NUREG–0847
Supplement No. 15

Safety Evaluation Report
related to the operation of
Watts Bar Nuclear Plant,
Units 1 and 2

Docket Nos. 50–390 and 50–391

Tennessee Valley Authority

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

June 1995

DISCLAIMER

ABSTRACT

This report supplements the Safety Evaluation Report (SER), NUREG-0847 (June 1982), Supplement No. 1 (September 1982), Supplement No. 2 (January 1984), Supplement No. 3 (January 1985), Supplement No. 4 (March 1985), Supplement No. 5 (November 1990), Supplement No. 6 (April 1991), Supplement No. 7 (September 1991), Supplement No. 8 (January 1992), Supplement No. 9 (June 1992), Supplement No. 10 (October 1992), Supplement No. 11 (April 1993), Supplement No. 12 (October 1993), Supplement No. 13 (April 1994), and Supplement No. 14 (December 1994) issued by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission with respect to the application filed by the Tennessee Valley Authority, as applicant and owner, for licenses to operate the Watts Bar Nuclear Plant, Units 1 and 2 (Docket Nos. 50-390 and 50-391). The facility is located in Rhea County, Tennessee, near the Watts Bar Dam on the Tennessee River. This supplement provides recent information regarding resolution of some of the outstanding and confirmatory items, and proposed license conditions identified in the SER.

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ABBREVIATIONS

ABGTS	auxiliary building gas treatment system
ACP	access control portal
ACR	auxiliary control room
AOI	abnormal operating instruction
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	anticipated transient without scram
BIT	boron injection tank
BOP	balance of plant
CAP	corrective action program
CAS	central alarm station
CCP	centrifugal charging pump
CCTV	closed circuit television
CFR	Code of Federal Regulations
CNPP	Corporate Nuclear Performance Plan
CRT	cathode ray tube
CSF	critical safety function
DBA	design-basis accident
DCRDR	detailed control room design review
DEC	Digital Equipment Corporation
DEI	dose equivalent iodine
DNB	departure from nucleate boiling
EAB	exclusion area boundary
ECCS	emergency core cooling system
EI	electrical and instrumentation
ENI	electromagnetic interference
EPRI	Electric Power Research Institute
ERCW	essential raw cooling water
ERFDS	emergency response facilities data system
FSAR	final safety analysis report
FWLB	feedwater line break
GDC	general design criterion
GL	generic letter
HED	human engineering discrepancy
HPI	high-pressure injection
IE	Office of Inspection and Enforcement
IEEE	Institute of Electrical and Electronics Engineers
INEL	Idaho National Engineering Laboratory
IPE	individual plant examination
ISA	Instrument Society of America
ITI	incore thermocouple indicator

LCO	limiting condition for operation
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
LPZ	low population zone
LTOP	low-temperature overpressure protection
MCR	main control room
MSF	member of the security force
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation (NRC)
NSSS	nuclear steam supply system
NUDOCS	NRC document control system
OBE	operating basis earthquake
PA	protected area
PORV	pilot-operated relief valve
PTS	pressurized thermal shock
PWR	pressurized-water reactor
QA	quality assurance
RCS	reactor coolant system
RFI	radio frequency interference
RG	regulatory guide
RHR	residual heat removal
RPS	reactor protection system
RTD	resistance temperature detector
RTV	room temperature vulcanized
SAMA	Scientific Apparatus Manufacturers Association
SAS	secondary alarm station
SBLOCA	small-break loss-of-coolant accident
SER	safety evaluation report
SGTR	steam generator tube rupture
SI	safety injection
SIAS	safety injection actuation signal
SP	special program
SPDS	safety parameter display system
SRP	standard review plan
SSER	supplement to safety evaluation report
SSI	soil-structure interaction
SWEC	Stone & Webster Engineering Corporation
TAC	technical assignment control
TER	technical evaluation report
TRM	Technical Requirements Manual
TSs	Technical Specifications
TTD	trip time delay
TVA	Tennessee Vally Authority
USI	unresolved safety issue
WBN	Watts Bar Nuclear (Plant)

WBNPP	Watts Bar Nuclear Performance Plan
WEMD	Westinghouse Electro-Mechanical Division
WISP	Workload Information and Scheduling Program

1 INTRODUCTION AND DISCUSSION

1.1 Introduction

In June 1982, the Nuclear Regulatory Commission staff (NRC staff or staff) issued a Safety Evaluation Report, NUREG-0847, regarding the application by the Tennessee Valley Authority (TVA or the applicant) for licenses to operate the Watts Bar Nuclear Plant, Units 1 and 2. The Safety Evaluation Report (SER) was followed by SER Supplement No. 1 (SSER 1, September 1982), Supplement No. 2 (SSER 2, January 1984), Supplement No. 3 (SSER 3, January 1985) Supplement No. 4 (SSER 4, March 1985), Supplement No. 5 (SSER 5, November 1990), Supplement No. 6 (SSER 6, April 1991), Supplement No. 7 (SSER 7, September 1991), Supplement No. 8 (SSER 8, January 1992), Supplement No. 9 (SSER 9, June 1992), Supplement No. 10 (SSER 10, October 1992), Supplement No. 11 (SSER 11, April 1993), Supplement No. 12 (October 1993), Supplement No. 13 (SSER 13, April 1994), and Supplement No. 14 (SSER 14, December 1994). As of this date, the staff has completed its review of the applicant's Final Safety Analysis Report (FSAR) up to Amendment 88.

The SER and its supplements were written to agree with the format and scope outlined in the Standard Review Plan (SRP, NUREG-0800). Issues raised by the SRP review that were not closed out when the SER was published were classified into outstanding issues, confirmatory issues, and proposed license conditions (see Sections 1.7, 1.8, and 1.9, respectively, which follow).

