the original 40-year licensed operating period. The projected number of cycles of the design basis events does not exceed the design basis limit during the extended licensed operating period. Therefore, the actual fatigue usage factor is not expected to approach the calculated value at the end of the extended licensed operating period.

The Fatigue Monitoring Program will ensure that the effects of aging will be adequately managed for the period of extended operation by assuring that a reanalysis or other appropriate corrective action is taken if a design basis cycle count limit is reached at any time during the extended licensed operating period.

A4.2.10 Pressurizer Auxiliary Spray Line Tee Fatigue Analysis in Response to NRC Bulletin 88-08

NRC Bulletin 88-08 and supplements describe observed effects of thermal cycling and thermal stratification in reactor coolant system pressure boundary components due to thermally-driven cyclic inleakage at isolation valves and similar phenomena. In 1989 a conservative, bounding analysis of the section of the Palisades auxiliary spray line from check valve CK 2118 to the pressurizer spray line tee demonstrated that fatigue due to these effects would be acceptable for the then-remaining 30 year licensed operating life.

The piping material is A 376 Type 316. The auxiliary spray line connects to the normal spray line vertically and below, so that the cooler auxiliary spray water does not cause a thermal stratification effect. The analysis assumed 500 lifetime operating basis earthquake (OBE) cycles and 500 lifetime full-range thermal expansion (heatup and cooldown) cycles. It modeled the Bulletin 88-08 phenomena (due to inleakage through the check valve) as a thermal cycle every two minutes, between 536 and 400 ºF, or 1.84×10^5 per year at a 70 percent availability factor.

The 500 lifetime OBE cycles are assumed to occur only once in a design lifetime, independent of its length. The analysis also conservatively attributed the entire 500 full-range thermal expansion to the remaining 30 year life. These OBE and full-range thermal cycles contributed a negligible 0.0063 usage factor. The assumed Bulletin 88-08 cycles contributed 0.4245 in 30 years, for an end-of-life cumulative usage factor (CUF) of 0.43

Analysis

License renewal will add an additional 20 years to the design life assumed by the analysis. Increasing the lifetime contribution of the OBE cycles and full-range thermal expansion cycles by 50% (a conservative assumption for OBE, and given the plant history, also for the full-range thermal cycles) results in a contribution of only 0.0095. Increasing the assumed remaining design life to 50 years results in a contribution from the Bulletin 88-08 cycles of 0.7075, for a 60-year CUF of 0.717, well within the allowable of 1.0.

The analysis assumed 70 percent plant availability. Recent experience has been about 90 percent, which would increase the contribution of the Bulletin 88-08 cycles to about 0.91, or a CUF of about 0.92.

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

The assumptions and methods of the analysis are otherwise very conservative, and this result is still within the allowable of 1.0.

A4.2.11 Assumed Thermal Cycle Count for Allowable Secondary Stress Range Reduction Factor in Piping and Components

This section addresses the issue of assumed thermal cycle counts which determine an allowable secondary stress range reduction factor for some Consumers Power Design Class 1 piping and components, and for non-Consumers Power Design Class 1 piping and components.

Only the primary coolant system piping and components have an ASME Class 1 fatigue analysis. Other Consumers Power Design Class 1 piping and components were designed to the ANSI B31.1 Power Piping Code, ASME Section VIII, or ASME III, Class 2 and 3, which requires a stress range reduction factor to the allowable stress range for secondary (expansion and displacement) stresses to account for thermal cycling. For ANSI B31.1 the allowable secondary stress range is 1.0 S_A for 7,000 equivalent full-range thermal cycles or less. The allowable secondary stress range is reduced to 0.5 S_A for thermal cycles greater than 100,000. Components designed to other codes, such as ASME VIII, have identical or very similar provisions. An increase in design life could increase the number full-range thermal cycles, therefore, design analyses under these codes are TLAAs.

Some piping within the scope of license renewal was originally designed and built to the American Standard (ASA) Code for Pressure Piping, Section 1, “Power Piping Systems,” 1955 edition. However, during the implementation of IE Bulletin 79-14 work, CP Co Design

Class 1 piping, except the main primary coolant piping, was designed to the USAS B31.1.0 (1967) Power Piping Code. In 1992, as a result of discussions between CP Co and the NRC, for new and existing CP Co Design Class 1 piping (except the main primary coolant piping), the code of record was changed to ANSI B31.1 (1973) Power Piping Code with the Summer (1973) Addenda (FSAR 5.10.1.1).

With regard to the stress range reduction factors and corresponding thermal cycle count assumptions, the Consumers Power Class 2 and 3 piping systems and components are designed to the same requirements as CP Co Design Class 1 piping.

The review of possible TLAAs found no Palisades piping and components design analyses to the B31.1 rules, which invoke lower stress range reduction factors for an increase in the equivalent full-range thermal and displacement cycles.

Analysis

The number of lifetime (full-range and equivalent) thermal and other displacement cycles applicable to most of the Palisades B31.1 piping and components are expected to be similar to the plant design basis events. Therefore, so long as the assumed number of the plant design basis event cycles is not exceeded, the secondary stress range reduction factors assumed for these B31.1 piping and components, and similar code designs remain valid.

Results of the TLAA fatigue review for B31.1 piping and similar code designs for mechanical systems within the scope of license renewal and with operating temperature in excess of 220º F for carbon steel or 270º F for austenitic stainless steel, revealed only two piping systems that have additional cycles that exceed the 7000-cycle limit of the B31.1 Code. These systems include the charging lines inboard of the regenerative heat exchanger, which experience an increase in partial-range thermal cycles due to cycling of the fixed-speed pumps. The original 6,000 events increased to 18,000 for 40-year and 27,000 for 60-year life. The effects of additional cycles have been evaluated for the Regenerative Heat Exchanger fatigue and for the Charging Inlet nozzle. The calculation will be revised to include the effects of the additional cycles on charging lines. The other system is the PCS hot leg sampling piping, which may exceed 7,000 cycles during the period of extended operation. A calculation will be performed to justify PCS sampling to occur at any reasonable frequency for 60 years of operation without exceeding the allowable number of cycles.

Disposition for the Charging Lines Inboard of the Regenerative Heat Exchanger: Revision, 10 CFR 54.21(c)(1)(ii)

NMC will evaluate the effect the increase in variable-speed-charging-pump-out-of-service events may have on these lines, and will take actions necessary to ensure these lines meet licensing basis design criteria for the extended operating period. NMC will complete this evaluation and will advise the NRC of the results, and of any necessary corrective actions, before the end of the current licensed operating period.

Disposition for Other Piping and Components: Validation, 10 CFR 54.21(c)(1)(i), and Aging Management, 10 CFR 54.21(c)(1)(iii)

The number of lifetime (full-range and equivalent) thermal and other displacement cycles applicable to most of the Palisades B31.1 piping and components are expected to be similar to the plant design basis events. The number of each of these events is not expected to exceed the existing 40-year design basis for the 60-year extended licensed operating period.

The Fatigue Monitoring Program will ensure reanalysis or other appropriate corrective action in the unlikely event that a design basis cycle count limit is reached at any time during the extended licensed operating period.

A4.2.12 Effects of Reactor Coolant System Environment on Fatigue Life of Piping and Components (Generic Safety Issue 190)

The effects of the reactor coolant environment may need to be included in the calculated fatigue life of components. GSI-190 addressed this issue. Although the parent GSI-190 safety issue has been resolved, NUREG-1800, Section 4.3.1.2, states that “The applicant's consideration of the effects of coolant environment on component fatigue life for license renewal is an area of review.” The GSI-190 review requirements are therefore imposed by the Standard Review Plan and do not depend on the individual plant licensing basis.

Analysis

NUREG/CR 6260, Table 5-43, identifies seven sample locations for older Combustion Engineering plants:

- Reactor Vessel (Lower Head to Shell Transition)
- Primary Coolant Inlet Nozzle
- Primary Coolant Outlet Nozzle
- Surge Line Elbow
- Charging Nozzles
- Safety Injection Nozzles
- Shutdown Cooling Line Inlet Transition.

Of these:

- Palisades has no “shutdown cooling line inlet transition.” The safety injection and shutdown cooling functions share a common nozzle.

- NUREG/CR-6260 evaluated a long-radius elbow in the surge line because, in the sample plant, this was the highest usage factor location in the surge line and nozzles subject to NRC Bulletin 88-11 reanalysis for thermal stratification cycles. In the NUREG/CR-6260 sample plant this component is SA-376 Type 316; at Palisades the surge line material is a similar Type 316. The only Palisades location on this line with a plant-specific fatigue analysis is the hot leg nozzle to the surge line. The analysis for the surge line elbow is for a typical C-E PWR with intermittent pressurizer spray, with thermal stratification transients; and the pressurizer surge nozzle analysis uses pipe loads from the same source. The results are therefore more conservative than would be expected for Palisades, which has a continuously-modulated pressurizer spray and less-severe thermal transients.
- The Palisades charging nozzles are SB 166 Ni-Cr-Fe Alloy 600 instead of the austenitic stainless of the NUREG/CR-6260 sample plant. However, Alloy 600 is evaluated the same as austenitic stainless for these purposes.

Of the seven NUREG/CR-6260, Section 5.2, sample locations for an older Combustion Engineering plant, six are therefore applicable to Palisades. See Table A4.2.12-1, below.

Environmental effects on cracking in the charging and other Alloy 600 nozzles are also addressed in Appendix A4.5.2, fatigue in the charging nozzles in Appendix A4.2.7, and fatigue in the surge nozzles in Appendix A4.2.8.

All of the primary coolant system at Palisades is stainless steel, Alloy 600, or carbon steel with stainless or Alloy 600 clad. Fatigue in clad components is evaluated using base material properties only; that is, as if the coolant is in contact with the base material, consistent with NUREG/CR-6260.

Table A4.2.12-1Summary of Fatigue Usage Factors at NUREG/CR-6260 Sample Locations Applicable to Palisades

Location	Material	CUF for Design Cycles	F_{en}	NUREG/ CR-5999 CUF for Design Cycles
Reactor Vessel (Lower Head to Shell Transition)	SA-302 Grade B	0.00364	2.53	0.009
Primary Coolant Inlet Nozzle	SA-302 Grade B	0.01702	2.53	0.043

Table A4.2.12-1Summary of Fatigue Usage Factors at NUREG/CR-6260 Sample Locations Applicable to Palisades

Location	Material	CUF for Design Cycles	F_{en}	NUREG/ CR-5999 CUF for Design Cycles
Primary Coolant Outlet Nozzle	SA-302 Grade B	0.115	2.53	0.29
Charging Inlet Nozzles (with Thermal Sleeves)	SB-166 Alloy 600	0.288	15.35	4.428
Surge Line Elbow	SA-376 Type 316	0.0343	15.35	0.526
Safety Injection-Shutdown Cooling Nozzles	SA-516 Grade 70	0.048	1.79	0.085

Disposition: Revision, 10 CFR 54.21(c)(1)(ii); and Aging Management, 10 CFR 54.21(c)(1)(iii)

A plant-specific calculation was performed for the six sample locations applicable to Palisades, adapted from the seven identified in NUREG/CR-6260 for older-vintage Combustion Engineering plants. Detailed environmental fatigue calculation use the appropriate F_{en} relationships from NUREG/CR-6583 for carbon and low-alloy steels and from NUREG/CR-5704 for stainless steels, as appropriate for the material at each of these seven locations.[1]

The calculation determines an appropriate $F_{en(i)}$ for each individual load pair in the governing fatigue calculation, so that an overall F_{en} multiplier on cumulative usage factor (CUF) for environmental effects can be determined for each location. The analysis shows that the fatigue usage factors at all NUREG/CR-6260 sample locations, but the charging nozzle, including the effects of the reactor coolant environment, will remain less than 1.0 for the extended operation period.

The Charging nozzle has an Alloy 600 safe end, and is one of the components in the Palisades Alloy 600 Inspection Program, which specifies inspection methods and inspection frequency. The Fatigue Monitoring Program will also ensure that the original design basis

1.Neither NUREG/CR-5704 nor NUREG/CR-5999 provide interim fatigue curves for Alloy 600 materials. The NUREG/CR-5704/5999 values for stainless steel will therefore be used, consistent with the practice of NUREG/CR-6260.

number of load cycles for each loading event is not exceeded. If this occurs, an evaluation will be made and appropriate actions will be taken to confirm the basis of the safety determination.

A4.3 Environmental Qualification of Electrical Equipment

Under 10 CFR 54.21(c)(1)(iii), a Plant EEQ Program which implements the requirements of 10 CFR 50.49 is viewed as an Aging Management Program (AMP) for license renewal. Re-analysis of an aging evaluation to extend the qualification of components under 10 CFR 50.49(f) is performed on a routine basis as part of the EEQ Program. Important attributes for the reanalysis of an aging evaluation include analytical methods, data collection and reduction methods, the underlying assumptions, the acceptance criteria, and corrective actions (if acceptance criteria are not met).

Analytical Methods: The analytical models used in the reanalysis of an aging evaluation are the same as those previously applied during the prior evaluation. The analytical method used for a radiation aging evaluation is to demonstrate qualification for the total integrated dose (that is, normal radiation dose for the projected installed life plus accident radiation dose). For license renewal, one acceptable method of establishing the 60-year normal radiation dose is to multiply the 40 year normal radiation dose by 1.5 (that is, 60 years/40 years). The result is added to the accident radiation dose to obtain the total integrated dose for the component. For cyclical aging, a similar approach may be used. Other models may be justified on a case-by-case basis.

Data Collection and Reduction Methods: Reducing excess conservatism in the component service conditions (for example, temperature, radiation, and cycles) used in the prior aging evaluation is a typical method used for a reanalysis. Plant temperature data can be obtained in several ways, including monitors used for technical specification compliance, other installed monitors, measurements made by plant operators during rounds, and temperature sensors on large motors (while the motor is not running). When used, a representative number of temperature measurements are conservatively evaluated to establish the temperatures used in an aging evaluation. Plant temperature data may be used in an aging evaluation in different ways, such as (a) directly applying the plant temperature data in the evaluation, or (b) using the plant temperature data to demonstrate conservatism when using plant design temperatures for an evaluation. Any changes to material activation energy values as part of a reanalysis are justified on a case-specific basis. Similar methods of reducing excess conservatism in the component service conditions used in prior aging evaluations may be used for radiation and cyclical aging.

Underlying Assumptions: EQ component aging evaluations contain sufficient conservatism to account for most environmental changes occurring due to plant modifications and events. When unexpected adverse conditions are identified during operational or maintenance activities that affect the normal operating environment of a qualified component, the affected EQ component is evaluated and appropriate corrective actions are taken, which may include changes to the qualification bases and conclusions.

Acceptance Criteria and Corrective Actions: The reanalysis of an aging evaluation could extend the qualification of the component. If the qualification cannot be extended by reanalysis, the component is maintained, replaced, or re-qualified prior to exceeding the period for which the current qualification remains valid.

Regulatory Issue Summary (RIS) 2003-09

On May 2, 2003, the staff issued NRC Regulatory Issue Summary (RIS) 2003-09, "Environmental Qualification of Low-Voltage Instrumentation and Control Cables", providing the results of the staff's technical assessment of GSI-168, following completion of the NRC-sponsored cable test research. The staff's technical assessment of GSI-168 in RIS-2003-09 is stated as follows:

> For license renewal, a re-analysis (based on the Arrhenius methodology) to extend the life of the cables by using the available margin based on a knowledge of the actual operating environment compared to the qualification environment, coupled with observations of the condition of the cables during walk-downs, was found to be an acceptable approach. Monitoring I&C cable condition could provide the basis for extending cable life.

The Palisades Plant Electrical Equipment Qualification Program allows re-analysis for maintaining qualification using the methods described above. In addition, the EEQ Program has procedural requirements in place to monitor and track aging effects of EQ equipment including cables. The requirements are listed below:

- Monitoring equipment condition and equipment performance
- Monitoring environmental conditions of plant areas, and
- Incorporating the results of testing and analysis into the plant maintenance and surveillance program.

Disposition

The EEQ Program will continue to be implemented for the extended operating period in accordance with 10 CFR 50.49. Continuing the existing EEQ Program provides reasonable assurance that the aging effects will be managed and that the EQ components will continue to perform their intended functions for the period of extended operation. Therefore, this TLAA is dispositioned under 10 CFR 54.21(c)(1)(iii), in that continuation of the existing EEQ

Program will adequately manage aging of affected components for the period of extended operation.

A4.4 Containment Liner Plate, Metal Containments and Penetrations Fatigue Analysis

A4.4.1 Concrete Containment Tendon Prestress Analysis

The original design included a calculation of expected loss of prestress for the plant design life In accordance with ACI 318-63. The calculation evaluated loss of prestress due to friction and initial seating loss, tendon relaxation, concrete elasticity, concrete shrinkage, and concrete creep. FSAR Section 5.8.5.3.1 lists the predicted values for remaining prestress at the end of the 40 year design life. This original analysis was conservative, as demonstrated by a regression analysis of tendon surveillance data from the twentieth and twenty-fifth-year tendon surveillances. This regression analysis indicated that the effective dome, hoop, and vertical tendon forces would remain significantly higher than values predicted by the original relaxation estimates beyond the 40-year licensed operating period.

Periodic surveillances of containment tendons for degradation are required by 10 CFR 50.55a and Palisades Technical Specification 5.5.5. See FSAR Section 5.8.8 for additional information on the existing surveillance program requirements and results.

Disposition

This issue is dispositioned under 10 CFR 54.21(c)(1)(iii), the effects of aging on the intended function(s) will be adequately managed for the extended operating period, in that the existing Palisades tendon surveillance program activities will be continued in accordance with 10 CFR 50.55a.

A4.4.2 Containment Liner Plate Load and Penetrations Load Cycles

The containment liner plate and penetrations were conservatively designed, in part, to the rules of the ASME Boiler and Pressure Vessel Code, Section III-1965. This code edition classifies containment as a Class B vessel. A fatigue analysis is required under this code edition only for Class A vessels (reactor coolant pressure boundary, etc.). However the Palisades containment liner and penetration designs use some of the methods and data from Section III, Article 4, for design of Class A vessels for fatigue loads.

The Palisades containment design relies on the liner only to maintain a leak-tight containment. There are no design conditions under which the liner plate is relied upon to assist the concrete in maintaining the integrity of the structure. Containment penetrations are designed to maintain the leak tightness of the containment structure under normal and accident conditions.

Analysis

The Palisades containment design relies on the liner to maintain a leak-tight containment. However, there are no design conditions under which the liner plate is relied upon to assist the concrete in maintaining the integrity of the structure. Forces are transmitted between the liner plate and the concrete through the anchorage system and through direct contact (pressure). At times, forces may also be transmitted by bond and/or friction. These forces cause, or are caused by, liner plate strains. The liner plate is designed to withstand the predicted strains. The effect of concrete cracking on the liner plate has also been considered.

The allowable liner plate strains/stress was conservatively based on the ASME B&PV Code, Section III, Article 4, 1965. Specifically, the following sections were adopted as guides in establishing allowable strain limits:

1. Paragraph N-412(m) - Thermal Stress, Subparagraph 2
2. Paragraph N-412(n) - Operational Cycle
3. Paragraph N-414.5, Table N-413, Figures N-414 and N-415(a) - Peak Stress Intensity
4. Paragraph N-415.1 - Vessels Not Requiring Analysis for Cyclic Operation

The liner strains/stresses due the (non-DBA) loads are relatively small such that the number of environmental and operational load cycles is insignificant compared to the allowable number of cycles on the fatigue curve in code Fig. N-415 (a), or compared to 3 times the S_m value of code of Table N-421 at the operational temperature. The results of the analysis confirm that the design of the containment liner complies with the provisions of code paragraph N-415.1 for not requiring a fatigue analysis for design load cycles.

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

Of the design basis fatigue load cycles of the containment liner, only the number of environmental and operational load cycles would increase due to the 60-year extended licensed operating period.

Of these two events, the effect of the assumed summer-winter annual cycles is negligible, and will remain negligible on increase from 40 to 60 cycles for the 60-year extended licensed operating period.

The assumed 500 containment interior operational heatup and cooldown cycles is very conservative, since it corresponds to an average of 8 1/3 cycles per year, or a PCS cooldown and heatup every 6 weeks. This is more than adequate to accommodate the 60-year extended licensed operating period.

Therefore, there will be negligible change in the fatigue resistance of the containment liner for the 60-year extended licensed operating period.

In addition, periodic inservice LLRTs required by Plant Technical Specifications monitor the continued inservice leaktight functionality of each individual penetration through the extended licensed operating period.

A4.5 Other Plant-Specific Time-Limited Aging Analyses

A4.5.1 Fuel Handling Crane Load Cycles

A crane evaluation to the Crane Manufacturers Association of America Standard CMAA-70 assumes a number of rated lifts in the design lifetime in order to establish the design Service Level, and hence the allowable stresses. At Palisades, two cranes have been reanalyzed to CMAA-70 design criteria. The NUREG-0612 heavy loads evaluation of the reactor building polar crane was performed to CMAA 70. A redesign of the Spent Fuel Pool Crane for dry fuel storage also included a NUREG-0612 evaluation to CMAA-70 design criteria. The limiting components of the containment polar crane (135 tons) and the redesigned spent fuel pool crane (110 tons) are now rated for CMAA-70 “Service Level A - Standby or Infrequent Service.” Service Level A cranes are designed to stress limits which assume either 20,000 to 100,000, or 200,000, rated lifts in a design lifetime.

Analysis

Containment Polar Crane: The polar crane was originally designed to Electric Overhead Crane Institute Specification 61. The subsequent NUREG-0612 heavy loads evaluation of the polar crane was performed to CMAA 70 (1975). The minimally-rated components are CMAA 70 Service Level A. Since the minimally-rated components are Service Level A, the effective crane design life for fatigue or allowed number of rated lifts depends on this classification, which assumes 20,000 to 100,000 rated lifts in a design lifetime.

Separate evaluations have been performed of polar crane planned engineered lifts (over the rated capacity). The evaluations were done to ANSI/ASME Standard B30.2 (1996). Lifts have been evaluated and approved up to 140 T, less than 4 percent over the 135 T rating.

Spent Fuel Pool Crane: The redesign to 110 T for dry cask storage included a NUREG-0612 evaluation to CMAA-70 Service Level A design criteria, and other considerations; and replacement of the trolley with a 110 T single-failure-proof trolley meeting NUREG 0554 guidelines. The structural redesign and evaluation considered the load combinations of ASME NOG 1, Section NOG 4140, “Load Combinations.”

Disposition: 10 CFR 54.21(c)(1)(i)

Polar Crane (L-1): Polar crane rated or near-rated lifts are limited to the reactor head plus CRDMs and insulation, and reactor internals. Only a few rated lifts are performed each refueling outage, and none during operation. Therefore this machine cannot realistically approach the 20,000 to 100,000 rated lifts, assumed for components evaluated to CMAA 70 (1975) Service Level A, during a 60 year licensed operating period.

Spent Fuel Pool Crane: Approximately 11 dry cask storage campaigns are expected between rerating and the end of the 60 year extended license. This will require loading about 64 casks. Each will require about two lifts of 100 T or more per cask, and some additional lifts of between 50 and 100 T. The total for 64 casks and 11 campaigns is about 140 lifts of 100 T or more, and about 162 lifts between 50 and 100 T. Other lifts, and lifts prior to rerating the crane, were determined to be inconsequential. Therefore, this machine can not realistically approach the 20,000 to 100,000 rated lifts assumed for its design evaluation during the 60 year extended licensed operating period.

A4.5.2 Alloy 600 Nozzle and Safe End Life Assessment Analyses

Alloy 600 (Ni-Cr-Fe alloy) was used to clad the pressurizer lower head and surge nozzle, for the pressurizer heater sleeves, and for smaller nozzles and safe ends and flanges on larger nozzles of the Palisades reactor vessel head, primary coolant system loop piping, and pressurizer. There are 250 Alloy 600 heater sleeves, nozzles, safe ends, and flanges in the Palisades primary coolant system.

Analysis

The inspection methods and intervals of the Palisades Alloy 600 aging management program were determined from evaluations of the susceptibility of all 250 remaining heater sleeves, nozzles, safe ends, and flanges to primary water stress corrosion cracking (PWSCC).

Disposition for Fatigue Analyses of the Weld Pad Repairs Installed in 1993 for the Pressurizer Temperature Element Nozzles: 10 CFR 54.21(c)(1)(iii)

NMC will monitor the cumulative number of pressurizer temperature element nozzle fatigue cycles within the Fatigue Monitoring Program, and maintain a special action level to ensure that appropriate actions are taken if at any time the cycle count for any design basis event since 1993 reaches the number assumed by these analyses.

For this purpose the fatigue management program will compare cycle counts since the repair in 1993 to appropriate action levels. Since the fatigue analyses were based on half of the 40-year pressurizer design basis event cycles, the action levels for cycles since then will be about half of the 40-year pressurizer design basis event cycles for each event.

Disposition for Corrosion Life Assessment of the TE 0101 Temperature Element Nozzle Bore in the Carbon Steel Pressurizer Wall: 10 CFR 54.21(c)(1)(i)

The evaluation estimated a repair lifetime of 52.3 years for this effect following initiation of leakage. Leakage was first detected in 1993, after 22 years of operation, which indicates that the pressurizer wall can withstand this effect for a total plant life of over 70 years. Therefore, the current analyses remain valid for the period of extended operation.

Disposition for Cycle-Dependent Aspects of the Bounding Fracture Mechanics Analysis of the Hot Leg, Piping RTD and Sampling Nozzles, Pressurizer Instrument Nozzles, and Pressurizer Heater Sleeves; and Disposition for Fatigue Portions of All Other Alloy 600 Fracture Mechanics Analyses for a 40-Year Design Life: 10 CFR 54.21(c)(1)(i) and 10 CFR 54.21(c)(1)(iii)

The Palisades plant-specific bounding fracture mechanics analysis demonstrates the validity of the cycle-dependent aspects of the generic bounding fracture mechanics analysis (WCAP-15973-P) by demonstrating that the plant-specific load and thermal events are within those assumed by the generic bounding analysis. The basis for the safety determination of the fracture mechanics evaluation calculation will therefore remain valid so long as the numbers of these events do not exceed the design basis values.

The fatigue cycle count program described in Appendix B, Fatigue Monitoring Program, will ensure a reanalysis or other appropriate corrective action if a design basis primary coolant system cycle count limit is reached at any time during the extended licensed operating period.

Disposition for All Alloy 600 Heater Sleeves, Nozzles, Safe Ends, and Flanges: 10 CFR 54.21(c)(1)(iii)

The Palisades Alloy 600 Program identifies the Alloy 600 components in the primary coolant system, ranks them according to PWSCC susceptibility, and establishes a program for inspection, repairs, and mitigation. All 250 remaining Alloy 600 heater sleeves, nozzles, safe ends, and flanges are subject to the inspection program. At all 250 locations the program requires at least an insulated VT-2 visual inspection for leakage every refueling outage. Locations which are more susceptible to PWSCC, or whose failure could result in a more-significant safety hazard, are also subject to initial or periodic bare-metal VT-2, volumetric, or penetrant inspections.

A4.5.3 ASME Code Case N-481 Relaxation of The Primary Coolant Pump Weld Category B-L-1 Inspection Interval from 10 Years to 40 Years

The Byron-Jackson DFSS (Diffuser Single Suction) pumps at Palisades are vertical, single stage, centrifugal, with top-mounted motors, bottom suction, and single discharge. The castings are Grade CF8M austenitic stainless. The casings were designed and fabricated to the ASME Boiler and Pressure Vessel Code, Section III, 1968. The casings met all requirements of Paragraph N-415.1 for exemption from a fatigue analysis. However, additional crack growth and thermal embrittlement analyses were performed to support relaxation of the ASME Section XI 10-year Category B-L-1 volumetric inspection interval for casing welds, under ASME Code Case N-481.

Analysis

The analysis includes support for its assumption of an initial flaw depth of 8 percent of the wall thickness (8% t), instead of the 25% t specified by Code Case N-481; based on the ASME code radiographic inspection standard used for these pumps which requires detection of a 2% t flaw.

Crack Growth: The crack growth analysis demonstrated that:

- The initial flaw would not exceed 25% t due to effects of the alternating stress for the design basis primary system cyclic events assumed for 40 year design life,
- The resulting 25% t flaw is stable for design, emergency, and faulted loads, and
- The resulting 25% t flaw would not grow to an unstable end-point size for another 5 years, assuming a uniform rate of design cyclic events equal to their assumed total number divided by 40 years.

Thermal Embrittlement: The evaluation of the reduction of fracture toughness due to thermal embrittlement assumed that weld material behaved like the CF8M castings, which is conservative, since weld materials are not as subject to these effects. The evaluation used the J-R curve methodology of NUREG/CR-4513, which yields an aged material toughness K_{Jc}.

The evaluation of aged material toughness K_{Jc} used the most-conservative infinite time-at-temperature or “saturation” J-R curve for each material. The results of this evaluation of thermal embrittlement effects therefore do not depend on design life, and this aspect of the analysis is therefore not a TLAA.

End-Point Crack Size Determination: A crack will become unacceptable if any of the following would occur under design, emergency, or faulted loads:

- The crack is unstable against non-ductile propagation (“brittle fracture”).
- The crack is unstable against ductile tearing.
- The remaining ligament cannot carry its design loads, based on its flow stress.

Stability against brittle fracture is in part determined by the material fracture toughness and therefore depends in part on any reduction of fracture toughness due to thermal embrittlement. Stability is indicated if the applied stress intensity factor K_I (ASME Section XI, Equation A-3300 (1)), for all load conditions at all locations of concern, remains less than the aged material toughness K_{Jc}.

At Palisades the maximum end-point values (at the volute crotch) is K_I = 130, minimum K_{Jc} = 152.3, and median K_{Jc} = 223.5 (all in ksi in$^{1/2}$). Therefore substantial margins would remain at the end of a 40 year life, even assuming K_I is the maximum that would result from a complete design basis complement of load cycles. Of these two parameters, K_I depends on the loading and the flaw sizes at the end of evaluation period, and the aged material toughness K_{Jc} in this analysis is the saturation J-R curve for each material. Thus the expected additional thermal aging effect will not diminish the K_{Jc} assumed by the analysis, the analysis will remain acceptable so long as the design basis complement of load cycles is not exceeded, which will ensure that the worst-location K_I remains below 130 ksi in$^{1/2}$.

Stability against ductile tearing is ensured if the applied crack extension J never reaches the J-integral required to initiate ductile tearing. This condition is also satisfied so long as K_I does not exceed K_{Jc}, as above.

The flow stress ligament capacity limit is defined as the crack size for which a ligament flow stress evaluation no longer demonstrates that the ligament will support the required loads.

At Palisades the flow stress ligament capacity limit is the most time limiting of these three criteria. The critical crack size based on this criterion would be exceeded in a little over 45 years, assuming the 40 year design basis cycle count limit is reached at 40 years for all load events, and that the load cycles continue to accumulate at a constant rate. Since the margin to the K_{Jc} vs. K_I brittle fracture stability criterion will not change, the cycle-dependent ligament flow stress criterion will remain limiting for the extended licensed operating period.

Disposition: 10 CFR 54.21(c)(1)(i) and 10 CFR 54.21(c)(1)(iii)

The projected number of each of the design basis events of the primary coolant pumps will approach its design basis limit during the extended licensed operating period. Therefore the maximum end-of-60-year life crack size is not expected to approach the critical crack size during the 60-year extended licensed operating period

The Fatigue Monitoring Program will ensure that the effects of aging will be adequately managed for the period of extended operation by assuring that a reanalysis or other appropriate corrective action is taken if a design basis cycle count limit is reached at any time during the extended licensed operating period.

A4.5.4 Risk-Informed Inservice Inspection Program Calculations

The scope and inspection intervals of the Palisades Risk-Informed Inservice Inspection (RI-ISI) Program depended on probabilities of failure of each pipe segment calculated at the time of program implementation (approximately year 29) and the time of initial license expiration (year 40).

Analysis

The scope of the RI-ISI program may increase somewhat when the extended operating period is considered.

Disposition: 10 CFR 54.21(c)(1)(ii)

The supporting calculations for the Palisades RI-ISI program will be reviewed, and updated as needed, to reflect a 60-year operating period; and the program inspection scope will be updated accordingly, before the period of extended operation.

APPENDIX B

AGING MANAGEMENT PROGRAMS

APPENDIX B
Contents

B1.0 Appendix B Introduction

B1.1 Overview

This appendix provides aging management program descriptions for each Palisades program credited for managing aging effects based upon the aging management review results provided in Sections 3.1 through 3.6. Many of the programs have already been implemented in some form at Palisades, although, often under another program name, or as groups of individual activities. To facilitate NRC review, many existing activities have been realigned within the program organization described herein. For License Renewal Purposes, a program is considered an existing program if the types of activities it manages are, for the most part, already in place, even though its title, organization, or scope has differed from those described in NUREG-1801.

Each of the new or existing aging management programs presented in this section has been evaluated for consistency with the ten elements described in the applicable Section X or XI of NUREG-1801. Evaluation results are provided for each program to indicate whether the element is consistent with, or consistent with exceptions, to the assumptions made in that program's description in NUREG-1801. In addition, for new or existing plant-specific programs that are not addressed in NUREG-1801, a discussion is provided to show how the program conforms with the ten elements described in NUREG-1800, Appendix A.1, “Aging Management Review - Generic,” Table A.1-1, “Elements of an Aging Management Program for License Renewal,” (SRP-LR).

Program enhancements are identified for some of the existing programs that will bring the programs into conformance with NUREG-1801. Each program in this appendix is described as if the identified enhancements have been implemented. Enhancements are discussed one time in the enhancement section of each program description, and are not repeated in the individual element discussions. New programs will not list specific enhancements, per se.

Each NUREG-1801 program description is considered to include the minimum features of the program that have been found by NRC to effectively manage aging of the SSC of interest. In some cases, Palisades takes exception to certain aspects of a program description provided in NUREG-1801. An exception to NUREG-1801 exists when a program feature described in NUREG-1801 is not implemented within the Palisades program. When the Palisades program includes additional features, or is applied to a wider scope of SSC, that go beyond the description in NUREG-1801, this is not considered an exception. Exceptions are typically taken if a NUREG-1801 program feature is not practical to implement at Palisades, or if NUREG-1801 program items are unnecessary to effectively manage aging of the plant-specific SSC for which the program is credited.

Inservice inspection code relief requests that have been approved by NRC are not considered exceptions to NUREG-1801, and are not listed in this application. The NRC has already reviewed the requests in accordance with existing regulations, and concluded that they are acceptable interpretations of, or deviations from, the code requirements applicable to the current inspection interval. When a new inspection code of record is adopted at the beginning of each subsequent inspection interval, in accordance with 10 CFR 50.55a, all relief requests that are applicable to the new interval must be reviewed and approved by the NRC at that time. This process is independent from license renewal requirements under 10 CFR 54, and is unrelated to the term of the operating license.

It should also be noted that if a Palisades program is fully consistent with NUREG-1801, but is based on a different revision (usually later) of a code or standard referenced in the NUREG-1801 program description, the program is not considered to have an exception to NUREG-1801. A number of the codes and standards referenced in NUREG-1801 have been superseded by later revisions since NUREG-1801 was written. Adoption of updated codes or standards for plant programs occurs frequently as the state of knowledge advances in the nuclear industry, and updates are sometimes mandated by NRC's own requirements. Therefore, differences in reference codes or standards, between a Palisades program and the applicable description in NUREG-1801, are not, in themselves, considered to be exceptions to NUREG-1801.

Method of Discussion

For existing or new aging management programs that are addressed in Sections X or XI of NUREG-1801, each program discussion is presented in the following format:

- A Program Description abstract of the overall program form and function is provided.
- A NUREG-1801 Consistency statement is made about the program.
- Exceptions to the NUREG-1801 program are outlined and a justification is provided, when applicable.

  For those aging management programs that are plant-specific (i.e., not in NUREG 1801), there will be no exceptions listed since there is no applicable NUREG-1801 reference standard for the program.

- Enhancements to each existing program are identified. These enhancements include changes initiated to achieve consistency with NUREG-1801, as well as changes to incorporate additional features needed for effective aging management of components which credit the program (e.g., manage aging for certain components or aging effects not assumed in NUREG-1801). Except when otherwise noted in an individual program discussion, enhancements are scheduled to be implemented prior to the period of extended operation. Note that the listed enhancements are defined at an overall program level, and do not include various administrative and editorial changes planned for the implementing documents and activity schedules.

Since new programs are, by definition, yet to be implemented, specific features that will be incorporated into the programs will not be listed as enhancements. The only “enhancement” that will be identified for any new program will be to develop and implement the new program as described in this section.

- A discussion of the ten elements is provided for each program. Comparisons are made to, and consistency evaluated for, each NUREG-1801 or NUREG-1800 program element, as applicable.
- A Conclusion Section provides a statement of reasonable assurance that the program, as described, will be effective.

B1.2 Quality Assurance Program and Administrative Controls

The Palisades Quality Program is described in the topical report, Quality Program Description for Nuclear Power Plants (Part 2) - Palisades Nuclear Plant (Reference 3). This NRC-approved program implements the requirements of 10 CFR 50, Appendix B, and is consistent with the summary in Appendix A.2 of NUREG-1800 (Reference 1).The Quality Program includes the Aging Management Program elements of corrective action, confirmation process and administrative controls.

The Quality Program implementation procedures will be expanded to apply the elements of corrective action, confirmation process, and administrative controls to both safety related and non-safety related systems, structures, and components that are subject to aging management review for license renewal.

Generically, these three elements will be applicable as follows:

Corrective Actions:

Corrective actions are implemented in accordance with the requirements of 10 CFR 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants” (Reference 4), and the NRC approved Quality Program Description for Nuclear Power Plants (Part 2) - Palisades Nuclear Plant (Reference 3). Controls are established to assure that conditions adverse to quality are identified and documented and that appropriate remedial action is taken. For significant conditions adverse to quality, necessary corrective action is promptly determined and recorded. Corrective action includes determining the cause and extent of the condition, and taking appropriate action to preclude similar problems in the future. The controls also assure that corrective action is implemented in a timely manner.

Corrective actions are implemented through the initiation of an Action Request in accordance with plant procedures. Equipment deficiencies may be initially documented by a work order, but the corrective action process specifies that an Action Request also be initiated if required. This approach ensures that identified problems are corrected in a timely manner.

Confirmation Process:

The confirmation process is part of the corrective action program, which is implemented in accordance with the requirements of 10 CFR 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants” (Reference 4), and the NRC approved Quality Program Description for Nuclear Power Plants (Part 2) - Palisades Nuclear Plant (Reference 3). The aging management activities required by this program would also uncover any unsatisfactory condition due to ineffective corrective action.

Administrative Procedures include provisions for identification, evaluation, assignment, tracking, monitoring, reviewing, verifying, and approving corrective actions, to ensure effective corrective actions are taken. The Corrective Action Process is also monitored for potentially adverse trends. The existence of an adverse trend due to recurring or repetitive adverse conditions is required to be documented in an Action Request. The post maintenance testing procedure includes provisions for verifying the completion and effectiveness of corrective actions for equipment deficiencies, establishes criteria for the selection and documentation of post-maintenance tests, and provides guidelines to ensure equipment will perform its intended function prior to return to service, and to ensure the original equipment deficiency is corrected and a new deficiency has not been created.

Administrative Controls:

Plant programs are implemented through a variety of procedures and other documents. These implementing documents are subject to administrative controls, including a formal review and approval process, in accordance with the requirements of 10 CFR 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants” (Reference 4), and the NRC approved Quality Program Description for Nuclear Power Plants (Part 2) - Palisades Nuclear Plant (Reference 3).

Administrative procedures provide guidance on procedures and other forms of administrative control documents. Uniform guidelines and requirements are provided for preparing, revising, reviewing and approving procedures. Usage and adherence requirements are also defined for plant procedures.

B1.3 Operating Experience

Industry operating experience was incorporated into the License Renewal process through a review of industry documents to identify aging effects and mechanisms that could challenge the intended function of structures and components within the scope of License Renewal. Review of plant-specific operating experience (corrective actions, maintenance history, etc.) was also performed to identify significant aging effects previously experienced by the plant, and significant issues that were considered relevant to program performance. The results of these searches were provided as inputs to both the aging management review process and development of the aging management programs. The OE applicable to each aging management program is discussed in this section as a part of each aging management program discussion.

It should be noted that a new program will not have OE that can be used to assess effectiveness of the program, since the program has not yet established a performance record. OE is still included in new program discussions, but only to indicate the types of issues that were considered relevant to the design of the new program.

B1.4 Aging Management Programs

The following aging management programs are described in the sections listed in this appendix. The programs are either generic in nature (i.e., similar features apply to multiple plants, including Palisades) as discussed in NUREG-1801, or are plant-specific (i.e., features are unique to Palisades). Plant-specific programs are listed near the end of the table in Section B2.0. All generic programs are identified in this section as either fully consistent with the program description provided in NUREG-1801, or as consistent except for certain identified exceptions that are listed.

1. Alloy 600 Inspection Program [Section B2.1.1]
2. ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program [Section B2.1.2]
3. Bolting Integrity Program [Section B2.1.3]
4. Boric Acid Corrosion Program [Section B2.1.4]
5. Buried Services Corrosion Monitoring Program [Section B2.1.5]
6. Closed Cycle Cooling Water Program [Section B2.1.6]
7. Containment Inservice Inspection Program [Section B2.1.7]
8. Containment Leakage Testing Program [Section B2.1.8]

9. Diesel Fuel Monitoring and Storage Program [Section B2.1.9]
10. Fire Protection Program [Section B2.1.10]
11. Flow Accelerated Corrosion Program [Section B2.1.11]
12. Non-EQ Electrical Commodities Condition Monitoring Program [Section B2.1.12]
13. One-Time Inspection Program [Section B2.1.13]
14. Open Cycle Cooling Water Program [Section B2.1.14]
15. Overhead Load Handling Systems Inspection Program [Section B2.1.15]
16. Reactor Vessel Integrity Surveillance Program [Section B2.1.16]
17. Reactor Vessel Internals Inspection Program [Section B2.1.17]
18. Steam Generator Tube Integrity Program [Section B2.1.18]
19. Structural Monitoring Program [Section B2.1.19]
20. System Monitoring Program [Section B2.1.20]
21. Water Chemistry Program [Section B2.1.21]

B1.5 Time Limited Aging Analyses Management Programs:

1. Electrical Equipment Qualification Program [Section B3.1]
2. Fatigue Monitoring Program [Section B3.2]

B2.0 Aging Management Programs

Correlations between NUREG-1801 (Generic Aging Lessons Learned (GALL)) programs and Palisades' programs are shown below. For the Palisades Programs, links to appropriate sections of this appendix are provided.

NUREG-1801 ID NUMBER	NUREG-1801 PROGRAM TITLE	PALISADES AGING MANAGEMENT PROGRAM
Chapter XI Programs		
XI.M1	ASME Section XI Inservice Inspection, Subsections IWB, IWC, & IWD	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program [Section B2.1.2]
XI.M2	Water Chemistry	Water Chemistry Program [Section B2.1.21]
XI.M3	Reactor Head Closure Studs	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program [Section B2.1.2]
XI.M4	BWR Vessel ID Attachment Welds	Not Applicable. PNP is a PWR.
XI.M5	BWR Feedwater Nozzle	Not Applicable. PNP is a PWR.
XI.M6	BWR Control Rod Drive Return Line Nozzle	Not Applicable. PNP is a PWR.
XI.M7	BWR Stress Corrosion Cracking	Not Applicable. PNP is a PWR.
XI.M8	BWR Penetrations	Not Applicable. PNP is a PWR.
XI.M9	BWR Vessel Internals	Not Applicable. PNP is a PWR.
XI.M10	Boric Acid Corrosion	Boric Acid Corrosion Program [Section B2.1.4]
XI.M11	Nickel-Alloy Nozzles and Penetrations	Alloy 600 Program [Section B2.1.1]
XI.M12	Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)	Not applicable. Palisades does not have CASS piping. CASS valves and pump casings are managed as indicated in Section [Section B2.1.2].
XI.M13	Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)	Not Applicable. Palisades does not have CASS reactor vessel internals.

NUREG-1801 ID NUMBER	NUREG-1801 PROGRAM TITLE	PALISADES AGING MANAGEMENT PROGRAM
XI.M14	Loose Part Monitoring	Not credited for aging management
XI.M15	Neutron Noise Monitoring	Not credited for aging management
XI.M16	PWR Vessel Internals	Reactor Vessel Internals Inspection Program [Section B2.1.17]
XI.M17	Flow-Accelerated Corrosion	Flow Accelerated Corrosion Program [Section B2.1.11]
XI.M18	Bolting Integrity	Bolting Integrity Program [Section B2.1.3]
XI.M19	Steam Generator Tube Integrity	Steam Generator Tube Integrity Program [Section B2.1.18]
XI.M20	Open-Cycle Cooling Water System	Open Cycle Cooling Water Program [Section B2.1.14]
XI.M21	Closed-Cycle Cooling Water System	Closed Cycle Cooling Water Program [Section B2.1.6]
XI.M22	Boraflex Monitoring	Not Credited for Aging Management
XI.M23	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems	Overhead Load Handling Systems Inspection Program [Section B2.1.15]
XI.M24	Compressed Air Monitoring	Not credited for aging management.
XI.M25	BWR Reactor Water Cleanup System	Not Applicable. Palisades is a PWR.
XI.M26	Fire Protection	Fire Protection Program [Section B2.1.10]
XI.M27	Fire Water System	Fire Protection Program [Section B2.1.10]
XI.M28	Buried Piping and Tanks Surveillance	Not credited for aging management.
XI.M29	Aboveground Carbon Steel Tanks	System Monitoring Program [Section B2.1.20] One-Time Inspection Program [Section B2.1.13]

NUREG-1801 ID NUMBER	NUREG-1801 PROGRAM TITLE	PALISADES AGING MANAGEMENT PROGRAM
XI.M30	Fuel Oil Chemistry	Diesel Fuel Monitoring and Storage Program [Section B2.1.9]
XI.M31	Reactor Vessel Surveillance	Reactor Vessel Integrity Surveillance Program [Section B2.1.16]
XI.M32	One-Time Inspection	One-Time Inspection Program [Section B2.1.13]
XI.M33	Selective Leaching of Materials	One-Time Inspection Program [Section B2.1.13]
XI.M34	Buried Piping and Tanks Inspection	Buried Services Corrosion Monitoring Program [Section B2.1.5]
XI.S1	ASME Section XI, Subsection IWE	Containment Inservice Inspection Program [Section B2.1.7]
XI.S2	ASME Section XI, Subsection IWL	Containment Inservice Inspection Program [Section B2.1.7]
XI.S3	ASME Section XI, Subsection IWF	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program [Section B2.1.2]
XI.S4	10 CFR 50, Appendix J	Containment Leakage Testing Program [Section B2.1.8]
XI.S5	Masonry Wall Program	Structural Monitoring Program [Section B2.1.19]
XI.S6	Structures Monitoring Program	Structural Monitoring Program [Section B2.1.19]
XI.S7	RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants	Structural Monitoring Program [Section B2.1.19]
XI.S8	Protective Coating Monitoring and Maintenance Program	Not credited for aging management.

NUREG-1801 ID NUMBER	NUREG-1801 PROGRAM TITLE	PALISADES AGING MANAGEMENT PROGRAM
XI.E1	Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Non-EQ Electrical Commodities Condition Monitoring Program [Section B2.1.12]
XI.E2	Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits	Non-EQ Electrical Commodities Condition Monitoring Program [Section B2.1.12]
XI.E3	Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Non-EQ Electrical Commodities Condition Monitoring Program [Section B2.1.12]
Chapter X Programs		
X.M1	Metal Fatigue of Reactor Coolant Pressure Boundary	Fatigue Monitoring Program [Section B3.2]
X.S1	Concrete Containment Tendon Prestress	Containment Inservice Inspection Program [Section B2.1.7]
X.E1	Environmental Qualification (EQ) of Electrical Components	Electrical Equipment Qualification Program [Section B3.1]
Plant-Specific Programs		
NA	NA	System Monitoring Program [Section B2.1.20]

B2.1 Aging Management Programs Details

B2.1.1 Alloy 600 Program

Program Description

The Alloy 600 Program is an existing program that manages aging due to PWSCC of the Primary Coolant System (PCS) pressure boundary Alloy 600 components, including Inconel 82/182 weld joints, reactor vessel head penetrations, etc. The program includes (a) PWSCC susceptibility assessment using industry models to identify susceptible components, (b) monitoring and control of primary coolant chemistry to mitigate PWSCC, (c) in-service inspections (ISI) of pressurizer penetrations, reactor vessel head penetrations and Alloy 82/182 PCS pressure boundary welds in accordance with ASME Section XI, Subsection IWB, Table IWB-2500-1, and (d) augmented inspections or preemptive repair/replacement of susceptible components or welds.

NUREG-1801 references a long-term inspection program for the vessel head penetrations based on the industry responses to NRC GL 97-01. Subsequent to issuance of NUREG-1801, events at another nuclear plant triggered establishment of new regulatory requirements that superseded NRC GL 97-01 for reactor pressure vessel head and penetration nozzle inspections. Palisades is committed to the revised NRC Order EA-03-009 (Reference 5). Because these new regulatory requirements were issued after NUREG-1801 was published, this is not considered an exception to NUREG-1801.

It is noted that the Palisades Section XI ISI Program is currently following ASME Section XI, in accordance with applicable provisions and requirements of 10 CFR 50.55a, while NUREG-1801, Section XI.M11, references the 1995 Edition through the 1996 Addenda. The ASME Section XI ISI edition and addenda in effect at any time will be those required by 10 CFR 50.55a. Therefore, the edition and addenda on which the Palisades program is based are not considered exceptions to NUREG-1801.

NUREG-1801 Consistency

The Alloy 600 Program is consistent with NUREG-1801, Section XI.M11, “Nickel-Alloy Nozzles and Penetrations.”

Exceptions to NUREG-1801

None.

Enhancements

None.

Aging Management Program Elements

The key elements, which are used in the Alloy 600 Program, are described below. The results of an evaluation of each key element against NUREG-1801, Section XI.M11, “Nickel-Alloy Nozzles and Penetrations,” are also provided below.

Scope of Program

All Alloy 600 components and Alloy 82/182 welds that are part of the Primary Coolant pressure boundary are within the scope of the Alloy 600 Program, and include the following:

- CRDM nozzles
- Reactor vessel and vessel head mounted instrumentation penetrations
- Pressurizer nozzles and penetrations
- Reactor Coolant System piping penetrations and nozzles
- Steam Generator penetrations (Steam Generator U-tubes are Alloy 600, and are managed by the Steam Generator Tube Integrity Program)

This program credits the ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program for the performance of required inservice inspections in accordance with ASME Section XI, Subsection IWB, Table IWB-2500-1. This program also credits the Water Chemistry Program for monitoring and control of primary coolant water chemistry in accordance with EPRI guideline TR-105714 to mitigate PWSCC.

Palisades has 251 Alloy 600 penetrations, all of which are contained within the Primary Coolant System (PCS). As a result of Alloy 600 cracking issues associated with the pressurizer power-operated relief valve (PORV) nozzle, a project was initiated in 1993 to identify and rank all Alloy 600 penetrations contained within the PCS. This project ranked all 251 Alloy 600 penetrations based on four main criteria: primary water stress corrosion cracking (PWSCC) susceptibility, failure consequence, leakage detection margin, and radiation dose rates.

The susceptibility of all Alloy 600 components and Alloy 82/182 welds was evaluated. On February 11, 2003, the NRC issued Order EA-03-009 to licensees operating PWRs, and revised the Order on February 20, 2004 (Reference 5) which NMC consented to on March 8, 2004 (Reference 6). The Order established a minimum set of reactor vessel head inspections based on susceptibility categories. Palisades’ long-term inspection requirements are based on the revised NRC Order EA-03-009 and the March 8, 2004 NMC response.

On May 28, 2004, the NRC issued Bulletin 2004-01, “Inspections of Alloy 82/182/600 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at Pressurized-Water reactors” requiring bare metal inspections of pressurizer heater sleeve locations and steam space piping connections and NDE if visual inspections indicate evidence of leakage. Palisades’ long-term inspection requirements for these components are based on the NMC response to that bulletin dated July 26, 2004.

This element is consistent with NUREG-1801, Section XI.M11, “Nickel-Alloy Nozzles and Penetrations.”

Preventive Actions

The Alloy 600 Program is a condition monitoring program and there are no preventive actions associated with this program. However, this program credits the Water Chemistry Program for monitoring and control of primary coolant water chemistry to mitigate PWSCC.

This element is consistent with NUREG-1801, Section XI.M11, “Nickel-Alloy Nozzles and Penetrations.”

Parameters Monitored, Inspected, and/or Tested

The Alloy 600 Program monitors the effects of PWSCC on the intended function of the Alloy 600 reactor vessel head penetrations by detection and sizing of cracks and detection of coolant leakage with the implementation of the Inservice Inspection Program as modified by the revised NRC Order EA-03-009.

Additionally, Palisades is committed to pressurizer and other primary coolant pressure boundary Alloy 600 penetration inspection requirements as discussed in the NMC response to Bulletin 2004-01 dated July 26, 2004.

This element is consistent with NUREG-1801, Section XI.M11, “Nickel-Alloy Nozzles and Penetrations.”

Detection of Aging Effects

The Palisades Alloy 600 Program includes a variety of inspection and testing activities that are designed to detect degradation due to aging effects prior to loss of intended function.

Inspections are based on the requirements prescribed by the revised NRC Order EA-03-009 and Bulletin 2004-01 rather than the model referenced in the

GL 97-01 industry response. This also includes the susceptibility assessment, program, scope, and schedule. Inspections consist of periodic visual and NDE examinations and pressure tests, that are capable of detecting cracks and leakage before the loss of intended function of the components.

Palisades is committed to following the requirements of NRC Order EA-03-009, which supersedes GL 97-01 requirements for leakage detection. Therefore, the "leakage detection system" referred to in the NUREG-1801 program, and attributed to GL 97-01, is satisfied by the visual inspections of Order EA-03-009.

This element is consistent with NUREG-1801, Section XI.M11, "Nickel-Alloy Nozzles and Penetrations."

Monitoring and Trending

Periodic reactor vessel head inspections, including examination schedule, scope expansion, and comparison of previous examination results, are performed per the requirements of revised NRC Order EA-03-009. Periodic pressurizer heater sleeve connections and primary coolant pressure boundary penetration inspections are performed per the requirements of NRC Bulletin 2004-01. Following completion of bare metal reactor vessel head, pressurizer, and primary coolant pressure boundary penetration inspections, all boric acid deposits, staining, or scaling is cleaned down to bare metal prior to installation of insulation. This provides assurance that any deposits discovered during reactor vessel head visual examinations resulted from leakage that occurred since the previous visual examination.

This element is consistent with NUREG-1801, Section XI.M11, "Nickel-Alloy Nozzles and Penetrations."

Acceptance Criteria

Indications and non-conformances detected will be evaluated in accordance with ASME Section XI Program requirements and the ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program. Any evidence of leakage noted during the reactor vessel head visual inspections are considered recordable indications requiring further evaluation and inspection, and are reportable to the NRC, in accordance with the revised NRC Order EA-03-009.

This element is consistent with NUREG-1801, Section XI.M11, "Nickel-Alloy Nozzles and Penetrations."

Corrective Actions, Confirmation Process, Administrative Controls

These elements are consistent with the corresponding NUREG-1801 aging management program elements. See Section B1.2 for further discussion.

Operating Experience

Palisades has a comprehensive Operating Experience Program (OEP) that monitors industry issues/events and assesses these for applicability to its own operations. In addition, the Palisades Corrective Action Program (CAP) is used to track, trend and evaluate plant issues/events. Those issues and events, whether external or plant specific, that are potentially significant to the Alloy 600 Program at Palisades are evaluated. The Alloy 600 Program is augmented, as appropriate, if these evaluations show that program changes will enhance program effectiveness.

Using the OEP and CAP to focus on industry and plant operating experience ensures that Alloy 600 Program issues are addressed in a timely manner and that age related deterioration of SSC within the scope of the Alloy 600 Program will be effectively managed throughout the license renewal period.

A review of the industry operating experience related to the Alloy 600 Program revealed instances where degradation of material has occurred as a result of PWSCC. The review also considered related issues which included degradation of PCS hot leg piping and nozzles, thermal sleeves and instrument nozzles, reactor vessel head nozzles, control rod drive mechanism and thermocouple nozzles and intrusion of demineralizer resins.

A review of the plant specific operating experience revealed four (4) instances where the Alloy 600 program has been instrumental in discovering material degradation. Degradation was discovered in the following items:

- Pressurizer Temperature Element Penetration
- Pressurizer Safe End
- CRD Nozzle penetration indications (2)

Two of the safe-ends on the primary coolant piping (12-inch, schedule 140 surge nozzle and 12-inch, schedule 140, shutdown cooling outlet nozzle) had MSIP (Mechanical Stress Improvement Process) applied to them in 1995. MSIP changes the residual stress patterns at these locations from tensile to compressive by plastically deforming the piping near the welds. The compressive residual stress is desired to help mitigate PWSCC.

The first bare metal visual examination of the reactor vessel head to meet the requirements of NRC Order EA-09-003 was completed during the spring, 2003 refueling outage, with results reported to the NRC. Qualified VT-2 examiners using direct visual techniques performed the examinations. Visual inspection was also performed to identify potential boric acid leaks from pressure retaining components from above the reactor vessel. Results found that there were no accumulations of boric acid in the vicinity of, and no leakage of boric acid through, any of the reactor vessel head penetrations. All visual examinations of the reactor vessel head penetrations had acceptable results.

The first volumetric examinations performed in accordance with the revised NRC order EA-03-009 were completed during the fall, 2004 refueling outage. Results of the inspections identified two CRD nozzle penetrations showing indications that required repair. These repairs were completed during the outage.

In response to commitments made by Palisades to NRC Bulletin 2004-01, “Inspection of Alloy 82/182/600 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at Pressurized-Water Reactors,” a bare metal visual examination of 100% of the pressurizer heater sleeve locations, including 360° around each sleeve, and 36 Alloy 82/182/600 primary system pressure boundary locations, normally operated at greater than or equal to 350°F, was performed during the fall 2004 refueling outage. Results of the bare metal visual examination of each of the penetrations were reported to the NRC and were acceptable, with no accumulation of boric acid in the vicinity of any of the penetrations.

The Palisades Alloy 600 Program has demonstrated that it provides reasonable assurance that aging effects are being adequately managed for Alloy 600 materials. This has been demonstrated through NRC inspection reports, INPO evaluations, audits, self-assessments, and the Corrective Action Program. The Alloy 600 Program has been effective in identifying material degradation in a timely manner, thus ensuring that age related degradation of Alloy 600 materials will be effectively managed throughout the license renewal period.

This element is consistent with NUREG-1801, Section XI.M11, “Nickel-Alloy Nozzles and Penetrations.”

Conclusion

The Alloy 600 Program is an existing program that uses as its bases, various industry and NRC standards. This program is consistent with NUREG-1801, Section XI.M11, “Nickel-Alloy Nozzles and Penetrations.”

Reports of recent internal and external assessments of the program have been reviewed. These reports demonstrate that the program has effectively identified and dispositioned issues that could have led to degraded conditions. It is concluded that the Alloy 600 Program has been effective in maintaining the intended functions of the affected long-lived, passive SSCs.

The continued implementation of the Alloy 600 Program provides reasonable assurance that aging effects will be managed such that SSCs within the scope of this program will continue to perform their intended functions consistent with the current licensing bases for the period of extended operation.

B2.1.2 ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program

Program Description

ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program is an existing program that facilitates inspections to identify and correct degradation in Class 1, 2, and 3 piping, components, their supports and integral attachments. The program includes periodic visual, surface and/or volumetric examinations and leakage tests of all Class 1, 2 and 3 pressure-retaining components, their supports and integral attachments, including welds, pump casings, valve bodies, pressure-retaining bolting, piping/component supports, and reactor head closure studs. These are identified in ASME Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” or commitments requiring augmented inservice inspections, and are within the scope of license renewal. This program is in accordance with 10 CFR 50.55a.

The Code of Federal Regulations, 10 CFR 50.55a, requires that inservice inspection of Class 1, 2, and 3 pressure retaining components, their integral attachments and supports be conducted in accordance with the latest edition of ASME Section XI approved by the NRC twelve months prior to the start of a ten year interval. The ISI Program for Palisades’ third (3rd) ten year interval, which began on May 12, 1995, implements ASME Section XI in accordance with applicable provisions and requirements of 10 CFR 50.55a. The IWB-2500 Category B-Q requirements to perform volumetric examinations of steam generator tubes is addressed by the Steam Generator Tube Integrity Program.

Class 1, 2 and 3 Component Supports

The examination scope provided by Table 2500-1 of Code Case N491-2 specifies the percentage of supports that must be examined. As specified by Table 2500-1, VT-3 examinations are conducted to determine the general mechanical and structural condition of components and their supports by verifying parameters such as clearances, settings, and physical displacement, and to detect discontinuities and imperfections, such as loss of integrity of bolted or welded connections, loose or missing parts, debris, corrosion, wear, or erosion. Acceptance standards for supports are given by Code Case N491-2. Unacceptable conditions include:

- Deformation or structural degradation of fasteners, springs, clamps, or other support items;
- Missing, detached, or loosened support items;
- Arc strikes, weld spatter, paint, scoring, roughness, or general corrosion on close tolerance machined or sliding surfaces;
- Improper hot or cold positions of spring supports and constant load supports;
- Misalignment of supports; and
- Improper clearances of guides and stops.

Class 1, 2 and 3 Pressure-Retaining Bolting

Nondestructive examination, repair and replacement of pressure retaining bolting are conducted as part of the ISI Program. Examination requirements are in accordance with ASME Section XI, Table IWB-2500-1 or IWC-2500-1. When Class 1, 2 and 3 bolting must be replaced as a result of degradation, ASME Section XI requirements for preservice examination are performed.

Reactor Vessel Head Closure Studs

Reactor vessel head closure studs are examined as required by Table IWB-2500-1, examination category B-G-1, “Pressure Retaining Bolting Greater than 2 inches in Diameter.” Volumetric examinations are performed using Performance Demonstration Initiative (PDI) techniques in accordance with ASME Section XI, Appendix VIII and 10 CFR 50.55a. Lubrication of the studs is controlled via maintenance procedures.

NUREG-1801 Consistency

The ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program is consistent with the following sections of NUREG-1801:

- Section XI.M1, “ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD”
- Section XI.M3, “Reactor Head Closure Studs”
- Section XI.S3, “ASME Section XI, Subsection IWF”

Exceptions to NUREG-1801

None.

Enhancements

None.

Aging Management Program Elements

The key elements, which are used in the ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program, are described below. The results of an evaluation of each key element against NUREG-1801, Section XI.M1, “ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD,” Section XI.M3, “Reactor Head Closure Studs,” and Section XI.S3, “ASME Section XI, Subsection IWF,” are also provided below.

Scope of Program

The ASME Section XI IWB, IWC, IWD and IWF Inservice Inspection Program implements the requirements for inservice inspection, repair, and replacement of ASME Class 1, 2, and 3 components, their integral attachments and supports, including all pressure retaining bolting. This program also includes the ISI of the reactor vessel closure studs and nuts, and includes the preventive measures of RG 1.65, “Material and Inspection for Reactor Vessel Closure Studs,” to mitigate cracking. Section XI provides rules for exempting components from volumetric or surface examinations and Federal Law allows the NRC to grant relief from specific portions of the code upon demonstrated need.

This element is consistent with NUREG-1801, Section XI.M1, “ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD,” Section XI.M3, “Reactor Head Closure Studs,” and Section XI.S3, “ASME Section XI, Subsection IWF.”

Preventive Actions

There are no preventative actions associated with the ASME Section XI IWB, IWC, IWD and IWF Inservice Inspection Program.

This element is consistent with NUREG-1801, Section XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," and Section XI.S3, "ASME Section XI, Subsection IWF."

For the reactor vessel head closure studs, preventive measures include the use of manganese phosphate surface treatment and stable lubricants to mitigate degradation. This element is consistent with Section XI.M3, "Reactor Head Closure Studs."

Parameters Monitored, Inspected, and/or Tested

The ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program detects degradation of components by using the examination and inspection techniques of ASME Section XI.

Components with pressure retaining bolting are inspected for signs of leakage during pressure testing.

For piping and component supports, the parameters monitored or inspected include corrosion, deformation, misalignment, improper clearances, improper spring settings, damage to close tolerance machined or sliding surfaces, and missing, detached, or loosened support items. Component support bolting is inspected for indication of potential problems, including obvious signs of corrosion.

This element is consistent with NUREG-1801, Section XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," Section XI.M3, "Reactor Head Closure Studs," and Section XI.S3, "ASME Section XI, Subsection IWF."

Detection of Aging Effects

The ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program includes a variety of inspection and testing activities that are designed to detect degradation due to aging effects prior to loss of intended function.

The extent and schedule of the inspection and test techniques prescribed by the program are designed to maintain structural integrity and ensure that aging effects will be discovered and repaired before the loss of a component intended function. Inspection can reveal crack initiation and growth, loss of material due to corrosion, leakage of coolant, and indications of degradation caused by wear or stress relaxation, such as verification of clearances, settings, physical displacements, loose or missing parts, debris, wear, erosion, or loss of integrity at bolted or welded connections.

The program uses three types of examination; visual, surface, and volumetric in accordance with the general requirements of Subsection IWA-2000. Visual VT-1 examination detects discontinuities and imperfections, such as cracks, corrosion, wear, or erosion, on the surface of components. Visual VT-2 examination detects evidence of leakage from pressure retaining components, as required during the system pressure test. Visual VT-3 examination (a) determines the general mechanical and structural condition of components and their supports by verifying parameters, such as clearances, settings, and physical displacements; and (b) detects discontinuities and imperfections, such as loss of integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear, or erosion. Surface examination uses magnetic particle, liquid penetrant, or eddy current examinations to indicate the presence of surface discontinuities and flaws. Volumetric examination uses radiographic, ultrasonic, or eddy current examinations to indicate the presence of discontinuities or flaws throughout the volume of material.

The ISI Program for the 3rd Inspection Interval meets the requirements of ASME Section XI, in accordance with applicable provisions and requirements of 10 CFR 50.55a. These include visual VT-2 examination of all pressure retaining components during the system leakage test. The Palisades ISI Program specifies performance of the system leakage test once per refueling outage or each period for these examination categories, whichever applies.

Examination requirements for Class 1, 2 and 3 pressure retaining bolting are in accordance with ASME Section XI, Table IWB-2500-1 or IWC-2500-1. For Class 1 components, Table IWB-2500-1, examination category B-G-1, for bolting greater than 2 inches in diameter, specifies volumetric examination of studs and bolts and visual VT-1 examination of surfaces of nuts, washers, bushings, and flanges. Examination category B-G-2, for bolting 2 inches or smaller requires only visual VT-1 examination of surfaces of bolts, studs, and nuts. For Class 2 components, Table IWC-2500-1, examination category C-D, for bolting greater than 2 inches in diameter, requires volumetric examination of studs and bolts. Examination Categories B-P and C-H, require VT-2 visual examination (IWA-5240) during system leakage testing of all pressure retaining Class 1 and 2 components, according to Tables IWB-2500-1 and IWC-2500-1 respectively. The extent and schedule of inspections, in accordance with Tables IWB-2500-1 and IWC-2500-1 ensure detection of aging degradation before the loss of the intended function of the closure bolting.

Reactor vessel head closure studs are examined as required by Table IWB-2500-1, examination category B-G-1, “Pressure Retaining Bolting Greater than 2 inches in Diameter.” Volumetric examinations are performed using Performance Demonstration Initiative (PDI) techniques in accordance with ASME Section XI, Appendix VIII and 10 CFR 50.55a.

This element is consistent with NUREG-1801, Section XI.M1, “ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD,” Section XI.M3, “Reactor Head Closure Studs,” and Section XI.S3, “ASME Section XI, Subsection IWF.”

Monitoring and Trending

The examination schedules contained in the Palisades ISI Program meet the requirements of ASME Section XI, IWB-2412, IWC-2412, and IWD-2412, respectively (Inspection Program B). The Palisades' ISI Program also meets the requirements for the extent and frequency of examinations specified by ASME Section XI, IWB-2500-1, IWC-2500-1, and IWD-2500-1.

In some cases, an evaluation in accordance with ASME Section XI, IWB-3100 or IWC-3100, may be used to qualify a component with flaw indications as acceptable for continued service. In such cases, the areas containing such flaw indications and relevant conditions are reexamined during the next three inspection periods of IWB-2410 for Class 1 components and for the next inspection period of IWC-2410 for Class 2 components. Examinations that reveal indications that exceed the acceptance standards are extended to include additional examinations in accordance with ASME Section XI, IWB-2430 or IWC-2430 for Class 1 or 2 components, respectively. The Palisades ISI Program meets ASME Section XI with respect to inspection schedules, extent, method, and frequency of examination, flaw evaluations, and additional examinations.

For reactor vessel head closure studs, the inspection schedule of IWB-2400, and the extent and frequency specified in Table IWB-2500-1 provide timely detection of stud and nut degradation.

For piping and component support inspections, unacceptable conditions, as described in Code Case N-491-2 Section 3400, are noted for correction or further evaluation.

This element is consistent with NUREG-1801, Section XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," Section XI.M3, "Reactor Head Closure Studs," and Section XI.S3, "ASME Section XI, Subsection IWF."

Acceptance Criteria

Indications or relevant conditions that are not removed by repair, replacement, or surface conditioning, where necessary, are evaluated in accordance with IWB-3000, IWC-3000, or IWD-3000 for Class 1, 2, or 3 components, respectively. Examination results are evaluated in accordance with IWB-3100 or IWC-3100 by comparing the results with the acceptance standards of IWB-3400 and IWB-3500 or IWC-3400 and IWC-3500 for Class 1 or Class 2 and 3 components, respectively. In rare cases, flaws exceeding the size of allowable flaws, as defined in IWB-3500 or IWC-3500, may be evaluated by using the analytical procedures of IWB-3600 or IWC-3600.

For reactor vessel head closure studs and Class 1 bolting, degradation is evaluated in accordance with IWB-3100 and the acceptance standards of IWB-3400 and IWB-3500.

The program for component supports utilizes the acceptance standards for visual examination specified in ASME Code Case N491-2, Section 3400. Discovery of unacceptable conditions that require corrective action triggers an expansion of the inspection scope in accordance with the code case. Reexamination of supports requiring corrective actions or acceptance by evaluation shall be performed during the next inspection period, in accordance with the code case.

This element is consistent with NUREG-1801, Section XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," Section XI.M3, "Reactor Head Closure Studs," and Section XI.S3, "ASME Section XI, Subsection IWF."

Corrective Actions, Confirmation Process, Administrative Controls

These elements are consistent with the corresponding NUREG-1801 aging management program elements. See Section B1.2 for further discussion.

Operating Experience

Palisades has a comprehensive Operating Experience Program (OEP) that monitors industry issues/events and assesses these for applicability to its own operations. In addition, the Palisades Corrective Action Program (CAP) is used to track, trend and evaluate plant issues/events. Those issues and events, whether external or plant specific, that are potentially significant to the ASME Section XI ISI Program at Palisades are evaluated. The ASME Section XI ISI Program is augmented, as appropriate, if these evaluations show that program changes will enhance program effectiveness.

Using the OEP and CAP to focus on industry and plant operating experience ensures that ASME Section XI ISI Program issues are addressed in a timely manner and that age related deterioration of SSC within the scope of the ASME Section XI ISI Program will be effectively managed throughout the license renewal period.

A review of the industry operating experience related to the ASME Section XI, Inservice Inspection (ISI) Aging Management Program revealed numerous instances where degradation of components, component supports, and bolting has occurred. In completing our review we looked at related issues which included stress corrosion cracking (SCC) and crack initiation and growth due to thermal loading.

A review of the plant specific operating experience revealed thirteen (13) instances where the ISI program has been instrumental in discovering degradation. Degradation was discovered in the following items:

- Control Rod Drive Housings
- Piping Welds
- Component Supports
- Bolting
- Temperature Element Penetration
- Reactor Coolant Pressurizer Safe End
- Engineered Safeguards Systems Check Valve

The Palisades ISI Aging Management Program has demonstrated on several occasions that it provides reasonable assurance that aging effects are being adequately managed for Class 1, 2, and 3 components, component supports, and bolting. This has been demonstrated through NRC inspection reports, INPO evaluations, audits, self-assessments, and the Corrective Action Program.

This element is consistent with NUREG-1801, Section XI.M1, “ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD,” Section XI.M3, “Reactor Head Closure Studs,” and Section XI.S3, “ASME Section XI, Subsection IWF.”

Conclusion

The ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program is an existing program that uses as its bases, various industry and NRC standards. This program is consistent with NUREG-1801, Section XI.M1, “ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD.” This program is also consistent with NUREG-1801, Section XI.M3, “Reactor Head Closure Studs,” and Section XI.S3, “ASME Section XI, Subsection IWF.”

Reports of recent internal and external assessments of the program have been reviewed. These reports demonstrate that the program has effectively identified and dispositioned issues that could have led to degraded conditions. It is concluded that the ASME Section XI ISI Program has been effective in maintaining the intended functions of the affected long-lived, passive SSCs.

The continued implementation of the ASME Section XI ISI Program provides reasonable assurance that aging effects will be managed such that SSCs within the scope of this program will continue to perform their intended functions consistent with the current licensing bases for the period of extended operation.

B2.1.3 Bolting Integrity Program

Program Description

The Bolting Integrity Program is an existing program that manages the aging effects associated with bolting through the performance of periodic inspections. The program also includes repair/replacement controls for ASME Section XI related bolting and generic guidance regarding material selection, thread lubrication and assembly of bolted joints. The program considers the guidelines delineated in NUREG-1339 for a bolting integrity program, EPRI NP-5769 (with the exceptions noted in NUREG-1339) for safety related bolting, and EPRI TR-104213 for non-safety related bolting.

The Bolting Integrity Program has been created to permit direct comparison with NUREG-1801. The program is considered to be an existing program since most of the activities addressed by the program are already being performed. The program credits activities performed under three separate aging management programs for the inspection of bolting. The three aging management programs are: (1) ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program, (2) Structural Monitoring Program, and (3) System Monitoring Program.

The scope of the credited programs for bolting is summarized below.

- The ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program provides the requirements for inservice inspection of ASME Class 1, 2, and 3 piping, supports, and their integral attachments, which includes pressure retaining and support bolting. This program specifically discusses the inspection and lubrication of the reactor vessel head closure studs. The program supplements the ASME Section XI (Code Case N491-2), Subsection IWF requirements, by applying the inspection requirements of Subsection IWB, Category B-G-1 to high yield strength (≥150 ksi) bolting used in Nuclear Steam Supply System (NSSS) component supports.
- The System Monitoring Program provides the requirements for the inspection of non-safety related bolting within the scope of license renewal.
- The Structural Monitoring Program provides the requirements for the inspection of all structural bolting within the scope of license renewal. Other bolting and fasteners are also included within the scope of this program, such as those used in supports for cable trays, conduits and cabinet supports.

NUREG-1801 states that, “It is noted that hot torquing of bolting is a leak preventive measure once the joint is brought to operating temperature and before or after it is pressurized.” The attributes of the Palisades Bolting Integrity Program are adequate to manage loss of preload without hot torquing. Therefore, hot torquing to establish a pre-load will not be credited for aging management of bolting.

NUREG-1801 Consistency

The Bolting Integrity Program is consistent with NUREG-1801, Section XI.M18, “Bolting Integrity.”

Exceptions to NUREG-1801

None

Enhancements

Two enhancements are planned to bring the Bolting Integrity Program into conformance with the NUREG-1801 program description. The enhancements are:

Review and revise ASME ISI Master Plan, procedures that implement credited License Renewal Programs, and plant maintenance procedures to reflect and reference the applicable guidance provided in NUREG-1339 and EPRI TR-104213 for safety and non-safety related bolting. These revisions should also include instructions for selection of bolting material and use of lubricants and sealants, in accordance with the guidelines of EPRI NP-5769 and the additional recommendations of NUREG-1339 to prevent or mitigate degradation and failure of safety-related bolting.

Evaluate the high strength bolting used for component supports for susceptibility to cracking as described in NUREG-1801, Section XI.M.18, “Parameters Monitored/Inspected,” and implement appropriate inspection requirements to provide adequate age-management for these bolts. This is to be completed prior to the end of the current operating license.

Note that the element descriptions describe the program as it will exist after the identified enhancements have been implemented. Enhancements are scheduled for completion prior to the period of extended operation.

Aging Management Program Elements

The key elements of the Bolting Integrity Program are described below. The results of an evaluation of each key element against the corresponding element of NUREG-1801, Section XI.M18, “Bolting Integrity,” are also provided.

Scope of Program

The program covers all bolting and fasteners within the scope of license renewal, including safety related bolting, bolting for NSSS component supports, bolting for other pressure retaining components, and structural bolting. The Bolting Integrity Program manages the aging effects associated with bolting through the performance of periodic inspections. The program also includes repair/replacement controls for ASME Section XI related bolting and generic guidance regarding thread lubrication and assembly of bolted flanges. The program considers the guidelines delineated in NUREG-1339 for a bolting integrity program, EPRI NP-5769 (with the exceptions noted in NUREG-1339) for safety related bolting, and EPRI TR-104213 for non-safety related bolting. The Bolting Integrity Program credits three separate aging management programs for the inspection of bolting.

NUREG-1801 implies conformance with the recommendations of NUREG-1339, EPRI NP-5769, and EPRI TR-104213 concerning material selection and testing, preload control, inservice inspection, plant operation and maintenance, and evaluation of structural integrity. NUREG-1339 documents the resolution of Generic Safety Issue 29 related to bolting degradation or failure in nuclear power plants. NUREG-1339 relies on the industry technical findings presented in EPRI NP-5769 as the basis for resolution of the generic safety issue. EPRI NP-5769 is a broadly scoped document intended to resolve the generic safety issue and was formatted in a manner to aid the utility engineer in addressing bolting problems. EPRI TR-104213 was developed with the intention of providing a consolidated source of generic technical information pertaining to the design, assembly, inspection, trouble shooting, and repair of bolted joints. EPRI TR-104213 also includes recommendations for the selection, specification, and procurement of threaded fasteners.

This element is consistent with NUREG-1801, Section XI.M18, “Bolting Integrity.”

Preventive Actions

The use of lubricants and sealants is in accordance with plant guidelines, which generally meet the recommendations of EPRI NP-5769 and NUREG-1339. Replacement of bolting in ASME Section XI components is performed in accordance with the plant's repair and replacement program. Replacement of other bolting is like-for-like or evaluated on a case-by-case basis in accordance with the applicable monitoring program identified in the program description above.

Palisades normal maintenance practices and quality verification procedures for pressure retaining bolting includes a check of bolt torque and uniformity of gasket compression, when applicable.

This element is consistent with NUREG-1801, Section XI.M18, “Bolting Integrity.”

Parameters Monitored, Inspected, and/or Tested

The program monitors aging effects such as loss of material, cracking, and loss of mechanical closure integrity. The specific parameters monitored or inspected are discussed in the three credited programs and include inspection of high yield strength ($\geq$150 ksi) bolting for NSSS component supports for cracking, pressure retaining bolted joints for signs of leakage, and structural bolting for signs of aging degradation.

This element is consistent with NUREG-1801, Section XI.M18, “Bolting Integrity.”

Detection of Aging Effects

The Palisades Bolting Integrity Program includes a variety of inspection and testing activities that are designed to detect degradation due to aging effects prior to loss of intended function.

Examination and inspection of ASME Section XI Class 1, 2, and MC bolting is performed as specified in Tables IWB-2500-1, IWC-2500-1, and IWE-2500-1 with guidance from the recommendations of EPRI NP-5769. The tables specify the extent of the examination, schedule, and methodology. Information on exemptions from examination requirements are described in the ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program.

The program uses two types of examination: visual, and volumetric. Visual VT-1 examination detects discontinuities and imperfections such as cracks, erosion, wear, corrosion, or physical damage on the surface of the bolting. Visual VT-2 examination detects evidence of leakage from pressure retaining components as a result of bolting degradation. Volumetric examinations detects the presence of discontinuities or flaws throughout the volume of the bolting. This program establishes an augmented inspection requirement for all high strength bolting (Actual Yield Strength $\geq$150 ksi.) used in NSSS component supports by applying the requirements of Examination Category B-G-1.