In addition to the guidance in the SRP, the staff would issue generic requirements or recommendations in the form of bulletins and generic letters. Each of these bulletins and generic letters carries its own applicability, work scope, and acceptance criteria; some are applicable to Watts Bar. The implementation status was addressed in Section 1.14 of SSER 6. The staff is reevaluating the status of implementation of all bulletins and generic letters.

Each of the following sections or appendices of this supplement is numbered the same as the section or appendix of the SER that is being updated, and the discussions are supplementary to, and not in lieu of, the discussion in the SER, unless otherwise noted. Accordingly, Appendix A continues the chronology of the safety review. Appendix C, originally published in the SER, is supplemented here. Appendix E lists principal contributors to this supplement. Appendices CC and DD are added in this supplement. The other appendices are not changed by this supplement.

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1.7 Summary of Outstanding Issues

In SER Section 1.7, the staff listed 17 outstanding issues (open items) that had not been resolved at the time the SER was issued. Additional outstanding issues were added in SER supplements that followed. In this section, the staff updates the status of those items. The completion status of each of the issues is tabulated below with the relevant document in which the issue was last addressed shown in parentheses. Detailed, up-to-date status information for still-unresolved issues is conveyed in the staff's summaries of the licensing status meetings.

Issue[1]	Status	Section
(1) Potential for liquefaction beneath ERCW pipelines and Class 1E electrical conduit	Resolved (SSER 3)	2.5.4.4
(2) Buckling loads on Class 2 and 3 supports	Resolved (SSER 4)	3.9.3.4
(3) Inservice pump and valve test program (TAC M74801)	Resolved (SSER 14)	3.9.6
(4) Qualification of equipment		
(a) Seismic (TAC M71919)	Resolved (SSER 9)	3.10
(b) Environmental (TAC M63591)	Resolved (SSER 15)	3.11
(5) Preservice inspection program (TAC M63627)	Resolved for Unit 1 (SSERs 10 and 12)	5.2.4, 6.6, App. Z
(6) Pressure-temperature limits for Unit 2 only	On hold (SER)	5.3.2, 5.3.3
(7) Model D-3 steam generator preheater tube degradation	Resolved (SSER 4)	5.4.2.2
(8) Branch Technical Position CSB 6-4	Resolved (SSER 3)	6.2.4
(9) H_2 analysis review	Resolved (SSER 4)	6.2.5
(10) Safety valve sizing analysis (WCAP-7769)	Resolved (SSER 2)	5.2.2
(11) Compliance of proposed design change to the offsite power system to GDC 17 and 18 (TAC M63649)	Resolved (SSER 13)	8.2
(12) Fire-protection program (TAC M63648)	Under review (SER)	9.5.1

[1] The TAC (technical assignment control) number that appears in parentheses after the issue title is an internal NRC control number by which the issue is managed through the Workload Information and Scheduling Program (WISP) and by which relevant documents are filed. Documents associated with each TAC number can be located by the NRC document control system, NUDOCS/AD.

Issue	Status	Section
(13) Quality classification of diesel generator auxiliary system piping and components (TAC M63638)	Resolved (SSER 5)	9.5.4.1
(14) Diesel generator auxiliary system design deficiencies (TAC M63638)	Resolved (SSER 5)	9.5.4, 9.5.5, 9.5.7
(15) Physical Security Plan (TAC M63657)	Resolved (SSER 15)	13.6
(16) Boron-dilution event	Resolved (SSER 4)	15.2.4.4
(17) QA Program (TAC M76972)	Resolved (SSER 13)	17
(18) Seismic classification of cable trays and conduit (TACs R00508, R00516)	Resolved (SSER 8)	3.2.1, 3.10
(19) Seismic design concerns (TACs M79717, M80346):		
(a) Number of OBE events	Resolved (SSER 8)	3.7.3
(b) 1.2 multi-mode factor	Resolved (SSER 9)	3.7.3
(c) Code usage	Resolved (SSER 8)	3.7.3
(d) Conduit damping values	Resolved (SSER 8)	3.7.3
(e) Worst case, critical case, bounding calculations	Resolved (SSER 12)	3.7.3
(f) Mass eccentricities	Resolved (SSER 8)	3.7.2.1.2
(g) Comparison of set A versus set B response	Resolved (SSER 11)	3.7.2.12
(h) Category 1(L) piping qualification	Resolved (SSER 8)	3.9.3
(i) Pressure relief devices	Resolved (SSER 7)	3.9.3.3
(j) Structural issues	Resolved (SSER 9)	3.8
(k) Update FSAR per 12/18/90 letter	Resolved (SSER 8)	3.7
(20) Mechanical systems and components (TACs M79718, M80345)		
(a) Feedwater check valve slam	Resolved (SSER 13)	3.9.1
(b) New support stiffness and deflection limits	Resolved (SSER 8)	3.9.3.4
(21) Removal of RTD bypass system (TAC M63599)	Resolved (SSER 8)	4.4.3
(22) Removal of upper head injection system (TAC M77195)	Resolved (SSER 7)	6.3.1
(23) Containment isolation using closed systems (TAC M63597)	Resolved (SSER 12)	6.2.4
(24) Main steamline break outside containment (TAC M63632)	Resolved (SSER 14)	3.6.1

Issue	Status	Section
(25) Health Physics Program (TAC M63647)	Resolved (SSER 10)	12
(26) Regulatory Guide 1.97, Instruments To Follow Course of Accident (TACs M77550, M77551)	Resolved (SSER 9)	7.5.2
(27) Containment sump screen design anomalies (TAC M77845)	Resolved (SSER 9)	6.3.3
(28) Emergency procedure (TAC M77861)	Resolved (SSER 9)	13.5.2.1

1.8 Summary of Confirmatory Issues

In SER Section 1.8, the staff listed 42 confirmatory issues for which additional information and documentation were required to confirm preliminary conclusions. Issue 43 was added in SSER 6. In this section, the staff updates the status of those items for which the confirmatory information has subsequently been provided by the applicant and for which review has been completed by the staff. The completion status of each of the issues is tabulated below, with the relevant document in which the issue was last addressed shown in parentheses.