Structural bolting and fasteners both inside and outside containment are inspected by visual examination in accordance with the Structural Monitoring Program.

This element is consistent with NUREG-1801, Section XI.M18, “Bolting Integrity.”

Monitoring and Trending

Examination schedules contained in the Palisades ISI programs meet the requirements of ASME Section XI, Subsections IWB, IWC, and IWE for Class 1, 2, and MC pressure retaining bolting. Examination schedules also meet the requirements of ASME Section XI, Code Case N491-2 for Class 1, 2, and 3 component support bolting. If bolting for pressure retaining components not covered by ASME Section XI is reported to be leaking, the bolting will be addressed in accordance with the System Monitoring Program and the Palisades Action Request Process. Corrective Actions may include monitoring and trending of leakage on a daily, weekly, or biweekly basis.

This element is consistent with NUREG-1801, Section XI.M18, “Bolting Integrity.”

Acceptance Criteria

Indications are evaluated in accordance with IWB-3000, IWC-3000, and IWE-3000 for Class 1, 2, and MC pressure retaining bolting, respectively. Class

1, 2, 3, and MC component support bolting is evaluated in accordance with the acceptance standards of ASME Code Case N491-2, Section 3410. All other bolting is evaluated in accordance with the requirements of the applicable monitoring program identified in the Program Description above.

Indications of cracking in component support bolting, or leaks at bolted joints of pressure retaining components are evaluated and repaired if the degradation may cause a loss of intended function or cause adverse effects, such as corrosion or contamination.

NUREG-1801 states that “immediate” repairs be done for major leaks that may cause corrosion or contamination. The Palisades Bolting Integrity Program makes no distinction regarding “immediate” repairs, but instead relies upon the plant inservice inspection program and corrective action process to evaluate, prioritize and schedule repairs. NUREG-1801 states that, “As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.” Therefore, Palisades meets the intent of this NUREG-1801 element.

This element is consistent with NUREG-1801, Section XI.M18, “Bolting Integrity.”

Corrective Actions, Confirmation Process, Administrative Controls

These elements are consistent with the corresponding NUREG-1801 aging management program elements. See Section B1.2 for further discussion.

Operating Experience

Palisades has a comprehensive Operating Experience Program (OEP) that monitors industry issues/events and assesses these for applicability to its own operations. In addition, the Palisades Corrective Action Program (CAP) is used to track, trend and evaluate plant issues/events. Those issues and events, whether external or plant specific, that are potentially significant to the Bolting Integrity Program at Palisades are evaluated. The Bolting Integrity Program is augmented, as appropriate, if these evaluations show that program changes will enhance program effectiveness.

Using the OEP and CAP to focus on industry and plant operating experience ensures that Bolting Integrity Program issues are addressed in a timely manner and that age related deterioration of SSC within the scope of the Bolting Integrity Program will be effectively managed throughout the license renewal period with the guidance and recommendation from EPRI NP-5769 and TR-104213.

A review of the industry operating experience related to the Bolting Integrity Program revealed numerous instances where degradation of bolting has occurred. In completing our review we looked at related issues which included degradation of threaded fasteners due to stress corrosion cracking and fatigue loading, plus stress corrosion cracking in high strength bolts used for NSSS component supports.

A review of the plant specific operating experience revealed six (6) instances where the ISI program has been instrumental in discovering bolting degradation.

Degradation was discovered in the following items:

- Piping Flange Bolts (1)
- Pump Studs (2)
- Tank Flange Bolts (1)
- Pipe Support Bolting (1)
- ESS Equipment Bolting (1)

The Bolting Integrity Program has been effective in identifying bolting degradation in a timely manner, thus ensuring that age related degradation of bolting will be effectively managed throughout the license renewal period.

This element is consistent with NUREG-1801, Section XI.M18, “Bolting Integrity.”

Conclusion

The Bolting Integrity Program is an existing program that uses as its bases, various industry and NRC standards. This program is consistent with NUREG-1801, Section XI.M18, “Bolting Integrity.”

Reports of recent internal and external assessments of the program have been reviewed. These reports demonstrate that the program has effectively identified and dispositioned issues that could have led to degraded conditions. It is concluded that the Bolting Integrity Program has been effective in maintaining the intended functions of the affected long-lived, passive SSCs.

The continued implementation of the Bolting Integrity Program provides reasonable assurance that aging effects will be managed such that SSCs within the scope of this program will continue to perform their intended functions consistent with the current licensing bases for the period of extended operation.

B2.1.4 Boric Acid Corrosion Program

Program Description

The Palisades Boric Acid Corrosion Program is an existing program that monitors component degradation due to boric acid leakage through the performance of periodic inspections. It implements the recommendations of NRC Generic Letter 88-05. The program requires periodic visual inspection of all systems within the scope of license renewal that contain borated water for evidence of leakage, accumulations of dried boric acid, or boric acid wastage. The program also provides for visual inspections and early discovery of borated water leaks such that structures, electrical and mechanical components that may be contacted by leaking borated water will not be adversely affected such that their intended functions are impaired.

Specifically, the Boric Acid Corrosion Program includes provisions for:

a. Identification of components exhibiting boric acid accumulations or leakage
b. Evaluation of the acceptability for continued service of components exhibiting boric acid accumulations or leakage
c. Trending and tracking of previously identified leaks or boric acid accumulations
d. Corrective actions per plant technical specifications, FSAR and administrative procedures

As part of the Boric Acid Corrosion Program, Palisades monitors operating experience relating to boric acid leaks and takes appropriate corrective actions. Thus, by conducting visual inspections, locating the source of the leaks when they are discovered, performing engineering evaluations, and reviewing internal and external operating experience, the program ensures that SSC within the scope of license renewal will continue to perform their intended functions.

NUREG-1801 Consistency

The Boric Acid Corrosion Program is consistent with NUREG-1801, Section XI.M10, “Boric Acid Corrosion.”

Exceptions to NUREG-1801

None.

Enhancements

Three enhancements are planned to bring the Boric Acid Corrosion Program into conformance with the NUREG-1801 program elements. The enhancements are:

Detection of Aging Effects: Revise applicable plant procedures to include criteria for observing susceptible SSC, within the scope of license renewal, for boric acid leakage and degradation, during system walkdown inspections.

Acceptance Criteria: Revise applicable plant procedure(s) to include explicit acceptance criteria for boric acid inspections.

Detection of Aging Effects: Revise applicable plant procedures to include inspection of structural steel and non-ASME component supports for evidence of boric acid residue and boric acid wastage/corrosion on a periodic frequency.

Note that the element descriptions describe the program as it will exist after the identified enhancements have been implemented. Enhancements are scheduled for implementation prior to the period of extended operation.

Aging Management Program Elements

The key elements of the Boric Acid Corrosion Program are described below. The results of an evaluation of each key element against the corresponding element of NUREG-1801, Section XI.M10, “Boric Acid Corrosion,” are also provided.

Scope of Program

The Boric Acid Corrosion Program includes structures or components, and electrical connectors within the scope of license renewal, which borated water may leak from or on. The program adheres to NRC Generic Letter 88-05. It includes (a) identification of the leakage source, (b) evaluations of the acceptability for continued service of components exhibiting boric acid accumulations or leakage (c) trending of boric acid leaks using the Palisades corrective action system and work request, work order system, and (d) corrective actions in accordance with Plant Technical Specifications, FSAR and administrative procedures. This will ensure that boric acid corrosion does not lead to degradation of the leakage source or adjacent structures or components, which could cause the loss of intended function of the structures or components.

This element is consistent with NUREG-1801, Section XI.M10, “Boric Acid Corrosion.”

Preventive Actions

Preventive actions include improving maintenance practices such as revising the valve packing program to improve packing design and techniques, performance of periodic walkdowns to identify those components that may require corrective maintenance, and monitoring of locations where potential leakage could occur. Timely repair of detected leakage prevents or mitigates boric acid corrosion, and is accomplished through the corrective action process. Electrical connectors are inspected on a periodic frequency for signs of boric acid residue and degradation per plant procedures.

Borated water spills or pooled water on concrete floors can cause corrosion of structural steel, rebar, and equipment floor mounted supports. These are identified in a timely manner through frequent operator rounds, and various building walkdowns/inspections. Any leaks that are identified are evaluated and dispositioned in accordance with plant procedures. Borated water spills and pooled water on concrete floors is cleaned up when identified, in accordance with the standards set forth in plant procedures, to prevent degradation of structural components.

This program credits the System Monitoring Program for periodic walkdowns and inspections for signs of boric acid leakage, residue or degradation.

This element is consistent with NUREG-1801, Section XI.M10, “Boric Acid Corrosion.”

Parameters Monitored, Inspected, and/or Tested

Visual inspections are conducted to monitor the effects of boric acid corrosion on the intended function(s) of an affected structure or component. Borated water leakage results in deposit of white boric acid crystals and presence of moisture that can be observed by visual inspections during system walkdowns.

This program credits the System Monitoring Program for periodic walkdowns and inspections for signs of boric acid leakage, residue or degradation.

This element is consistent with NUREG-1801, Section XI.M10, “Boric Acid Corrosion.”

Detection of Aging Effects

The Palisades Boric Acid Corrosion Program includes visual inspections performed in conjunction with ISI Section XI pressure testing activities that are designed to detect degradation due to aging effects prior to loss of intended function. Degradation of components due to boric acid corrosion cannot occur without borated water leakage. Visual inspections and primary coolant pressure boundary leakage tests are conducted to identify necessary repairs and minimize the potential of a leak not being discovered and developing into a larger leak. Guidelines for detecting small leaks by visual inspections are utilized, and engineering evaluations are conducted when leaks are detected. Post maintenance inspections are conducted to verify that repairs were properly performed. Therefore, the Boric Acid Corrosion Program, which follows the guidelines of NRC GL 88-05, will assure the detection of leakage before a loss of intended function(s).

Electrical connectors are inspected on a periodic frequency for signs of boric acid residue and degradation per plant procedure.

This program also credits the System Monitoring Program for the visual inspection of other SSC outside the primary coolant pressure boundary that may be subject to the degrading effects of any borated water leakage.

This element is consistent with NUREG-1801, Section XI.M10, “Boric Acid Corrosion.”

Monitoring and Trending

This program relies on visual inspections conducted both during normal plant operation and when the plant is shutdown for refueling. The program follows the guidelines in NRC GL 88-05 and provides for timely detection of leakage by observance of boric acid crystal deposits during plant walkdowns and maintenance.

This element is consistent with NUREG-1801, Section XI.M10, “Boric Acid Corrosion.”

Acceptance Criteria

Plant procedures establish acceptance criteria and require corrective action or further evaluation if any leakage or boric acid residue is noted.

This element is consistent with NUREG-1801, Section XI.M10, “Boric Acid Corrosion.”

Corrective Actions, Confirmation Process, Administrative Controls

These elements are consistent with the corresponding NUREG-1801 aging management program elements. See Section B1.2 for further discussion.

Operating Experience

Palisades has a comprehensive Operating Experience Program (OEP) that monitors industry issues/events and assesses these for applicability to its own operations. In addition, the Palisades Corrective Action Program (CAP) is used to track, trend and evaluate plant issues/events. Those issues and events, whether external or plant specific, that are potentially significant to the Boric Acid Corrosion Program at Palisades are evaluated. The Boric Acid Corrosion Program is augmented, as appropriate, if these evaluations show that program changes will enhance program effectiveness.

Using the OEP and CAP to focus on industry and plant operating experience ensures that Boric Acid Corrosion Program issues are addressed in a timely manner and that age related deterioration of SSC within the scope of the Boric Acid Corrosion Program will be effectively managed throughout the license renewal period.

A review of industry operating experience associated with the Boric Acid Corrosion Program and aging revealed issues related to:

- Boric acid wastage of reactor coolant system piping and nozzles
- Boric acid corrosion of reactor vessel head and closure studs from leaking borated water
- Failure of valve packing gland bolts due to boric acid wastage
- Failure of valve body to bonnet studs/nuts due to boric acid wastage
- Boric acid wastage of reactor coolant pump closure flange studs
- Boric acid corrosion of steam generator manway closure studs
- Boric acid corrosion of high pressure safety injection pump casing

These issues were addressed in various NRC and industry communications, which have been incorporated into the program as applicable.

A review of plant specific operating experience related to the Boric Acid Corrosion Program and aging revealed that the following issues had been addressed:

- Boric acid leaks in the containment spray header in containment at flanges with carbon steel bolting and a threaded spray nozzle connection
- Boric acid wastage of primary coolant pump studs
- Boric acid wastage of manual valve body-to-bonnet bolts

- Corrosion of flanges for primary coolant pump component cooling water connections due to external boric acid leakage

Recent assessments have shown that the Boric Acid Corrosion Program provides reasonable assurance that aging effects are being managed for Boric Acid Corrosion Program SSCs. This has been demonstrated through NRC inspection reports, audits, self-assessments, and the Corrective Action Program.

This element is consistent with NUREG-1801, Section XI.M10, “Boric Acid Corrosion.”

Conclusion

The Boric Acid Corrosion Program is an existing program that uses as its bases, various industry and NRC standards, including GL 88-05. This program is consistent with NUREG-1801, Section XI.M10, “Boric Acid Corrosion.”

Reports of recent internal and external assessments of the program have been reviewed. These reports demonstrate that the program has effectively identified and dispositioned issues that could have led to degraded conditions. It is concluded that the Boric Acid Corrosion Program has been effective in maintaining the intended functions of the affected long-lived, passive SSCs.

The continued implementation of the Boric Acid Corrosion Program provides reasonable assurance that aging effects will be managed such that SSCs within the scope of this program will continue to perform their intended functions consistent with the current licensing bases for the period of extended operation.

B2.1.5 Buried Services Corrosion Monitoring Program

Program Description

The Buried Services Corrosion Monitoring Program is a new program that manages aging effects on the external surfaces of carbon steel, low-alloy steel, and stainless steel components that are buried in soil or sand. This program includes (a) visual inspections of external surfaces of buried components for evidence of coating damage and substrate degradation to manage the effects of aging, (b) visual inspection of the external surfaces of buried stainless steel components for evidence of crevice corrosion, pitting, and MIC. The periodicity of these inspections for carbon, low-alloy, and stainless steel will be based on opportunities for inspection such as scheduled maintenance work.

Age-related degradation of buried components susceptible to selective leaching is managed by the One-Time Inspection Program.

NUREG-1801 Consistency

The Buried Services Corrosion Monitoring Program is consistent with NUREG1801, Section XI.M34, “Buried Piping and Tanks Inspection.”

Exceptions to NUREG-1801

None.

Enhancements

A Buried Services Corrosion Monitoring Program will be developed and implemented. Features of the program will include development and implementation of procedures for inspection of selected buried SSCs for corrosion, pitting and MIC. The periodicity of these inspections will be based on opportunities for inspection such as scheduled excavation and maintenance work.

Note that the element descriptions describe the program as it will exist after the program has been implemented. The program is scheduled to be implemented prior to the period of extended operation.

Aging Management Program Elements

The key elements of the Buried Services Corrosion Monitoring Program are described below. The results of an evaluation of each key element against NUREG-1801, Section XI.M34, “Buried Piping and Tank Inspection,” are also provided below.

Scope of Program

The scope of this program addresses buried components that are within the scope of License Renewal. The program relies on preventive measures, such as coating and wrapping, and periodic inspection for loss of material caused by corrosion of the external surface of buried carbon, low alloy, and stainless steel piping and components. Palisades has no buried tanks in sand or soil. Loss of materials in these components, which may be exposed to a soil or sand environment, is caused by general, pitting, crevice corrosion, and microbiologically influenced corrosion (MIC). The periodicity of inspections for carbon, low-alloy, and stainless steel to detect degradation due to these aging mechanisms will be based on opportunities for inspection such as scheduled maintenance work.

This element is consistent with NUREG-1801, Section XI.M34, “Buried Piping and Tanks Inspection.”

Preventive Actions

Buried components, except for stainless steel, are coated per industry practice prior to installation in order to protect the component outer surfaces from corrosion. The buried pipe specifications for Palisades Nuclear Plant required coating and wrapping in accordance with AWWA C203. The coatings include a primer, coal tar enamel, asbestos mat, coal tar enamel, paper wrap layered system that finishes at approximately 3/16” thickness.

This element is consistent with NUREG-1801, Section XI.M34, “Buried Piping and Tanks Inspection.”

Parameters Monitored, Inspected, and/or Tested

The program monitors parameters such as coating and wrapping integrity that are directly related to loss of material due to corrosion on the external surfaces of buried carbon steel, and low alloy steel components. Coatings and wrappings are visually inspected when access becomes available. Any evidence of damage to the coating or wrapping, such as coating perforation, holidays, or other damage will cause the protected components to be inspected for evidence of loss of material. If no evidence of damage to the coating or wrapping is detected, then the coating or wrapping will not be removed for further inspection. Buried stainless steel components are typically not coated, and the outer surface is visually inspected for evidence of crevice corrosion, pitting, and MIC.

This element is consistent with NUREG-1801, Section XI.M34, “Buried Piping and Tanks Inspection.”

Detection of Aging Effects

The Palisades Buried Services Corrosion Monitoring Program includes inspection activities that are designed to detect degradation due to aging effects prior to loss of intended function. Visual inspections of buried carbon, low-alloy, and stainless steel components will be performed based on plant operating experience and when components are excavated for maintenance or any other reason.

This element is consistent with NUREG-1801, Section XI.M34, “Buried Piping and Tanks Inspection.”

Monitoring and Trending

The results of previous inspections will be evaluated, and used to assess the condition of the external surfaces of other buried carbon, low-alloy, and stainless steel components, and to identify susceptible locations that warrant further inspections.

This element is consistent with NUREG-1801, Section XI.M34, “Buried Piping and Tanks Inspection.”

Acceptance Criteria

Any coating and wrapping degradations, or components identified with significant corrosion, will be documented and evaluated under the corrective action program, which includes provisions for a root cause analysis, if appropriate.

This element is consistent with NUREG-1801, Section XI.M34, “Buried Piping and Tanks Inspection.”

Corrective Actions, Confirmation Process, Administrative Controls

These elements are consistent with the corresponding NUREG-1801 aging management program elements. See Section B1.2 for further discussion.

Operating Experience

Palisades has a comprehensive Operating Experience Program (OEP) that monitors industry issues/events and assesses these for applicability to its own operations. In addition, the Palisades Corrective Action Program (CAP) is used to track, trend and evaluate plant issues/events. Those issues and events, whether external or plant specific, that are potentially significant to the Buried Services Corrosion Monitoring Program at Palisades are evaluated. The Buried Services Corrosion Monitoring Program is augmented, as appropriate, if these evaluations show that program changes will enhance program effectiveness.

Using the OEP and CAP to focus on industry and plant operating experience ensures that Buried Services Corrosion Monitoring Program issues are addressed in a timely manner and that age related deterioration of SSC's within the scope of the Buried Services Corrosion Monitoring Program will be effectively managed throughout the license renewal period.

A review of industry operating experience applicable to aging of buried services reveals issues related to Diesel fuel line leakage from the absence of required coating leading to corrosion. None of the industry operating issues or instances reflect any new program issues attributed to exterior corrosion to buried services component materials.

A review of plant specific operating experience related to the Buried Services Corrosion Monitoring Program and aging revealed that the following issues have been addressed:

- Through wall leak in buried steam line.
- Generic program deficiencies from internal Engineering Programs audit.
- See the Fire Protection Program for OE related to buried fire main ruptures.

None of the plant operating issues or instances resulted from normal aging, or reflect significant program deficiencies.

This element is consistent with the portions of NUREG-1801, Section XI.M34, “Buried Piping and Tanks Inspection,” that are applicable to SSC's that are buried in soil or sand.

Conclusion

The Buried Services Corrosion Monitoring Program is a new program that uses as its bases, various industry and NRC standards. This program is consistent with NUREG-1801, Section XI.M34, “Buried Piping and Tanks Inspection.”

Implementation of the Buried Services Corrosion Monitoring Program provides reasonable assurance that aging effects will be managed such that SSC's within the scope of this program will continue to perform their intended functions consistent with the current licensing bases for the period of extended operation.

B2.1.6 Closed Cycle Cooling Water Program

Program Description

The Closed Cycle Cooling Water Program is an existing program that manages aging effects in closed cycle cooling water systems that are not subject to significant sources of contamination, in which water chemistry is controlled and heat is not directly rejected to the ultimate heat sink. The program includes (a) maintenance of system corrosion inhibitor concentrations to minimize degradation, and (b) periodic or one-time testing and inspections to assess SSC aging.

The program scope includes activities to manage aging in the Component Cooling Water (CCS) System, Emergency Diesel Generator (EDG) Jacket Cooling Water (Emergency Power System), and Shield Cooling System (SCS).

The program is based on requirements delineated in EPRI TR-107396, “Closed Cooling Water Chemistry Guideline,” and relies on mitigative measures to minimize corrosion through the addition of corrosion inhibitors and maintenance of water chemistry within specified limits.

The program credits the One-Time Inspection Program for the inspection of selected Shield Cooling System and Emergency Diesel Generator system heat exchangers and a representative sample of stagnant portions of the system piping. The inspections will check for fouling and evidence of corrosion or cracking. Nondestructive examinations will be used, if practical and warranted, to verify pipe wall thickness at selected locations where loss of material has been experienced.

NUREG-1801 Consistency

The Closed Cycle Cooling Water Program is consistent with, but includes exceptions to, NUREG-1801, Section XI.M21, “Closed-Cycle Cooling Water System.”

Exceptions to NUREG-1801

Exceptions are taken to the selected NUREG-1801 Program elements listed below. The specific exceptions being taken are also discussed in the corresponding element discussions below. They are repeated here for ease of review.

1. Parameters Monitored, Tested and/or Inspected, Monitoring and Trending, and Acceptance Criteria: NUREG-1801, Section XI.M21.3, states that, “For pumps, the parameters monitored include flow and discharge and suction pressures. For heat exchangers, the parameters monitored include flow, inlet and outlet temperatures, and differential pressure.” Palisades does not credit active flow testing for managing age-related degradation of CCCW components. However, performance of selected heat exchangers is monitored in accordance with the Master Heat Exchanger Testing Plan. The performance and operability testing of selected pumps, including flow, suction and discharge pressure, is monitored in accordance with ASME Section XI, Subsection IWP, Inservice Testing Program. This is generally in accordance with the position taken by Fort Calhoun Station and V.C. Summer Nuclear Station in their license renewal applications, and accepted by the NRC in their respective Safety Evaluation Reports.
2. Detection of Aging Effects, Monitoring and Trending, and Acceptance Criteria: NUREG-1801 states “The extent and schedule of inspections and testing in accordance with EPRI TR-107396, assure detection of corrosion before the loss of intended function on the component. Performance and functional testing in accordance with EPRI TR-107396 ensures acceptable functional testing of the CCCW System or components serviced by the CCCW System.” Palisades does not credit active performance and functional testing for managing age-related

degradation of CCCW components. However, performance of selected heat exchangers is monitored in accordance with the Master Heat Exchanger Testing Plan. The performance and operability testing of selected pumps, including flow, suction and discharge pressure, is monitored in accordance with ASME Section XI, Subsection IWP, Inservice Testing Program. This is generally in accordance with the position taken by Fort Calhoun Station, and V.C. Summer Nuclear Station, and accepted by the NRC in their respective Safety Evaluation Reports.

Enhancements

None.

Aging Management Program Elements

The key elements, which are used in the Closed Cycle Cooling Water Program, are described below. The results of an evaluation of each key element against NUREG-1801, Section XI.M21, “Closed-Cycle Cooling Water System,” are also provided below.

Scope of Program

The Closed Cycle Cooling Water Program pertains to closed systems that are not subject to significant sources of contamination, in which the water chemistry is controlled, monitored, and kept within specified limits, and in which the heat is not directly rejected to the ultimate heat sink. The program is applicable to the Component Cooling Water System, EDG Jacket Cooling Water, and the Shield Cooling System.

This element is consistent with NUREG-1801, Section XI.M21, “Closed-Cycle Cooling Water System.”

Preventive Actions

The program relies on the use of appropriate materials and a water treatment program, including corrosion inhibitors to inhibit general, crevice, and pitting corrosion. Sodium nitrite and tolyltriazole (TTA) are used for carbon steel and copper corrosion control. Applicable chemistry procedures include provisions for monitoring CCCW Systems water chemistry and adding chemicals as necessary to maintain chemistry parameters within limits. These procedures, as well as the administrative limits for corrosion inhibitors sodium nitrite and TTA, are based on the requirements and guidelines in EPRI TR-107396.

This element is consistent with NUREG-1801, Section XI.M21, “Closed-Cycle Cooling Water System.”

Parameters Monitored, Tested, and/or Inspected

Heat exchangers are inspected in accordance with plant procedures. Included in this program for selected heat exchangers is Eddy Current Testing and/or tube side/shell side inspections prior to, and following tube cleaning, an inspection of channels, covers and welds, and an exterior bundle and shell interior inspection, if accessible, by the program or system engineer.

This element is consistent with, but includes an exception to, NUREG-1801, Section XI.M21, “Closed-Cycle Cooling Water System.” The exception is:

- NUREG-1801, Section XI.M21.3 states that, “For pumps, the parameters monitored include flow and discharge and suction pressures. For heat exchangers, the parameters monitored include flow, inlet and outlet temperatures, and differential pressure.” Palisades does not credit active flow testing for managing age-related degradation of CCCW components. However, performance of selected heat exchangers is monitored in accordance with the Master Heat Exchanger Testing Plan. The performance and operability testing of selected pumps, including flow, suction and discharge pressure, is monitored in accordance with ASME Section XI, Subsection IWP, Inservice Testing Program. This is generally in accordance with the position taken by Fort Calhoun Station and V.C. Summer Nuclear Station in their license renewal applications, and accepted by the NRC in their respective Safety Evaluation Reports.

Detection of Aging Effects

The Palisades Closed Cycle Cooling Water Program includes a variety of inspection activities that are designed to detect degradation due to aging effects prior to loss of intended function.

For the Shield Cooling system, Diesel Generator Jacket Water Cooling System, and the Component Cooling System piping inside containment, the One-Time Inspection Program is credited with the detection of corrosion in areas of stagnant flow conditions in the CCCW Systems. Periodic internal inspections of selected heat exchangers will provide indications of fouling and degradation.

This element is consistent with, but includes an exception to, NUREG-1801, Section XI.M21, “Closed-Cycle Cooling Water System.” The exception is:

- NUREG-1801 states “The extent and schedule of inspections and testing in accordance with EPRI TR-107396, assure detection of corrosion before the loss of intended function on the component. Performance and functional testing in accordance with EPRI TR-107396, ensures acceptable functional testing of the CCCW System or components serviced by the CCCW System." Palisades does not credit active performance and functional testing for managing age-related degradation of CCCW components. However, performance of selected heat exchangers is monitored in accordance with the Master Heat Exchanger Testing Plan. The performance and operability testing of selected pumps, including flow, suction and discharge pressure, is monitored in accordance with ASME Section XI, Subsection IWP, Inservice Testing Program. This is generally in accordance with the position taken by Fort Calhoun Station, and V.C. Summer Nuclear Station, and accepted by the NRC in their respective Safety Evaluation Reports.

Monitoring and Trending

Chemistry procedures provide for monitoring and analysis of the following chemistry parameters:

- Nitrite
- PH
- Conductivity
- Iron
- Copper
- Ammonia
- Tolyltriazole (TTA)
- Gamma Scan (CCS only)
- Microbiological Activity (Component Cooling & Emergency Diesel Generator Systems only)

Normal ranges for each parameter are identified, as well as administrative limits, sampling frequency, and corrective actions. Sampling frequencies are based on plant operating conditions and experience.

This element is consistent with, but includes exceptions to, the corresponding element of NUREG-1801, Section XI.M21, "Closed-Cycle Cooling Water System." The exceptions include:

- NUREG-1801, Section XI.M21.3 states that, "For pumps, the parameters monitored include flow and discharge and suction pressures. For heat exchangers, the parameters monitored include flow, inlet and outlet temperatures, and differential pressure." Palisades does not credit active flow testing for managing age-related degradation of CCCW components. However, performance of selected heat exchangers is monitored in accordance with the Master Heat Exchanger Testing Plan. The performance and operability testing of selected pumps, including flow, suction and discharge pressure, is monitored in accordance with ASME Section XI, Subsection IWP, Inservice Testing Program. This is generally in accordance with the position taken by Fort Calhoun Station, and V.C. Summer Nuclear Station, and accepted by the NRC in their respective Safety Evaluation Reports.

- NUREG-1801 states “The extent and schedule of inspections and testing in accordance with EPRI TR-107396, assure detection of corrosion before the loss of intended function on the component. Performance and functional testing in accordance with EPRI TR-107396, ensures acceptable functional testing of the CCCW System or components serviced by the CCCW System.” Palisades does not credit active performance and functional testing for managing age-related degradation of CCCW components. However, performance of selected heat exchangers is monitored in accordance with the Master Heat Exchanger Testing Plan. The performance and operability testing of selected pumps, including flow, suction and discharge pressure, is monitored in accordance with ASME Section XI, Subsection IWP, Inservice Testing Program. This is generally in accordance with the position taken by Fort Calhoun Station, and V.C. Summer Nuclear Station, and accepted by the NRC in their respective Safety Evaluation Reports.

Acceptance Criteria

Nitrite and Tolyltriazole concentrations are maintained within the limits specified in the EPRI water chemistry guidelines for Closed Cycle Cooling Water Systems. System and component performance test results are evaluated in accordance with plant procedures and system design parameters and functions.

This element is consistent with, but includes exceptions to, the corresponding element of NUREG-1801, Section XI.M21, “Closed-Cycle Cooling Water System.” The exceptions include:

- NUREG-1801, Section XI.M21.3 states that, “For pumps, the parameters monitored include flow and discharge and suction pressures. For heat exchangers, the parameters monitored include flow, inlet and outlet temperatures, and differential pressure.” Palisades does not credit active flow testing for managing age-related degradation of CCCW components.

However, performance of selected heat exchangers is monitored in accordance with the Master Heat Exchanger Testing Plan. The performance and operability testing of selected pumps, including flow, suction and discharge pressure, is monitored in accordance with ASME Section XI, Subsection IWP, Inservice Testing Program. This is generally in accordance with the position taken by Fort Calhoun Station, and V.C. Summer Nuclear Station, and accepted by the NRC in their respective Safety Evaluation Reports.

- NUREG-1801 states “The extent and schedule of inspections and testing in accordance with EPRI TR-107396, assure detection of corrosion before the loss of intended function on the component. Performance and functional testing in accordance with EPRI TR-107396, ensures acceptable functional testing of the CCCW System or components serviced by the CCCW System.” Palisades does not credit active performance and functional testing for managing age-related degradation of CCCW components. However, performance of selected heat exchangers is monitored in accordance with the Master Heat Exchanger Testing Plan. The performance and operability testing of selected pumps, including flow, suction and discharge pressure, is monitored in accordance with ASME Section XI, Subsection IWP, Inservice Testing Program. This is generally in accordance with the position taken by Fort Calhoun Station, and V.C. Summer Nuclear Station, and accepted by the NRC in their respective Safety Evaluation Reports.

Corrective Actions, Confirmation Process, Administrative Controls

These elements are consistent with the corresponding NUREG-1801 aging management program elements. See Section B1.2 for further discussion.

Operating Experience

Palisades has a comprehensive Operating Experience Program (OEP) that monitors industry issues/events and assesses them for applicability to its own operations. In addition, the Palisades Corrective Action Program (CAP) is used to track, trend and evaluate plant issues/events. Those issues and events, whether external or plant specific, that are potentially significant to the Closed Cycle Cooling Water Program at Palisades are evaluated. The Closed Cycle Cooling Water Program is augmented, as appropriate, if these evaluations show that program changes are needed to enhance program effectiveness.

Using the OEP and CAP to focus on industry and plant operating experience ensures that Closed Cycle Cooling Water Program issues are addressed in a timely manner and that age related deterioration of SSC within the scope of the Closed Cycle Cooling Water Program will be effectively managed throughout the license renewal period.

A review of industry operating experience associated with the Closed Cycle Cooling Water Program and aging reveals issues and instances related to:

- SCC in reactor coolant pump oil cooler discharge piping.
- Corroded solder connections in diesel lube oil cooler due to inadequate corrosion inhibitor
- Inoperable check valves (stuck open) due to corrosion product buildup
- Cracks in Component Cooling Water piping
- Fouling of diesel cooling water heat exchangers

Various related NRC and/or industry generic communications have been issued, and, in turn, have been incorporated into the program as applicable.

A review of plant specific operating experience related to the Closed Cycle Cooling Water Program and aging revealed that the following issues have been addressed:

- Tube blockage and fouling in Component Cooling Heat Exchanger
- Fuel Pool Heat Exchanger tube breakage due to high Component Cooling Water flow
- Through wall flaw in Spent Fuel Pool Cooling pipe

The Palisades Closed Cycle Cooling Water Program has demonstrated that it provides reasonable assurance that aging effects are being managed for Closed Cycle Cooling Water Program SSCs. Additionally, this has been demonstrated through NRC inspection reports, audits, self-assessments, and the Corrective Action Program.

This element is consistent with NUREG-1801, Section XI.M21, “Closed-Cycle Cooling Water System.”

Conclusion

The Closed Cycle Cooling Water Program is an existing program that uses as its bases, various industry and NRC standards. This program is consistent with, but includes exceptions to, NUREG-1801, Section XI.M21, “Closed-Cycle Cooling Water System.” A summary of each exception is provided in the discussion of the affected program element above.

Reports of recent internal and external assessments of the program have been reviewed. These reports demonstrate that the program has effectively identified and dispositioned issues that could have led to degraded conditions. It is concluded that the Closed Cycle Cooling Water Program has been effective in maintaining the intended functions of the affected long-lived, passive SSCs.

The continued implementation of the Closed Cycle Cooling Water Program provides reasonable assurance that aging effects will be managed such that SSCs within the scope of this program will continue to perform their intended functions consistent with the current licensing bases for the period of extended operation.

B2.1.7 Containment Inservice Inspection Program

Program Description

The Containment Inservice Inspection (ISI) Program is an existing program that is designed to ensure that containment shell concrete, the post-tensioning system and steel pressure retaining elements continue to provide an acceptable level of structural integrity. In addition, it is designed to ensure that the liner (with associated moisture barriers), other leakage limiting steel barriers and pressure retaining bolted connections have not degraded. This program does not demonstrate actual containment leak tightness; that is done under the Containment Leakage Testing Program.

This Program incorporates, with some differences in code editions, elements of several applicable programs identified in NUREG-1801. These are as follows:

- XI.S1 - ASME Section XI, Sub-Section IWE
- XI.S2 - ASME Section XI, Sub-Section IWL
- X.S1 - Concrete Containment Tendon Prestress

NUREG-1801 Consistency

The Containment Inservice Inspection Program is consistent with NUREG-1801, Section XI.S1, “ASME Section XI, Subsection IWE,” Section XI.S2, “ASME Section XI, Subsection IWL,” and Section X.S1, “Concrete Containment Tendon Prestress.”

Exceptions to NUREG-1801

None.

Enhancements

XI.S1 ASME Section XI, Subsection IWE

None

XI.S2 ASME Section XI, Subsection IWL

None

X.S1 Concrete Containment Tendon Prestress

None

Aging Management Program Elements

The key elements, which are used in the Containment Inservice Inspection Program, are described below. The results of an evaluation of each key element against NUREG-1801, Section XI.S1, “ASME Section XI, Subsection IWE,” Section XI.S2, “ASME Section XI, Subsection IWL,” and Section X.S1, “Concrete Containment Tendon Prestress” are also provided below.

Scope of Program

Scope of Program - XI.S1 ASME Section XI, Subsection IWE

This part of the Palisades Containment ISI Program covers inspection of the steel pressure boundary and corresponds to the NUREG-1801 ASME Section XI, Sub-Section IWE, Program. This Program specifies visual examination of the entire accessible steel pressure boundary to uncover age related and other conditions that represent damage or deterioration. The steel pressure boundary includes: The liner with adjacent parts of integral structural attachments; mechanical penetration sleeves and transition pieces up to the Class 2 piping boundaries; electrical & spare penetration sleeves with closures; the fuel transfer tube & closure flange; the equipment hatch; air locks; and, pressure retaining bolting. The accessible part of the boundary is any surface that is not in contact with concrete or otherwise obstructed from view by plant structures and equipment.

The Palisades program follows the 1998 Code Edition (no addenda), which does not include a requirement to examine seals and gaskets. Seals and gaskets are almost entirely hidden from view by the mating flanges and cannot be meaningfully examined unless the flanged closure is opened or disassembled. Disassembly is not required by the 92 / 92 Code. As the continuing quality of seals and gaskets is verified by periodic leakage testing, there is no need to require such examinations. This is not considered an exception to NUREG-1801.

Scope of Program - XI.S2 ASME Section XI, Subsection IWL

This part of the Palisades Containment ISI Program covers inspection of the post-tensioned concrete containment structure and corresponds to the NUREG-1801 ASME Section XI, Sub-Section IWL Program.

The Containment ISI Program specifies visual examination of the entire accessible surface of the concrete structure, examination of all accessible tendon end anchorage areas and examination of and tests on a small sample of pre-stressing tendons. The accessible part of the concrete surface is that part not obstructed from view by foundation material, backfill, the liner and permanent plant structures and equipment.

Scope of Program - X.S1 Concrete Containment Tendon Prestress

The program addresses the assessment of Palisades' containment tendon wire prestressing forces. During the scheduled surveillances, the tendon prestress force is measured for a random sample of each tendon group. One tendon in each group is designated as the common tendon. The prestress force in the common tendon is measured during each surveillance and is used to establish the trend of prestress loss for the group. The trend of prestress force for each group is compared to the minimum required value to ensure that the prestress force will not fall below the minimum required value prior to the next scheduled inspection.

This element is consistent with NUREG-1801, Section XI.S1, “ASME Section XI, Subsection IWE,” Section XI.S2, “ASME Section XI, Subsection IWL,” and Section X.S1, “Concrete Containment Tendon Prestress.”

Preventive Actions

The Containment Inservice Inspection Program is a condition monitoring program and there are no preventive actions associated with this program. As long as the tendon wire prestressing values are found to be within the acceptance criteria no actions are required. Maintaining the prestress above the minimum required value will ensure that the structural and functional adequacy of the containment are maintained.

This element is consistent with NUREG-1801, Section XI.S1, "ASME Section XI, Subsection IWE," Section XI.S2, "ASME Section XI, Subsection IWL," and Section X.S1, "Concrete Containment Tendon Prestress."

Parameters Monitored, Inspected, and/or Tested

Parameters Monitored, Inspected and/or Tested - XI.S1 ASME Section XI, Subsection IWE

The Palisades Containment ISI Program specifies a general visual examination to find evidence of age related and other conditions that represent damage to, or deterioration of, the steel pressure boundary and moisture barrier at the interface between the inner liner and the containment fill concrete.

General and Detailed Visual Examination, as defined in the 1998 Code (no addenda), are further defined in the Palisades program to correspond to VT-3 and VT-1 examinations as defined in the 92 / 92 Code. Therefore, the examination techniques used at the Palisades Plant are the same as those identified in NUREG-1801.

The examination categories identified in the Palisades program are those defined in the 98 Code (no addenda). The Palisades program categories effectively encompass NUREG-1801 categories. The differences between the examination categories listed in NUREG-1801 and those included in the Palisades program are also discussed below, and are a result of code editions called out in NUREG-1801 vs. Palisades' approved program. These are not considered to be exceptions to NUREG-1801.

- Pressure retaining welds and pressure retaining dissimilar metal welds receive a General Visual Examination and not a VT-1 examination. This approach is consistent with 10 CFR 50.55a which (as noted in NUREG-1801) states that the weld examinations specified in the 92 / 92 Code are optional.
- Neither the 98 Code (no addenda) nor the Palisades program requires seal and gasket examinations. As explained above, elimination of seal & gasket examination requirement does not impact the assessment of containment leak-tight integrity.
- Neither the 98 Code (no addenda) nor the Palisades program requires bolt torque tests. Elimination of torque tests does not impact the assessment of containment leak tight integrity.

10 CFR 50, Appendix J tests are covered by the Containment Leakage Testing Program and are not specified in the Containment ISI Program except as post-repair activities.