Issue	Status	Section
(1) Design-basis groundwater level for the ERCW pipeline	Resolved (SSER 3)	2.4.8
(2) Material and geometric damping effect in SSI analysis	Resolved (SSER 3)	2.5.4.2
(3) Analysis of sheetpile walls	Resolved (SSER 3)	2.5.4.2
(4) Design differential settlement of piping and electrical components between rock-supported structures	Resolved (SSER 3)	2.5.4.3
(5) Upgrading ERCW system to seismic Category I (TAC M63617)	Resolved (SSER 5)	3.2.1, 3.2.2
(6) Seismic classification of structures, systems, and components important to safety (TAC M63618)	Resolved (SSER 5)	3.2.1
(7) Tornado-missile protection of diesel generator exhaust	Resolved (SSER 2)	3.5.2, 9.5.4.1, 9.5.8
(8) Steel containment building buckling research program	Resolved (SSER 3)	3.8.1
(9) Pipe support baseplate flexibility and its effects on anchor bolt loads (IE Bulletin 79-02) (TAC M63625)	Resolved (SSER 8)	3.9.3.4

Issue	Status	Section
(10) Thermal performance analysis	Resolved (SSER 2)	4.2.2
(11) Cladding collapse	Resolved (SSER 2)	4.2.2
(12) Fuel rod bowing evaluation	Resolved (SSER 2)	4.2.3
(13) Loose-parts monitoring system	Resolved (SSER 3)	4.4.5
(14) Installation of residual heat removal flow alarm	Resolved (SSER 5)	5.4.3
(15) Natural circulation tests (TACs M63603, M79317, M79318)	Resolved (SSER 10)	5.4.3
(16) Atmospheric dump valve testing	Resolved (SSER 2)	5.4.3
(17) Protection against damage to containment from external pressure	Resolved (SSER 3)	6.2.1.1
(18) Designation of containment isolation valves for main and auxiliary feedwater lines and feedwater bypass lines (TAC M63623)	Resolved (SSER 5)	6.2.4
(19) Compliance with GDC 51	Resolved (SSER 4)	6.2.7, App. H
(20) Insulation survey (sump debris)	Resolved (SSER 2)	6.3.3
(21) Safety system setpoint methodology	Resolved (SSER 4)	7.1.3.1
(22) Steam generator water level reference leg	Resolved (SSER 2)	7.2.5.9
(23) Containment sump level measurement	Resolved (SSER 2)	7.3.2
(24) IE Bulletin 80-06	Resolved (SSER 3)	7.3.5
(25) Overpressure protection during low-temperature operation	Resolved (SSER 4)	7.6.5
(26) Availability of offsite circuits	Resolved (SSER 2)	8.2.2.1
(27) Non-safety loads powered from the Class 1E ac distribution system	Resolved (SSER 2)	8.3.1.1
(28) Low and/or degraded grid voltage condition (TAC M63649)	Resolved (SSER 13)	8.3.1.2
(29) Diesel generator reliability qualification testing (TAC M63649)	Resolved (SSER 7)	8.3.1.6
(30) Diesel generator battery system	Resolved (SSER 2)	8.3.2.4

Issue	Status	Section
(31) Thermal overload protective bypass	Resolved (SSER 2)	8.3.3.1.2
(32) Update FSAR on sharing of dc and ac distribution systems (TAC M63649)	Resolved (SSER 13)	8.3.3.2.2
(33) Sharing of raceway systems between units	Resolved (SSER 2)	8.3.3.2
(34) Testing Class 1E power systems	Resolved (SSER 2)	8.3.3.5.2
(35) Evaluation of penetration's capability to withstand failure of overcurrent protection device (TAC M63649)	Resolved (SSER 7)	8.3.3.6
(36) Missile protection for diesel generator vent line (TAC M63639)	Resolved (SSER 5)	9.5.4.2
(37) Component cooling booster pump relocation	Resolved (SSER 5)	9.2.2
(38) Electrical penetrations documentation (TAC M63648)	Under review (SER)	9.5.1.3
(39) Compliance with NUREG/CR-0660 (TAC M63639)	Resolved (SSER 5)	9.5.4.1
(40) No-load, low-load, and testing operations for diesel generator (TAC M63639)	Resolved (SSER 5)	9.5.4.1
(41) Initial test program	Resolved (SSER 3)	14
(42) Submergence of electrical equipment as result of a LOCA (TAC M63649)	Resolved (SSER 13)	8.3.3.1.1
(43) Safety parameter display system (TAC M73723)	Resolved (SSER 15)	18.2

1.9 Summary of Proposed License Conditions

In Section 1.9 of the SER and in SSERs that followed, the staff listed 43 proposed license conditions. Since these documents were issued, the applicant has submitted additional information on some of these items, thereby removing the necessity to impose a condition. The completion status of the proposed license conditions is tabulated below, with the relevant document in which the issue was last addressed shown in parentheses. Detailed, up-to-date status of still-unresolved issues is conveyed in the staff's summaries of the licensing status meetings.

Proposed Condition	Status	Section
(1) Relief and safety valve testing (II.D.1)	Resolved (SSER 3)	3.9.3.3, 5.2.2

Proposed Condition	Status	Section
(2) Inservice testing of pumps and valves (TAC M74801)	Resolved (SSER 12)	3.9.6
(3) Detectors for inadequate core cooling (II.F.2) (TACs M77132, M77133)	Resolved (SSER 10)	4.4.8
(4) Inservice Inspection Program (TAC M76881)	Resolved (SSER 12)	5.2.4, 6.6
(5) Installation of reactor coolant vents (II.B.1)	Resolved (SSER 5)	5.4.5
(6) Accident monitoring instrumentation (II.F.1)		
(a) Noble gas monitor (TAC M63645)	Resolved (SSER 5)	11.7.1
(b) Iodine particulate sampling (TAC M63645)	Resolved (SSER 6)	11.7.1
(c) High-range in-containment radiation monitor (TAC M63645)	Resolved (SSER 5)	12.7.2
(d) Containment pressure	Resolved (SSER 5)	6.2.1
(e) Containment water level	Resolved (SSER 5)	6.2.1
(f) Containment hydrogen	Resolved (SSER 5)	6.2.5
(7) Modification to chemical feedlines (TAC M63622)	Resolved (SSER 5)	6.2.4
(8) Containment isolation dependability (II.E.4.2) (TAC M63633)	Resolved (SSER 5)	6.2.4
(9) Hydrogen control measures (NUREG-0694, II.B.7) (TAC M77208)	Resolved (SSER 8)	6.2.5, App. C
(10) Status monitoring system/BISI (TACs M77136, M77137)	Resolved (SSER 7)	7.7.2
(11) Installation of acoustic monitoring system (II.D.3)	Resolved (SSER 5)	7.8.1
(12) Diesel generator reliability qualification testing at normal operating temperature	Resolved (SSER 2)	8.3.1.6
(13) DC monitoring and annunciation (TAC M63649)	Resolved (SSER 13)	8.3.2.2
(14) Possible sharing of dc control power to ac switchgear	Resolved (SSER 3)	8.3.3.2.4
(15) Testing of associated circuits	Resolved (SSER 3)	8.3.3.3
(16) Testing of non-Class 1E cables	Resolved (SSER 3)	8.3.3.3

Proposed Condition	Status	Section
(17) Low-temperature overpressure protection/power supplies for pressurizer relief valves and level indicators (II.G.1) (TAC M63649)	Resolved (SSER 7)	8.3.3.4
(18) Testing of reactor coolant pump breakers	Resolved (SSER 2)	8.3.3.6
(19) Postaccident sampling system (TAC M77543)	Resolved (SSER 14)	9.3.2
(20) Fire protection program (TAC M63648)	Under review (SER)	9.5.1.8
(21) Performance testing for communications systems (TAC M63637)	Resolved (SSER 5)	9.5.2
(22) Diesel generator reliability (NUREG/CR-0660) (TAC M63640)	Resolved (SSER 5)	9.5.4.1
(23) Secondary water chemistry monitoring and control program	Resolved (SSER 5)	10.3.4
(24) Primary coolant outside containment (III.D.1.1) (TACs M63646, M77553)	Resolved (SSER 10)	11.7.2
(25) Independent safety engineering group (I.B.1.2) (TAC M63592)	Resolved (SSER 8)	13.4
(26) Use of experienced personnel during startup (TAC M63592)	Resolved (SSER 8)	13.1.3
(27) Emergency preparedness (III.A.1.1, III.A.1.2, III.A.2) (TAC M63656)	Resolved (SSER 13)	13.3
(28) Review of power ascension test procedures and emergency operating procedures by NSSS vendor (I.C.7) (TAC M77861)	Resolved (SSER 10)	13.5.2
(29) Modifications to emergency operating instructions (I.C.8) (TAC M77861)	Resolved (SSER 10)	13.5.2
(30) Report on outage of emergency core cooling system (II.K.3.17)	Resolved (SSER 3)	13.5.3
(31) Initial test program (TAC M79872)	Resolved (SSER 7)	14.2
(32) Effect of high-pressure injection for small-break LOCA with no auxiliary feedwater (II.K.2.13)	Resolved (SSER 4)	15.5.1

Proposed Condition	Status	Section
(33) Voiding in the reactor coolant system (II.K.2.17)	Resolved (SSER 4)	15.5.2
(34) PORV isolation system (II.K.3.1, II.K.3.2) (TAC M63631)	Resolved (SSER 5)	15.5.3
(35) Automatic trip of the reactor coolant pumps during a small-break LOCA (II.K.3.5)	Resolved (SSER 4)	15.5.4
(36) Revised small-break LOCA analysis (II.K.3.30, II.K.3.31) (TAC M77298)	Resolved (SSER 5)	15.5.5
(37) Detailed control room design review (I.D.1) (TAC M63655)	Resolved (SSER 15)	18.1
(38) Physical security of fuel in containment (TACs M63657, M83973)	Resolved (SSER 10)	13.6.4
(39) Control of heavy loads (NUREG-0612) (TAC M77560)	Resolved (SSER 13)	9.1.4
(40) Anticipated transients without scram (Generic Letter 83-28, Item 4.3) (TAC M64347)	Resolved (SSER 5)	15.3.6
(41) Steam generator tube rupture (TAC M77569)	Resolved (SSER 14)	15.4.3
(42) Loose-parts monitoring system (TAC M77177)	Resolved (SSER 5)	4.4.5
(43) Safety parameter display system (TAC M73723)	Opened (SSER 5)	18.2
(44) Physical Security Plan (TAC M63657, M83973)	Opened (SSER 15)	13.6

1.12 Approved Technical Issues for Incorporation in the License as Exemptions

The applicant applied for exemptions from certain provisions of the regulations. These have been reviewed by the staff and approved in appropriate sections of the SER and SSERs. These technical issues are listed below and the actual exemptions will be incorporated in the operating license:

(1) Seal leakage test instead of full-pressure test (Section 6.2.6, SSER 4) (TAC M63615)

(2) Criticality monitor (Section 9.1, SSER 5) (TAC M63615)

(3) Schedule to implement the vehicle bomb rule (Section 13.6.9, SSER 15) (TAC M90696)

In addition to these, the staff granted an exemption to the applicant on December 15, 1994, which will also be incorporated in the operating license:

(4) Issuance, storage, and retrieval of badges for personnel (TAC M90729)

The staff reevaluated three technical issues previously approved for exemption from various provisions of Appendix G to 10 CFR Part 50 in SSER 14. As a result, Section 5.3.1.1 of SSER 14 reports that these exemptions are no longer needed.