The containment steel pressure boundary serves primarily as a leakage limiting barrier. That part of the boundary not backed by concrete also functions as a structural pressure retaining boundary. The inspections and tests (if required) performed under the Containment ISI Program are designed to detect age related (and other) deterioration that could reduce the leak tightness and structural strength of this boundary.

<u>Parameters Monitored, Inspected and/or Tested - XI.S2 ASME Section XI, Subsection IWL</u>

The Palisades Containment ISI Program specifies a General Visual Examination of 100% of the accessible concrete surface, a Detailed Visual Examination of all accessible tendon end anchorage areas and a Detailed Visual Examination and tests on a small sample of randomly selected tendons. These examinations and tests are performed to find evidence of damage to, or deterioration of, the concrete structure and the post-tensioning system.

<u>Parameters Monitored, Inspected, and/or Tested - X.S1 Concrete Containment Tendon Prestress</u>

The parameters monitored are the containment tendon wire prestressing forces in accordance with the requirements specified in Subsection IWL of Section XI of the ASME Code.

This element is consistent with NUREG-1801, Section XI.S1, “ASME Section XI, Subsection IWE,” Section XI.S2, “ASME Section XI, Subsection IWL,” and Section X.S1, “Concrete Containment Tendon Prestress.”

Detection of Aging Effects

The Palisades Containment Inservice Inspection Program includes a variety of examination and testing activities that are designed to detect degradation due to aging effects prior to loss of intended function.

<u>Detection of Aging Effects - XI.S1 ASME Section XI, Subsection IWE</u>

The Palisades Plant follows Inspection Program B which stipulates consecutive 10 year ISI intervals, each divided (per Table IWE-2412-1) into 3 successive periods having nominal durations of 3, 4 and 3 years, respectively. The Palisades program requires a visual examination of 100% of the accessible surfaces in each category during each inspection interval. These frequent, comprehensive examinations provide a high degree of assurance that age related deterioration will be detected long before it has a significant impact on either the structural or leak tight integrity of the containment. Areas that exhibit damage or deterioration must meet the requirements of IWE-3122, Acceptance.

Detection of Aging Effects - XI.S2 ASME Section XI, Subsection IWL

Palisades performs containment concrete, tendon end anchorage and post-tensioning system inspections at 5 year intervals. The entire accessible concrete surface and all accessible tendon end anchorage areas are examined during each inspection.

Detection of Aging Effects - X.S1 Concrete Containment Tendon Prestress

The loss of tendon wire prestressing forces is detected by performance of the tendon inspections and analyses conducted per plant procedures and surveillance tests.

This element is consistent with NUREG-1801, Section XI.S1, “ASME Section XI, Subsection IWE,” Section XI.S2, “ASME Section XI, Subsection IWL,” and Section X.S1, “Concrete Containment Tendon Prestress.”

Monitoring and Trending

Monitoring and Trending - XI.S1 ASME Section XI, Subsection IWE

All accessible surfaces are examined and monitored on a regular schedule per requirements of IWE - 2500.

The Palisades program requires augmented examination of areas that exhibit signs of significant damage or deterioration. Conditions in these areas are trended and corrective actions are scheduled as required to ensure that an acceptable level of containment integrity is maintained. Augmented examinations are ended, in accordance with 98 Code (no addenda) Table IWE-2500-1 / Category E-C, if there is no change in the monitored condition over the course of one inspection period. NUREG-1801 references the 92 / 92 Code requirement, which requires that augmented examinations be continued over three consecutive periods.

The 98 Code (no addenda) requirement is reasonably conservative since any condition that remains stable over a three to four year period is not experiencing a significant rate of ongoing deterioration. Conditions that remain stable over one inspection period are either the result of a manufacturing defect, a construction oversight, a singular incident or an earlier aggressive environment that has been corrected. Therefore, if there is no change in an observed condition over one inspection period, no benefit is gained by continuing augmented examination. This is not considered to be an exception to NUREG-1801.

Trending of observed conditions as prescribed in the Containment ISI Program ensures that age related deterioration of the steel pressure boundary is identified and controlled in timely manner throughout the license renewal period.

Monitoring and Trending - XI.S2 ASME Section XI, Subsection IWL

The entire accessible concrete surface, all accessible tendon end anchorage areas, and selected tendons meeting the requirements of IWL 2520 are examined and, in the case of tendons, tested, to detect damage or degraded conditions. Unacceptable conditions are documented on action requests and are subject to engineering evaluation. The evaluation may require more frequent and detailed monitoring to determine if the observed condition is deteriorating with time. Trending of action requests is in accordance with the corrective action program.

Monitoring and Trending - X.S1 Concrete Containment Tendon Prestress

The tendon wire prestressing forces are compared to the predicted lower limit and upper bound lines that were developed for the period of extended operation in engineering analyses. Prestress force trends are also evaluated to ensure that the prestress force will not fall below the minimum required value prior to the next scheduled inspection. If the trend is not acceptable, an engineering evaluation is performed to determine a proper course of action (typically, immediate re-tensioning or reducing the length of time to the next inspection).

This element is consistent with NUREG-1801, Section XI.S1, “ASME Section XI, Subsection IWE,” Section XI.S2, “ASME Section XI, Subsection IWL,” and Section X.S1, “Concrete Containment Tendon Prestress.”

Acceptance Criteria

Acceptance Criteria - XI.S1 ASME Section XI, Subsection IWE

The Palisades program uses the acceptance standards of IWE 3000. Any item found not meeting the acceptance standards of IWE - 3000 is subject to an engineering evaluation. The evaluation must state and justify one of the following conclusions:

- The observed condition is acceptable as is and no further action is required.
- The observed condition is acceptable as is but has the potential for further deterioration and requires augmented examination.
- The observed condition is not acceptable and repair or other remedial action is required.

If the observed condition includes a reduction in metal thickness in excess of 10% of the nominal thickness, an evaluation is performed to determine if the reduced section satisfies design criteria.

If augmented examination is required, the evaluation report must specify the type of examination (Detailed Visual, surface NDE, volumetric NDE or other) and the frequency (once per period at a minimum) at which this is to be conducted.

If repair or other remedial action is required, the evaluation report must specify the nature of the action and the deadline for completion.

Acceptance Criteria - XI.S2 ASME Section XI, Subsection IWL

The Palisades program uses the acceptance criteria of IWL - 3000. Any item found not meeting the acceptance standards of IWL - 3000 is subject to an engineering evaluation. The acceptance criteria for concrete surfaces, corrosion protection medium leakage, or end-cap deformation are qualitative, and rely on the determination of the Responsible Engineer using the guidance of IWL 2510.

If augmented examination is required, the evaluation report must specify the type of examination (Detailed Visual, subsurface by concrete removal, subsurface by volumetric NDE or other) and the frequency (once per period at a minimum) at which this is to be conducted. The report must also identify and discuss the cause of the unacceptable condition.

Acceptance Criteria - X.S1 Concrete Containment Tendon Prestress

An evaluation is required if the acceptance criteria of ASME Section XI, Subsection IWL is not met, or if the trend indicates that tendon force(s) would be less than the minimum required value before the next scheduled inspection, as required by 10 CFR 50.55a(b)(2)(viii)(B).

This element is consistent with NUREG-1801, Section XI.S1, “ASME Section XI, Subsection IWE,” Section XI.S2, “ASME Section XI, Subsection IWL,” and Section X.S1, “Concrete Containment Tendon Prestress.”

Corrective Actions, Confirmation Process, Administrative Controls

These elements are consistent with the corresponding NUREG-1801 aging management program elements. See Section B1.2 for further discussion.

Operating Experience

Palisades has a comprehensive Operating Experience Program (OEP) that monitors industry issues/events and assesses these for applicability to its own operations. In addition, the Palisades Corrective Action Program (CAP) is used to track, trend and evaluate plant issues/events. Those issues and events, whether external or plant specific, that are potentially significant to the Containment Inservice Inspection Program at Palisades are evaluated. The Containment Inservice Inspection Program is augmented, as appropriate, if these evaluations show that program changes will enhance program effectiveness.

Using the OEP and CAP to focus on industry and plant operating experience ensures that Containment Inservice Inspection Program issues are addressed in a timely manner and that age related deterioration within the scope of the Containment Inservice Inspection Program will be effectively managed throughout the license renewal period.

A review of the industry operating experience related to the Containment Inservice Inspection Program revealed instances where degradation has occurred within containments. In completing our review we looked at related issues which included degradation of containment liner plates, concrete, coatings, moisture barriers, bellows, tendons and tendon wires, tendon anchor heads, and penetrations.

A review of the plant specific operating experience revealed some instances where the Containment Inservice Inspection Program has been instrumental in discovering material degradation.

Containment degradation included:

- Liner plate corrosion
- Unacceptable tendon liftoff value
- Tendon gallery corrosion

- Tendon grease leakage
- Moisture barrier not in place
- Tendon sheath water intrusion

The Containment Inservice Inspection Program has been effective in identifying material degradation in a timely manner, thus ensuring that age related degradation of the containment will be effectively managed throughout the license renewal period.

The Palisades Containment Inservice Inspection Program has demonstrated that it provides reasonable assurance that aging effects are being adequately managed for the containment. This has been demonstrated through NRC inspection reports, INPO evaluations, audits, self-assessments, and the Corrective Action Program.

This element is consistent with NUREG-1801, Section XI.S1, “ASME Section XI, Subsection IWE,” Section XI.S2, “ASME Section XI, Subsection IWL,” and Section X.S1, “Concrete Containment Tendon Prestress.”

Conclusion

The Containment Inservice Inspection Program is an existing program that uses as its bases, various industry and NRC standards. This program is consistent with NUREG-1801, Sections XI.S1, “ASME Section XI, Subsection IWE,” XI.S2, “ASME Section XI, Subsection IWL,” and X.S1, “Concrete Containment Tendon Prestress.”

Reports of recent internal and external assessments of the program have been reviewed. These reports demonstrate that the program has effectively identified and dispositioned issues that could have led to degraded conditions. It is concluded that the Containment Inservice Inspection Program has been effective in maintaining the intended functions of the affected long-lived, passive containment SSCs.

The continued implementation of the Containment Inservice Inspection Program provides reasonable assurance that aging effects will be managed such that the SSCs within the scope of this program will continue to perform their intended functions consistent with the current licensing bases for the period of extended operation.

B2.1.8 Containment Leakage Testing Program

Program Description

The Containment Leakage Testing Program is an existing program that ensures that containment leakage is maintained below the upper acceptance limit of La = 0.1% / day.

This testing program, in conjunction with the Containment Inservice Inspection Program, provides assurance that age related (and other) deterioration of the containment leakage limiting boundary is appropriately managed to ensure that postulated post-accident releases are limited to an acceptable level. The program is implemented through the following testing and examination activities:

- Overall containment leakage (integrated leakage rate or Type A) test to assess the leak tight integrity of the entire pressure boundary.
- Visual examinations of the containment exterior and interior.
- Local (Type B & C) tests to assess the leak tight integrity of individual penetrations.

NUREG-1801 Consistency

The Containment Leakage Testing Program is consistent with NUREG-1801, Section XI.S4, “10 CFR 50, Appendix J.”

Exceptions to NUREG-1801

None.

Enhancements

None.

Aging Management Program Elements

The elements of the Containment Leakage Testing Program are described below. The results of an evaluation of each element against NUREG-1801, Section XI.S4, “10 CFR 50, Appendix J,” are also provided.

Scope of Program

The scope of the Palisades Containment Leakage Testing Program satisfies the requirements of 10 CFR 50, Appendix J, Option B, but takes the following three approved exceptions to the guidance given in RG 1.163, NEI 94-01, and ANSI/ANS 56.8:

- Regulatory Guide 1.163 stipulates that containment purge valves are to be tested at least every 30 months. The NRC has approved application of performance based test interval extension criteria to the Palisades containment purge valves. However, Palisades has retained the Improved Technical Specification requirement to perform a leakage test on these valves every 184 days.

- 10 CFR 50, Appendix J, Option B states that Type B tests are to be done at accident pressure. The NRC has approved testing of the Palisades personnel air lock door seals at 10 psig rather than P_a = 53 psig.
- NEI 94-01 specifically requires that air lock door seals be tested following air lock door use when containment integrity is required. The NRC has approved the Palisades request to verify seal contact in lieu of performing a leakage test on emergency escape air lock door seals.

The above exceptions are documented in Technical Specifications Section 5.5.14.

This element is consistent with NUREG-1801, Section XI.S4, “10 CFR 50, Appendix J.”

Preventive Actions

The Palisades Containment Leakage Testing Program is a monitoring program and does not specify preventive actions.

This element is consistent with NUREG-1801, Section XI.S4, “10 CFR 50, Appendix J.”

Parameters Monitored, Inspected, and/or Tested

The parameters monitored or inspected under the Containment Leakage Testing Program include the following.

- Overall leakage through the containment pressure boundary (Type A test).
- General condition, with respect to leak tightness and structural integrity, of the accessible exterior and interior surfaces of the containment (visual examination associated with the Type A test).
- Leakage through individual penetrations, access openings and fittings (Type B & C tests).

The containment leakage limiting boundary is designed to provide an essentially leak tight barrier against the release of fission products to the outside environment. The tests and inspections performed under the Containment Leakage Testing Program are designed to detect age related (and other) deterioration that reduces the leak tightness of this boundary.

The parameters monitored and inspected meet the NUREG 1801 intent that the program detect the effects of aging on the intended function (to limit leakage at postulated post-accident pressure) of the containment throughout the license renewal period.

This element is consistent with NUREG-1801, Section XI.S4, “10 CFR 50, Appendix J.”

Detection of Aging Effects

The Palisades Containment Leakage Testing Program is effective in detecting age related and other deterioration of containment pressure boundary leak tight integrity. Specifically, the program is effective in detecting such deterioration in the following components and items:

- The liner, the secondary system boundary, transition pieces connecting the liner to penetration assemblies, the fuel transfer tube assembly, air lock assemblies, the equipment hatch assembly, other blind flange penetration closures, other fixed steel pressure boundary components and associated welds.
- Resilient seals, gaskets and electrical penetration conductor feed-through seals.
- Passive isolation valves (defined in the Technical Specifications as manual isolation valves and deactivated, locked shut automatic valves).

In addition, this program detects age related and other deterioration that affects the leak tightness of active isolation valves and other active components that are not within the scope of the aging management program.

This program also demonstrates (during the Type A test) that the containment has the structural capacity to withstand design basis accident pressure (Pa).

Age related and other deterioration of the containment structural and leakage limiting boundaries are monitored through the Containment Leakage Testing Program and Containment Inservice Inspection Program. Implementing the visual examinations, non-destructive tests as applicable & leakage tests on the schedules mandated by the two programs provides a high degree of assurance that age related and other deterioration mechanisms having a potential impact on containment integrity are detected prior to the loss of the intended function and are effectively managed during the plant license renewal period.

This element is consistent with NUREG-1801, Section XI.S4, “10 CFR 50, Appendix J.”

Monitoring and Trending

Overall containment leakage (Type A test) and individual penetration leakages (Type B & C tests) are monitored at intervals determined in accordance with the requirements of 10 CFR 50, Appendix J, Option B and the guidance given in Regulatory Guide 1.163 & NEI 94-01. The leakage history of each tested item is evaluated considering both trended performance and risk aspects. The date of the subsequent test is set based on the evaluation result.

Trending of overall containment and component leakages as prescribed in the Containment Leakage Testing Program ensures that age related deterioration of leak tightness is identified and controlled in timely manner throughout the license renewal period.

This element is consistent with NUREG-1801, Section XI.S4, “10 CFR 50, Appendix J.”

Acceptance Criteria

Acceptance criteria are stated in Technical Specification Section 5.5.14. These criteria are derived from NEI 94-01 requirements using the plant specific value (0.1% / day) of maximum allowable leakage rate, L_a. Criteria include:

- Limit on overall leakage rate: < 0.1% / day
- Maximum overall as-left leakage at the end of a Type A test: < 0.075% / day
- As-found performance leakage limit: < 0.1% / day
- Limit on as-left penetration minimum pathway leakage summation: < 0.06% / day

In addition, administrative leakage limits are assigned to each component subject to Type B or C testing.

This element is consistent with NUREG-1801, Section XI.S4, “10 CFR 50, Appendix J.”

Corrective Actions, Confirmation Process, Administrative Controls

These elements are consistent with the corresponding NUREG-1801 aging management program elements. See Section B1.2 for further discussion.

Operating Experience

Palisades has a comprehensive Operating Experience Program (OEP) that monitors industry issues / events and assesses these for applicability to its own operations. In addition, the Palisades Corrective Action Program (CAP) is used to track, trend and evaluate plant issues / events. Those issues and events, whether external or plant specific, that are potentially significant to the leak-tight integrity of the Palisades containment are evaluated. The Containment Leakage Testing Program is augmented, as appropriate, if these evaluations show that program changes will enhance leak-tight integrity and, as a consequence, operational safety.

Using the OEP and CAP to focus on industry and plant operating experience ensures that containment leakage issues are addressed in a timely manner and that age related deterioration of containment leak-tightness is effectively managed throughout the license renewal period.

NUREG-1493, which provided the technical justification for the performance based leakage testing program defined in Appendix J, Option B, includes a summary of industry testing experience. This summary demonstrates that performance based leakage testing programs will almost always detect problems at an early stage and are, therefore, acceptable for managing containment leak tight integrity. The NUREG does include a few contrary examples. However, in most cases, these examples illustrate lack of administrative control rather than any technical deficiency in performance based programs. While the NUREG does not specifically address license renewal, it does show, by inference, that performance based leakage testing programs are effective as an aging management tool.

A review of additional industry operating experience associated with the Containment Leakage Testing Program and aging reveals issues and instances related to:

- Type B LLRT performed on containment penetration bellows, which was later invalidated by a subsequent containment ILRT.

No significant problems have been found during periodic Type A tests at Palisades. This confirms that the local leakage rate testing program (in conjunction with periodic containment examinations) has always detected developing deterioration before this could result in a loss of containment leak tight integrity (as defined by overall leakage exceeding L_a).

Instances of excessive (in excess of the assigned administrative limit) component leakage have been uncovered during the performance of Type B & C tests over the operating lifetime of the plant. Most instances of excessive leakage are the result of active isolation valve seat deterioration. Some are the result of air lock door seal misalignment or damage. Active isolation valves and air lock door seals (which are replaced at least once every three refueling outages), however, are not long lived, passive components which are subject to aging management under this program.

Instances of problems with passive components are relatively rare. Two reported instances were found in a search through records going back through the mid 1980's. These are summarized below.

- In 1983 a leak of about 0.1 L_a was found at one conductor seal in electrical penetration EZ-104. The leak affected only one barrier so that minimum pathway leakage through the penetration was still essentially nil. The entire penetration was replaced during the 1985 outage.
- During a September 2001 Type C test on penetration MZ-66, a measured leak of about 0.15 L_a was identified primarily to leakage through a manual isolation gate valve. The cause of the leakage was determined to be debris on the seat. The problem was corrected and leakage restored to an acceptable level. Since the measured leakage was identified primarily to a single barrier, minimum pathway leakage through the penetration remained at a relatively low level.

The Palisades Containment Leakage Testing Program has demonstrated that it provides reasonable assurance that aging effects are being managed for in-scope SSC. Additionally, this has been demonstrated through inspection reports, Program Health Reports, and the Corrective Action Program.

This element is consistent with NUREG-1801, Section XI.S4, “10 CFR 50, Appendix J.”

Conclusion

The Containment Leakage Testing Program is an existing program that is consistent with NUREG-1801, Section XI.S4, “10 CFR 50, Appendix J.”

Reports of recent internal and external assessments of the program have been reviewed. These reports demonstrate that the program has effectively identified and dispositioned issues that could have led to degraded conditions. It is concluded that the Palisades Containment Leakage Testing Program has been demonstrated to be an effective aging management tool. It is further concluded that this program, when implemented in conjunction with the Containment Inservice Inspection Program, provides a high degree of assurance that the containment will retain acceptable levels of structural and leak tight integrity throughout the period of license renewal.

The continued implementation of the Containment Leakage Testing Program provides reasonable assurance that aging effects will be managed such that the SSCs within the scope of this program will continue to perform their intended functions consistent with the current licensing bases for the period of extended operation.

B2.1.9 Diesel Fuel Monitoring and Storage Program

Program Description

The Diesel Fuel Monitoring and Storage Program is an existing program that assures the continued availability and quality of fuel oil to be used in diesel generators and diesel fire pumps. The program includes (a) monitoring and trending of fuel oil chemistry to maintain fuel oil quality and mitigate corrosion, (b) periodic draining, cleaning, and internal inspection of fuel oil storage tanks, and (c) verification of program effectiveness by a one-time measurement of fuel oil storage tank bottom thickness confirming the absence of an aging effect. Fuel oil quality is maintained by monitoring and controlling fuel oil contamination in accordance with the guidelines of the American Society for Testing Materials (ASTM) Standards D 1796, D 2276, D 2709, and D 4057.

Exposure to fuel oil contaminants, such as water and microbiological organisms, is minimized by periodic draining and cleaning of tanks and by verifying the quality of new oil before its introduction into the storage tanks. However, corrosion may occur at locations in which contaminants may accumulate, such as tank bottoms. Accordingly, the effectiveness of the program is verified, through visual inspection and onetime ultrasonic thickness measurement of fuel oil storage tank bottom surface, to ensure that significant degradation is not occurring and that applicable component intended functions will be maintained during the period of extended operation.

Per plant procedures, samples of new fuel oil are obtained and analyzed to verify quality prior to off-loading into storage tanks. Additionally, samples of new fuel are obtained for later analysis for particulates.

NUREG-1801 Consistency

The Diesel Fuel Monitoring and Storage Program is consistent with, but includes exceptions to, NUREG-1801, Section XI.M30, “Fuel Oil Chemistry.”

Exceptions to NUREG-1801

NUREG-1801 Section XI.M30 states, “The quality of fuel oil is maintained by additions of biocides to minimize biological activity, stabilizers to prevent biological breakdown of the diesel fuel, and corrosion inhibitors to mitigate corrosion.”

Palisades tests the fuel in T-10A periodically for microbiological growth.

There is no operational experience that indicates a positive test for microbiological growth. In the event a test would come back positive for microbiological growth, an evaluation would be performed, per the corrective action program, to determine whether addition of biocides should be performed.

Palisades performs periodic tests for particulates on the fuel oil storage tanks.

Additionally, the stored fuel in T-10A Fuel Oil Storage Tank is filtered approximately every three years, or as needed. The fuel oil storage tanks have a relatively fuel oil high turnover rate. T-10A typically has an operating volume of 38,000 gallons, with an average fuel consumption of 76,000 gallons per year. T-926 has a typical operating volume of 15,000 gallons, with an average fuel consumption rate of 35,000 gallons per year. Based on these activities, and high fuel turnover in the storage tanks, an assessment has determined that there is no need to add fuel oil stabilizers to the diesel fuel.

Palisades fuel oil is procured to meet ASTM D975 standards, which include specifications and acceptance criteria for a Copper Strip Corrosion Test. Additionally, samples are periodically analyzed by an off-site facility for relative corrosivity of the fuel by a Copper Strip Corrosion Test. All Copper Strip Corrosion tests performed in the last 5 years have returned results that meet the ASTM standard. Consequently, Palisades does not add corrosion inhibitors to the diesel fuel.

Enhancements

Enhancements are planned to bring the Diesel Fuel Monitoring and Storage Program into conformance (with exception noted above) with the NUREG-1801 program description. The enhancements are:

1. Preventive Actions: Develop and implement procedures for periodic draining and cleaning of Fuel Oil Storage Tanks, Emergency Diesel Generator Day Tanks, and Diesel Fire Pump Day Tanks. These procedures shall include steps to perform a visual inspection of interior tank surfaces for signs of degradation or corrosion, with acceptance criteria, corrective actions, and documentation of inspection results.

2. Preventive Actions: Develop and implement procedures for periodic draining of water accumulated in the bottom of the Fuel Oil Storage Tanks, Emergency Diesel Generator Day Tanks, and Diesel Fire Pump Day Tanks.

3. Detection of Aging Effects: Develop and implement procedures for periodic ultrasonic measurement of thickness of the bottom of Fuel Oil Storage Tanks, Emergency Diesel Generator Day Tanks, and Diesel Fire Pump Day Tanks.

Note that the element descriptions describe the program as it will exist after the identified enhancements have been implemented. Enhancements are scheduled for completion prior to the period of extended operation.

Aging Management Program Elements

The key elements of the Diesel Fuel Monitoring and Storage Program are described below. The results of an evaluation of each key element against the corresponding element of NUREG-1801, Section XI.M30, “Fuel Oil Chemistry,” are also provided.

Scope of Program

Components in the Fuel Oil System associated with the Emergency Diesel Generators and Diesel Fire Pumps, including storage tanks, day tanks, piping, valve bodies, and other passive components that rely on the Diesel Fuel Monitoring and Storage Program to minimize potential for degradation and loss of intended function. The program manages the conditions that would cause general, pitting and microbiological influenced corrosion (MIC) of the diesel fuel tank internal surfaces.

This element is consistent with NUREG-1801, Section XI.M30, “Fuel Oil Chemistry.”

Preventive Actions

The following activities are performed to prevent loss of intended function for Fuel Oil System components that take credit for the Diesel Fuel Monitoring and Storage Program:

- Periodic draining and cleaning of fuel oil storage tanks to remove accumulated sludge, and inspect/correct any locations where corrosion may have started.
- Periodic draining of water collected at the bottom of the fuel oil storage tanks.

There are no coatings on the interior surfaces of the fuel oil storage tanks that are credited with protecting the internal surfaces of the tank from contact with water and microbiological organisms.

This element is consistent with, but includes an exception to, NUREG-1801, Section XI.M30, “Fuel Oil Chemistry.” NUREG-1801 Section XI.M30 states, “The quality of fuel oil is maintained by additions of biocides to minimize biological activity, stabilizers to prevent biological breakdown of the diesel fuel, and corrosion inhibitors to mitigate corrosion.”

Palisades tests the fuel in T-10A periodically for microbiological growth.

There is no operational experience that indicates a positive test for microbiological growth. In the event a test would come back positive for microbiological growth, an evaluation would be performed, per the corrective action program, to determine whether addition of biocides should be performed.

Palisades performs periodic tests for particulates on the fuel oil storage tanks.

Additionally, the stored fuel in T-10A Fuel Oil Storage Tank is filtered approximately every three years, or as needed. The fuel oil storage tanks have a relatively fuel oil high turnover rate. T-10A typically has an operating volume of 38,000 gallons, with an average fuel consumption of 76,000 gallons per year. T-926 has a typical operating volume of 15,000 gallons, with an average fuel consumption rate of 35,000 gallons per year. Based on these activities, and high fuel turnover in the storage tanks, an assessment has determined that there is no need to add fuel oil stabilizers to the diesel fuel.

Palisades fuel oil is procured to meet ASTM D975 standards, which include specifications and acceptance criteria for a Copper Strip Corrosion Test. Additionally, samples are periodically analyzed by an off-site facility for relative corrosivity of the fuel by a Copper Strip Corrosion Test. All Copper Strip Corrosion tests performed in the last 5 years have returned results that meet the ASTM standard. Consequently, Palisades does not add corrosion inhibitors to the diesel fuel.

Parameters Monitored, Inspected, and/or Tested

Palisades Fuel Oil Monitoring and Storage program references ASTM D4057, “Manual Sampling of Petroleum and Petroleum Products,” for fuel oil sampling, and ASTM D975 “Diesel Fuel Oils,” for monitoring fuel oil quality and the levels of water and microbiological organisms in the fuel oil. ASTM D2709, “Standard Test Method for Water and Sediment in Distillate Fuels by Centrifuge”, and ASTM D1796 “Standard Test Method for Water and Sediment in Fuel Oils by the Centrifuge Method (laboratory procedure)” are used as the basis for the water and sediment analysis. Modified ASTM D2276, “Standard Test Method for Particulate Contaminant in Middle Distillate Fuel by laboratory filtration,” is used as the basis for the particulate analysis.

This element is consistent with NUREG-1801, Section XI.M30, “Fuel Oil Chemistry.”

Detection of Aging Effects

The Palisades Diesel Fuel Monitoring and Storage Program includes a variety of inspection and testing activities that are designed to detect signs of degradation due to aging effects prior to loss of intended function. The Palisades Diesel Fuel Monitoring and Storage Program conducts periodic multilevel sampling, tank draining/cleaning, internal tank visual inspections, and tank bottom thickness testing to detect any signs of potential adverse degradation.

This element is consistent with NUREG-1801, Section XI.M30, “Fuel Oil Chemistry.”

Monitoring and Trending

Results of fuel oil analyses for Fuel Oil Storage Tanks and Truck Transport Tankers are documented within the analysis procedures, and reviewed against acceptance criteria. Results are also logged into the Chemistry Department trending database to facilitate long term trending. Additionally, clear acceptance criteria exist within procedures to facilitate corrective actions if parameters exceed these criteria.

Parameters monitored and trended include:

- Water and sediment
- Viscosity
- Specific Gravity
- Particulates
- Biological Activity

This element is consistent with NUREG-1801, Section XI.M30, “Fuel Oil Chemistry.”

Acceptance Criteria

Acceptance criteria are derived from the referenced ASTM standard D975-74, “Classification of Diesel Fuel Oils,” which contains limits for percent water and sediment, and viscosity. Additionally, Chemistry procedures impose further acceptance criteria based on ASTM D975, which references ASTM D4057 for oil sampling, ASTM D1796 and D2709 for water and sediment, and Modified ASTM D2276 for particulates.

This element is consistent with NUREG-1801, Section XI.M30, “Fuel Oil Chemistry.”

Corrective Actions, Confirmation Process, Administrative Controls

These elements are consistent with the corresponding NUREG-1801 aging management program elements. See Section B1.2 for further discussion.

Operating Experience

Palisades has a comprehensive Operating Experience Program (OEP) that monitors industry issues/events and assesses these for applicability to its own operations. In addition, the Palisades Corrective Action Program (CAP) is used to track, trend and evaluate plant issues/events. Those issues and events, whether external or plant specific, that are potentially significant to the Diesel Fuel Monitoring and Storage Program at Palisades are evaluated. The Diesel Fuel Monitoring and Storage Program is augmented, as appropriate, if these evaluations show that program changes will enhance program effectiveness.

Using the OEP and CAP to focus on industry and plant operating experience ensures that Diesel Fuel Monitoring and Storage Program issues are addressed in a timely manner, and that age related deterioration of SSC within the scope of the Diesel Fuel Monitoring and Storage Program will be effectively managed throughout the license renewal period.

A review of industry operating experience associated with the Diesel Fuel Monitoring and Storage Program and aging reveals issues and instances related to:

- Fuel contamination leading to corrosion of fuel oil system components.
- Improper zinc coating curing and epoxy application by the manufacturer leads to zinc-fuel reaction creating adverse corrosion.
- Fuel oil leak caused by improper outer coating application.

Various related NRC and/or industry generic communications have been issued, and, in turn, have been incorporated into the program as applicable.

A review of plant specific operating experience related to the Diesel Fuel Monitoring and Storage Program was performed, and no aging issues were identified.

The Palisades Diesel Fuel Monitoring and Storage Program has demonstrated that it provides reasonable assurance that aging effects are being managed for the Diesel Fuel Monitoring and Storage Program SSCs.

This element is consistent with NUREG-1801, Section XI.M30, “Fuel Oil Chemistry.”

Conclusion

The Diesel Fuel Monitoring and Storage Program is an existing program that uses as its bases, various industry and NRC standards. This program is consistent with, but includes exceptions to, NUREG-1801, Section XI.M30, “Fuel Oil Chemistry.”

Although no recent internal or external assessments of the program have been performed, reviews for plant operating experience, and a general lack of equipment problems related to aging demonstrate that the program has effectively managed aging mechanisms that could have led to degraded conditions. It is concluded that the Diesel Fuel Monitoring and Storage Program has been effective in maintaining the intended functions of the affected long-lived, passive SSCs.

The continued implementation of the Diesel Fuel Monitoring and Storage Program provides reasonable assurance that aging effects will be managed such that SSCs within the scope of this program will continue to perform their intended functions consistent with the current licensing bases for the period of extended operation.

B2.1.10 Fire Protection Program

Program Description

The Fire Protection Program is an existing program that includes (a) fire barrier inspections, (b) electric and diesel-driven fire pump tests, and (c) periodic maintenance, testing, and inspection of water-based fire protection systems. Periodic visual inspections of fire barrier penetration seals, fire dampers, fire barrier walls, ceilings and floors, and periodic visual inspections and functional tests of fire-rated doors are performed to ensure that functionality and operability is maintained. Periodic testing of the fire pumps ensures that an adequate flow of firewater is supplied and that there is no degradation of diesel fuel supply lines. Periodic maintenance, testing and inspection activities of water-based fire protection systems provides reasonable assurance that fire water systems are capable of performing their intended function. Inspection and testing include periodic hydrant inspections, fire main flushing, sprinkler inspections, pipe wall thickness testing and flow tests.

This program manages aging of the fire protection components through detailed fire barrier inspections of fire barrier penetration seals, and fire rated doors. Aging related degradation of fire barrier walls, ceilings and floors are managed by the Structural Monitoring Program. Aging of the diesel-driven fire pump's fuel oil supply line is managed through regularly scheduled fire pump performance tests.

The Fire Protection Program also manages aging of fire water systems through periodic hydrant inspections, flushes, and flow tests, fire main flushing, sprinkler system inspections, and pipe wall thickness testing. In addition, fire water system pressure and flow is continuously monitored in the plant control room through annunciator alarms and personnel.

Also included within the scope of the Fire Protection Program is aging management of spare cables for Appendix R required equipment. These spare cables are located in various storage locations in the event they are needed for repairs to key components damaged by an Appendix R type fire. The Fire Protection Program credits the Non-EQ Electrical Commodities Condition Monitoring Program for aging management of these cables.

The NRC issued ISG-4, “Aging Management of Fire Protection Systems for License Renewal,” in December 2002. This staff position clarified the guidance of NUREG-1801 Section XI.M26, “Fire Protection,” and XI.M27, “Fire Water System,” with regard to wall thinning of fire protection piping due to internal corrosion, testing of sprinkler heads, and valve line-up inspections of Halon/Carbon Dioxide fire suppression systems. The Fire Protection Program is based on the aging management program guidance presented in ISG-4.

NUREG-1801 Consistency

The Fire Protection Program is consistent with, but includes exceptions to, NUREG-1801, Section XI.M26, “Fire Protection.” The program is consistent with, but includes exceptions to, Section XI.M27, “Fire Water System,” as clarified by ISG-4.

Exceptions to NUREG-1801

Exceptions are taken to the selected NUREG-1801 Program elements listed below. The specific exceptions being taken are also discussed in the corresponding element discussions below. They are repeated here for ease of review.

1. XI.M26, Detection of Aging Effects: NUREG-1801, Section XI.M26, as clarified by ISG-4, specifies that visual inspections as follows: “Visual inspection (VT-1 or equivalent) of approximately 10% of each type of seal ... Visual inspection (VT-1 or equivalent) of the fire barrier walls, ceilings, and floors ... Visual inspection (VT-3 or equivalent) detects any signs of degradation of the fire door ....” Palisades does not qualify the personnel performing the visual inspections of fire barrier walls, ceilings, floors, penetration seals and fire doors to the ASME Code type of qualification as stated in NUREG-1801 and ISG -04. Per plant procedures, inspectors for fire barriers/doors/fire seals are appropriately qualified to perform those inspections, but are not necessarily qualified to VT-1 or VT-3. There are no regulatory or other requirements specifying that these inspections be performed to VT-1 or VT-3

standards. This is generally in accordance with the position taken by the Dresden and Quad Cities Nuclear Plants in their application for License Renewal, and subsequent NRC approval of this position in the associated Safety Evaluation Report.

2. XI.M27, Monitoring and Trending: NUREG-1801, Section XI.M27, as clarified by ISG-4, states that the results of system performance testing are monitored and trended as specified by the NFPA codes and standards. At Palisades, inspection and testing is performed as outlined in Fire Protection Implementing Procedures. NFPA codes of record are identified in the Palisades Fire Protection Program Report (FPPR) and/or FSAR. This is generally in accordance with the position taken by the Dresden and Quad Cities Nuclear Power Stations in their License Renewal Application and approved by the NRC in the associated Safety Evaluation Report.
3. XI.M26, Acceptance Criteria: NUREG-1801, as clarified by ISG-4, states “Inspection results are acceptable if there are no visual indications of cracking, separation of seals from walls and components, separation of layers of material, or ruptures or punctures of seals.” Palisades inspection acceptance criteria states that no cracks of ¼” wide or greater are allowed. The acceptance criteria is derived from fire test reports, and is acceptable.

Enhancements

Enhancements are planned to bring the Fire Protection Program into conformance (with exceptions noted above) with the NUREG-1801 program description. The enhancements are:

1. Detection of Aging Effects, and Monitoring and Trending: The Structural Monitoring Program implementing procedures shall be revised to include specific inspection criteria and documentation requirements for verifying that walls, ceilings and floors that serve as Fire Protection Program fire barriers are verified to be free from aging related degradation that would impact the fire barrier's intended function.

2. Scope of Program, Detection of Aging Effects, and Monitoring and Trending: Plant procedures shall be revised to more specifically address aging related degradation and expectations for documentation of fire door condition.

3. Detection of Aging Effects, and Monitoring and Trending: Develop and implement procedures to perform visual inspections for fire door clearances.

4. Detection of Aging Effects, and Monitoring and Trending: Revise diesel-driven fire pump performance test procedures to more specifically address requirement to inspect

and monitor fuel oil supply line for aging related degradation, and to document inspection results.

5. Detection of Aging Effects: Develop and implement procedures for inspection of below grade fire protection system piping. Inspections shall occur when below grade piping is excavated for maintenance, and shall include pipe wall thickness (NDE or direct measurement) and documentation of aging related degradation of pipes. Procedures shall include acceptance criteria, and criteria for further corrective actions if acceptance criteria are not met.

6. Parameters Monitored, Inspected and Tested: Plant procedures shall be revised to more specifically address identification of aging related degradation and expectations for documentation of fire hydrant condition. Also, these revisions shall include provisions to perform flow testing for fire hydrants within the scope of License Renewal that are credited for fire suppression in the Palisades current licensing basis.

7. Detection of Aging Effects: Develop and implement procedures to replace all sprinkler heads prior to the end of the 50 year service life, or for testing of a representative sample of sprinkler heads prior to the end of the 50 year service life and at 10 year intervals thereafter, per requirements of NFPA 25, Section 5.3.

Note that the element descriptions describe the program as it will exist after the identified enhancements have been implemented. Enhancements are scheduled for completion prior to the period of extended operation.

Aging Management Program Elements

The key elements of the Fire Protection Program are described below. The results of an evaluation of each key element against NUREG-1801, Sections XI.M26, “Fire Protection,” and XI.M27, “Fire Water System” as clarified by ISG-4, are also provided.

Scope of Program

SSCs included within the scope of the Fire Protection Program include both fire suppression and fire mitigation components. These include fire barriers, Appendix R spare cables, fire detection and fire suppression systems, as discussed in current licensing basis documents. The program focuses on managing loss of material due to corrosion, MIC, or biofouling of carbon steel and cast-iron components exposed to water and fuel oil, and aging/degradation of fire barrier components.

This element is consistent with NUREG-1801, Sections XI.M26, “Fire Protection” as clarified by ISG-4.

This element is consistent with NUREG-1801, Section XI.M27, “Fire Water System” as clarified by ISG-4.

Preventive Actions

The Palisades Fire Protection Program incorporates many activities that serve to prevent or manage aging of the Fire Protection System. These include regular inspections of fire barrier penetration seals, fire rated doors, and sprinkler systems. Performance tests and flushes are performed on fire pumps and fire hydrants. These inspections and tests ensure that aging related degradation will be detected in the early stages to prevent loss of intended function.

The Fire Hazards Analysis Report (FHAR) quantifies the combustible loading and assesses the fire severity for all plant fire areas. It also specifies measures for fire suppression, fire containment, and safe shutdown capability for each fire area. Palisades fire prevention is administered through the plant procedures.

This element is consistent with NUREG-1801, Sections XI.M26, “Fire Protection,” as clarified by ISG-4.

This element is consistent with NUREG-1801, Sections XI.M27, “Fire Water System,” as clarified by ISG-4.