1.13 Implementation of Corrective Action Programs and Special Programs

On September 17, 1985, the NRC sent a letter to the applicant, pursuant to Title 10 of the Code of Federal Regulations, Section 50.54(f), requesting that the applicant submit information on its plans for correcting problems concerning the overall management of its nuclear program as well as on its plans for correcting plant-specific problems. In response to this letter, TVA prepared a Corporate Nuclear Performance Plan (CNPP) that identified and proposed corrections to problems concerning the overall management of its nuclear program, and a site-specific plan for Watts Bar entitled "Watts Bar Nuclear Performance Plan" (WBNPP). The staff reviewed both plans and documented results in two safety evaluation reports, NUREG-1232, Vol. 1 (July 1987), and NUREG-1232, Vol. 4 (January 1990).

In a letter of September 6, 1991, the applicant submitted Revision 1 of the WBNPP. In SSER 9, the staff concluded that Revision 1 of the WBNPP does not necessitate any revision of the staff's safety evaluation report, NUREG-1232, Vol. 4.

In NUREG-1232, Vol. 4, the staff documented its general review of the corrective action programs (CAPs) and special programs (SPs) through which the applicant would effect corrective actions at Watts Bar. When the report was published, some of the CAPs and SPs were in their initial stages of implementation. The staff stated that it will report its review of the implementation of all CAPs and SPs and closeout of open issues in future supplements to the licensing SER, NUREG-0847; accordingly, the staff prepared Temporary Instructions (TIs) 2512/016-043 for the Inspection Manual and adhered to the TIs to perform inspections of the CAPs and SPs. This new section was introduced in SSER 5 and will be updated in subsequent SSERs. The current status of all CAPs and SPs follows. The status described here fully supersedes that described in previous SSERs.

1.13.1 Corrective Action Programs

(1) Cable Issues (TAC M71917; TI 2512/016)

Program review status:	Complete: NUREG-1232, Vol. 4; Letter, P. S. Tam (NRC) to D. A. Nauman (TVA), April 25, 1991 (the safety evaluation was reproduced in SSER 7 as Appendix P); supplemental safety evaluation dated April 24, 1992 (Appendix T of SSER 9); letter, P. S. Tam (NRC) to M. O. Medford (TVA), February 14, 1994.
Implementation status:	Full implementation expected by August 1995.

NRC inspections: Inspection Reports 50-390, 391/90-09 (June 22, 1990); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/90-22 (November 21, 1990); 50-390, 391/90-24 (December 17, 1990); 50-390, 391/90-27 (December 20, 1990); 50-390, 391/90-30 (February 25, 1991); 50-390, 391/91-07 (May 31, 1991); 50-390, 391/91-09 (July 15, 1991); 50-390, 391/91-12 (July 12, 1991); 50-390, 391/91-31 (January 13, 1992); 50-390, 391/92-01 (March 17, 1992); audit report of June 12, 1992 (Appendix Y of SSER 9); 50-390, 391/92-05 (April 17, 1992); 50-390, 391/92-13 (July 16, 1992); 50-390, 391/92-18 (August 14, 1992); 50-390, 391/92-22 (September 18, 1992); 50-390, 391/92-26 (October 16, 1992); 50-390, 391/92-30 (November 13, 1992); 50-390, 391/92-35 (December 15, 1992); 50-390, 391/92-40 (January 15, 1993); 50-390, 391/93-10 (March 19, 1993); 50-390, 391/93-11 (March 25, 1993); 50-390, 391/93-35 (June 10, 1993); 50-390, 391/93-40 (July 15, 1993); 50-390, 391/93-48 (August 13, 1993); 50-390, 391/93-56 (September 20, 1993); 50-390, 391/93-63 (October 18, 1993); 50-390, 391/93-70 (November 12, 1993); 50-390, 391/93-74 (December 20, 1993); 50-390, 391/93-85 (January 14, 1994); 50-390, 391/93-91 (February 17, 1994); 50-390, 391/94-11 (March 16, 1994); 50-390, 391/94-18 (April 18, 1994); 50-390, 391/94-32 (May 16, 1994); 50-390, 391/94-35 (June 20, 1994); 50-390, 391/94-45 (July 15, 1994); 50-390, 391/94-51 (August 11, 1994); 50-390, 391/94-53 (September 20, 1994); 50-390, 391/94-55 (September 16, 1994); 50-390, 391/94-61 (October 12, 1994); 50-390, 391/94-66 (November 16, 1994); 50-390, 391/94-75 (December 19, 1994); 50-390, 391/94-82 (January 13, 1995); 50-390, 391/94-88 (February 15, 1995); 50-390, 391/95-17 (April 13, 1995); to come.

(2) <u>Cable Tray and Tray Supports (TAC R00516; TI 2512/017)</u>

Program review status: Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 13, 1989; NUREG-1232, Vol. 4; SSER 6, Section 3.

Implementation status: Full implementation expected by August 1995.

NRC inspections: Inspection Reports 50-390, 391/89-14 (December 18, 1989); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/90-22 (November 21, 1990); 50-390, 391/92-02 (March 17, 1992); audit report of May 14, 1992 (Appendix S of SSER 9); 50-390, 391/92-13 (July 16, 1992); 50-390, 391/92-201 (September 21, 1992); 50-390, 391/93-07 (February 19, 1993); 50-390/94-64 (December 15, 1994); 50-390, 391/94-88 (February 15, 1995); 50-390, 391/95-23 (May 2, 1995); to come.