Parameters Monitored, Inspected, and/or Tested

Fire Protection Program activities monitor a variety of parameters to prevent loss of intended function due to age-related degradation. These parameters include:

- Visual inspections of at least 10% of the fire barrier penetration seals for signs of cracking, discontinuities, spalling, shrinkage, gaps, voids, loose or dislodged pieces, on a refueling cycle frequency. Damming board inspections look for breaks, unsealed holes, discontinuities and pulling away from the fire barrier.
- Fire rated hollow metal doors are visually inspected to verify integrity of door surfaces and proper clearances. Doors are inspected for proper functioning, including hardware.
- Fire rated assemblies, and fire barrier walls, ceilings and floors are inspected for missing components, spalling, cracking and loss of material caused by freeze-thaw, chemical attack and reaction with aggregates.
- Fire hydrants are flushed to test for flow restriction and proper hydrant operation and drainage. Hydrants are visually inspected for corrosion and

damage, and proper thread/valve lubrication. Hydrants within the scope of license renewal that are credited for fire suppression in the Palisades current licensing basis are flow tested.

- Sprinkler system flow switches are functionally tested for blockage or biofouling.
- Sprinkler systems are inspected for external corrosion, damage, paint, dirt, and blockage
- Diesel-driven fire pump fuel supply line is inspected during pump flow tests to detect degradation such as corrosion.
- Pressure alarms and flow switches are continuously monitored by control room annunciators and personnel to detect fire suppression system operation or abnormalities.
- Fire water system piping is volumetrically examined as part of the Palisades ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program.
- Appendix R spare cables are inventoried, inspected and tested to ensure satisfactory condition and readiness, while in storage. The Fire Protection Program credits the Non-EQ Electrical Commodities Condition Monitoring Program for aging management of these components.

This element is consistent with NUREG-1801, Sections XI.M26, “Fire Protection,” and XI.M27, “Fire Water System” as clarified by ISG-4.

Detection of Aging Effects

The Palisades Fire Protection Program includes a variety of inspection and testing activities that are designed to detect degradation due to aging effects prior to loss of intended function.

At least 10% of the fire barrier penetration seals are visually inspected by qualified inspectors every 18 months for signs of age-related degradation, such as seal separation from walls and components, cracking, rupture and puncture of seals. The Fire Protection Program credits the Structural Monitoring Program for aging management of fire barrier walls, ceilings and floors. Fire doors are periodically tested and visually inspected by qualified inspectors for signs of corrosion, wear, or missing parts to ensure that functionality and operability is maintained.

For fire barrier penetration seals visual inspections, fire pump performance tests, and sprinkler system visual inspections, NUREG-1801 specifies a frequency of once per refueling outage. Palisades Fire Protection Program specifies a frequency of once every 18 months. This is considered acceptable because Palisades has an 18 month operating cycle, which corresponds to the same amount of time between inspections.

Palisades credits the One-Time Inspection Program for aging management of RCP oil collection tank, piping and valve bodies for wall thickness and aging related degradation.

Testing of the fire pumps is performed every 18 months to ensure that an adequate flow of fire water is supplied and that there is no degradation of the fuel line to the diesel-driven fire pump.

Continuous fire water system pressure monitoring, and periodic functional testing including fire detection/actuation devices, ensure that corrosion and biofouling are not occurring to an extent that an intended function would be compromised. To determine the ability of fire water system piping to perform its intended pressure boundary function, volumetric inspections of selected fire water system piping segments are performed as part of the Palisades ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program.

Sprinkler heads will be replaced, or tested in accordance with NFPA 25, prior to exceeding their 50 year service life. The required testing will be repeated at ten year intervals.

Below grade fire protection system piping will be inspected for pipe wall thickness and age related degradation during inspections of opportunity when the below grade systems are excavated for maintenance.

Annual visual inspections, flushes, and flow tests of fire hydrants within the scope of license renewal that are credited for fire suppression in the Palisades current licensing basis provide the opportunity for degradation to be detected before a loss of intended function can occur.

This element is consistent with, but includes an exception to, NUREG-1801, Sections XI.M26, “Fire Protection” as clarified by ISG-4. The exception is:

- NUREG-1801, Section XI.M26, as clarified by ISG-4, specifies that visual inspections as follows: “Visual inspection (VT-1 or equivalent) of approximately 10% of each type of seal ... Visual inspection (VT-1 or equivalent) of the fire barrier walls, ceilings, and floors ... Visual inspection (VT-3 or equivalent) detects any signs of degradation of the fire door ....”Palisades does not qualify the personnel performing the visual inspections of fire barrier walls, ceilings, floors, penetration seals and fire doors to the ASME Code type of qualification as stated in NUREG-1801 and

ISG -04. Per plant procedures, inspectors for fire barriers/doors/fire seals are qualified to perform those inspections, but are not necessarily qualified to VT-1 or VT-3. There are no regulatory or other requirements specifying that these inspections be performed to VT-1 or VT-3 standards. This is generally in accordance with the position taken by the Dresden and Quad Cities Nuclear Plants in their application for License Renewal, and subsequent NRC approval of this position in the associated Safety Evaluation Report.

This element is consistent with NUREG-1801, Section XI.M27, “Fire Water System” as clarified by ISG-4.

Monitoring and Trending

The Fire Protection Program credits the Structural Monitoring Program for monitoring the condition of fire barrier walls, ceilings and floors. At least 10% of the fire barrier penetration seals are visually inspected every 18 months for signs of age-related degradation, such as seal separation from walls and components, cracking, rupture and puncture of seals. Fire doors are tested and/or visually inspected by qualified inspectors semi-annually for signs of corrosion, wear, missing parts, and proper clearances to ensure that functionality and operability is maintained.

For fire barrier penetration seal visual inspections, NUREG-1801 specifies a frequency of once per refueling outage. Palisades Fire Protection Program specifies a frequency of once every 18 months. This is considered acceptable because Palisades has an 18 month operating cycle, which corresponds to the same amount of time between inspections.

Testing of the fire pumps is performed every 18 months to ensure that an adequate flow of fire water is supplied and that there is no degradation of the fuel line to the diesel-driven fire pump.

The fire protection system pressure is continuously monitored. Test results from the various surveillance tests are evaluated. Periodic full flow flushing of the main fire system underground piping is performed to assure that corrosion is not occurring and the system function is maintained. Any degradation identified either by visual inspections or as a result of testing is evaluated and corrected.

This element is consistent with NUREG-1801, Sections XI.M26, “Fire Protection” as clarified by ISG-4.

This element is consistent with, but includes an exception to, NUREG-1801, Sections XI.M27, “Fire Water System” as clarified by ISG-4. The exception is:

- NUREG-1801, Section XI.M27, as clarified by ISG-4, states that the results of system performance testing are monitored and trended as specified by the NFPA codes and standards. At Palisades, inspection and testing is performed as outlined in Fire Protection Implementing Procedures. This is generally in accordance with the position taken by the Dresden and Quad Cities Nuclear Power Stations in their License Renewal Application and approved by the NRC in the associated Safety Evaluation Report.

Acceptance Criteria

Acceptance criteria are defined in the Palisades procedures used to perform tests and inspections of the Fire Protection System. Fire seal and conduit wrapping inspection results are acceptable if there are no visual indication of cracking > ¼", separation of seals from building structures and components, rupture or puncture of seals. Fire door inspection results are acceptable if there are no visual indications of wear, holes, damaged or missing parts, and clearances are within limits. Diesel-driven fire pump inspections are acceptable if there is no evidence of corrosion or leaks on the fuel oil supply line. Acceptance criteria for the diesel-driven fire pump capacity is contained within the test procedure.

This element is consistent with, but includes an exception to, NUREG-1801, Sections XI.M26, "Fire Protection" as clarified by ISG-4. The exception is:

- NUREG-1801, as clarified by ISG-4, states "Inspection results are acceptable if there are no visual indications of cracking, separation of seals from walls and components, separation of layers of material, or ruptures or punctures of seals." Palisades inspection acceptance criteria states that no cracks of ¼" wide or greater are allowed. The acceptance criteria is derived from fire test reports, and is acceptable.

This element is consistent with NUREG-1801, Sections XI.M27, "Fire Water System" as clarified by ISG-4.

Corrective Actions, Confirmation Process, Administrative Controls

These elements are consistent with the corresponding NUREG-1801 aging management program elements. See Section B1.2 for further discussion.

Operating Experience

Palisades has a comprehensive Operating Experience Program (OEP) that monitors industry issues/events and assesses these for applicability to its own operations. In addition, the Palisades Corrective Action Program (CAP) is used to track, trend and evaluate plant issues/events. Those issues and events, whether external or plant specific, that are potentially significant to fire protection at Palisades are evaluated. The Fire Protection Program is augmented, as appropriate, if these evaluations show that program changes will enhance fire protection and operational safety.

Using the OEP and CAP to focus on industry and plant operating experience ensures that Fire Protection Program issues are addressed in a timely manner and that age related deterioration of SSC within the scope of the Fire Protection Program will be effectively managed throughout the license renewal period.

A review of industry operating experience associated with the Fire Protection Program and aging reveals issues and instances related to:

- Fire water system piping corrosion and ruptures
- Fire retardant coatings and materials
- Fouling of components in contact with raw water
- Problems with fire barriers.

Various related NRC and/or industry generic communications have been issued, and, in turn, have been incorporated into the program as applicable.

A review of plant specific operating experience related to the Fire Protection Program and aging revealed that the following issues have been addressed:

- Blockage of Fire Protection piping with corrosion products
- Deluge valve trim piping failures due to corrosion
- Underground fire main rupture due to cyclic loadings
- Water tight fire door seal degradation

The Palisades Fire Protection Program has demonstrated that it provides reasonable assurance that aging effects are being managed for Fire Protection Program SSCs. This has been demonstrated through NRC inspection reports, audits, self-assessments, and the Corrective Action Program.

This element is consistent with NUREG-1801, Sections XI.M26, “Fire Protection,” and XI.M27, “Fire Water System” as clarified by ISG-4.

Conclusion

The Fire Protection Program is an existing program that uses as its bases, various industry and NRC standards. The program is consistent with, but includes exceptions to, NUREG-1801, Section XI.M26, “Fire Protection,” and Section XI.M27, “Fire Water System,” as clarified by ISG-4. A summary of each exception is provided in the discussion of the affected program element above.

Reports of recent internal and external assessments of the program have been reviewed. These reports demonstrate that the program has effectively identified and dispositioned issues that could have led to degraded conditions. It is concluded that the Fire Protection Program has been effective in maintaining the intended functions of the affected long-lived, passive SSCs.

The continued implementation of the Fire Protection Program provides reasonable assurance that aging effects will be managed such that SSCs within the scope of this program will continue to perform their intended functions consistent with the current licensing bases for the period of extended operation.

B2.1.11 Flow Accelerated Corrosion Program

Program Description

The Flow Accelerated Corrosion Program is an existing program that manages aging effects due to flow-accelerated corrosion (FAC) on the internal surfaces of carbon or low alloy steel piping, elbows, reducers, expanders, and valve bodies which contain high energy fluids (both single phase and two phase). The program implements the EPRI guidelines in NSAC-202L-R2 for an effective FAC program and includes (a) an analysis using a predictive code such as CHECWORKS™ to determine critical locations, (b) baseline inspections to determine the extent of thinning at these locations, (c) follow-up inspections to confirm the predictions, and (d) repairing or replacing components, as necessary.

NUREG-1801 Consistency

The Flow Accelerated Corrosion Program is consistent with NUREG-1801, Section XI.M17, “Flow Accelerated Corrosion.”

Exceptions to NUREG-1801

None.

Enhancements

None.

Aging Management Program Elements

The elements of the Flow Accelerated Corrosion Program are described below. The results of an evaluation of each element against NUREG-1801, Section XI.M17, "Flow Accelerated Corrosion," are also provided.

Scope of Program

The Palisades FAC program implements EPRI guidelines in NSAC-202L-R2, and includes procedures and administrative controls to assure that the structural integrity of all carbon steel and low alloy steel lines containing high-energy fluids (two phase as well as single phase) is maintained.

This program predicts, detects, and monitors flow accelerated corrosion for the systems and components that are susceptible to flow accelerated corrosion and that are within the scope of License Renewal. These systems and components are modeled in CHECWORKS in accordance with the EPRI guidelines in NSAC-202L-R2. This program predicts, detects, and monitors flow accelerated corrosion wear for the systems modeled in CHECWORKS that are within the scope of License Renewal.

The following systems that are within the scope of license renewal credit the FAC program for aging management of flow accelerated corrosion:

- Condensate
- Feedwater
- Main Steam
- Steam generators - secondary side
- Heater Extraction and Drain
- HVAC (heating steam)
- Air Ejector and Gland Seal

The program encompasses the following component types:

- Elbows
- Tees
- Straight Pipe
- Concentric Reducers/Expanders
- Nozzles
- Valves

- Pumps
- Traps
- Tanks
- Strainers

The program includes the following activities:

- Conduct appropriate analysis and limited baseline inspections;
- Determine the extent of thinning and repair or replace components as appropriate;
- Perform follow up inspections to confirm or quantify previously discovered conditions; and
- Take long term corrective actions if necessary.

This element is consistent with NUREG-1801, Section XI.M17, "Flow -Accelerated Corrosion."

Preventive Actions

There are no preventive actions associated with the FAC Program. The program is an inspection, analysis, and verification program. Mitigation of aging due to FAC is by maintaining high water quality, as described in the Water Chemistry Program. Components with a calculated remaining life of less than one operating cycle are either replaced, repaired, or re-evaluated to ensure that the intended function of maintaining the component pressure boundary will be preserved.

This element is consistent with NUREG-1801, Section XI.M17, "Flow -Accelerated Corrosion."

Parameters Monitored, Inspected, and/or Tested

The FAC program monitors the effects of FAC on the intended function of piping and components by measuring the wall thickness using nondestructive examinations and by performing analytical evaluations.

This element is consistent with NUREG-1801, Section XI.M17, "Flow -Accelerated Corrosion."

Detection of Aging Effects

The Palisades Flow Accelerated Corrosion Program includes a variety of inspection and testing activities that are designed to detect degradation due to aging effects caused by FAC prior to loss of intended function. The methods of inspection for FAC include visual observation, ultrasonic (UT), and radiography (RT).

The inspection schedule provides for timely detection of degradation of susceptible piping and components. The CHECWORKS^{TM} Model is used to select components for inspection. The extent and schedule of inspections ensures detection of wall thinning before the loss of the intended function of the component.

This element is consistent with NUREG-1801, Section XI.M17, “Flow -Accelerated Corrosion.”

Monitoring and Trending

The CHECWORKS^{TM} code is used to predict component degradation in systems susceptible to FAC. Plant data, including material composition, system flow characteristics, and operating conditions are used by CHECWORKS^{TM} to determine the remaining service life, which is recalculated after each inspection. Per plant procedures, if inspection results indicate a higher than expected wear rate that is inconsistent with the predicted wear rates, the reason for those inconsistencies are investigated. As part of the evaluation, an updated FAC analysis is performed, and additional investigations conducted, to determine the extent of the unpredicted wear rates. CHECWORKS^{TM} provides a bounding analysis for FAC. The inspection schedule developed on the basis of the results of this predictive code provides reasonable assurance that adequate wall thickness will be maintained between inspections.

This element is consistent with NUREG-1801, Section XI.M17, “Flow -Accelerated Corrosion.”

Acceptance Criteria

The acceptance criteria are defined in plant procedures. Criteria include minimum requirements for remaining life, program expansion criteria, repairing and replacing components, and engineering analyses to support actions and conclusions. CHECWORKS^{TM} is used to determine if the remaining life is shorter than the amount of time until the next inspection. An engineering evaluation is also performed to determine acceptability. Activities are then planned and executed if the engineering evaluation determines repair or replacement is necessary.

This element is consistent with NUREG-1801, Section XI.M17, “Flow -Accelerated Corrosion.”

Corrective Actions, Confirmation Process, Administrative Controls

These elements are consistent with the corresponding NUREG-1801 aging management program elements. See Section B1.2 for further discussion.

Operating Experience

Palisades has a comprehensive Operating Experience Program (OEP) that monitors industry issues/events and assesses these for applicability to its own operations. In addition, the Palisades Corrective Action Program (CAP) is used to track, trend and evaluate plant issues/events. Those issues and events, whether external or plant specific, that are potentially significant to the Flow Accelerated Corrosion Program at Palisades are evaluated. The Flow Accelerated Corrosion Program is augmented, as appropriate, if these evaluations show that program changes will enhance program effectiveness.

A review of industry operating experience associated with the Flow Accelerated Corrosion Program and aging reveals issues and instances related to:

- Feedwater heater shell degradation and ruptures
- Feedwater and Condensate line ruptures
- Pipe wall thinning downstream of control valves and flow restricting devices
- Valve body erosion
- Extraction steam line ruptures
- Moisture Separator Reheater Drain Tank drain line ruptures
- Steam Generator Feedwater distribution piping and J-tube damage
- Erosion of carbon steel ribs and tube supports in Steam Generators

Various related NRC and/or industry generic communications have been issued, and, in turn, have been incorporated into the program as applicable.

A review of plant specific operating experience related to the Flow Accelerated Corrosion Program and aging revealed that the following issues have been addressed:

- FAC on 2 inch main steam line elbows
- Higher than expected wear rates on 8 inch steam pipes and elbows on the outlet of Moisture Separator Reheater

- Main Condenser tube leaks caused by FAC
- Higher than expected wear rates on high pressure extraction steam piping to high pressure feedwater heater
- FAC on end-bell of low pressure feedwater heater
- Valve body FAC on control valves and check valves
- FAC of feedwater heater shell side capped drains
- FAC damage to low pressure turbine extraction sleeves
- FAC damage to extraction steam lines to high pressure feedwater heaters
- FAC damage to Moisture Separator Reheater vent line
- FAC of feedwater piping
- FAC of reducer downstream of control valve
- Through wall steam leak on steam generator flash tank

Using the OEP and CAP to focus on industry and plant operating experience ensures that Flow Accelerated Corrosion Program issues are addressed in a timely manner and that age related deterioration of SSC's within the scope of the Flow Accelerated Corrosion Program will be effectively managed throughout the license renewal period.

In addition, NRC inspection reports, audits, self-assessments, and the Corrective Action Program were reviewed for relevant information. No significant findings were identified that would indicate that the program is ineffective. Some weaknesses have been identified that have resulted in appropriate corrective actions and enhancements.

This element is consistent with NUREG-1801, Section XI.M17, “Flow -Accelerated Corrosion.”

Conclusion

The Flow Accelerated Corrosion Program is an existing program that uses as its bases, various industry and NRC standards. This program is consistent with NUREG-1801, Section XI.M17, “Flow -Accelerated Corrosion.”

Reports of recent internal and external assessments of the program have been reviewed. These reports demonstrate that the program has effectively identified and dispositioned issues that could have led to degraded conditions. It is concluded that the Flow Accelerated Corrosion Program has been effective in maintaining the intended functions of the affected long-lived, passive SSC's.

The continued implementation of the Flow Accelerated Corrosion Program provides reasonable assurance that aging effects will be managed such that SSC's within the scope of this program will continue to perform their intended functions consistent with the current licensing bases for the period of extended operation.

B2.1.12 Non-EQ Electrical Commodities Condition Monitoring Program

Program Description

The Non-EQ Electrical Commodities Condition Monitoring Program is a new program that manages aging in selected non-EQ commodity groups within the scope of 10 CFR 54. Program activities are responsive to the NRC guidance provided in NUREG-1801 and industry standards.

Palisades has identified each electrical commodity group requiring aging management for the three applicable sections of NUREG-1801 with the additional guidance provided in ISG-2 and draft ISGs-5, 15, 17 and 18, as follows:

- NUREG-1801 Program XI.E1, “Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements” requires a periodic inspection program that visually inspects accessible cables and connections in adverse localized environments with any identified degradation being evaluated and, as appropriate per plant procedures, entered into the plant corrective action process. The Non-EQ Electrical Commodities Condition Monitoring Program predominantly inspects for adverse aging from temperature, radiation, or moisture in the presence of oxygen.

  Electrical pinned connectors are subject to pin corrosion from boric acid leakage, and periodic inspections are conducted in the Boric Acid Corrosion Program to preclude failures resulting from leakage.

  The non-segregated bus in-scope of License Renewal was conservatively assessed, as discussed in draft ISG-17, to require aging management. The “weak link” in maintaining a non-aging environment was identified to be unchecked water leakage through the housing seals and bus bar connections due to thermal cycling. Appropriate inspection activities are included in the periodic inspections.
- NUREG-1801 Program XI.E2, “Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits” requires routine calibration tests to be performed to identify potential existence of aging degradation of cables and connections used in low-level signal applications that are sensitive to reduction in insulation resistance (IR) such as radiation monitoring and nuclear instrumentation. This is revised as discussed in draft ISG-15 which allows

testing once every 10 years in lieu of TS surveillance test trending. The Non-EQ Electrical Commodities Condition Monitoring Program does subject sensitive instrumentation circuits, identified as requiring aging management, to periodic testing.

- NUREG-1801 Program XI.E3, “Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements” requires a periodic test to provide an indication of the condition of the conductor insulation for those cables in-scope of License Renewal exposed to long periods of high moisture (greater than a few days at a time) and subjected to voltage stress (energized greater than 25% of the time). Periodic testing will be performed on these medium voltage cables to provide an indication of the insulation condition. The Non-EQ Electrical Commodities Condition Monitoring Program includes input from draft ISG-18 for periodic inspections of underground raceway manholes for the accumulation of water over the medium-voltage cables. Periodic inspections of underground manholes for the accumulation of water in the medium-voltage cable manholes will minimize the effects of water inside the underground manholes.

  The fuse holders that were not inside active equipment were evaluated per draft ISG-5 and determined to have no aging effect that required management.

NUREG-1801 Consistency

The Non-EQ Electrical Commodities Condition Monitoring Program is consistent with NUREG-1801, Sections XI.E1, “Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements,” XI.E2, “Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits,” and XI.E3, “Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements,” and guidance provided in ISG-2 and draft ISGs-5, 15, 17 and 18.

Exceptions to NUREG-1801

None.

Enhancements

A Non-EQ Electrical Commodities Condition Monitoring Program will be developed and implemented. Features of the program will include development and implementation of procedures to conduct periodic inspection of insulated cables and connectors, test sensitive instrumentation circuits, test medium voltage cables, and inspect manhole water levels.

Note that the element descriptions describe the program as it will be implemented. The program will be implemented prior to the period of extended operation.

Aging Management Program Elements

The key elements of the Non-EQ Electrical Commodities Condition Monitoring Program are described below. The results of an evaluation of each key element against NUREG-1801, Sections XI.E1, “Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements,” XI.E2, “Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits,” and XI.E3, “Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements,” and guidance provided in ISG-2 and draft ISGs-5, 15, 17 and 18, are also provided below.

Scope of Program

Commodities within the scope of the Non-EQ Electrical Commodities Condition Monitoring Program include all non-EQ insulated cables and connections in-scope of License Renewal identified as requiring aging management, and any identified in the future to be in a newly-discovered localized adverse environment. The non-segregated phase bus, in-scope due to being a component of the SBO restoration path, is also included.

The specific commodity groups included within the scope of the Non-EQ Electrical Commodities Condition Monitoring Program are as follows:

- Low-Voltage Cables and Connections
- Low-Voltage Electrical Pinned Connectors
- Low-Voltage Sensitive Instrumentation Cables
- Inaccessible Medium-Voltage Cables and Connections
- Non-Segregated Phase Bus and Connections

The aging effects/mechanisms managed by the Non-EQ Electrical Commodities Condition Monitoring Program for these electrical commodities are as follows:

- Cable and connection jacket/insulation degradation, such as embrittlement and cracking, on a circuit section or component exposed to a localized adverse environment typically involving high temperature, radiation, and moisture levels
- Corrosion (Loss of Material) of the non-segregated bus duct internal metal components, and loose connections creating localized heating leading to electrical failure

- Loss of dielectric (insulation material) properties
- Promotion of water trees

This element is consistent with NUREG-1801, Section XI.E1, “Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements,” Section XI.E2, “Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits,” Section XI.E3, “Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements,” and guidance provided in ISG-2 and draft ISGs-5, 15, 17 and 18.

Preventive Actions

- Accessible Non-EQ Electrical Cables and Connections

  This is a periodic visual inspection program and no actions are taken as part of this program to prevent or mitigate aging degradation. This element is consistent with NUREG-1801, Section XI.E1, “Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements.”

- Non-EQ Electrical Cables Used in Sensitive Instrumentation Circuits

  This is a periodic testing program to provide an indication of the condition of the conductor insulation and no actions are taken as part of this program to prevent or mitigate aging degradation. This element is consistent with NUREG-1801, Section XI.E2, “Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits” and guidance provided in draft ISG-15.

- Inaccessible Non-EQ Medium-Voltage Cables

  The periodic testing program provides an indication of the condition of the conductor insulation. Periodic inspections of underground manholes, for the accumulation of water over the medium-voltage cables, will be conducted to minimize prolonged moisture conditions that promote the growth of water trees. This element is consistent with NUREG-1801, Section XI.E3, “Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements” and guidance provided in draft ISG-18.

Parameters Monitored, Inspected, and/or Tested

- Accessible Non-EQ Electrical Cables and Connections

  A periodic visual inspection will be performed of accessible insulated cables and connections in-scope of License Renewal that may be subjected to a localized adverse environment. A localized adverse environment is defined as when any electrical insulation material is exposed to an aging environment that is significantly greater than the bounding design parameter value.

  The periodic inspection shall also include visual inspection for signs of water leakage or contamination into the non-segregated bus through the housing seals and signs of localized heating potentially from loose internal electrical connections that may lead to electrical failure.

  This element is consistent with NUREG-1801, Section XI.E1, "Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements" and guidance provided in ISG-2 and draft ISGs-17 and 18.

- Non-EQ Electrical Cables Used in Sensitive Instrumentation Circuits

  This is a periodic testing program to check insulation condition and no actions are taken as part of this testing program to prevent or mitigate aging degradation.

  This element is consistent with NUREG-1801, Section XI.E2, "Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits" and guidance provided in draft ISG-15.

- Inaccessible Non-EQ Medium-Voltage Cables

  The periodic testing program provides an indication of the condition of the conductor insulation. Periodic inspections of underground manholes, for the accumulation of water over the medium-voltage cable levels, will be conducted to minimize the prolonged moisture conditions that promotes the growth of water trees.

  This element is consistent with NUREG-1801, Section XI.E3, "Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements," and guidance provided in draft ISG-18

Detection of Aging Effects

- Accessible Non-EQ Electrical Cables and Connections

  A periodic inspection of accessible insulated cables and connections in-scope of License Renewal will be performed at least once every 10 years. As stated in NUREG-1801, this is an adequate inspection period to preclude failures of the conductor insulation/jacket and connection since experience has shown that aging degradation is a slow process. The first inspection for License Renewal will be completed before the period of extended operation.

  The accessible insulated cable and connections shall be visually inspected for insulation/jacket surface anomalies, such as discoloration, swelling, cracking, or surface contamination. Surface anomalies are a precursor indication of insulation degradation. If an unacceptable condition is identified for an insulated cable or connection, a determination would be made as to whether the same condition or situation is applicable to other accessible or inaccessible insulated cables and connections exposed to the same type of localized adverse environment.

  This element is consistent with NUREG-1801, Section XI.E1, “Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements.”

- Non-EQ Electrical Cables Used in Sensitive Instrumentation Circuits

  Cables used in nuclear instrumentation circuits are to be periodically tested (such as insulation resistance tests, time domain reflectometry tests, or other tests effective in determining cable insulation condition), at least once every ten years to provide an indication of the condition of the insulated conductor and connection, and the ability of the circuit to perform its intended function. This is an adequate period to identify cable and connection degradation to preclude excessive leakage currents since experience has shown that aging degradation is a slow process. The first tests for License Renewal will be completed before the period of extended operation. If an unacceptable condition or situation is identified, a determination shall be made as to applicability of the condition on other cables used in the nuclear instrumentation circuits.

  This element is consistent with NUREG-1801, Section XI.E2, “Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits” and guidance provided in draft ISG-15.

- Inaccessible Non-EQ Medium-Voltage Cables

  Identified in-scope inaccessible medium-voltage insulated cables (not designed for submergence) subject to long periods of high moisture conditions and voltage stress are tested (such as insulation resistance tests, time domain reflectometry tests, or other tests effective in determining cable insulation condition) at least once every 10 years to provide an indication of the condition of the conductor insulation and the ability of the cable to perform its intended function. As stated in NUREG-1801, this is an adequate period to preclude failures of the conductor insulation since experience has shown that aging degradation is a slow process. Periodic inspections (periodicity will be based on inspection results) of underground manholes, for the accumulation of water over the medium-voltage cable levels, will be conducted to minimize the prolonged moisture conditions that promotes the growth of water trees. The first tests and inspections for License Renewal will be completed before the period of extended operation. If an unacceptable condition or situation is identified, a determination would be made as to whether the same condition or situation is applicable to other inaccessible, in-scope, medium-voltage cables.

  This element is consistent with NUREG-1801, Section XI.E3, “Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements,” and guidance provided in draft ISG-18.

Monitoring and Trending

Trending actions are not included as part of this program because the ability to trend inspection or test results is limited and dependant on the specific type of test selected.

This element is consistent with NUREG-1801, Section XI.E1, “Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements,” Section XI.E2, “Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits,” Section XI.E3, “Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements,” and guidance provided in ISG-2 and draft ISGs-5, 15, 17 and 18.

Acceptance Criteria

- Accessible Non-EQ Electrical Cables and Connections

  The accessible insulated cables and connections are to be free from unacceptable levels of surface anomalies, which indicate conductor or connection insulation degradation. Unacceptable degradation is defined as a noted condition or situation that, if left unmanaged, could lead to a loss of intended function.

  This element is consistent with NUREG-1801, Section XI.E1, “Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements.”

- Non-EQ Electrical Cables Used in Sensitive Instrumentation Circuits

  The acceptance criteria for each test is defined by the specific type of test performed and the specific circuit tested in the sensitive instrumentation circuits.

  This element is consistent with NUREG-1801, Section XI.E2, “Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits” and guidance provided in draft ISG-15.

- Inaccessible Non-EQ Medium-Voltage Cables

  The acceptance criteria for each test is defined for the specific type of test performed and the specific cable tested. Periodic inspections of underground manholes, for the accumulation of water around medium-voltage cables, shall minimize time periods exposed to water.

  This element is consistent with NUREG-1801, Section XI.E3, “Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements” and guidance provided in draft ISG-18.

Corrective Actions, Confirmation Process, Administrative Controls

These elements are consistent with the corresponding NUREG-1801 aging management program elements. See Section B1.2 for further discussion.

Operating Experience

Palisades has a comprehensive Operating Experience Program (OEP) that monitors industry issues/events and assesses these for applicability to its own operations. In addition, the Palisades Corrective Action Program (CAP) is used to track, trend and evaluate plant issues/events. Those issues and events, whether external or plant specific, that are potentially significant to the Non-EQ Electrical Commodities Condition Monitoring Program at Palisades are evaluated. The Non-EQ Electrical Commodities Condition Monitoring Program is augmented, as appropriate, if these evaluations show that program changes will enhance program effectiveness.

Using the OEP and CAP to focus on industry and plant operating experience ensures that Non-EQ Electrical Commodities Condition Monitoring Program issues are addressed in a timely manner and that age related deterioration of commodities within the scope of the Non-EQ Electrical Commodities Condition Monitoring Program will be effectively managed throughout the license renewal period.

Industry experience, as documented in SAND96-0344 “Aging Management Guideline for Commercial Nuclear Plants - Electrical Cables and Terminations” (Reference 7), has shown three main causes of cable and connection failures well before a nominal 40 or 60-year service life:

- Cables routed/installed in abnormal configurations, outside the prescribed or normal design guidelines and installation design criteria, may fail due to being exposed to temperatures well above the expected normal ambient temperature. PVC insulated cable insulation failures are the most common cable insulation failures to occur due to high temperature and/or radiation environments.
- Sensitive instrumentation cable insulations (nuclear instrumentation & radiation monitoring) have less tolerance for “loss of material properties” that adversely affect the circuit signals.
- Medium voltage power cable failures occur because of water-treeing (moisture & voltage stress).

Site-specific experience has shown that existing routine switchyard inspections detect loose connections in the switchyard. Existing periodic and routine switchyard inspections preclude failures of connections in the switchyard.

Abnormal plant configurations at Palisades were found to produce localized adverse environments in some specific cases. A corrective action document identified signs of cable jacket damage from improper design/installation that led to a localized adverse environment for the cables. In addition, LER 84-10 resulted from improper design and installation outside expected normal cable configurations. The Corrective Action Program corrected both plant configurations to eliminate the identified localized adverse temperature environments.

A medium-voltage cable failure has occurred at Palisades from the possible effects of water-treeing. LER 96-002 did demonstrate that this commodity group warrants periodic testing to preclude or minimize future failures. Palisades has also experienced that the underground manholes for the medium-voltage cables in-scope of License Renewal have experienced moisture for periods greater than a few days at a time.

One cable commodity-related assessment was conducted to address over-loaded cable trays. This analysis calculated power cable ohmic heating temperatures in those overloaded tray sections and compared it against the respective cable temperature rating to ensure that proper operating conditions exist and are maintained. The results of this analysis were considered when reviewing the plant electrical cables and connections, and were addressed when assessing and identifying those cables requiring aging management during the extended period of operation.

Since the Non-EQ Electrical Commodities Condition Monitoring Program is a new program, no NRC inspection reports, audits, self assessments, or program-specific corrective actions are available.

This element is consistent with NUREG-1801, Section XI.E1, “Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements,” Section XI.E2, “Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits,” and Section XI.E3, “Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements” with the additional guidance provided in ISG-2 and draft ISGs-5, 15, 17 and 18.

Conclusion

The Non-EQ Electrical Commodities Condition Monitoring Program is a new program that uses as its bases, various industry and NRC standards. A number of existing activities are being consolidated into this new program. This program is consistent with NUREG-1801, Section XI.E1, “Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements,” Section XI.E2, “Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits,” and Section XI.E3, “Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements” with the additional guidance provided in ISG-2 and draft ISGs-5, 15, 17 and 18.

The implementation of the Non-EQ Electrical Commodities Condition Monitoring Program provides reasonable assurance that aging effects will be managed such that the electrical commodities within the scope of this program will continue to perform their intended functions consistent with the current licensing bases for the period of extended operation.

B2.1.13 One-Time Inspection Program

Program Description

The One-Time Inspection Program is a new program that addresses potentially long incubation periods for certain aging effects, including various corrosion mechanisms, cracking, and selective leaching, and provides a means of verifying that an aging effect is either not occurring or progressing so slowly as to have negligible effect on the intended function of the structure or component. Hence, the One-Time Inspection Program provides methods for verifying an aging management program is not needed, verifying the effectiveness of an existing program, or determining that degradation is occurring which will require evaluation and corrective action.

The program includes (a) determination of appropriate inspection sample size, (b) identification of inspection locations, (c) selection of examination technique, with acceptance criteria, and (d) evaluation of results to determine the need for additional inspections or other corrective actions. The inspection sample includes locations where the most severe aging effect(s) would be expected to occur. Inspection methods may include visual (or remote visual), surface or volumetric examinations, or other established NDE techniques.

This program is used for a variety of purposes, including the following:

- To verify the effectiveness of water chemistry control for managing the effects of aging in stagnant or low-flow portions of piping exposed to a treated water environment.
- To manage the aging effects of loss of material due to aging mechanisms such as general, crevice, pitting, and galvanic corrosion; selective leaching; and MIC.
- To verify that cracking due to stress corrosion cracking or cyclic loading, in small bore (< 4" NPS) ASME class 1 piping, is not occurring.
- To verify, for components in the Compressed Air System, that there are no aging effects requiring management in the dry air environment.

The following Aging Management Programs credit the One-Time Inspection Program:

- Closed Cycle Cooling Water Program
- Water Chemistry Program

- Fire Protection Program
- System Monitoring Program

NUREG-1801 Consistency

The One-Time Inspection Program is a new program that is consistent with, but includes exceptions to, NUREG-1801, Section XI.M32, “One-Time Inspection.”

The One Time Inspection Program is consistent with NUREG 1801, Section XI.M33, “Selective Leaching of Materials,” and the portions of NUREG-1801, Section XI.M29, “Above Ground Carbon Steel Tanks” that are associated with the thickness measurement of tank bottom surfaces. The balance of Section XI.M29, “Above Ground Carbon Steel Tanks,” is implemented by the System Monitoring Program.

Exceptions to NUREG-1801

One exception is taken to the selected NUREG-1801 Program elements listed below. The specific exception being taken is also discussed in the corresponding element discussions below. It is repeated here for ease of review.

<u>XI.M32, One Time Inspection</u>

1. Section XI.M32.4 of NUREG-1801 requires plant-specific destructive examination of replaced piping due to plant modifications, or NDE that will detect cracking on the inside surfaces of the small bore piping. The current state of technology does not provide for an effective, reliable method of performing volumetric examinations of small bore socket welds. The combination of these one-time volumetric examinations of a 10% sample of Class 1 butt welds, 4” NPS and smaller, and the 100% VT-2 examinations of all Class 1 and 2 HSS socket welds 2” NPS and under each refueling outage meets the intent of NUREG-1800 and -1801 to provide aging management for small-bore class 1 piping. This is generally in accordance with the position approved by the NRC for the Dresden/Quad Cities Nuclear Plants in their License Renewal Application and associated Safety Evaluation Report.

Enhancements

A One Time Inspection Program will be developed and implemented. Features of the program will include:

- Controlling procedure and implementing documents for activities associated with the program. This procedure will include a listing of all SSCs that credit this program for aging management, the aging effects and mechanisms being managed, the materials and environments for the SSCs, grouping and inspection sampling techniques to be used, identification of inspection locations, acceptance criteria, inspection scope expansion criteria, and required actions for inspection results that fall outside acceptance criteria. Inspection results and evaluation of results should be documented, and records retrievable for the life of the plant.

- Controls to ensure that at least 10% of all Class 1 butt welds less than 4” NPS receive a volumetric examination prior to the end of, and within the last 5 years of, the current operating period, with the welds to be inspected chosen from the population of Class 1 HSS butt welds from the RI-ISI Program. In addition, ensure that 100% of all Class 1 and 2 HSS socket welds 2” NPS and under receive a VT-2 visual inspection each refueling outage.

Note that the element descriptions describe the program as it will exist after it has been implemented. The program is scheduled to be implemented prior to the period of extended operation.

Aging Management Program Elements

The key elements of the One-Time Inspection Program are described below. The results of an evaluation of each key element against NUREG-1801, Sections XI.M32, “One-Time Inspection,” XI.M33, “Selective Leaching of Materials,” and the portions of XI.M29, “Above Ground Carbon Steel Tanks” associated with thickness measurement of tank bottom surfaces, are also provided below.

Scope of Program

<u>Scope Related to XI.M32, One Time Inspection</u>

The One-Time Inspection Program will be used to determine the acceptability of components that may be susceptible to various aging effects and to verify that unacceptable degradation is not occurring, thereby validating the effectiveness of an existing aging management program or confirming that there is no need to manage age-related degradation for the period of extended operation.

The scope of this program includes the following:

- Verification of the effectiveness of the Water Chemistry Program for managing the effects of aging in stagnant or low flow portions of piping or components exposed to a treated water environment.
- Verification, for components in the Compressed Air System, that there are no aging effects requiring management in the dry air environment.

- Verification that cracking is not occurring on small-bore (< 4" NPS) class 1 piping.
- Verification of the effectiveness of the Closed Cycle Cooling Water Program for managing the effects of aging in stagnant or low flow portions of closed-cycle systems.
- For susceptible components, verification that selective leaching is not occurring, that selective leaching is occurring so slowly that the intended function will remain intact throughout the period of extended operation, or to determine that the rate of selective leaching will require corrective actions to maintain the intended function.
- Verification that primary coolant pump lube-oil leakoff piping and components are not degrading.
- Verification that existing programs are effective in controlling aging effects for the applicable components.

This element is consistent with the corresponding element of NUREG-1801, Section XI.M32, “One-Time Inspection.”

Scope Related to XI.M33, Selective Leaching

For selective leaching, which is a slow acting corrosion process, visual examinations and hardness tests are performed on a selected set of accessible components from each material type susceptible to selective leaching, including cast iron, brass, bronze, and aluminum-bronze, exposed to raw water, treated water, or groundwater environments, to verify that this aging mechanism is not occurring. Results of examinations and measurements are then evaluated to assess ability of components to perform their intended functions during the period of extended operation.

This element is consistent with the corresponding element of NUREG-1801, Section XI.M33, “Selective Leaching of Materials.”

Scope Related to XI.M29, Above-Ground Carbon Steel Tanks

For storage tanks supported on earthen or concrete foundations, corrosion may occur at inaccessible locations, such as the tank bottom. This program provides for a one-time verification that significant corrosion is not occurring at these inaccessible locations, by providing for measurement of storage tank bottom surface thickness for those storage tanks that credit this program for aging management during the extended operating period.

This element is consistent with the portions of NUREG-1801, Section XI.M29, “Above Ground Carbon Steel Tanks,” that are associated with the thickness measurement of tank bottom surfaces.

Preventive Actions

Preventive Actions Related to XI.M32, One Time Inspection

The one-time inspection activities conducted as part of this program are independent of methods to mitigate or prevent degradation.

This element is consistent with NUREG-1801, Section XI.M32, “One-Time Inspection.”

Preventive Actions Related to XI.M33, Selective Leaching

The one-time visual inspections and any hardness measurements conducted as part of this program are independent of methods to mitigate or prevent degradation. It is noted that for those systems where fluid chemistry is controlled and monitored, this inspection will verify that fluid chemistry is effective in mitigating the applicable aging effects.

This element is consistent with NUREG-1801, Section XI.M33, “Selective Leaching of Materials.”

Preventive Actions Related to XI.M29, Above-Ground Carbon Steel Tanks

There are no measures implemented by the One-Time Inspection program for mitigating or preventing aging of the bottom surface of above-ground carbon steel tanks. It is noted that for those systems where fluid chemistry is controlled and monitored, this internal inspection will verify that fluid chemistry is effective in mitigating the applicable aging effects.

Sealants or caulking at the tank/support structure interfaces, if used, are inspected for degradation. However, Palisades does not credit sealants or caulking for prevention of water intrusion underneath tanks.

This element is consistent with the portions of NUREG-1801, Section XI.M29, “Above-Ground Carbon Steel Tanks,” that are associated with the thickness measurement of tank bottom surfaces.

Parameters Monitored, Inspected, and/or Tested

<u>Parameters Monitored, Inspected, and/or Tested Related to XI.M32, One Time Inspection</u>

For verification of the effectiveness of the Water Chemistry Program and the Closed Cycle Cooling Water Program for stagnant or low flow areas and for verification of the effectiveness of the Diesel Fuel Monitoring and Storage Program, a visual examination or other appropriate NDE methodology, in accordance with the ASME code and 10 CFR 50 Appendix B, will be used to verify that degradation due to the applicable aging effects is not occurring. For verification that corrosion on the surfaces of piping and components is not occurring, visual (or remote visual) or volumetric examinations will be performed.

This element is consistent with NUREG-1801, Section XI.M32, “One-Time Inspection.”

<u>Parameters Monitored, Inspected, and/or Tested Related to XI.M33, Selective Leaching</u>

For selective leaching, which is a slow acting corrosion process, visual examinations and hardness testing are performed on a selected set of accessible components from each material type susceptible to selective leaching, including cast iron, brass, bronze, and aluminum bronze exposed to raw water, treated water, or ground water environments, to verify that this aging mechanism is not occurring. Results of examinations and measurements are then evaluated to assess ability of components to perform their intended functions during the period of extended operation. Any corrective actions will be implemented through the Corrective Action Program.

This element is consistent with NUREG-1801, Section XI.M33, “Selective Leaching of Materials.”

<u>Parameters Monitored, Inspected, and/or Tested Related to XI.M29, Above-Ground Carbon Steel Tanks</u>

The One-Time Inspection Program provides for the inspection and thickness measurement of the bottom surface of selected above ground carbon steel tanks.

This element is consistent with the portions of NUREG-1801, Section XI.M29, “Above-Ground Carbon Steel Tanks,” that are associated with the thickness measurement of tank bottom surfaces.

Detection of Aging Effects

Detection of Aging Effects Related to XI.M32, One Time Inspection

The Palisades One-Time Inspection Program includes a variety of inspection and testing activities that are designed to detect degradation due to aging effects prior to loss of intended function. The examination techniques will be visual, volumetric, or other appropriately established NDE methods. The NDE will be performed by qualified personnel following procedures consistent with the ASME Code and 10 CFR 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.” A representative sample of the component population will be chosen for inspection. The focus will be placed on bounding or lead components. Factors that will be considered when choosing components for inspection are time in service, accessibility, severity of operating conditions, and operating experience.

To verify that the Water Chemistry Program and the Closed Cycle Cooling Water Program are mitigating the applicable aging effects in stagnant or low flow areas, visual examinations or other appropriate NDE methodology will be used when components are inspected.

The One-Time Inspection Program will ensure that at least 10% of all Class 1 butt welds less than 4” NPS receive a volumetric examination prior to the end of the current operating period, with the welds to be inspected chosen from the population of Class 1 HSS butt welds from the RI-ISI Program.

This element is consistent with, but includes an exception to, NUREG-1801, Section XI.M32, “One-Time Inspection.” The exception is:

- Section XI.M32.4 of NUREG-1801 requires plant-specific destructive examination of replaced piping due to plant modifications, or NDE that will detect cracking on the inside surfaces of the small bore piping. The current state of technology does not provide for an effective, reliable method of performing volumetric examinations of small bore socket welds. The combination of these one-time volumetric examinations of a 10% sample of Class 1 butt welds, 4” NPS and smaller, and the 100% VT-2 examinations of all HSS socket welds each refueling outage meets the intent of NUREG-1800 and -1801 to provide aging management for small-bore class 1 piping. This is generally in accordance with the position approved by the NRC for the Dresden/Quad Cities Nuclear Plants in their License Renewal Application and associated Safety Evaluation Report.

Detection of Aging Effects Related to XI.M33, Selective Leaching

For selective leaching which is a slow acting process, visual examinations and hardness tests are performed on a selected set of accessible components from each material type susceptible to selective leaching, including cast iron, brass, bronze, and aluminum-bronze, exposed to raw water, treated water, or ground water environments, to verify this aging mechanism is not occurring.

This element is consistent with NUREG-1801, Section XI.M33, “Selective Leaching of Materials.”

Detection of Aging Effects Related to XI.M29, Above-Ground Carbon Steel Tanks

The One-Time Inspection Program provides for internal inspection and thickness measurements of the bottom surface of selected above-ground carbon steel tanks.

This element is consistent with the portions of NUREG-1801, Section XI.M29, “Above-Ground Carbon Steel Tanks,” that are associated with the thickness measurement of tank bottom surfaces.

Monitoring and Trending

Monitoring and Trending Related to XI.M32, One Time Inspection

This new One-Time Inspection Program does not provide for monitoring and trending. However, follow up examinations will be required if unacceptable conditions are discovered, thus expanding the sample size and locations of inspections.

This element is consistent with NUREG-1801, Section XI.M32, “One-Time Inspection.”

Monitoring and Trending Related to XI.M33, Selective Leaching

There are no monitoring and trending activities for the one-time visual inspections, or any hardness tests performed, unless inspection or test results indicate degradation. However, follow up examinations will be required if unacceptable conditions are discovered, thus expanding the sample size and locations of inspections.

This element is consistent with NUREG-1801, Section XI.M33, “Selective Leaching of Materials.”

Monitoring and Trending Related to XI.M29, Above-Ground Carbon Steel Tanks

The effects of corrosion of the inaccessible external surfaces of above-ground carbon steel tanks are detectable by thickness measurement of the tank bottom and are monitored and trended if significant material loss is detected during the one-time inspection.

This element is consistent with the portions of NUREG-1801, Section XI.M29, “Above-Ground Carbon Steel Tanks,” that are associated with the thickness measurement of tank bottom surfaces.

Acceptance Criteria

Acceptance Criteria Related to XI.M32, One Time Inspection

Any indications of degradation or unacceptable thickness measurements are evaluated through the corrective action program. The need to increase the sample population will also be evaluated when indications or relevant conditions of degradation are found.

This element is consistent with NUREG-1801, Section XI.M32, “One-Time Inspection.”

Acceptance Criteria Related to XI.M33, Selective Leaching

Any indications of degradation or unacceptable thickness measurements are evaluated through the corrective action program. The need to increase the sample population will also be evaluated when indications or relevant conditions of degradation are found.

This element is consistent with NUREG-1801, Section XI.M33, “Selective Leaching of Materials.”

Acceptance Criteria Related to XI.M29, Above-Ground Carbon Steel Tanks

Any indications of degradation or unacceptable thickness measurements are evaluated through the corrective action program. The need to increase the sample population will also be evaluated when indications or relevant conditions of degradation are found.

This element is consistent with the portions of NUREG-1801, Section XI.M29, “Above-Ground Carbon Steel Tanks,” that are associated with the thickness measurement of tank bottom surfaces.

Corrective Actions, Confirmation Process, Administrative Controls

These elements are consistent with the corresponding NUREG-1801 aging management program elements. See Section B1.2 for further discussion.

Operating Experience

Palisades has a comprehensive Operating Experience Program (OEP) that monitors industry issues/events and assesses these for applicability to its own operations. In addition, the Palisades Corrective Action Program (CAP) is used to track, trend and evaluate plant issues/events. Those issues and events, whether external or plant specific, that are potentially significant to the One-Time Inspection Program at Palisades are evaluated.

The One-Time Inspection Program is a new program to be implemented before the current operating license expires. The NDE inspection methods that will be used, such as visual (or remote visual), surface or volumetric, or other established techniques, are consistent with industry practice.

Using the OEP and CAP to focus on industry and plant operating experience ensures that One-Time Inspection Program issues will be addressed in a timely manner and that age related deterioration of SSC within the scope of the One-Time Inspection Program will be effectively managed.

This element is consistent with NUREG-1801, Section XI.M32, “One-Time Inspection.”

This element is consistent with NUREG-1801, Section XI.M33, “Selective Leaching of Materials.”

This element is consistent with the portions of NUREG-1801, Section XI.M29, “Above-Ground Carbon Steel Tanks,” that are associated with the thickness measurement of tank bottom surfaces.

Conclusion

The One-Time Inspection Program is a new program that uses as its bases, various industry and NRC standards. This program is consistent with, but includes exceptions to, NUREG-1801, Section XI.M32, “One-Time Inspection.” This program is consistent with NUREG-1801, Section XI.M33, “Selective Leaching of Materials.” This program is consistent with the portions of NUREG-1801, Section XI.M29, “Above-Ground Carbon Steel Tanks,” that are associated with the thickness measurement of tank bottom surfaces. A summary of each exception is provided above in the discussion of the affected program element.

The implementation of the One-Time Inspection Program provides reasonable assurance that aging effects will be managed such that SSCs within the scope of this program will continue to perform their intended functions consistent with the current licensing bases for the period of extended operation.

B2.1.14 Open Cycle Cooling Water Program

Program Description

The Open Cycle Cooling Water Program is an existing program that manages aging effects such as loss of material due to general, pitting, and crevice corrosion, erosion, MIC, and loss of heat transfer due to biological/corrosion product fouling (e.g., sedimentation, silting) caused by exposure of internal surfaces of metallic components to raw, untreated (e.g., service) water. The program scope includes activities to manage aging in the Service Water System (SWS) and Circulating Water system (CWS). The aging effects are managed through (a) monitoring and control of biofouling, (b) flow balancing and flushing, (c) heat exchanger testing (d) routine inspection and maintenance program activities to ensure that aging effects do not impair component intended function. Inspection methods include visual (VT), ultrasonic (UT), radiographic (RT), and eddy current (ECT). This program is responsive to NRC GL 89-13.

Biofouling monitoring is accomplished by having the service water pump intake bay inspected each refueling outage by divers for biofouling species. Periodic maintenance activities for the inspection of safety-related heat exchangers include steps to take samples for analysis and inspection for biofouling species and Microbiologically Influenced Corrosion (MIC). Biofouling control is accomplished by chlorinating the SWS and CWS to mitigate the effects of biological fouling. Chemical concentration conforms to the requirements set by the Michigan Department of Environmental Quality in Palisades' National Pollutant Discharge Elimination System (NPDES) permit MI0001457.

To verify that components receive FSAR required flows, the Service Water System is balanced each refueling outage as per plant procedures. The performance of this test assures that flows set during testing meet the requirements set by the plant's FSAR for a Design Basis Accident. The testing also fulfills the recommended actions that other components be tested once per refueling outage to assure that components are not fouled or clogged. The actions to periodically flush piping and flow balance the Service

Water System fulfill the recommendation suggested by control technique 2 of GL 89-13 to assure that fouling sediment is flushed from piping.

The heat exchanger testing program main goals are to establish a framework to assess plant heat exchanger condition:

- To select heat exchangers requiring immediate attention and apply realistic plugging criteria.
- To identify problems and perform repairs where necessary to maintain the integrity of heat exchangers.
- To establish a sound basis for heat exchanger inspection intervals.
- To apply effective Non-Destructive Evaluation and Testing (NDE)
- To utilize a database management program to acquire, analyze, trend, and store data.
- To plan heat exchanger replacement.

Palisades has established a routine inspection and maintenance monitoring program for service water piping and components to ensure that corrosion, erosion, silting, and biofouling cannot degrade the performance of the safety related systems supplied by service water to where they are unable to perform their intended functions. This program has the following objectives:

- To remove excessive accumulations of biofouling agents, corrosion products, and silt.
- To repair defective protective coatings and corroded piping and components that could adversely affect performance of their intended safety functions.

NUREG-1801 Consistency

The Open Cycle Cooling Water Program is consistent with NUREG-1801, Section XI.M20, “Open-Cycle Cooling Water System.”

Exceptions to NUREG-1801

None.

Enhancements

None

Aging Management Program Elements

The key elements of the Open Cycle Cooling Water Program are described below. The results of an evaluation of each key element against NUREG-1801, Section XI.M20, “Open-Cycle Cooling Water System” are also provided.

Scope of Program

The purpose of the Open Cycle Cooling Water Program is to perform periodic inspections and/or tests to detect degradation, monitoring of system integrity, and evaluations required to maintain reliability and operability of systems and components within its scope. This is accomplished by following the guidelines provided in GL 89-13, which include (a) monitoring and control of biofouling, (b) flow balancing and flushing, (c) heat exchanger testing, and (d) routine inspection and maintenance program activities to ensure that aging effects due to fouling and various corrosion mechanisms do not impair component intended function.

GL 89-13 required a walkdown inspection to ensure compliance with the licensing basis and a review of maintenance, operating, and training practices and procedures. These requirements were satisfied as part of the response to the generic letter. These requirements are not associated with the on-going management of aging effects and there are no corresponding activities in the Open Cycle Cooling Water Program.

This element is consistent with NUREG-1801, Section XI.M20, “Open-Cycle Cooling Water System.”

Preventive Actions

Plant procedures implement commitments to GL 89-13 at Palisades. This program includes activities for condition and performance monitoring, and biofouling monitoring and control. Activities are in place to have the service water pump intake bay inspected and cleaned each refueling outage by divers. Periodic maintenance activities for the inspection of safety related heat exchangers include steps for sampling and analysis for biofouling species and MIC. Additionally, programs are in place for Chlorination and molluscicide treatment of the Service Water and Circulating Water Systems.

Sections of SWS that have been identified as being susceptible to fouling due to sediment settling have been evaluated and are periodically flushed where necessary. Radiography (RT) is utilized in selected piping sections to validate that significant silting/sanding or corrosion product buildup is not occurring that would cause flow blockage in excess of design requirements.

NUREG-1801, Section XI.M20.2 states that, “The system components are constructed of appropriate materials and lined or coated to protect the underlying metal surfaces from being exposed to aggressive cooling water environments.” NUREG-1801, Section XI.M20.3 states that, “Cleanliness and material integrity of piping, components, heat exchangers, and their internal linings or coatings (when applicable) that are part of the OCCW system or that are cooled by the OCCW system are periodically inspected, monitored or tested to ensure heat transfer capabilities”. Neither SWS nor CWS has internal linings or coatings on in-scope components. The CWS and SWS piping included within the scope of this program are constructed of carbon steel that is not lined or coated on the interior side.

This element is consistent with NUREG-1801, Section XI.M20, “Open-Cycle Cooling Water System.”

Parameters Monitored, Inspected, and/or Tested

The parameters monitored, inspected, or tested vary depending on the component and are based on commitments to GL 89-13 and operating experience. Some heat exchangers are visually inspected, some have the temperature of the affected components monitored for trending and acceptability of performance of Service Water System cooling, some are inspected to verify that service water is flowing through the coolers, and some are tested for heat transfer capability. Pressure drop across some components is measured and trended, while some components are periodically cleaned and inspected.

Various actions are undertaken to verify system or component performance will not be compromised by the accumulation of biofouling agents, corrosion products, and silt. Cleanliness and material integrity of piping, components, and heat exchangers are periodically inspected, monitored, or tested to ensure heat transfer capabilities. The program ensures (a) removal of biofouling agents, corrosion products, and silt, and (b) detection of corroded piping and components that could adversely affect the performance of their intended safety function. The activities performed under this program were implemented as a result of GL 89-13, and their continued performance will ensure that the Service Water System will continue to perform its intended function.

This element is consistent with NUREG-1801, Section XI.M20, “Open-Cycle Cooling Water System.”

Detection of Aging Effects

Plant procedures implement commitments to GL 89-13, which requires establishing a routine inspection and maintenance program for Service Water System piping and components such that corrosion, erosion, silting, and biofouling do not degrade the performance of the safety-related systems supplied by service water. The inspection program verifies that the system will perform its design basis heat removal function by ensuring that sufficient water flow is maintained.

In addition to the inspections described above, the plant has a biofouling and Microbiologically Influenced Corrosion (MIC) control program described in plant procedures. The program consists of system inspections, vendor reports, deposit, and coupon analyses, etc., which are performed and subsequently documented in corrosion reports to aid in early detection and mitigation of biofouling and MIC-related concerns.

This element is consistent with NUREG-1801, Section XI.M20, “Open-Cycle Cooling Water System.”

Monitoring and Trending

The scope of inspections and heat transfer tests are in accordance with Palisades' commitments under GL 89-13. Selected components have the service water flow checked, and/or are flushed, to reduce biofouling or sedimentation build-up. Component degradation, such as pipe wall thinning and silt build-up, is measured and recorded to predict the expected remaining life of the component such that corrective actions can be taken prior to a loss of intended function.

Examination and Maintenance Frequency and Locations:

Safety related heat exchangers are inspected, cleaned and tested at various periodicities specified in plant procedures. The service water pump intake bay is inspected each refueling outage by divers for biofouling species. Selected locations in the Service Water system are flushed periodically as described in plant procedures. The SWS and Circulating Water System (CWS) are normally chlorinated when on line or during outages to control MIC and biofouling. As water conditions change, continuous chlorination and/or molluscicide treatments will be performed on an as needed basis.

Analysis of Results:

Results for visual, chemical and mechanical inspection of heat exchangers are documented and retained in plant records. Inspection results of safety-related service water piping are documented and retained in plant records. A summary Raw Water Corrosion Program Report on the effectiveness of the program is completed after each refueling outage and includes the previous operational cycle. Inspection activity results are reviewed for trending purposes to adjust inspection, performance testing, and maintenance frequencies, as necessary.

This element is consistent with NUREG-1801, Section XI.M20, "Open-Cycle Cooling Water System."

Acceptance Criteria

The acceptance criteria are specified in the procedures that control the inspections of components. Pipe and tubing wall thickness is measured and compared to minimum required wall thickness. Biofouling is removed or reduced as part of the activities performed under this program. Acceptance criteria are based on maintaining the system free of significant sediment and biofouling build-up, and able to perform its intended functions.

This element is consistent with NUREG-1801, Section XI.M20, "Open-Cycle Cooling Water System."

Corrective Actions, Confirmation Process, Administrative Controls

These elements are consistent with the corresponding NUREG-1801 aging management program elements. See Section B1.2 for further discussion.

Operating Experience

Palisades has a comprehensive Operating Experience Program (OEP) that monitors industry issues/events, and assesses these for applicability to its own operations. In addition, the Palisades Corrective Action Program (CAP) is used to track, trend and evaluate plant issues/events. Those issues and events, whether external or plant specific, that are potentially significant to the Open Cycle Cooling Water Program at Palisades are evaluated. The Open Cycle Cooling Water Program is augmented, as appropriate, if these evaluations show that program changes will enhance program effectiveness.

A review of industry operating experience associated with the Open Cycle Cooling Water Program and aging reveals issues and instances related to:

- Accumulations of silt and corrosion products in service water piping, valves, and heat exchangers
- Accumulation of biological growth (mussels, clams, and shells) in service water piping, valves and heat exchangers
- MIC causing pitting attack of carbon steel and stainless steel service water piping, pump casings, and 90/10 Cu/Ni heat exchanger tubes

A review of plant specific operating experience related to the Open Cycle Cooling Water Program and aging revealed that the following issues have been addressed:

- Defective tubes in the Main Condenser that required plugging due to MIC
- Control Room Condensing Unit Condenser Drain Plug severely corroded due to MIC
- Large Zebra Mussel accumulation near traveling screens and inside intake piping
- Blockage of heat exchanger and cooler tubing
- Corroded service water piping at threaded connections
- Pinhole leaks in service water piping due to MIC
- Switch failure due to sediment and corrosion (galvanic) blocking sensing line
- Tubercles growing in carbon steel service water piping
- Erosion of pipes, cooling coils, and heat exchanger tubes causing service water leaks

Using the OEP and CAP to focus on industry and plant operating experience ensures that Open Cycle Cooling Water Program issues are addressed in a timely manner and that age related deterioration of SSC within the scope of the Open Cycle Cooling Water Program will be effectively managed throughout the license renewal period.

The Palisades Open Cycle Cooling Water Program has demonstrated on several occasions that it provides reasonable assurance that aging effects are being managed for Open Cycle Cooling Water Program SSC. This has been demonstrated through NRC inspection reports, audits, self-assessments, and the Corrective Action Program.

This element is consistent with NUREG-1801, Section XI.M20, “Open-Cycle Cooling Water System.”

Conclusion

The Open Cycle Cooling Water Program is an existing program that uses as its bases, various industry and NRC standards. This program is consistent with NUREG-1801, Section XI.M20, “Open-Cycle Cooling Water System.”

Reports of recent internal and external assessments of the program have been reviewed. These reports demonstrate that the program has effectively identified and dispositioned issues that could have led to degraded conditions. It is concluded that the Open Cycle Cooling Water Program has been effective in maintaining the intended functions of the affected long-lived, passive SSCs.

The continued implementation of the Open Cycle Cooling Water Program provides reasonable assurance that aging effects will be managed such that SSCs within the scope of this program will continue to perform their intended functions consistent with the current licensing bases for the period of extended operation.

B2.1.15 Overhead Load Handling Systems Inspection Program

Program Description

The Overhead Load Handling Systems Inspection Program is an existing program that provides for inspections of the structural components and rails of cranes and fuel handling machines associated with heavy load handling that are subject to the requirements of NUREG-0612 and are within the scope of license renewal requiring aging management. For Palisades these are the Containment Building Polar Crane, the Spent Fuel Pool Overhead Crane, the Containment Building jib and boom cranes, and the reactor and spent fuel pool fuel handling machines. These cranes comply with the Maintenance Rule requirements provided in 10 CFR 50.65. The Overhead Load Handling Systems Inspections Program is primarily focused on structural components that make up the bridge and trolley of the overhead cranes that are within the scope of NUREG-0612.

NUREG-1801 Consistency

The Overhead Load Handling Systems Inspection Program is an existing program that is consistent with, but includes exceptions to, NUREG-1801, Section XI.M23, “Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems.”

Exceptions to NUREG-1801

One exception is taken to selected NUREG-1801 Program elements. The specific exception is also discussed in the corresponding element discussion. The exception is repeated here for ease of review.

1. Parameters Monitored, Inspected, and/or Tested: NUREG-1801, Section XI.M23, Parameters Monitored/Inspected section, states, "The number and magnitude of lifts made by the crane are also reviewed." Palisades does not track the number and magnitude of all lifts made by cranes. Administrative controls are implemented to ensure that only allowable loads are handled and fatigue failure of structural elements is not expected. A time-limited aging analysis report (Section 4.7.1) concludes that, at the current service level, there are no fatigue concerns for the Containment polar crane and the Spent Fuel Pool crane as both the Containment polar and Spent Fuel Pool crane can not realistically approach the 20,000 to 100,000 rated lifts assumed for its design evaluation during the extended operating period. Frequent inspections of cranes for indications of functional failures are conducted. However, Palisades does track the number and magnitude of lifts made that exceed the rated capacity of the cranes. These are called "engineered lifts," follow the requirements of ANSI B30.2, and are generally only used for the polar crane lifting the reactor head with lead shielding. These lifts are numerically restricted, and evaluated by engineering analysis. This exception is generally in accordance with the position taken by the Dresden and Quad Cities Nuclear Power Stations in their License Renewal Application, and subsequently approved by the Nuclear Regulatory Commission in the associated Safety Evaluation Report.

Enhancements

One enhancement is planned to bring the Overhead Load Handling Systems Inspection Program into conformance (with exception noted above) with the NUREG-1801 program description. The enhancement is:

Scope of Program, Detection of Aging Effects, and Acceptance Criteria: Revise crane and fuel handling machine inspection procedures to specifically inspect for general corrosion on passive components making up the bridge, trolley, girders, etc., and to inspect rails of Bridge Cranes for wear. Revision should also include documentation of results of these inspections, acceptance criteria, and qualification requirements for inspectors and crane supervisors.

Note that the element descriptions describe the program as it will exist after the identified enhancements have been implemented. Enhancements are scheduled for completion prior to the period of extended operation.

Aging Management Program Elements

The key elements of the Overhead Load Handling Systems Inspection Program are described below. The results of an evaluation of each key element against NUREG-1801, Section XI.M23, “Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems” are also provided.

Scope of Program

This program manages the effects of general corrosion on the crane and trolley structural components, and the effects of wear on the rails, for those overhead cranes, and fuel handling machines that are within the scope of license renewal and subject to the requirements of NUREG-0612. The aging effects/mechanisms managed by the Overhead Load Handling Systems Inspection Program are loss of material due to general corrosion and/or wear.

This element is consistent with NUREG-1801, Section XI.M23, “Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems.”

Preventive Actions

There are no preventive actions associated with this program. This program is an inspection program.

This element is consistent with NUREG-1801, Section XI.M23, “Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems.”

Parameters Monitored, Inspected, and/or Tested

For the SSC within the scope of license renewal, the inspections include structural bolting, rail wear, and corrosion of structural components of the bridge and trolley.

This element is consistent with, but includes an exception to, NUREG-1801, Section XI.M23, “Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems.” The exception is:

- NUREG-1801, Section XI.M23, Parameters Monitored/Inspected section, states, “The number and magnitude of lifts made by the crane are also reviewed.” Palisades does not track the number and magnitude of all lifts made by cranes. Administrative controls are implemented to ensure that only allowable loads are handled and fatigue failure of structural elements is not expected. A time-limited aging analysis report (Section 4.7.1) concludes that, at the current service level, there are no fatigue concerns for the Containment polar and the Spent Fuel Pool cranes, as both the Containment polar and Spent Fuel Pool cranes can not realistically approach the 20,000 to 100,000 rated lifts assumed for its design evaluation during the 60 year extended operating period. Frequent inspections of cranes for indications of functional failures are conducted. However, Palisades does track the number and magnitude of lifts made that exceed the rated capacity of the cranes. These are called “engineered lifts,” follow the requirements of ANSI B30.2, and are generally only used for the polar crane lifting the reactor head with lead shielding. These lifts are numerically restricted, and evaluated by engineering analysis. This exception is generally in accordance with the position taken by the Dresden and Quad Cities Nuclear Power Stations in their License Renewal Application, and subsequently approved by the Nuclear Regulatory Commission in the associated Safety Evaluation Report.

Detection of Aging Effects

The Palisades Overhead Load Handling Systems Inspection Program includes visual inspection and testing activities that are designed to detect degradation due to aging effects prior to loss of intended function.

Crane rails and structural components are visually inspected on a routine basis for degradation. Functional tests are also performed to assure their integrity prior to use. Quarterly and Annual Inspections and Functional Tests of overhead load handling systems are performed under the direct supervision of trained Palisades Supervisors.

This element is consistent with NUREG-1801, Section XI.M23, “Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems.”

Monitoring and Trending

Monitoring and trending are not required as part of the Overhead Load Handling Systems Inspection Program.

This element is consistent with NUREG-1801, Section XI.M23, “Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems.”

Acceptance Criteria

The Containment Polar Crane and the Spent Fuel Pool Overhead Crane were originally designed and installed in accordance with Electric Overhead Crane Institute (EOCI) Specification #61. The subsequent NUREG-0612 heavy loads evaluation of the polar crane was performed to CMAA 70 (1975). The minimally-rated components are CMAA 70 Service Level A.

The Spent Fuel Pool Crane was upgraded to 110 T for dry cask storage. This included a NUREG-0612 evaluation to CMAA-70 Service Level A design criteria, and replacement of the trolley with a 110 T single-failure-proof trolley.

Any significant visual indication of loss of material due to corrosion of crane structural members or rail wear is evaluated according to vendor recommendations and/or applicable industry good practice and standards.

This element is consistent with NUREG-1801, Section XI.M23, “Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems.”

Corrective Actions, Confirmation Process, Administrative Controls

These elements are consistent with the corresponding NUREG-1801 aging management program elements. See Section B1.2 for further discussion.

Operating Experience

Palisades has a comprehensive Operating Experience Program (OEP) that monitors industry issues/events and assesses these for applicability to its own operations. In addition, the Palisades Corrective Action Program (CAP) is used to track, trend and evaluate plant issues/events. Those issues and events, whether external or plant specific, that are potentially significant to the Overhead Load Handling Systems Inspection Program at Palisades are evaluated. The Overhead Load Handling Systems Inspection Program is augmented, as appropriate, if these evaluations show that program changes will enhance program effectiveness.

Using the OEP and CAP to focus on industry and plant operating experience ensures that Overhead Load Handling Systems Inspection Program issues are addressed in a timely manner and that age related deterioration of SSC within the scope of the Overhead Load Handling Systems Inspection Program will be effectively managed throughout the license renewal period.

A review of industry operating experience associated with the Overhead Load Handling Systems Inspection Program revealed no issues and instances related to aging.

A review of plant specific operating experience related to the Overhead Load Handling Systems Inspection Program and aging revealed that the following issues have been addressed:

- Damage to Spent Fuel Handling Machine Hoist Cable
- Movement identified on Containment Hatch Crane's base structure
- Containment crane rail attachment bolt grout pads cracked
- Load limit of containment crane exceeded during head lift
- Containment crane bridge rail splice weld cracks

The Palisades Overhead Load Handling Systems Inspection Program has demonstrated that it provides reasonable assurance that aging effects are being managed for the Overhead Load Handling Systems Inspection Program SSC. No recent external or internal audits/assessments were conducted on this program.

This element is consistent with NUREG-1801, Section XI.M23, “Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems.”

Conclusion

The Overhead Load Handling Systems Inspection Program is an existing program that uses as its bases, various industry and NRC standards. This program is consistent with, but includes exceptions to, NUREG-1801, Section XI.M23, “Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems.” A summary of each exception is provided in the discussion of the affected program attribute above.

It is concluded that the Overhead Load Handling Systems Inspection Program has been effective in maintaining the intended functions of the affected long-lived, passive SSCs.

The continued implementation of the Overhead Load Handling Systems Inspection Program provides reasonable assurance that aging effects will be managed such that SSCs within the scope of this program will continue to perform their intended functions consistent with the current licensing bases for the period of extended operation.

B2.1.16 Reactor Vessel Integrity Surveillance Program

Program Description

The Reactor Vessel Integrity Surveillance Program is an existing program that manages the aging effect reduction of fracture toughness due to neutron embrittlement of the low alloy steel reactor vessel. Monitoring methods will be in accordance with 10 CFR 50, Appendix H. This program includes (a) capsule insertion, withdrawal and materials testing/evaluation, (including upper shelf energy and RT_{NDT} determinations), (b) fluence and uncertainty calculations, (c) monitoring of Effective Full Power Years (EFPY), (d) development of pressure temperature limitations, and (e) determination of low temperature overpressure protection (LTOP) set points. The program ensures the reactor vessel materials (a) meet the fracture toughness requirements of 10 CFR 50, Appendix G, and (b) have adequate margins against brittle fracture caused by Pressurized Thermal Shock (PTS) in accordance with 10 CFR 50.61.

NUREG-1801 Consistency

The Reactor Vessel Integrity Surveillance Program is consistent with NUREG-1801, Section XI.M31, “Reactor Vessel Surveillance.” The Reactor Vessel Integrity Surveillance Program is also an existing program that contains the applicable elements of an acceptable aging management program as described in Branch Technical Position RLSB-1, “Aging Management Review-Generic,” which is included in Appendix A of NUREG-1800, “Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants.”

Exceptions to NUREG-1801

None.

Enhancements

1. The Reactor Vessel Integrity Surveillance Program will ensure that pressure-temperature and LTOP curves are updated to reflect the additional neutron fluence accumulated during the extended operating period. Curves will be updated and submitted to NRC for approval prior to the period of extended operation

2. Document and establish the requirement to save and store all pulled and tested reactor vessel surveillance capsules for future reconstitution use.

3. Evaluate and revise as necessary, the surveillance capsule withdrawal and testing schedule of FSAR Table 4-20 such that at least one capsule remains in the reactor vessel

and is tested during the period of extended operation to monitor the effects of long-term exposure to neutron irradiation.

4. Develop a program level procedure to implement and control Technical Specification and FSAR activities associated with the Reactor Vessel Integrity Surveillance Program, including activities associated with surveillance capsules, pressure-temperature limit curves, LTOP setpoints, neutron embrittlement calculation methodology, neutron fluence calculations and control, and documentation requirements. The procedure title should be “Reactor Vessel Integrity Surveillance Program.”

Aging Management Program Elements

The elements of the Reactor Vessel Integrity Surveillance Program are described below. The results of an evaluation of each element against NUREG-1801, Section XI.M31, “Reactor Vessel Surveillance,” are provided below. However, the NUREG-1801 program does not define the ten elements similar to other programs. Therefore, the results of an evaluation of each key element against the appropriate ten elements described in Branch Technical Position RLSB-1, “Aging Management Review-Generic,” which is included in Appendix A of NUREG-1800, “Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants,” is also provided below.

Scope of Program

The Reactor Vessel Integrity Surveillance Program applies to the Palisades reactor pressure vessel.

The program controls the development of surveillance capsule insertion and withdrawal schedules and capsule materials testing. The surveillance program meets the requirements of ASTM E 185. The results of capsule materials testing, fluence analysis, and EFPY monitoring are used to predict the effects of neutron embrittlement through the end of extended life (EOEL). The results of capsule tests, fluence analysis, and EFPY monitoring are also used to determine compliance with the PTS screening criteria of 10 CFR 50.61.

Fluence and Uncertainty Calculations are performed for the Palisades reactor vessel in accordance with RG 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence.” The results are used as an input to embrittlement predictions.

EFPY monitoring is accomplished using operations data for the Palisades reactor. The results are used to project the fluence corresponding to specific values of EFPY.

The Reactor Vessel Integrity Surveillance Program controls the development of pressure and temperature limit curves in accordance with 10 CFR 50, Appendix G requirements. The methods of ASME Section XI, Appendix G are used to determine pressure and temperature limits. The fracture toughness used in calculating PT limits is determined as a function of the difference in temperature from RT_{NDT}. RG 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials” is used to determine RT_{NDT}.

The Reactor Vessel Integrity Surveillance Program requires the calculation of LTOP set points for the Palisades primary coolant system. These set points ensure that an LTOP event will not increase the probability of brittle fracture of the reactor vessel. LTOP set points include the maximum pressure allowed before the LTOP system actuates to relieve the pressure, and the temperature below which the LTOP system must be effective. These pressures and temperatures are determined using the method of ASME Section XI, Appendix G.

This element is consistent with NUREG-1801, Section XI.M31, “Reactor Vessel Surveillance,” and with the corresponding element described in the Branch Technical Position.

Preventive Actions

This surveillance program determines neutron embrittlement for upper shelf energy and pressure temperature limits for 60 years in accordance with the RG 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials,” and is not preventive in nature. However, there are certain activities within this program that provide necessary information to prevent operation of the reactor pressure vessel outside limits.

Calculations demonstrating that the Palisades reactor vessel meets the PTS screening criteria of 10 CFR 50.61 ensure the probability of brittle fracture of the reactor vessels during a PTS event is acceptably low.

Fluence and uncertainty calculations do not constitute preventative actions.

EFPY monitoring is a monitoring activity, not a preventative action.

The development of, and operation within, P-T limit curves minimizes the probability of brittle fracture of the reactor vessel during normal operation.

The LTOP system with the actuation setpoints and operational restrictions established by the LTOP analysis, minimizes the probability of an LTOP event, and therefore, helps to minimize the probability of reactor vessel brittle fracture.

This element is consistent with NUREG-1801, Section XI.M31, “Reactor Vessel Surveillance,” and with the corresponding element described in the Branch Technical Position.

Parameters Monitored, Inspected, and/or Tested

The program monitors the effects of neutron irradiation on the Palisades reactor vessel beltline materials. Fracture toughness of beltline materials is indirectly monitored through measurement of the impact energy of Charpy V Notch (CV) specimens, made from representative materials from the reactor vessel beltline regions. The surveillance capsules also contain neutron dosimetry that monitors the amount of neutron fluence received by the test specimens. The vessel fluence is then used to calculate RT_{NDT} using the method of RG 1.99, Revision 2.

Effective Full Power Years (EFPY) are monitored and used to predict the fluence that the vessel will accumulate at some future time, which is then used to predict change in RT_{NDT} and upper shelf energy (USE).

LTOP system relief valve operation is monitored to determine whether an LTOP event could have occurred had the LTOP system been inoperable. Operation within the P-T limits is also monitored.

This element is consistent with NUREG-1801, Section XI.M31, “Reactor Vessel Surveillance,” and with the corresponding element described in the Branch Technical Position.

Detection of Aging Effects

Aging effects are detected through testing of surveillance materials. CV tests are performed to determine the decrease in USE and increase in transition temperature RT_{NDT}, for materials that closely match reactor vessel beltline.

This element is consistent with NUREG-1801, Section XI.M31, “Reactor Vessel Surveillance,” and with the corresponding element described in the Branch Technical Position.

Monitoring and Trending

Monitoring of reactor vessel beltline fracture toughness is accomplished through testing of surveillance specimens from surveillance capsules that are periodically withdrawn from the vessel. Trending is accomplished through the RG 1.99, Revision 2 methods for projection of RT_{NDT} and USE. Projection of the increase in RT_{NDT} and the decrease in USE provides early indication if the fracture toughness properties of the reactor vessel beltline materials will fail to meet regulatory requirements. The RT_{NDT} projection is compared to the PTS screening criteria of 270°F for plates, forgings, and axial welds and 300°F for circumferential welds specified in 10 CFR 50.61. USE projections are compared against the requirement to maintain 50 ft-lbs or greater given by 10 CFR 50, Appendix G (see Section 4.2.1).

Fluence estimates from capsules are trended to verify that results are adequately represented by fluence models, and to project fluence for future dates.

EFPY are monitored and trended to allow the EFPY for particular calendar dates, such as the end of the current and extended license periods, to be projected, and to establish deadlines for revising P-T curves that are valid only to a particular number of EFPY. These projections will be extended to a number of EFPY corresponding to the end of life - extended (EOLE).

This element is consistent with NUREG-1801, Section XI.M31, “Reactor Vessel Surveillance,” and with the corresponding element described in the Branch Technical Position.

Acceptance Criteria

The upper shelf energy of the most limiting material in the reactor vessel beltline must remain above 50 ft-lbs until the end of extended life, using the methods of RG 1.99, Revision 2. The RT_{NDT} of the most limiting material in the reactor vessel beltline must not exceed the PTS screening criteria specified by 10 CFR 50.61 (270°F for plates, forgings, and axial welds, and 300°F for circumferential welds).