(3) Design Baseline and Verification Program (TAC M63594; TI 2512/019)

Program review status: Complete: Inspection Report 50-390, 391/89-12 (November 20, 1989); NUREG-1232, Vol. 4.

Implementation status: Full implementation expected by August 1995.

NRC inspections: Inspection Reports 50-390, 391/89-12 (November 20, 1989); 50-390, 391/90-09 (June 22, 1990); 50-390, 391/90-20; (September 25, 1990); 50-390/91-201 (March 22, 1991); 50-390, 391/91-20 (October 8, 1991); 50-390, 391/91-25 (December 13, 1991); 50-390, 391/92-06 (April 3, 1992); 50-390, 391/92-201 (September 21, 1992); 50-390, 391/93-29 (May 14, 1993); 50-390, 391/93-66 (October 29, 1993); 50-390, 391/94-69 (November 18, 1994); to come.

(4) Electrical Conduit and Conduit Support (TAC R00508; TI 2512/018)

Program review status: Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 1, 1989; NUREG-1232, Vol. 4; SSER 6, Section 3.

Implementation status: Full implementation expected by August 1995.

NRC inspections: Inspection Reports 50-390, 391/89-05 (May 25, 1989); 50-390, 391/89-07; (July 11, 1989); 50-390, 391/89-14 (December 18, 1989); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/91-31 (January 13, 1992); 50-390, 391/92-02 (March 17, 1992); audit report of May 14, 1992 (Appendix S of SSER 9); 50-390, 391/92-05 (April 17, 1992); 50-390, 391/92-09 (June 29, 1992); 50-390, 391/92-201 (September 21, 1992); 50-390, 391/92-26 (October 16, 1992); 50-390, 391/93-07 (February 19, 1993); 50-390, 391/93-35 (June 10, 1993); 50-390, 391/93-70 (November 12, 1993); 50-390, 391/93-74 (December 20, 1993); 50-390, 391/93-91 (February 17, 1994); 50-390, 391/94-11 (March 16, 1994); 50-390, 391/94-32 (May 16, 1994); 50-390/94-64 (December 15, 1994); 50-390, 391/94-82 (January 13, 1995); 50-390, 391/94-88 (February 15, 1995); 50-390, 391/95-23 (May 2, 1995); to come.

(5) Electrical Issues (TAC M74502; TI 2512/020)

Program review status: Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 11, 1989; NUREG-1232, Vol. 4.

Implementation status: Full implementation expected by August 1995.

NRC inspections: Inspection Reports 50-390, 391/90-30 (February 25, 1991); 50-390, 391/92-22 (September 18, 1992); 50-390, 391/92-40 (January 15, 1993); 50-390, 391/93-35 (June 10, 1993); 50-390, 391/93-40 (July 15, 1993); 50-390, 391/93-63 (October 18, 1993); 50-390, 391/94-11 (March 16, 1994); 50-390, 391/94-18 (April 18, 1994); 50-390, 391/94-31 (May 11, 1994); 50-390, 391/94-45 (July 15, 1994); 50-390, 391/94-53 (September 20, 1994); 50-390, 391/94-66 (November 16, 1994); 50-390, 391/94-82 (January 13, 1995); 50-390, 391/94-88 (February 15, 1995); to come.

(6) Equipment Seismic Qualification (TAC M71919; TI 2512/021)

Program review status: Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 11, 1989; NUREG-1232, Vol. 4; SSER 6, Section 3.10.

Implementation status: Full implementation expected by August 1995.

NRC inspections: Inspection Reports 50-390, 391/90-05 (May 10, 1990); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/90-28 (January 11, 1991); 50-390, 391/91-03 (April 15, 1991); audit report of May 14, 1992 (Appendix S of SSER 9); 50-390, 391/92-201 (September 21, 1992); 50-390, 391/93-07 (February 19, 1993); 50-390, 391/93-79 (March 4, 1994); to come.

(7) Fire Protection (TAC M63648; TI 2512/022)

Program review status: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 7, 1989; NUREG-1232, Vol. 4; review in progress, results to be published in Section 9.5.1 of a future SSER.

Implementation status: Full implementation expected by August 1995.

NRC inspections: Inspection Reports 50-390, 391/94-45 (July 15, 1994); 50-390, 391/94-63 (November 2, 1994); 50-390, 391/94-62 (November 16, 1994); 50-390, 391/94-66 (November 16, 1994); 50-390, 391/94-78 (December 21, 1994); 50-390, 391/94-82 (January 13, 1995); 50-390, 391/95-03 (January 31, 1995); 50-390, 391/95-13 (March 1, 1995); 50-390, 391/95-16 (April 6, 1995); 50-390, 391/95-26 (May 1, 1995); to come.

(8) Hanger and Analysis Update Program (TAC R00512; TI 2512/023)

Program review status: Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), October 6, 1989; NUREG-1232, Vol. 4; SSER 6, Section 3.

Implementation status: Full implementation expected by August 1995.

NRC inspections: Inspection Reports 50-390, 391/89-14 (December 18, 1989); 50-390, 391/90-14 (August 3, 1990); 50-390, 391/90-18 (September 20, 1990); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/90-28

(January 11, 1991); 50-390, 391/91-03 (April 15, 1991); audit report of May 14, 1992 (Appendix S of SSER 9); 50-390, 391/92-201 (September 21, 1992); 50-390, 391/92-26 (October 16, 1992); 50-390, 391/92-35 (December 15, 1992); 50-390, 391/93-07 (February 19, 1993); 50-390, 391/93-35 (June 10, 1993); 50-390, 391/93-45 (July 20, 1993); 50-390, 391/93-56 (September 20, 1993); 50-390, 391/93-70 (November 12, 1993); 50-390, 391/93-74 (December 20, 1993); 50-390, 391/94-11 (March 16, 1994); 50-390, 391/94-32 (May 16, 1994); 50-390, 391/94-55 (September 16, 1994); 50-390, 391/95-06 (March 16, 1995); 50-390, 391/95-23 (May 2, 1995); to come.

(9) Heat Code Traceability (TAC M71920; TI 2512/024)

Program review status: Complete: Inspection Report 50-390, 391/89-09 (September 20, 1989); NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to D. A. Nauman (TVA), March 29, 1991.

Implementation status: 100% (certified by letter, E. Wallace (TVA) to NRC, July 31, 1990); staff concurrence in SSER 7, Section 3.2.2.

NRC inspections: Complete: Inspection Reports 50-390, 391/90-02 (March 15, 1990); 50-390, 391/89-09 (September 20, 1989).

(10) Heating, Ventilation, and Air-Conditioning Duct and Duct Supports (TAC R00510; TI 2512/025)

Program review status: Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), October 24, 1989; NUREG-1232, Vol. 4; SSER 6, Section 3.

Implementation status: Full implementation expected by August 1995.

NRC inspections: Inspection Reports 50-390, 391/89-14 (December 18, 1989); 50-390, 391/90-05 (May 10, 1990); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/91-01 (April 4, 1991); 50-390, 391/92-02 (March 17, 1992); audit report of May 14, 1992 (Appendix S of SSER 9); 50-390, 391/92-08 (May 15, 1992); 50-390, 391/92-13 (July 16, 1992); 50-390, 391/92-201 (September 21, 1992); 50-390, 391/93-07 (February 19, 1993); 50-390, 391/93-91 (February 17, 1994); 50-390, 391/94-08 (March 11, 1994); 50-390, 391/95-23 (May 2, 1995); to come.

(11) Instrument Lines (TAC M71918; TI 2512/026)

Program review status: Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 8, 1989; NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to O. D. Kingsley (TVA), October 26, 1990 (Appendix K of SSER 6) and May 5, 1994.

Implementation status: Full implementation expected by August 1995.

NRC inspections: Inspection Reports 50-390, 391/90-14 (August 3, 1990); 50-390, 391/90-23 (November 19, 1990); 50-390, 391/90-29 (January 29, 1991); 50390, 391/91-02 (March 6, 1991); 50-390, 391/91-03 (April 15, 1991); 50-390, 391/91-26 (December 6, 1991); 50-390, 391/93-74 (December 20, 1993); 50-390, 391/94-11 (March 16, 1994); 50-390, 391/94-24 (July 1, 1994); 50-390, 391/94-32 (May 16, 1994); 50-390, 391/94-55 (September 16, 1994); 50-390, 391/95-23 (May 2, 1995); to come.

(12) Prestart Test Program (TAC M71924)

Program review status: Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), October 17, 1989; NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to D. A. Nauman (TVA), March 27, 1991.

Implementation status: Withdrawn by letter (J. H. Garrity (TVA) to NRC, February 13, 1992). Applicant will re-perform preoperational test program per Regulatory Guide 1.68, Revision 2.

(13) Quality Assurance Records (TAC M71923; TI 2512/028)

Program review status: Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), December 8, 1989; NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to M. O. Medford (TVA) June 9, 1992 (Appendix X of SSER 9); letter, P. S. Tam (NRC) to M. O. Medford (TVA), January 12, 1993; letter, F. J. Hebdon (NRC) to M. O. Medford (TVA), August 12, 1993; letter, P. S. Tam (NRC) to O. D. Kingsley (TVA), April 25, 1994.

Implementation status: 100% (certified by letter, W. J. Museler (TVA), to NRC, April 27, 1994); staff concurrence in Inspection Report 50-390, 391/94-40 (June 24, 1994).

NRC inspections: Complete: Inspection Reports 50-390, 391/90-06 (April 25, 1990); 50-390, 391/90-08 (September 13, 1990); 50390, 391/91-08 (May 30, 1991); 50-390, 391/91-15 (September 5, 1991); 50-390, 391/91-29 (December 27, 1991); 50-390, 391/92-05 (April 17, 1992); 50-390, 391/92-10 (June 11, 1992); 50-390, 391/92-21 (September 18, 1992); 50-390, 391/93-11 (March 25, 1993); 50-390, 391/93-21 (April 9, 1993); 50-390, 391/93-29 (May 14, 1993); 50-390, 391/93-34 (July 5, 1993); 50-390, 391/93-35 (June 10, 1993); 50-390, 391/93-50 (September 3, 1993); 50-390, 391/93-59 (October 25, 1993); 50-390, 391/93-69 (November 12, 1993); 50-390, 391/93-70

(November 12, 1993); 50-390, 391/93-78 (December 16, 1993); 50-390, 391/93-86 (January 24, 1994); 50-390, 391/94-04 (February 23, 1994); 50-390, 391/94-09 (March 11, 1994); 50-390, 391/94-17 (April 1, 1994); 50-390, 391/94-28 (May 5, 1994); 50-390, 391/94-40 (June 24, 1994).

(14) Q-List (TAC M63590; TI 2512/029)

Program review status:	Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 11, 1989; NUREG-1232, Vol. 4; letters, P. S. Tam (NRC) to O. D. Kingsley (TVA), January 23, 1991 and March 17, 1994 (enclosure of this letter reproduced as Appendix AA in SSER 13).
Implementation status:	100% (certified by letter, W. J. Museler (TVA), to NRC, January 28, 1994); staff concurrence in Inspection Report 50-390, 391/94-27 (April 21, 1994).
NRC inspections:	Complete: Inspection Reports 50-390, 391/90-08 (September 13, 1990); 50-390, 391/91-08 (May 30, 1991); 50-390, 391/91-29 (December 27, 1991); 50-390, 391/91-31 (January 13, 1992); 50-390, 391/93-20 (April 16, 1993); 50-390, 391/93-68 (November 12, 1993); 50-390, 391/94-27 (April 21, 1994).