The acceptance criteria for P-T curves is that the flaw stability criteria of ASME Section XI, Appendix G, are met for all normal operating conditions as required by 10 CFR 50, Appendix G.

This element is consistent with NUREG-1801, Section XI.M31, “Reactor Vessel Surveillance,” and with the corresponding element described in the Branch Technical Position.

Corrective Actions, Confirmation Process, Administrative Controls

These elements are consistent with the corresponding NUREG-1801 aging management program elements and with the corresponding elements described in the Branch Technical Position. See Section B1.2 for further discussion.

Operating Experience

Palisades has a comprehensive Operating Experience Program (OEP) that monitors industry issues/events and assesses these for applicability to its own operations. In addition, the Palisades Corrective Action Program (CAP) is used to track, trend and evaluate plant issues/events. Those issues and events, whether external or plant specific, that are potentially significant to the Reactor Vessel Integrity Surveillance Program at Palisades are evaluated. The Reactor Vessel Integrity Surveillance Program is augmented, as appropriate, if these evaluations show that program changes will enhance program effectiveness.

Using the OEP and CAP to focus on industry and plant operating experience ensures that Reactor Vessel Integrity Surveillance Program issues are addressed in a timely manner and that age related deterioration of SSC within the scope of the Reactor Vessel Integrity Surveillance Program will be effectively managed throughout the license renewal period.

A search of industry and plant operating experience related to the Reactor Vessel Integrity Surveillance Program was performed. This search revealed the following issues relevant to Palisades Reactor Vessel Integrity Surveillance Program:

GL 92-01, Revision 1, “Reactor Vessel Structural Integrity,” and Supplement 1 to GL 92-01, Revision 1, “Reactor Vessel Structural Integrity.” Palisades' response to these documents has been incorporated into the Reactor Vessel Integrity Surveillance Program.

A review of NRC Inspection Reports, QA Audit/Surveillance Reports, and Self Assessments since 1999 revealed no issues or findings that could impact the effectiveness of the Reactor Vessel Surveillance Program.

This element is consistent with NUREG-1801, Section XI.M31, “Reactor Vessel Surveillance,” and with the corresponding element described in the Branch Technical Position.

Conclusion

The Reactor Vessel Integrity Surveillance Program is an existing program that uses as its bases, various industry and NRC standards. This program is consistent with NUREG-1801, Section XI.M31, “Reactor Vessel Surveillance,” and with the corresponding elements described in the Branch Technical Position.

Reports of recent internal and external assessments of the program have been reviewed. These reports demonstrate that the program has effectively identified and dispositioned issues that could have led to degraded conditions. It is concluded that the Reactor Vessel Integrity Surveillance Program has been effective in maintaining the intended functions of the affected long-lived, passive SSCs.

The continued implementation of the Reactor Vessel Integrity Surveillance Program provides reasonable assurance that aging effects will be managed such that SSCs within the scope of this program will continue to perform their intended functions consistent with the current licensing bases for the period of extended operation.

B2.1.17 Reactor Vessel Internals Inspection Program

Program Description

The Reactor Vessel Internals Inspection Program is an existing program that manages the aging effects for reactor vessel internals. The program provides for (a) Inservice Inspection (ISI) in accordance with ASME Section XI requirements, including examinations performed during the 10-year ISI examination; (b) Participation in industry initiatives to evaluate the significance of void swelling; (c) Monitoring and control of reactor coolant water chemistry in accordance with the EPRI guidelines in TR-105714 (see Water Chemistry Program) to mitigate SCC or IASCC; and (d) Participation in industry initiatives that will generate additional data on aging mechanisms relevant to RVI and develop appropriate inspection techniques to permit detection and characterization of features of interest.

Void Swelling is an aging mechanism for Reactor Vessel Internal components that has the potential to cause two aging effects: 1. Reduction in Fracture Toughness, and 2. Changes in Dimensions.

<u>Reduction in Fracture Toughness due to Void Swelling</u>

The Aging Management Review for the Reactor Vessel Internals includes a discussion of Reduction in Fracture Toughness due to Void Swelling, and concludes that Palisades does not consider void swelling to be a factor in the evaluation of reduction in fracture toughness.

Changes in Dimensions due to Void Swelling

The potential significance of void swelling will be assessed through monitoring industry operating experience. The PWR industry (including Palisades), through EPRI, is currently researching the potential significance of void swelling. If judged to be significant, evaluations of the need for augmented examinations for the effects of void swelling will be performed, and the results will be reported to the NRC at least two years prior to the end of the current operating license.

The ASME Section XI ISI Program is currently following ASME Section XI as required by 10 CFR 50.55a. However, NUREG-1801, Section XI.M16, “PWR Vessel Internals,” references the 1995 Edition through the 1996 Addenda. The ASME Section XI ISI Program will be updated to later editions and addenda as required by 10 CFR 50.55a; the code edition and addenda will not necessarily be those referenced by this revision of NUREG 1801.

NUREG-1801 Consistency

The Reactor Vessel Internals Inspection Program is consistent with NUREG-1801, Section XI.M16, “PWR Vessel Internals.”

Exceptions to NUREG-1801

None

Enhancements

Two enhancements are planned to bring the Reactor Vessel Internals Inspection Program into conformance with the NUREG-1801 program requirements. The enhancements are:

Scope of Program and Detection of Aging Effects: Palisades will participate in the industry initiatives to evaluate the effect of Changes in Dimensions due to Void Swelling, and will report to the NRC at least two years prior to the end of the current operating license the results of the industry initiative and a schedule for augmented inspections that will be required, if any.

Palisades will participate in industry initiatives that will generate additional data on aging mechanisms relevant to RVI and develop appropriate inspection techniques to permit detection and characterization of features of interest. Palisades will incorporate any recommended augmented inspections as appropriate.

Aging Management Program Elements

The elements of the Reactor Vessel Internals Inspection Program are described below. The results of an evaluation of each element against NUREG-1801, Sections XI.M.16, “PWR Vessel Internals,” are also provided.

Scope of Program

The program relies on the ASME Section XI examinations of the RVI to detect the applicable aging effects. For some components, the VT-3 examination required by ASME Section XI may not be adequate to detect the effects before the component intended function is compromised. The need for augmented examinations will be determined by participation in industry initiatives that will generate additional data on aging mechanisms relevant to RVI and develop appropriate inspection techniques to permit detection and characterization of features of interest.

The program is focused on managing the effects of crack initiation and growth due to stress corrosion cracking (SCC) or irradiation assisted stress corrosion cracking (IASCC), and loss of fracture toughness due to neutron irradiation embrittlement. The program contains preventive measures to mitigate SCC or IASCC; ISI to monitor the effects of cracking on the intended function of the components; and repair and/or replacement as needed to maintain the ability to perform the intended function. Loss of fracture toughness is of consequence only if cracks exist.

This element is consistent with NUREG-1801, Section XI.M16, “PWR Vessel Internals.”

Preventive Actions

This program credits the Water Chemistry Program for monitoring and control of primary coolant water chemistry to reduce the susceptibility to cracking due to SCC.

This element is consistent with NUREG-1801, XI.M16, “PWR Vessel Internals.”

Parameters Monitored, Inspected, and/or Tested

The program monitors the effects of crack initiation and growth by implementing the requirements of ASME Section XI Table IWB-2500-1.

This element is consistent with NUREG-1801, Section XI.M16, “PWR Vessel Internals.”

Detection of Aging Effects

The Palisades Reactor Vessel Internals Inspection Program includes a variety of inspection and testing activities that are designed to detect degradation due to aging effects prior to loss of intended function. Visual examinations are utilized as applicable and will be, as a minimum, the VT-3 examination specified by ASME Section XI, Subsection IWB, Category B-N-3. The need for augmented examinations will be determined by participation in industry initiatives that will generate additional data on aging mechanisms relevant to RVI and develop appropriate inspection techniques to permit detection and characterization of features of interest.

NUREG-1801, Section XI.M16, Detection, states, “For bolted components, augmented ISI is to include other demonstrated acceptable inspection methods to detect cracks between the bolt head and the shank. Alternatively, the applicant may perform a component-specific evaluation, including a mechanical loading assessment to determine the maximum tensile loading on the component during ASME Code Level A, B, C, and D conditions. If the loading is compressive or low enough (<5 ksi) to preclude fracture, then supplemental inspection of the component is not required.”

Based on operating experience, a CEOG plant and component-specific evaluation, and CE's position that normal in-service inspections will detect any problems with these bolts, Palisades has determined that augmented inspections of this bolting are not necessary. This is identical to the position taken by the Fort Calhoun Station in their License Renewal Application, and subsequently approved by the NRC in the associated Safety Evaluation Report, NUREG-1782.

This element is consistent with NUREG-1801, Section XI.M16, “PWR Vessel Internals.”

Monitoring and Trending

The reactor vessel internals are inspected per the requirements of ASME Section XI, subsection IWB, Table IWB-2500-1. The need for augmented examinations will be determined by participation in industry initiatives that will generate additional data on aging mechanisms relevant to RVI and develop appropriate inspection techniques to permit detection and characterization of features of interest. The inspection schedule is in accordance with IWB-2400.

This element is consistent with the corresponding element of NUREG-1801, Section XI.M16, “PWR Vessel Internals.”

Acceptance Criteria

Indications or relevant conditions of degradation are evaluated in accordance with ASME Section XI, subsection IWB-3100, by comparing the examination results with the acceptance standards of IWB-3400.

This element is consistent with NUREG-1801, Section XI.M16, “PWR Vessel Internals.”

Corrective Actions, Confirmation Process, Administrative Controls

These elements are consistent with the corresponding NUREG-1801 aging management program elements. See Section B1.2 for further discussion.

Operating Experience

Palisades has a comprehensive Operating Experience Program (OEP) that monitors industry issues/events and assesses these for applicability to its own operations. In addition, the Palisades Corrective Action Program (CAP) is used to track, trend and evaluate plant issues/events. Those issues and events, whether external or plant specific, that are potentially significant to the Reactor Vessel Internals Inspection Program at Palisades are evaluated. The Reactor Vessel Internals Inspection Program is augmented, as appropriate, if these evaluations show that program changes will enhance program effectiveness.

Using the OEP and CAP to focus on industry and plant operating experience ensures that Reactor Vessel Internals Inspection Program issues are addressed in a timely manner and that age related deterioration of SSC within the scope of the Reactor Vessel Internals Inspection Program will be effectively managed throughout the license renewal period.

In completing our review of Operating Experience we looked at related issues which included NRC Information Notices 84-18, “Stress Corrosion Cracking in PWR Systems,” and 98-11, “Cracking of Reactor Vessel Internal Baffle Former Bolts in Foreign Plants.”

A review of the industry operating experience related to the Reactor Vessel Internals Inspection Program revealed several instances where degradation has occurred within the reactor vessel internals. The review considered a variety of issues related to reactor internals which included degradation of baffle former bolts, barrel former bolts, guide bar bolts, core support shield to core barrel bolts, guide funnels, guide tube support pins, and rod cluster control assemblies.

A review of the plant specific operating experience revealed two (2) instances where the Reactor Vessel Internals Inspection program has been instrumental in discovering material degradation. Degradation was discovered in the following items:

- Core barrel
- CRDM seal housings

The Reactor Vessel Internals Inspection Program has been effective in identifying material degradation in a timely manner, thus ensuring that age related degradation of Reactor Vessel Internals will be effectively managed by the Reactor Vessel Internals Inspection Program throughout the license renewal period.

This element is consistent with NUREG-1801, Section XI.M16, “PWR Vessel Internals.”

Conclusion

The Reactor Vessel Internals Inspection Program is an existing program that uses as its bases, various industry and NRC standards. This program is consistent with NUREG-1801, Section XI.M16, “PWR Vessel Internals.”

Reports of recent internal and external assessments of the program have been reviewed. These reports demonstrate that the program has effectively identified and dispositioned issues that could have led to degraded conditions. It is concluded that the Reactor Vessel Internals Inspection Program has been effective in maintaining the intended functions of the affected long-lived, passive SSCs.

The continued implementation of the Reactor Vessel Internals Inspection Program provides reasonable assurance that aging effects will be managed such that SSCs within the scope of this program will continue to perform their intended functions consistent with the current licensing bases for the period of extended operation.

B2.1.18 Steam Generator Tube Integrity Program

Program Description

The Steam Generator Tube Integrity Program is an existing program that manages the aging effects of steam generator tubes and tube repairs. The Program also manages the aging effects of accessible steam generator secondary side internal components and incorporates the guidance of NEI 97-06. The program manages aging effects through a balance of mitigation, inspection, evaluation, repair, and leakage monitoring measures. Component degradation is mitigated by controlling primary and secondary water chemistry. Eddy current testing is used to detect steam generator tube flaws and degradation. Visual examinations are performed to identify degradation of accessible steam generator secondary side internal components. Primary to secondary leakage is monitored during plant operation.

The program credits the Water Chemistry Program for primary and secondary water chemistry control. The program also satisfies ASME Section XI, IWB- 2500, Category B-Q requirements to perform volumetric examinations of steam generator tubes in the Inservice Inspection (ISI) Program.

NUREG-1801 Consistency

The Steam Generator Tube Integrity Program is consistent with NUREG-1801, Section XI.M19, “Steam Generator Tube Integrity.”

Exceptions to NUREG-1801

None.

Enhancements

None

Aging Management Program Elements

The elements of the Steam Generator Tube Integrity Program are described below. The results of an evaluation of each element against NUREG-1801, Section XI.M19, “Steam Generator Tube Integrity,” are also provided.

Scope of Program

The scope of the Steam Generator Tube Integrity Program is specific to the steam generator primary side U-tubes and the accessible areas of the secondary side internal components. The secondary side internal components include the top of tube sheet, stay dome region, outer bundle region, top of tube bundle, and upper internals which includes feed ring piping, tube supports, J nozzles, secondary separators, and vertical strap and diagonal bar supports. Tube inspection scope and frequency, plugging or repair, and leakage monitoring is in accordance with Technical Specification 5.5.8.

This element is consistent with the corresponding element of NUREG-1801, Section XI.M19, “Steam Generator Tube Integrity.”

Preventive Actions

The Steam Generator Tube Integrity Program is a condition monitoring program. However, this program credits the Water Chemistry Program for monitoring and control of primary and secondary water chemistry, which is in accordance with the EPRI Guidelines of TR-105714 for monitoring and control of primary coolant water chemistry and TR-102134 for monitoring and control of secondary water chemistry. The Program also includes the guidance from NEI 97-06 regarding foreign material exclusion as a means to inhibit fretting and wear degradation. Periodic sludge lancing is performed to minimize pitting on the outside diameter of the steam generator tubes due to oxidizing conditions in the sludge piles.

This element is consistent with the corresponding element of NUREG-1801, Section XI.M19, “Steam Generator Tube Integrity.”

Parameters Monitored, Inspected, and/or Tested

The Steam Generator Tube Integrity Program monitors steam generator tube indications and growth, loss of section thickness, loss of material, tube denting, and stainless steel egg crate lattice support cracking. The Program inspection activities include detection of flaws in the steam generator tubes and degradation of the accessible areas of the secondary side internal components. The steam generator tubes are tested using the eddy current volumetric test method. The examinations for the accessible areas of the secondary side internal components are performed visually.

This element is consistent with the corresponding element of NUREG-1801, Section XI.M19, “Steam Generator Tube Integrity.”

Detection of Aging Effects

The Palisades Steam Generator Tube Integrity Program includes a variety of examination and testing activities that are designed to detect degradation due to aging effects prior to loss of intended function. The Program complies with the plant Technical Specification 5.5.8 and NEI 97-06 for the examination and test methods, sample size, and inspection frequency. The examination and test methods, sample size, and frequency of the inspections prescribed by the Program are designed to ensure that flaws or degradation do not exceed established performance criteria, and are documented in a degradation assessment completed prior to each steam generator inspection. In response to NRC Generic Letter 95-03, Palisades further committed to a more rigorous inspection plan defined by EPRI TR-107569.

The steam generator nondestructive examination (NDE) requirements are based on the applicable requirements and references in the ASME Boiler and Pressure Vessel Code, Section XI, used for the ISI Program.

This element is consistent with the corresponding element of NUREG-1801, Section XI.M19, "Steam Generator Tube Integrity."

Monitoring and Trending

The inspection schedule and scope expansion requirements are based on Technical Specification 5.5.8 and NEI 97-06. These requirements are expected to provide timely detection of tube degradation prior to any loss of intended function or challenge to tube integrity. Results of inspections are documented, evaluated, and compared with previous inspection results to identify adverse trends.

Following steam generator inspections, a condition monitoring assessment of the as-found condition is performed to assess degradation, and then a forward looking operational assessment is conducted to ensure that performance criteria will not be exceeded during the next operating cycle. The current condition monitoring assessment is also compared with the previous operational assessment to gain feedback and insight for the next operational assessment.

This element is consistent with the corresponding element of NUREG-1801, Section XI.M19, "Steam Generator Tube Integrity."

Acceptance Criteria

The acceptance criteria for steam generator inspections are in accordance with plant Technical Specification 5.5.8 and NEI 97-06. Loose parts or foreign objects that are found are removed from the steam generators unless it is shown by evaluation that these objects will not cause unacceptable tube damage.

Tube inspections are followed by tube integrity assessments that compare the as-found inspection results with the performance criteria for structural integrity and accident leakage. These performance criteria are expressed in terms of parameters that are directly measurable or that may be calculated on the basis of direct measurements. When steam generator tubes do not meet the acceptance criteria specified in the plant Technical Specifications or NEI 97-06, they are repaired or removed from service by plugging. The criteria for plugging or repairing steam generator tubes is consistent with NRC RG 1.121. The acceptance criteria for secondary-side internal components are delineated in the applicable program implementing documents.

This element is consistent with the corresponding element of NUREG-1801, Section XI.M19, “Steam Generator Tube Integrity.”

Corrective Actions, Confirmation Process, Administrative Controls

These elements are consistent with the corresponding NUREG-1801 aging management program elements. See Section B1.2 for further discussion.

Operating Experience

Palisades has a comprehensive Operating Experience Program (OEP) that monitors industry issues/events and assesses these for applicability to its own operations. In addition, the Palisades Corrective Action Program (CAP) is used to track, trend and evaluate plant issues/events. Those issues and events, whether external or plant specific, that are potentially significant to the Steam Generator Tube Integrity Program at Palisades are evaluated. The Steam Generator Tube Integrity Program is augmented, as appropriate, if these evaluations show that program changes will enhance program effectiveness.

Using the OEP and CAP to focus on industry and plant operating experience ensures that Steam Generator Tube Integrity Program issues are addressed in a timely manner and that age related deterioration within the scope of the Steam Generator Tube Integrity Program will be effectively managed throughout the license renewal period.

A review of the industry operating experience related to the Steam Generator Tube Integrity Program revealed instances where degradation has occurred within the steam generators. In completing our review we looked at related issues which included degradation of steam generator tubes, tube sheet, mechanical plugs, tube support plates, girth welds, antivibration bars, etc., plus degradation associated with loose parts, foreign objects, sludge, water chemistry, and wear.

A review of the plant specific operating experience revealed several instances where the Steam Generator Tube Integrity Program has been instrumental in discovering material degradation. Steam Generator tube degradation was discovered in the following areas:

- Top of tubesheet
- Within the tubesheet
- U-bends
- Mechanical wear at eggcrate supports, vertical straps, and diagonal bars

The Steam Generator Tube Integrity Program has been effective in identifying material degradation in a timely manner, thus ensuring that age related degradation of steam generator subcomponents will be effectively managed throughout the license renewal period.

The steam generators at Palisades were replaced in late 1990. The new steam generators are improved in design, material selection, and construction. Included in the new design was a change in the tube support from solid plate to egg crate dividers along with other features to minimize corrosion crevices and denting.

This element is consistent with NUREG-1801, Section XI.M19, “Steam Generator Tube Integrity.”

Conclusion

The Steam Generator Tube Integrity Program is an existing program that uses as its bases, various industry and NRC standards. This program is consistent with NUREG-1801, Section XI.M19, “Steam Generator Tube Integrity.”

Reports of recent internal and external assessments of the program have been reviewed. These reports demonstrate that the program has effectively identified and dispositioned issues that could have led to degraded conditions. The Steam Generator Tube Integrity Program has been effective in maintaining the intended functions of the affected long-lived, passive steam generator subcomponents.

The continued implementation of the Steam Generator Tube Integrity Program provides reasonable assurance that aging effects will be managed such that the components within the scope of this program will continue to perform their intended functions consistent with the current licensing bases for the period of extended operation.

B2.1.19 Structural Monitoring Program

Program Description

The Structural Monitoring Program is an existing program that is designed to ensure that age related (as well as other) deterioration of plant structures (including masonry walls) and components within its scope is appropriately managed to ensure that each such structure or component retains the ability to perform its intended function. The program is implemented through visual examination of these structures, components and other specified items. Damage or degradation found during visual examination may be further evaluated by measurements and testing techniques as appropriate.

This program also implements provisions of the Maintenance Rule, 10 CFR 50.65, that relate to masonry walls and water-control structures. It conforms to the guidance contained in RG 1.160 and NUMARC 93-01 as well as Nuclear Energy Institute publication NEI 96-03. This NEI document, which supplements NUMARC 93-01, contains additional guidance specific to the monitoring of structures.

Initial baseline inspections under the Structural Monitoring Program were performed, as required by 10 CFR 50.65, starting in late 1996. A second complete inspection was performed in 1999 to validate the initial inspection results. Subsequent inspections follow a 10 year interval schedule that is similar to Inspection Plan B defined in the ASME Boiler & Pressure Vessel Code, Section XI, Table IWE-2412-1.

The 10 year inspection interval is divided into three 40 month periods. Approximately one third of the items in the program scope are examined in each period and all items are examined at least once during the 10 year interval. The first interval, first period, inspections have been completed, and Palisades is currently in the first interval, second period inspection cycle. Other features may have greater inspection frequencies such as watertight/flood barrier inspections (at least once per 5 years) and below-the-waterline water-control structures (once every 5 years).

Augmented inspection is required for items that have been repaired or that exhibit significant damage or deterioration. It may also be required for items subject to aggressive environments. Items that are tagged for augmented inspection following repair or for reasons of damage / deterioration are examined, at a minimum, in the period immediately following the one during which the repair was performed or the deleterious condition was found. Augmented inspection may be performed on a 40 month period basis or at more closely spaced intervals as specified by the Structural Monitoring Coordinator or in plant procedures.

The Structural Monitoring Program includes requirements for the inspection of water-control structures and structural elements that are accessible above the waterline. Palisades is not committed to Regulatory Guide 1.127, and NUREG-1801 Program XI.S7 provides guidance to plants that are not committed to RG 1.127. This guidance includes addressing those structures above the waterline in NUREG-1801 Program XI.S6, and addressing below-the-waterline structures in the applicable sections of NUREG-1801 Program XI.S7.

NUREG-1801 Consistency

The Structural Monitoring Program is consistent with the following NUREG-1801 programs:

- NUREG-1801 Section XI.S6, “Structures Monitoring Program”
- The corresponding structural elements in NUREG-1801 Section XI.S5, “Masonry Wall Program”
- The corresponding structural elements in NUREG-1801 Section XI.S7, “RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants”

Exceptions to NUREG-1801

None.

Enhancements

One enhancement is planned to bring the Structural Monitoring Program into conformance with the NUREG-1801 program requirements. The enhancement is:

Scope of Program: Incorporate into the Structural Monitoring Program all structural members listed in Tables 3.5.2-1 through 3.5.2-10 that will use the Structural Monitoring Program as an AMP.

Note that the element descriptions describe the program as it will exist after the identified enhancement has been implemented. Enhancements are scheduled for completion prior to the period of extended operation.

Aging Management Program Elements

The elements of the Structural Monitoring Program are described below. The results of an evaluation of each key element against NUREG-1801, Section XI.S6, “Structures Monitoring Program,” and the applicable structural elements of Sections XI.S5, “Masonry Wall Program,” and XI.S7, “RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants,” are also provided.

Scope of Program

The Palisades Structural Monitoring Program implements the requirements of 10 CFR 54 and 10 CFR 50.65.

The NUREG-1801 Structures Monitoring (XI.S6) portion of this program applies to most structural components within the scope of license renewal, including those water-control structures and structural elements that are accessible above the waterline.

The NUREG-1801 Masonry Wall (XI.S5) portion of this program applies to all masonry walls within the scope of license renewal. The Structural Monitoring Program includes requirements for the examination of in-scope masonry walls for cracking and degradation of steel supports, and does not maintain a separate Masonry Wall Program.

The NUREG-1801 Water-Control Structures (XI.S7) portion of this program applies to below-the-waterline structures and structural elements within the scope of license renewal. These structures include the Intake Crib and Intake / Discharge Structures. Continuously submerged structures and structural elements are also age managed under the this program.

This element of the Palisades Structural Monitoring Program is consistent with NUREG-1801, Section XI.S6, "Structures Monitoring Program", and the applicable structural elements described in NUREG-1801, Sections XI.S5, "Masonry Wall Program," and XI.S7, "R.G.1.127, Inspection of Water Control Structures Associated With Nuclear Power Plants" (below-the-waterline).

Preventive Actions

The Palisades Structural Monitoring Program is a monitoring program and does not specify preventive actions.

This element of the Palisades Structural Monitoring Program is consistent with NUREG-1801, Section XI.S6, and the structural elements described in NUREG-1801, Sections XI.S5 and XI.S7 (below-the-waterline).

Parameters Monitored, Inspected, and/or Tested

<u>XI.S6, Structures Monitoring</u>

The NUREG-1801 evaluation of Program XI.S6 references ACI 349.3R-96 and ANSI / ASCE 11-90 as industry standard publications that provide acceptable guidance on the selection of parameters to monitor. The Palisades Structures Monitoring Program conforms to the guidance and basis provided in these two documents.

The NUREG-1801 evaluation also identifies monitoring of site de-watering systems to ensure continuing control over erosion of porous concrete sub-foundations and building settlement. Palisades' buildings are founded on dense sand and are not subject to significant settlement. Neither porous concrete sub-foundations nor de-watering systems are used. Therefore, the issues of de-watering systems, erosion of porous concrete sub-foundations and settlement are not applicable.

XI.S5, Masonry Walls

The NUREG-1801 evaluation of Program XI.S5 specifies the monitoring of cracking and degradation of steel edge supports and bracings. The Palisades Structural Monitoring Program requires monitoring of cracking as well as other signs of damage / deterioration. The Palisades program is, therefore, in conformance with the Masonry Wall Program monitoring elements identified in NUREG-1801.

XI.S7, Inspection of Water-Control Structures Associated with Nuclear Power Plants

The NUREG-1801 evaluation of Program XI.S7 specifies the monitoring of cracking, settlement & deflection and numerous other parameters that apply only to dams and related structures which are not used at Palisades. It references additional parameters as listed in USNRC Regulatory Guide 1.127, Section C.2. Similar to NUREG-1801 Program XI.S7, this regulatory guide addresses primarily dams and related structures. It does, however, list general guidelines for the examination of concrete structures and references ACI 201.1R for detailed descriptions of concrete deficiencies. The Palisades Structural Monitoring Program conforms to these recommendations and incorporates the additional specific monitoring tasks addressed in the XI.S7 program evaluation. Steel structures or portions of structures are examined for loss of material. This applies to the steel intake crib, and carbon steel and cast iron items in the intake structure.

In addition, RG 1.127 provides guidelines for examining and assessing the condition of conduits and inlet canals. These guidelines refer principally to continuously submerged surfaces and are implemented through the program underwater inspections.

This element of the Palisades Structural Monitoring Program is consistent with NUREG-1801, Section XI.S6, “Structures Monitoring Program”, and the applicable structural elements described in NUREG-1801, Sections XI.S5, “Masonry Wall Program,” and XI.S7, “R.G.1.127, Inspection of Water Control Structures Associated With Nuclear Power Plants” (below-the-waterline).

Detection of Aging Effects

The Palisades Structural Monitoring Program is implemented in accordance with a 10 year interval schedule that requires completion of approximately 1/3 of the specified examinations within each of three consecutive 40 month periods. The program requires that 100% of the items included within its scope be examined at least once (and more often for those items requiring augmented inspection) during each interval. These requirements for frequent and comprehensive examinations provide a high degree of assurance that age related deterioration of an item will be detected and corrected long before it has a significant impact on the item's intended function.

Examinations of structures and structural elements are performed by qualified personnel using techniques appropriate for the item, its environment and its intended function. Each individual selected to perform these examinations must have the following qualifications:

- A civil engineering degree from an accredited university.
- Familiarity with the design and performance requirements applicable to nuclear power plant structures and experience in the in-service examination and evaluation of these structures.
- A minimum of 5 years experience in engineering design and / or analysis of nuclear power plant structures.

System engineers and plant operators augment the formal examinations by noting the conditions of structures / structural elements during periodic system walk downs, and reporting observed damage / degradation to the Structural Monitoring Coordinator.

<u>XI.S6, Structures Monitoring</u>

The NUREG-1801 evaluation of Program XI.S6 references ACI 349.3R-96 and ANSI / ASCE 11-90 as industry standard publications (although not required) that provide acceptable guidance on the use of examination techniques suitable to detect significant aging effects. The Palisades program conforms to the guidance given in these two documents.

<u>XI.S5, Masonry Walls</u>

The NUREG-1801 evaluation of Program XI.S5 identifies visual examination of the masonry walls for cracking and degradation of steel edge supports and bracings that could cause loss of intended functions prior to the next scheduled examination. The Palisades program applies the same periodic visual examination techniques to masonry walls and to concrete structural elements as prescribed by this program element. Periodicity of examinations may vary, according to different reinforcement masonry configurations.

XI.S7, Inspection of Water-Control Structures Associated with Nuclear Power Plants

NUREG-1801 evaluation of Program XI.S7 identifies no specific requirements relative to detecting age related deterioration of water-control structure elements above the waterline. The Palisades program applies the same visual examination techniques to water-control structural components that it applies to other concrete structural components in the XI.S5 program. For structures and components below-the-waterline, Palisades maintains those elements consistent with those identified to the XI.S7 program in NUREG-1801.

This element of the Palisades Structural Monitoring Program is consistent with NUREG-1801, Section XI.S6, “Structures Monitoring Program”, and the applicable structural elements described in NUREG-1801, Sections XI.S5, “Masonry Wall Program,” and XI.S7, “R.G.1.127, Inspection of Water Control Structures Associated With Nuclear Power Plants” (below-the-waterline).

Monitoring and Trending

XI.S6, Structures Monitoring

Items included within the scope of this program that do not meet the acceptance criteria listed are evaluated for corrective action and, at a minimum, are scheduled for augmented examination. Items that are repaired are also scheduled for augmented examination. The condition of any item subject to augmented examination is trended to provide a basis for scheduling future inspections or corrective action. This approach is consistent with that described 10 CFR 50.65 & Regulatory Guide 1.160 and referenced in NUREG-1801 evaluation of the Structures Monitoring Program (XI.S6).

XI.S5, Masonry Walls

NUREG-1801 evaluation of Program XI.S5 states that trending is not required and that monitoring is achieved by the periodic examinations to detect cracking. However, for the evaluation of aging effects that do not meet acceptance criteria at Palisades, augmented inspections may be specified to identify if continued degradation is occurring and if corrective action may be required.

XI.S7, Inspection of Water-Control Structures Associated with Nuclear Power Plants

NUREG-1801 evaluation of Program XI.S7 states that no trending is required. Monitoring is to be performed in accordance with this guide, which contains no requirements that differ from those applied by the Palisades program to other structures and structural elements. Therefore, the Palisades program includes no special requirements to monitor or trend water-control structure condition.

This element of the Palisades Structural Monitoring Program is consistent with NUREG-1801, Section XI.S6, "Structures Monitoring Program", and the applicable structural elements described in NUREG-1801, Sections XI.S5, "Masonry Wall Program," and XI.S7, "R.G.1.127, Inspection of Water Control Structures Associated With Nuclear Power Plants" (below-the-waterline).

Acceptance Criteria

XI.S6, Structures Monitoring

NUREG-1801 evaluation of the Structures Monitoring Program, XI.S6, states that, "… acceptance criteria are selected to ensure that the need for corrective actions will be identified before loss of intended functions. Acceptance criteria are to be commensurate with industry codes, standards and guidelines, and are also to consider industry and plant-specific operating experience. Although not required, ACI-349.3R-96 provides an acceptable basis for developing acceptance criteria …" The Palisades Structural Monitoring Program satisfies the prescriptive requirements of NUREG-1801 evaluation and includes guidelines provided in the referenced ACI publication.

XI.S5, Masonry Walls

The NUREG-1801 evaluation of Program XI.S5 requires an engineering evaluation (and subsequent corrective action if found to be necessary) of any observed cracking and/or degradation of steel edge supports and bracing. This engineering evaluation is done to determine if the wall retrains the capability to fulfill its intended function. The Palisades program requires such an evaluation (and corrective action, if specified in the evaluation report) if cracking or degraded steel edge supports and bracing are found during the examination.

XI.S7, Inspection of Water-Control Structures Associated with Nuclear Power Plants

The Palisades program applies the same acceptance criteria to water-control structure elements as it does similar elements in other structures per

NUREG-1801, Program XI.S6. Therefore, Palisades maintains elements consistent with those below-the-waterline SSC's identified to the XI.S7 program in NUREG-1801.

This element of the Palisades Structural Monitoring Program is consistent with NUREG-1801, Section XI.S6, “Structures Monitoring Program”, and the applicable structural elements described in NUREG-1801, Sections XI.S5, “Masonry Wall Program,” and XI.S7, “R.G.1.127, Inspection of Water Control Structures Associated With Nuclear Power Plants” (below-the-waterline).

Corrective Actions, Confirmation Process, Administrative Controls

These elements are consistent with the corresponding NUREG-1801 aging management program elements. See Section B1.2 for further discussion.

Operating Experience

Palisades has a comprehensive Operating Experience Program (OEP) that monitors industry issues / events and assesses these for applicability to its own operations. In addition, the Palisades Corrective Action Program (CAP) is used to track, trend and evaluate plant issues / events. Those issues and events, whether external or plant specific, that are potentially significant to the structures and other items within the scope of the Palisades Structures Monitoring Program are evaluated. The Structural Monitoring Program is augmented, as appropriate, if these evaluations show that program changes will enhance item performance and, as a consequence, operational safety.

Using the OEP and CAP to focus on industry and plant operating experience ensures that Structural Monitoring Program issues are addressed in a timely manner and that age related deterioration of the items within the program scope is effectively managed throughout the license renewal period.

A review of industry operating experience associated with the Structures Monitoring Program and aging reveals issues and instances related to:

- Corrosion of steel ice condenser containment vessels caused by boric acid and condensation
- Cracks in concrete floors caused by flexing and shrinkage

Various related NRC and/or industry generic communications have been issued, and, in turn, have been incorporated into the program as applicable.

A review of plant specific operating experience related to the Structural Monitoring Program and aging revealed that the following issues have been addressed:

- Settling of air compressor foundations
- Watertight barrier degradation
- Spalled concrete and exposed anchor bolts
- Intake crib damage due to ice and to wave action
- Cracking of concrete beams in the Auxiliary Building
- Corrosion of condenser rock anchors caused by standing water and debris
- Degradation of snubber anchor support structure concrete and grout
- Deterioration of floor plugs due to leaking water
- Moisture Separator Reheater foundation cracking
- Cracks in concrete duct bank
- Cracks in West ESS room west wall
- Spalled concrete on wall of 1-2 Diesel Generator Exhaust Plenum
- Groundwater leaks in Auxiliary Feedwater Pump room floor

The Palisades Structural Monitoring Program has demonstrated that it provides reasonable assurance that aging effects are being managed for Structural Monitoring Program SSCs. Additionally, this has been demonstrated through inspection reports, Program Health Reports, and the Corrective Action Program.

This element of the Palisades Structural Monitoring Program is consistent with NUREG-1801, Section XI.S6, “Structures Monitoring Program”, and the applicable structural elements described in NUREG-1801, Sections XI.S5, “Masonry Wall Program,” and XI.S7, “R.G.1.127, Inspection of Water Control Structures Associated With Nuclear Power Plants” (below-the-waterline).

Conclusion

The Structural Monitoring Program is based on an existing program (Maintenance Rule Structural Monitoring) and is consistent with NUREG-1801, Section XI.S6, and the structural elements described in NUREG-1801, Sections XI.S5 and XI.S7 (below-the-waterline).

Reports of recent internal and external assessments of the program have been reviewed. These reports demonstrate that the program has effectively identified and dispositioned issues that could have led to degraded conditions. Palisades has demonstrated that the

Palisades Structural Monitoring Program will effectively manage aging of structural SSC's in-scope of License Renewal.

It is concluded that continued implementation of the Structural Monitoring Program provides reasonable assurance that the effects of aging will be managed such that the affected components will continue to perform their intended function(s) during the period of extended operation.

B2.1.20 System Monitoring Program

Program Description

The System Monitoring Program is an existing plant-specific program that manages aging effects for normally accessible, external surfaces of piping, tanks, and other components and equipment within the scope of License Renewal. These aging effects are managed through visual inspection and monitoring of external surfaces for leakage and evidence of material degradation. The program relies upon periodic system walkdowns to monitor degradation of the protective paint or coating, and/or the exterior steel surface area (if no paint or coatings exist, or if the existing protective paint and coatings are degraded to a point whereby the exterior steel surface is exposed). Palisades does not take credit for any above ground coating or paint for mitigating corrosion even though the tanks may be painted or coated. However, inspections of the above ground coating or paint will provide an indication of the condition of the material underneath the coating or paint.

NUREG-1801 Consistency

The System Monitoring Program is a plant-specific program that contains the applicable elements of an acceptable aging management program as described in Branch Technical Position RLSB-1, “Aging Management Review-Generic,” which is included in Appendix A of NUREG-1800, “Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants.”

The System Monitoring Program is also consistent with the portions of NUREG-1801, Section XI.M29, “Above Ground Carbon Steel Tanks,” that are applicable to the accessible external surfaces of the applicable tanks.

Exceptions to NUREG-1801

None.

Enhancements

One enhancement is planned to bring the System Monitoring Program into conformance with the NUREG-1801 program description. The enhancement is:

Parameters Monitored, Inspected, and/or Tested, Detection of Aging Effects, and Monitoring and Trending: Enhance system walkdown procedures to more specifically address the types of components to be inspected, and to specifically describe the relevant degradation mechanisms and effects of interest, and for use of the Corrective Action Program to document aging related degradation, identified during the inspections, that may affect the ability of the SSC to perform its intended function.

Note that the element descriptions describe the program as it will exist after the identified enhancements have been implemented. Enhancements are scheduled for completion prior to the period of extended operation.

Aging Management Program Elements

The elements of the System Monitoring Program are described below. An evaluation of each element against the corresponding element described in Branch Technical Position RLSB-1, “Aging Management Review-Generic,” which is included in Appendix A of NUREG-1800, “Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants,” is provided. Since this program includes visual inspections of the external surfaces of above ground carbon steel tanks, the results of an evaluation of each element against NUREG-1801, Section XI.M.29, “Above Ground Carbon Steel Tanks,” are also provided.

Scope of Program

The System Monitoring Program is credited for managing the aging effects for normally accessible surfaces of piping, tanks, and other components and equipment within the scope of License Renewal. These aging effects are managed through visual inspection and monitoring of external surfaces for leakage and evidence of material degradation.

The System Monitoring Program includes periodic visual inspections (walkdowns) of the external surfaces of various types of components to identify age related degradation.

NUREG-1801 for above ground carbon steel tanks states that the program consists of “preventive measures to mitigate corrosion by protecting the external surfaces of carbon steel tanks protected with paint or coatings.” Palisades Nuclear Plant does not take credit for any above ground coating or paint for mitigating corrosion even though the tanks may be painted or coated. However, inspections of the above ground coating or paint will provide an indication of the condition of the material underneath the coating or paint. Therefore, periodic system walkdowns will detect the effects of corrosion if present. This meets the intent of NUREG-1801.

This element is consistent with the corresponding element of NUREG-1800, Appendix A.

This element is also consistent with the portions of NUREG-1801, Section XI.M29, “Above Ground Carbon Steel Tanks,” that are applicable to the accessible external surfaces of above ground carbon steel tanks.

Preventive Actions

There are no preventive actions associated with this program. The objective of this program is to identify the aging effects of concern before a loss of intended function occurs (i.e., condition monitoring). Palisades Nuclear Plant does not take credit for any above ground coating or paint for mitigating corrosion even though the tanks may be painted or coated. However, inspections of the above ground coating or paint will provide an indication of the condition of the material underneath the coating or paint. Therefore, periodic system walkdowns will detect the effects of corrosion if present. This meets the intent of NUREG-1801. Sealants or caulking at the tank/support structure interface, if used, are also inspected for degradation. However, Palisades does not credit sealants or caulking for prevention of water intrusion underneath tanks.

This element is consistent with the corresponding element of NUREG-1800, Appendix A.

The System Monitoring Program is also consistent with the portions of NUREG-1801, Section XI.M29, “Above Ground Carbon Steel Tanks,” that are applicable to the accessible external surfaces of above ground carbon steel tanks.

Parameters Monitored, Inspected, and/or Tested

The program utilizes periodic plant system walkdowns to monitor for leakage and evidence of material degradation. Above ground carbon steel tank external coatings or paint are inspected to provide an indication of the condition of the material underneath the coating or paint. Sealants or caulking at the tank/support structure interface, if used, are also inspected for degradation. However, Palisades does not credit sealants or caulking for prevention of water intrusion underneath tanks.

This element is consistent with the corresponding element of NUREG-1800, Appendix A.

This element is also consistent with the portions of NUREG-1801, Section XI.M29, “Above Ground Carbon Steel Tanks,” that are applicable to the accessible external surfaces of above ground carbon steel tanks.

Detection of Aging Effects

The System Monitoring Program uses visual inspection activities to detect degradation due to aging effects prior to loss of intended function. The external surfaces of various component types (e.g., pump casings, valve bodies, piping, expansion joints) are visually inspected for leakage and evidence of material degradation, such as loss of material due to corrosion. The outer surfaces of above ground carbon steel tanks are visually inspected for signs of coating or paint degradation to provide an indication of the condition of the material underneath the coating or paint. The sealant or caulking at the tank/support structure interface, if used, is also inspected for degradation. However, Palisades does not credit sealants or caulking for prevention of water intrusion underneath tanks. This program credits the One-Time Inspection Program for the inspection of inaccessible portions of above ground (non-Diesel Fuel Oil) carbon steel tanks. Diesel Fuel Oil Tank bottom thickness tests are conducted periodically per the Diesel Fuel Monitoring and Storage Program.

Degradation of bolted connections is detected by visual inspections of the bolted components during system walkdowns. Bolted connections are inspected for missing fasteners and degradation such as damaged threads and evidence of corrosion. The minimum walkdown frequency is annual for those systems and components that are accessible during normal plant operation. Systems and components that are only accessible during plant outages, are inspected at least once per refueling interval. The inspection frequency may be increased based on the safety significance, production significance, discovery and/or operating experience of each system.

This element is consistent with the corresponding element of NUREG-1800, Appendix A.

This element is also consistent with the portions of NUREG-1801, Section XI.M29, “Above Ground Carbon Steel Tanks,” that are applicable to the accessible external surfaces of above ground carbon steel tanks.

Monitoring and Trending

Visual inspections are performed at least annually for those systems and components that are accessible during normal plant operation. Systems and components that are only accessible during plant outages, are inspected at least once per refueling interval. The inspection frequency may be increased or reduced based on the safety significance and operating experience of each system. These system walkdown inspections provide for timely detection of aging effects (i.e., prior to a loss of intended function). Walkdown results are documented to provide a historical record of items monitored during the walkdowns. This program credits the One-Time Inspection Program for the inspection of inaccessible portions of above ground (non-Diesel Fuel Oil) carbon steel tanks. Diesel Fuel Oil Tank bottom thickness tests are conducted periodically per the Diesel Fuel Monitoring and Storage Program. The results are monitored and trended if significant material loss is detected.

This element is consistent with the corresponding element of NUREG-1800, Appendix A.

This element is also consistent with the portions of NUREG-1801, Section XI.M29, “Above Ground Carbon Steel Tanks,” that are applicable to the accessible external surfaces of above ground carbon steel tanks.

Acceptance Criteria

System walkdown procedures require that signs of significant degradation to paint, coatings or exposed external steel surfaces found that may affect the components ability to perform its intended function be entered into the Corrective Action Program for resolution. Other types of degradation are recorded for further evaluation.

When bolted joints for pressure retaining components are observed to have significant degradation or to be leaking, corrective actions are taken in accordance with the corrective action program. Significant degradation of tank coatings or paint, and sealants or caulking (if applicable) are also addressed within the Corrective Action Program. Significant degradation consists of cracking, flaking, or peeling of paint or coatings, and cracked sealant or caulking (if applicable).

This element is consistent with the corresponding element of NUREG-1800, Appendix A.

This element is also consistent with the portions of NUREG-1801, Section XI.M29, “Above Ground Carbon Steel Tanks,” that are applicable to the accessible external surfaces of above ground carbon steel tanks.

Corrective Actions, Confirmation Process, Administrative Controls

These elements are consistent with the corresponding NUREG-1800 and NUREG-1801 aging management program elements. See Section B1.2 for further discussion.

Operating Experience

Palisades has a comprehensive Operating Experience Program (OEP) that monitors industry issues/events and assesses these for applicability to its own operations. In addition, the Palisades Corrective Action Program (CAP) is used to track, trend and evaluate plant issues/events. Those issues and events, whether external or plant specific, that are potentially significant to the System Monitoring Program at Palisades are evaluated. The System Monitoring Program is augmented, as appropriate, if these evaluations show that program changes will enhance program effectiveness.

Using the OEP and CAP to focus on industry and plant operating experience ensures that System Monitoring Program issues are addressed in a timely manner and that age related deterioration of SSC within the scope of the System Monitoring Program will be effectively managed throughout the license renewal period.

A review of industry operating experience associated with the System Monitoring Program and aging reveals issues related to:

- Service Water Pump flange welds and bolting found excess rusting leading to leakage (Inadequate/infrequent system walkdowns were cited).

Various related NRC and/or industry generic communications have been issued, and, in turn, have been incorporated into the program as applicable.

A review of plant specific operating experience related to the System Monitoring Program and aging revealed that the following issues have been addressed:

- Various pump and valve flange welds and bolting (carbon steel) were found having significant material loss due to high moisture environment or boric acid accumulations.

- Floor-mounted pipe supports were discovered with excessive corrosion of bolts. (Concrete failure may have contributed from vibration and/or concrete boric acid contamination)

None of the industry or plant operating issues or instances have led to a system component functional failure due to aging.

The Palisades System Monitoring Program has demonstrated that it provides reasonable assurance that aging effects are being managed for System Monitoring Program SSCs. Additionally, this has been demonstrated through NRC inspection reports, audits, self-assessments, and the Corrective Action Program.

This element is consistent with the corresponding element of NUREG-1800, Appendix A.

This element is also consistent with the portions of NUREG-1801, Section XI.M29, “Above Ground Carbon Steel Tanks,” that are applicable to the accessible external surfaces of above ground carbon steel tanks.

Conclusion

The System Monitoring Program is an existing plant-specific program that uses as its bases, various industry and NRC standards. The elements of this program are consistent with the corresponding elements of NUREG-1800, Appendix A.

This program is consistent with the portions of NUREG-1801, Section XI.M29, “Above Ground Carbon Steel Tanks,” that are applicable to the accessible external surfaces of above ground carbon steel tanks.

Reports of recent internal and external assessments of the program have been reviewed. These reports demonstrate that the program has effectively identified and dispositioned issues that could have led to degraded conditions. It is concluded that the System Monitoring Program will be effective in maintaining the intended functions of the affected long-lived, passive SSCs.

Implementation of the System Monitoring Program provides reasonable assurance that aging effects will be managed such that SSCs within the scope of this program will continue to perform their intended functions consistent with the current licensing bases for the period of extended operation.

B2.1.21 Water Chemistry Program

Program Description

The Water Chemistry Program is an existing program that is credited for managing aging effects such as loss-of-material due to general, pitting and crevice corrosion; cracking due to SCC; and steam generator tube degradation caused by denting, intergranular attack (IGA) and outer diameter stress corrosion cracking (ODSCC), by controlling the environment to which internal surfaces of systems and components are exposed. The aging effects are minimized by controlling the chemical species that cause the underlying mechanisms that result in these aging effects. The program provides assurance that an elevated level of contaminants and, where applicable, oxygen does not exist in the systems and components covered by the program, thus minimizing the occurrences of aging effects, and maintaining each component’s ability to perform the intended functions. The program is based on the guidelines in EPRI TR-105714, Rev. 5, and TR-102134, Rev. 5. The One-Time Inspection Program verifies that the Water Chemistry Program is managing the effects of aging of selected components in low flow or stagnant areas.

It is important to note that both the EPRI Primary And Secondary Water Chemistry Guidelines make a clear distinction between “control parameters” and “diagnostic parameters.” Strict adherence to control parameters is expected, whereas diagnostic parameters are suggested, but can be plant specific. Deviations from EPRI recommended diagnostic parameters are not considered exceptions to NUREG-1801.

NUREG-1801 states that the water chemistry control is based on guidelines in EPRI report TR-105714, Rev. 3 for primary water chemistry, and TR-102134, Rev. 3 for secondary water chemistry. Palisades has adopted TR-105714, Rev. 5 and TR-102134, Rev. 5 which are later revisions of the same documents.

The Revision 5 changes to TR-105714 consider the most recent operating experience and laboratory data. It reflects increased emphasis on plant-specific optimization of primary water chemistry to address individual plant circumstances and the impact of the Nuclear Energy Institute (NEI) steam generator initiative, NEI 97-06, which requires utilities to meet the intent of the EPRI Guidelines. TR-105714, Rev. 5 attempts to clearly distinguish between prescriptive requirements and non-prescriptive guidance.

Revision 4 of TR-102134 was issued in November 1996 and provided an increased depth of detail regarding the corrosion mechanisms affecting steam generators and the balance of plant, and also provided additional guidance on how to integrate these and other concerns into the plant-specific optimization process. Revision 5 provides additional details regarding plant-specific optimization and clarifies which portions of the EPRI Guidelines are mandatory under NEI 97-06.

Future revisions of the EPRI Primary and Secondary Water Chemistry Guidelines will be adopted as required, commensurate with industry standards. This is not considered an exception to NUREG-1801.

NUREG-1801 Consistency

The Water Chemistry Program is an existing program that is consistent with NUREG-1801, Section XI.M2, “Water Chemistry.”

Exceptions to NUREG-1801

None.

Enhancements

None.

Aging Management Program Elements

The key elements of the Water Chemistry Program are described below. The results of an evaluation of each key element against NUREG-1801, Section XI.M2, “Water Chemistry,” are also provided below.

Scope of Program

The Water Chemistry Program is credited for managing aging effects such as loss of material due to general, pitting and crevice corrosion; cracking due to SCC; and steam generator tube degradation caused by denting, intergranular attack (IGA), and outer diameter stress corrosion cracking (ODSCC). The program accomplishes this task by monitoring and controlling known detrimental contaminants such as chlorides, fluorides, dissolved oxygen and sulfate concentrations for water chemistry based on guidelines in EPRI TR-105714 for primary water chemistry, and TR-102134 for secondary water chemistry. The program includes sampling activities for primary, borated, secondary, and makeup water systems.

The aging effects/mechanisms managed by the Water Chemistry Program for these SSCs are:

- SCC of austenitic stainless steel, Alloy 600 and Alloy 690 components
- S/G tube degradation and balance of plant materials of construction loss of material, due to denting, IGA, SCC, general, crevice and pitting corrosion
- Loss of material due to crevice and pitting corrosion, MIC, IGA, and fouling of selected components, as applicable.

This element is consistent with NUREG-1801, Section XI.M2, "Water Chemistry."

Preventive Actions

The NUREG-1801 element states: "The program includes specifications for chemical species, sampling and analysis frequencies, and corrective actions for control of reactor water chemistry." The specifications, sampling and analysis frequencies, and corrective actions are discussed in the Parameters Monitored, Inspected, and/or Tested; Monitoring and Trending; and Acceptance Criteria Sections below.

Plant procedures establish limits and controls for harmful chemicals and contaminants on materials of construction for components within the scope of license renewal. The Water Chemistry Program is primarily a mitigative program used to minimize contaminant concentration and mitigate aging effects such as loss of material, cracking due to SCC, and loss of heat transfer.

This element is consistent with NUREG-1801, Section XI.M2, "Water Chemistry."

Parameters Monitored, Inspected, and/or Tested

The concentration of corrosive impurities are monitored and trended to ensure concentrations are below levels known to result in loss of material or crack initiation and growth. Examples of impurities monitored include chlorides, fluorides, sulfates, and dissolved oxygen. Water quality, including pH and conductivity, is verified by continuous monitoring or through sampling, and controlled in accordance with the EPRI guidelines and plant procedures discussed previously.

Plant procedures define the chemicals, impurities and parameters that are monitored and trended for the various water chemistry programs, and provide for administrative controls to assure the integrity, sampling methodology, and storage of samples will not cause a change in the concentration of the chemical species in the samples.

NUREG-1801, Section XI.M2.3 states, "The concentration of corrosive impurities listed in the EPRI guidelines discussed above, which include … hydrogen peroxide, are monitored to mitigate degradation of structural material." The addition of hydrogen peroxide to the Primary Coolant System (PCS) is an industry-accepted practice to reduce the source term, and provide for chemical degassing, and is recommended in the EPRI PWR Primary Water Chemistry

Guideline, Volume 2. Additions and monitoring of hydrogen peroxide in the PCS is accomplished by plant procedures, and is consistent with the NUREG-1801 requirement to monitor the PCS concentration of this chemical.

This element is consistent with NUREG-1801, Section XI.M2, “Water Chemistry.”

Detection of Aging Effects

The Water Chemistry Program mitigates aging effects such as loss of material due to general, pitting, and crevice corrosion; cracking due to SCC; and steam generator tube degradation caused by denting, intergranular attack (IGA) and outer diameter stress corrosion cracking (ODSCC), by controlling the chemical species that cause the underlying aging mechanisms that result in the aging effects. The chemistry parameters measured are defined and listed in the plant water chemistry program procedures for all modes of operation. The Water Chemistry Program does not detect aging effects directly; however in selected areas it does monitor for iron and copper presence, which could indicate loss of material in some components. In addition, inspections of selected components at susceptible locations in low-flow or stagnant portions of a system performed under the One-Time Inspection Program provide verification of the effectiveness of the Water Chemistry Program.

This element is consistent with NUREG-1801, Section XI.M2, “Water Chemistry.”

Monitoring and Trending

Monitoring and trending requirements for all parameters controlled by the Water Chemistry Program are controlled by Plant procedures. Monitoring and trending are based on the EPRI guidelines and plant operating conditions. Whenever corrective actions are taken to address an abnormal chemistry condition, increased sampling and monitoring are typically utilized to verify the effectiveness of these actions. This is a normal operating practice to verify that the corrective action has satisfactorily corrected the parameter that was out of the acceptable range.

This element is consistent with NUREG-1801, Section XI.M2, “Water Chemistry.”

Acceptance Criteria

The acceptance criteria for the chemistry parameters required to be monitored and controlled are based on the EPRI guidelines, Palisades Technical Specifications and the Operating Requirements Manual. Some of the parameters monitored are used for diagnostic purposes only and do not have acceptance criteria recommended by the EPRI guidelines. Water chemistry acceptance criteria are listed in plant procedures.

Any evidence of unacceptable water chemistry results is evaluated, the root cause identified, and the condition corrected. Plant procedures contain steps to be taken when parameters are out of acceptable range.

This element is consistent with, NUREG-1801, Section XI.M2, “Water Chemistry.”

Corrective Actions, Confirmation Process, Administrative Controls

These elements are consistent with the corresponding NUREG-1801 aging management program elements. See Section B1.2 for further discussion.

Operating Experience

Palisades has a comprehensive Operating Experience Program (OEP) that monitors industry issues/events and assesses these for applicability to its own operations. In addition, Palisades relies on the EPRI organization to collect and interpret, in accordance with the Water Chemistry Guidelines, industry operating experience which may have an impact on chemistry control. The Palisades Corrective Action Program (CAP) is used to track, trend and evaluate plant and industry issues/events. Those issues and events, whether external or plant specific, that are potentially significant to water chemistry at Palisades are evaluated. The Water Chemistry Program is augmented, as appropriate, if these evaluations show that program changes will enhance water chemistry and operational safety.

Using the OEP and CAP to focus on industry and plant operating experience ensures that Water Chemistry Program issues are addressed in a timely manner and that age related deterioration of SSC within the scope of the Water Chemistry Program will be effectively managed throughout the license renewal period.

A review of industry operating experience associated with the Water Chemistry Program and aging reveals issues and instances related to:

- Cracking in steam generator welds
- Cracking and pitting of steam generator tubes and components
- Alloy 600 cracking
- Thinning of pipe and components due to erosion/corrosion
- Cracking in safety injection accumulator nozzles
- High wear of Reactor Coolant Pump Aluminum Oxide coated seals
- Cracking of Control Rod Drive Housings
- Cracking of pressurizer instrument tap nozzles
- Cracking of safety injection piping
- Cracking in feedwater piping
- Chemical impurity intrusions into primary and secondary systems
- Resin intrusions into the primary coolant systems

Various related NRC and/or industry generic communications have been issued, and, in turn, have been incorporated into the program as applicable.

A review of plant specific operating experience related to the Water chemistry Program and aging revealed that the following issues have been addressed:

- Defective tubes in the Main Condenser due to steam impingement wear and Microbiologically Influenced Corrosion (MIC) pitting
- Exceeding Action Level 3 Limits for Steam Generator Cation Conductivity

The second item involved exceeding of Action Level 3 limits for Steam Generator Cation Conductivity, which resulted in a shutdown of the plant. The cause of the high conductivity was traced to intrusion of glass-blasting material left in the turbine following a major overhaul/replacement of the turbine. Although there were project oversight weaknesses identified in the events leading up to this chemistry excursion, proper chemistry monitoring quickly identified the rising Cation Conductivity levels, and subsequent actions prevented long term age-related degradation of components as action level 3 limits were exceeded for less than 6 hours. Compensatory actions were taken over the next cycle to ensure a high degree of contaminant removal or neutralization.

The Palisades Water Chemistry Program has demonstrated on several occasions that it provides reasonable assurance that aging effects are being managed for Water Chemistry Program SSC. This has been demonstrated through NRC inspection reports, audits, self-assessments, and the Corrective Action Program.

This element is consistent with NUREG-1801, Section XI.M2, “Water Chemistry.”

Conclusion

The Water Chemistry Program is an existing program that uses as its bases, various industry and NRC standards. This program is consistent with NUREG-1801, Section XI.M2, “Water Chemistry.”

Reports of recent internal and external assessments of the program have been reviewed. These reports demonstrate that the program has effectively identified and dispositioned issues that could have led to degraded conditions. It is concluded that the Water Chemistry Program has been effective in maintaining the intended functions of the affected long-lived, passive SSCs.

The continued implementation of the Water Chemistry Program provides reasonable assurance that aging effects will be managed such that SSCs within the scope of this program will continue to perform their intended functions consistent with the current licensing bases for the period of extended operation.

B3.0 Time Limited Aging Analyses Management Programs

B3.1 Electrical Equipment Qualification Program

Program Description

The Electrical Equipment Qualification Program is an existing program that implements the requirements of 10 CFR 50.49, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants,” at Palisades. 10 CFR 50.49 defines the scope of components to be included, requires the preparation and maintenance of a list of in-scope components, and requires the preparation and maintenance of a qualification file that includes component performance specifications, electrical characteristics, the environmental conditions to which the components could be subjected, and the basis for qualification. 10 CFR 50.49(e)(5) contains provisions for aging that require, in part, consideration of all significant types of aging degradation that can affect component functional capability. 10 CFR 50.49(e)(5) also requires replacement or refurbishment of qualified components prior to the end of its designated life, unless additional life is established through ongoing qualification. EQ programs manage component thermal, radiation, and cyclical aging through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods.

The Electrical Equipment Qualification Program is explicitly governed by regulation independent from license renewal, and NUREG-1801 merely credits that required program as sufficient for license renewal purposes. EEQ Program activities that satisfy license renewal considerations will continue to be managed in accordance with 10 CFR 50.49.

NUREG-1801 Consistency

The Electrical Equipment Qualification Program is consistent with NUREG-1801, Section X.E1, “Environmental Qualification of Electric Components.”

Exceptions to NUREG-1801

None.

Enhancements

None.

Aging Management Program Elements

The following discussion provides the results of an evaluation of each key element with the appropriate ten elements described in Branch Technical Position RLSB-1, “Aging Management Review-Generic,” which is included in Appendix A of NUREG-1800, “Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants,” and

NUREG-1801, Section X.E1, “Environmental Qualification (EQ) of Electric Components,” are provided below.

Scope of Program

The Electrical Equipment Qualification Program is credited for managing the aging effects (thermal, radiation, mechanical cycling, etc.) in components important to safety. The program implements the requirements of 10 CFR 50.49, “Environmental Qualification of Equipment Important to Safety for Nuclear Power Plants.”

The components (Criterion 1) within the scope of the program are safety related electrical equipment that will experience the direct effects of the environment created by a Loss of Coolant Accident (LOCA), High Energy Break (HELB), Main Steam Line Break (MSLB), or Feedwater Line Break (FWLB), and is required to function or must not fail to ensure:

a. The capability to mitigate the consequences of LOCA, HELB, MSLB, or FWLB accident that could result in potential offsite exposures comparable to 10 CFR Part 100 guidelines.
b. The ability to shutdown the reactor and maintain it in a safe shutdown condition.

This program scope also includes non-safety related electrical equipment whose failure under the postulated accident environmental conditions could prevent satisfactory accomplishment of safety functions provided by safety related equipment. In addition, the scope of this program includes certain post-accident monitoring equipment. This program scope is as defined in 10 CFR 50.49.

This element is consistent with NUREG-1801, Section X.E1, “Environmental Qualification (EQ) of Electrical Components.”

Preventive Actions

10 CFR 50.49 does not require actions that prevent aging effects. NUREG-1801 Program X.E1 states that EEQ Program actions that could be viewed as preventive actions include (a) establishing the component service condition tolerance and aging limits (e.g., qualified life or condition limit), and (b) where applicable, requiring specific installation, inspection, monitoring or periodic maintenance actions to maintain component aging effects within the bounds of the qualification bases. The Palisades EEQ Program includes these features.

This element is consistent with NUREG-1801, Section X.E1, “Environmental Qualification (EQ) of Electrical Components.”

Parameters Monitored, Inspected, and/or Tested

The qualified life of a component in the EEQ Program is not based on condition or performance monitoring. However, pursuant to R.G. 1.89, such monitoring programs can provide an acceptable basis to modify a qualified life through reanalysis. Monitoring or inspection of certain environmental conditions or component parameters may be used to ensure that a component is within the bounds of its qualification bases, or as a means to modify the qualified life during the extended period of operation.

This element is consistent with NUREG-1801, Section X.E1, “Environmental Qualification (EQ) of Electrical Components.”

Detection of Aging Effects

10 CFR 50.49 does not require the detection of aging effects for in-service components. Refurbishment, replacement or re-qualification of qualified components is based on the analyses and test reports documented in each component's qualification file. As implemented by the EEQ Program, monitoring or inspection of certain environmental conditions or component parameters may be used to ensure that the component is within the bounds of its qualification bases, or as a means to modify the qualified life.

This element is consistent with NUREG-1801, Section X.E1, “Environmental Qualification (EQ) of Electrical Components.”

Monitoring and Trending

10 CFR 50.49 does not require monitoring and trending of component condition or performance parameters of in-service components to manage the effects of aging. NUREG-1801, Section X.E1, states that EEQ Program actions that could be viewed as monitoring include monitoring how long qualified components have been installed. In addition, monitoring or inspection of certain environmental conditions or component parameters may be used to ensure that a component is within the bounds of its qualification bases, or as a means to modify the qualified life during the period of extended operation.

This element is consistent with NUREG-1801, Section X.E1, “Environmental Qualification (EQ) of Electrical Components.”

Acceptance Criteria

10 CFR 50.49 acceptance criteria, as implemented in the EEQ Program, requires each in-service EQ component to be maintained within the bounds of its qualification bases, including (a) its established qualified life, and (b) continued qualification for the projected accident conditions. 10 CFR 50.49 requires maintenance (refurbishment, replacement), or re-qualification prior to exceeding the qualified life of each installed device in-scope of the program. When monitoring is used to modify a component qualified life (e.g., extend qualified life based on actual operating environment), in accordance with applicable 10 CFR 50.49(f) qualification methods, the revised analysis assumptions, in effect, establish the new acceptance criteria for the component's operating environment.

This element is consistent with NUREG-1801, Section X.E1, “Environmental Qualification (EQ) of Electrical Components.”

Corrective Actions, Confirmation Process, Administrative Controls

These elements are consistent with the corresponding NUREG-1801 aging management program elements. See Section B1.2 for further discussion.

Operating Experience

Palisades has a comprehensive Operating Experience Program (OEP) that monitors industry issues/events and assesses these for applicability to its own operations. In addition, the Palisades Corrective Action Program (CAP) is used to evaluate, track, and trend plant issues/events. Those issues and events, whether industry or plant specific, that are potentially significant to the Electrical Equipment Qualification Program at Palisades, are evaluated. The Electrical Equipment Qualification Program is augmented, as appropriate, if these evaluations show that program changes are required to enhance program effectiveness.

Using the OEP and CAP to focus on industry and plant operating experience ensures that Electrical Equipment Qualification Program issues are addressed in a timely manner and that age-related deterioration of components within the scope of the Electrical Equipment Qualification Program will be effectively managed throughout the period of extended operation.

A review of applicable operating experience was performed to determine if there were deficiencies or recurring failures that would raise questions about EEQ Program effectiveness. No significant items were found. Similar reviews were performed of inspection, assessment, and audit reports. No significant findings were identified. Some issues have been identified that have resulted in appropriate corrective actions and enhancements.

The Palisades Electrical Equipment Qualification Program has demonstrated that it provides reasonable assurance that aging effects are being managed for all EQ components. This has been demonstrated through NRC inspection reports, audits, self-assessments, and the Corrective Action Program.

This element is consistent with NUREG-1801, Section X.E1, “Environmental Qualification (EQ) of Electrical Components.”

Conclusion

The Electrical Equipment Qualification Program is a plant specific program, implemented in accordance with 10 CFR 50.49, that uses as its bases, various industry and NRC standards. This program is consistent with NUREG-1801, Section X.E1, “Environmental Qualification (EQ) of Electrical Components.”

Reports of recent internal and external assessments of the program have been reviewed. These reports demonstrate that the program has effectively identified and dispositioned issues that could have led to degraded conditions. It is concluded that the Electrical Equipment Qualification Program has been effective in maintaining the intended functions of the affected long-lived, passive SSCs.

The continued implementation of the Electrical Equipment Qualification Program provides reasonable assurance that aging effects will be managed such that SSCs within the scope of this program will continue to perform their intended functions consistent with the current licensing bases for the period of extended operation.

B3.2 Fatigue Monitoring Program

Program Description

The Fatigue Monitoring Program is a new program that ensures that limits on fatigue usage are not exceeded during the renewal term. The program monitors and tracks selected cyclic loading transients (cycle counting) and their effects on susceptible components. Palisades has selected this option under 10 CFR 54.21 to manage cracking due to metal fatigue of the reactor coolant pressure boundary during the extended period of operation.

The Fatigue Monitoring Program provides the cycle counting activities credited in Section 4.3 for confirming analytically derived cumulative usage values for applicable locations. Specific locations that may be subject to cyclic loading that could cause fatigue cracking are monitored using a computer-based monitoring program provided by EPRI, called

FatiguePro. If warranted, other monitoring methods in addition to cycle counting may also be employed under this program to monitor specific locations.

NUREG-1801 Consistency

10 CFR 54.21 allows disposition of TLAAs using an aging management program (AMP). The new Palisades Fatigue Monitoring Program is consistent with the description presented in NUREG-1801, Section X.M1, “Metal Fatigue of Reactor Coolant Pressure Boundary,” as an acceptable option for managing metal fatigue for the reactor coolant pressure boundary, including consideration of environmental effects.

Exceptions to NUREG-1801

None.

Enhancements

A Fatigue Monitoring Program will be developed and implemented. Features of the program will include monitoring and tracking of selected cyclic loading transients (cycle counting) and their effects on critical reactor pressure boundary components and other selected components.

Note that the element descriptions describe the program as it will exist after it has been implemented. The program is scheduled to be implemented prior to the period of extended operation.

Aging Management Program Elements

The key elements of the Fatigue Monitoring Program are described below. The results of the evaluation of each element against NUREG-1801, Section X.M1, “Metal Fatigue of Reactor Coolant Pressure Boundary,” is also provided.

Scope of Program

The aging effect/mechanism managed by the Fatigue Monitoring Program for components is cumulative fatigue damage/fatigue. The specific components and locations encompassed by the cycle counting activities performed under the Fatigue Monitoring Program have been identified. These locations are verified by analysis to be the leading locations for potential cumulative fatigue damage during the period of extended operation. These components include those shown for older vintage Combustion Engineering plants in NUREG/CR-6260, as applicable to Palisades. The transients to be counted under the cycle-based fatigue monitoring program include design transients as well as those not considered in the original design.

The Fatigue Monitoring Program also addresses the effects of the coolant environment on component fatigue life by analyzing the impact of the reactor coolant environment on a sample of critical components and adjusting the CUF accordingly for those locations. Critical components are evaluated by applying environmental correction factors to the existing fatigue analyses. Formulas for calculating the environmental life correction factors are those contained in NUREG/CR-6583 for carbon and low-alloy steels and in NUREG/CR-5704 for austenitic stainless steels.

This program element is consistent with NUREG-1801, Section X.M1, "Metal Fatigue of Reactor Coolant Pressure Boundary."

Preventive Actions

The Palisades Fatigue Monitoring Program includes measures to prevent cracking due to cumulative fatigue damage in metal components within the reactor coolant pressure boundary and other selected locations. Cracking attributable to anticipated cyclic strains in the reactor coolant pressure boundary or other selected locations are prevented by monitoring actual incremental fatigue usage to ensure that the cumulative usage factor (CUF) remains below the Code requirement of <1.0 during the period of extended operation. Below this value, fatigue cracks are not expected to occur.

This element is consistent with NUREG-1801, Section X.M1, "Metal Fatigue of Reactor Coolant Pressure Boundary."

Parameters Monitored, Inspected, and/or Tested

The Palisades Fatigue Monitoring program monitors selected system temperatures, pressures and flows to determine the number and severity of cyclic loading events anticipated to be significant contributors to cumulative fatigue damage. If conditions warrant, detailed local monitoring of cyclic loading stresses (stress-based fatigue monitoring) at specific locations may be used to compute the fatigue usage for certain components.

This element is consistent with NUREG-1801, Section X.M1, "Metal Fatigue of Reactor Coolant Pressure Boundary."

Detection of Aging Effects

The Fatigue Monitoring Program provides for periodic updates of the plant cycle count and fatigue usage calculations. The metal fatigue aging effect is monitored using FatiguePro, which is an EPRI software product for plant transient monitoring and fatigue usage calculations. Plant operating cycles are tracked against design limits. Fatigue usage factors are computed on an on-going basis for bounding components using plant instrument data.

This element is consistent with NUREG-1801, Section X.M1, “Metal Fatigue of Reactor Coolant Pressure Boundary.”

Monitoring and Trending

The Palisades Fatigue Monitoring Program is used to monitor a sample of high fatigue usage locations. This sample includes the applicable locations identified in NUREG/CR 6260 for older vintage CE plants.

Use of FatiguePro enables actual fatigue usage to be trended, real time, based upon actual plant data. FatiguePro is loaded with initial baseline data which includes the total transient cycles counted under the prior cycle counting activities, as well as the initial cumulative fatigue usage values calculated for monitored locations. Thereafter, the incremental contribution for defined transient cycles are added to the initial usage values to produce running totals of cumulative fatigue usage at each monitored location. These totals, as well as the incremental cumulative fatigue usage values for the period, are available to the user on demand.

This element is consistent with NUREG-1801, Section X.M1, “Metal Fatigue of Reactor Coolant Pressure Boundary.”

Acceptance Criteria

The acceptance criteria consists of maintaining the fatigue usage factor less than or equal to the design code allowable limit of 1.0, considering environmental fatigue effects, and the cumulative number of plant cycles less than or equal to the cyclic design basis. For the components for which an environmentally-assisted fatigue evaluation is performed, the fatigue usage factor limit is taken to be a value of $1/F_{en}$. The acceptance criteria ensures that all original structural margins considered in the plant design are maintained through the extended period of operation.

This element is consistent with NUREG-1801, Section X.M1, “Metal Fatigue of Reactor Coolant Pressure Boundary.”

Corrective Actions, Confirmation Process, Administrative Controls

These elements are consistent with the corresponding NUREG-1801 aging management program elements. See Section B1.2 for further discussion.

Operating Experience

Palisades has a comprehensive Operating Experience Program (OEP) that monitors industry issues/events and assesses these for applicability to its own operations. In addition, the Palisades Corrective Action Program (CAP) is used to track, trend and evaluate plant issues/events. Those issues and events, whether external or plant specific, that are potentially significant to the Fatigue Monitoring Program at Palisades will be evaluated. The Fatigue Monitoring Program will be augmented, as appropriate, if these evaluations show that program changes or monitoring of additional locations will enhance program effectiveness.

Using the OEP and CAP to focus on industry and plant operating experience ensures that Fatigue Monitoring Program issues are addressed in a timely manner and that age related deterioration of SSC within the scope of the Fatigue Monitoring Program will be effectively managed throughout the license renewal period.

Industry Operating Experience has shown that significant thermal stresses in piping connected to reactor coolant systems have caused piping failures and subsequent leakage from reactor coolant systems at some plants. NRC Bulletin No. 88-08 and its supplements brought wide visibility to these issues beginning in June 1988. Later that same year, NRC Bulletin No. 88-11 covered issues pertaining specifically to thermal stratification and unexpected thermal cyclic loadings in the pressurizer surge line. NRC Bulletins 88-08 and 88-11 highlighted the need for evaluation and monitoring of cyclic loading events that were not considered in the original design of Class 1 piping and components. Subsequent to the identification of thermal stratification issues, significant resources have been expended in an effort to monitor cumulative fatigue damage due thermal fatigue.

While actual failures attributable to cumulative fatigue damage/thermal fatigue have been relatively rare, failures have occurred. EPRI has captured the significant operating experience in MRP-85, Materials Reliability Program: Operating Experience Regarding Thermal Fatigue of Piping Connected to PWR Reactor Coolant Systems. This experience was reviewed for development of the Fatigue Monitoring Program, but is not repeated here.

Since this is a new program, there is no significant plant-specific programmatic operating experience with which to make a determination as to its effectiveness. However, Palisades has been tracking and logging plant-specific transient cycles. A review was performed of plant-specific analytical results and transient logs for all tracked transients, as well as industry operating experience related to metal fatigue. The need for more rigorous cycle counting methodology has been determined. In addition, review of the tracked transients revealed the following issues:

Feedwater Flow Cycling

The use of auxiliary feedwater to slug feed feedwater into the Steam Generator feedwater nozzles during hot standby conditions has been shown to be a significant thermal stratification stress cycling event, not included in original design bases. The replacement steam generators, installed in 1990, incorporated separate auxiliary feedwater nozzles which minimize the potential for water hammer and thermal stratification. Use of these separate auxiliary feedwater nozzles during hot standby has been shown to minimize the occurrence of thermally stratified feedwater flow into the steam generators. Therefore Palisades has determined that this transient need not be counted.

Charging and Letdown Transients

Charging nozzles generally show a high design basis fatigue usage due to the relatively large number of rapid temperature transients that can occur when either charging or letdown flow, or both, are terminated and reinitiated. These transients can occur during normal plant operation, including during surveillance testing of the control and isolation valves. Unlike many of the design basis transients, the charging and letdown transients occur fairly often in normal plant operation and can be nearly as severe as the design basis transient. Therefore, these plant transients will be added to the cycle tracking list. If warranted, the charging nozzles may be included as stress-based fatigue-monitored locations, as these nozzles tend to be the highest fatigue locations in the system and could serve to bound the other system locations.

Pressurizer Spray Line Cycling

The Palisades pressurizer spray nozzle can be anticipated to show a relatively high design fatigue usage due to the large temperature shocks that regularly occur during heatup and cooldown conditions. Stratified flow in the pressurizer spray line was not considered to be significant because Palisades operates with continuous pressurizer spray flow of 60 to 80 gpm. Palisades assumes that one spray actuation will occur during each plant cooldown. The fatigue damage to the spray nozzle could conceivably be tracked just by counting the number of plant cooldowns. However, it is expected that main spray actuations of varying severities and duration could also occur during other plant conditions, such as plant heatup. Also, under certain conditions, the temperature difference between the source of the auxiliary spray and the main spray could be significant, thereby having the potential to cause significant thermal shocking at the auxiliary spray-to-main spray tee. Based upon the uncertainty in attempting to precisely monitor the fatigue damage of the spray nozzle, Palisades will not track spray cycles, but may include the spray nozzle as a stress-based fatigue-monitored location.

<u>Pressurizer Surge Line Cycling</u>

The issue of thermal stratification cycling in the surge line was addressed extensively in the industry's response to NRC IE Bulletin 88-11. CE evaluated the surge line and concluded that the fatigue usage of the pressurizer surge nozzle and the hot leg piping surge nozzle will be relatively high, but less than the ASME CUF limit of 1.0. It is a complex undertaking to identify the occurrence and magnitude of insurge or outsurge flow in the surge line. Based upon the complexity and uncertainty in the ability to adequately monitor the fatigue damage of the surge lines and the connected nozzles, Palisades does not intend to track surge cycles, but may monitor both surge line nozzles using stress-based fatigue-monitoring

<u>Failures in Unisolable Piping</u>

Intermittent valve leakage, causing thermally stratified conditions in unisolable sections of piping, have caused failures in safety injection nozzles and in socket welds near the reactor coolant piping in several plants. Such conditions were the subject of NRC IE Bulletin 88-08. Based upon that experience, the high pressure safety injection nozzles may be included in the Fatigue Monitoring Program as either cycle-based or stress-based fatigue locations.

This element is consistent with NUREG-1801, Section X.M1, “Metal Fatigue of Reactor Coolant Pressure Boundary.”

Conclusion

The Fatigue Monitoring Program is a new program that uses as its bases various industry and NRC documents, as well as industry operating experience, plant-specific design and plant-specific analysis. This program is consistent with NUREG-1801, Section X.M1, “Metal Fatigue of Reactor Coolant Pressure Boundary.”

Implementation of the Fatigue Monitoring Program will provide reasonable assurance that cumulative fatigue usage will be managed such that systems and components within the scope of this program will continue to perform their intended functions, consistent with the current licensing bases, during the period of extended operation.

Appendix B References

1. NUREG-1800, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, U.S. Nuclear Regulatory Commission, July 2001.

2. NUREG-1801, Generic Aging Lessons Learned (GALL) Report, U.S. Nuclear Regulatory Commission, July 2001.

3. Quality Program Description for Nuclear Power Plants (Part 2) - Palisades Nuclear Plant (CPC-2A)

4. 10 CFR 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants”

5. EA-03-009, “Issuance of First Revised NRC Order Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors” (dated 2/20/04)

6. Palisades Plant Response to Revised NRC Order EA-03-009, “Issuance of First Revised NRC Order Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors” (dated 3/8/04)

7. 45. SAND96-0344, UC-523, “Aging Management Guideline for Commercial Nuclear Power Plants - Electrical Cables and Terminations” (DOE and EPRI, dated September 1996)

APPENDIX C

(Not Used for This Application)

APPENDIX C
Contents

C1.0 Appendix C - Not Used

Appendix C is not used in this application.

APPENDIX D

TECHNICAL SPECIFICATION CHANGES

APPENDIX D
Contents

D1.0 Appendix D - Technical Specifications Changes

10 CFR 54.22 requires that an application for license renewal include any Technical Specification changes or additions that are necessary to manage the effects of aging during the period of extended operation. A review of the information provided in this License Renewal Application and the Palisades Technical Specifications confirms that no changes to the Technical Specifications are necessary.

APPENDIX E

(Provided as Linked Document)

APPENDIX E
Contents

E1.0 Appendix E - Environmental Report

This page provides a link to the Environmental Report.