(15) Replacement Items Program (TAC M71922; TI 2512/027)

Program review status:	Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), November 22, 1989; NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to O. D. Kingsley (TVA), February 11, 1991 (Appendix N of SSER 6); letter, P. S. Tam (NRC) to M. O. Medford (TVA), July 27, 1992, April 5, 1994, and February 6, 1995.
Implementation status:	Full implementation expected by August 1995.
NRC inspections:	Inspection Reports 50-390, 391/91-08 (May 30, 1991); 50-390, 391/91-29 (December 27, 1991); 50-390, 391/92-03 (March 16, 1992); 50-390, 391/92-11 (June 12, 1992); 50-390, 391/92-17 (July 22, 1992); 50-390, 391/92-21 (September 18, 1992); 50-390, 391/92-40 (January 15, 1993); 50-390, 391/93-22 (April 25, 1993); 50-390, 391/93-34 (July 9, 1993); 50-390, 391/93-38 (June 24, 1993); to come.

(16) Seismic Analysis (TAC R00514; TI 2512/030)

Program review status:	Complete: Letters, S. C. Black (NRC) to O. D. Kingsley (TVA), September 7 and October 31, 1989; NUREG-1232, Vol. 4; SSER 6, Section 3.7.

Implementation status:	100% (certified by letter, J. H. Garrity (TVA) to NRC, December 2, 1991); staff concurrence in SSER 9, Section 3.7.1.
NRC inspections:	Complete: Inspection Reports 50-390, 391/89-21 (May 10, 1990); 50-390, 391/90-20 (September 25, 1990); audit report by L. B. Marsh, October 10, 1990.

(16)(a) Civil Calculation Program (TAC R00514)

Program review status:	No program review. A number of civil calculation categories are required by the Design Baseline and Verification Program CAP and constitute parts of the applicant's corrective actions. This program is regarded as complementary to but not part of the Seismic Analysis CAP. Staff efforts consist mainly of audits performed at the site and in the office.
Implementation status:	100%. Final calculations transmitted by letter, W. J. Museler (TVA) to NRC, July 27, 1992.
NRC audits:	Complete: Memorandum (publicly available), T. M. Cheng (NRC) to P. S. Tam, January 23, 1992; letter, P. S. Tam (NRC) to D. A. Nauman (TVA), January 31, 1992; letters, P. S. Tam (NRC) to M. O. Medford (TVA), May 26 and December 18, 1992 and July 2, 1993; 50-390, 391/93-07 (February 19, 1993); letter, P. S. Tam (NRC) to M. O. Medford (TVA), November 26, 1993.

(17) Vendor Information Program (TAC M71921; TI 2512/031)

Program review status:	Complete: Letter, P. S. Tam (NRC) to O. D. Kingsley (TVA), September 11, 1990 (Appendix I of SSER 5); Appendix I of SSER 11.
Implementation status:	Full implementation expected by August 1995.
NRC inspections:	Inspection Reports 50-390, 391/91-08 (May 30, 1991); 50-390, 391/91-29 (December 27, 1991); 50-390, 391/93-27 (May 14, 1993); 50-390, 391/95-10 (March 17, 1995); to come.

(18) Welding (TAC M72106; TI 2512/032)

Program review status:	Complete: Inspection Reports 50-390, 391/89-04 (August 9, 1989); 50-390, 391/90-04 (May 17, 1990); NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to D. A. Nauman (TVA), March 5, 1991; these inspection reports also address recurrence control: 50-390, 391/93-02 (February 2, 1993); 50-390, 391/93-84 (December 21, 1993); 50-390, 391/94-79 (January 11, 1995).

Implementation status:	100% (certified by letter, W. J. Museler (TVA) to NRC, January 9, 1993); staff concurrence in Inspection Report 50-390, 391/94-79 (January 11, 1995).
NRC inspections:	Complete: Inspection Reports 50-390, 391/89-04 (August 9, 1989); 50-390, 391/90-04 (May 17, 1990); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/91-05 (May 28, 1991); 50-390, 391/91-18 (October 8, 1991); 50-390, 391/91-23 (November 21, 1991); 50390, 391/91-32 (February 10, 1992); 50-390, 391/9220 (August 12, 1992); 50-390, 391/92-28 (October 9, 1992); 50-390, 391/93-02 (February 2, 1993); 50-390, 391/93-19 (March 15, 1993); 50-390, 391/93-38 (June 24, 1993); 50-390, 391/93-84 (December 21, 1993); 50-390, 391/94-05 (February 19, 1994); 50-390, 391/94-16 (March 15, 1994); 50-390, 391/94-49 (July 21, 1994); 50-390, 391/94-79 (January 11, 1995).

1.13.2 Special Programs

(1) Concrete Quality (TAC M63596; TI 2512/033)

Program review status:	Complete: NUREG-1232, Vol. 4.
Implementation status:	100% (certified by letter, E. Wallace (TVA) to NRC, August 31, 1990); staff concurrence in SSER 7, Section 3.8.2.1.
NRC inspections:	Complete: NUREG-1232, Vol. 4; Inspection Reports 50-390, 391/89-200 (December 12, 1989); 50-390, 391/90-26 (January 8, 1991).

(2) Containment Cooling (TAC M77284; TI 2512/034)

Program review status:	Complete: NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to D. A. Nauman (TVA), May 21, 1991 (Section 6.2.2 of SSER 7).
Implementation status:	100% (certified by letter, W. J. Museler (TVA) to NRC, December 30, 1993); staff concurrence to come.
NRC inspections:	Inspection Report 50-390, 391/93-56 (September 20, 1993); to come.

(3) Detailed Control Room Design Review (TAC M63655; TI 2512/035)

Program review status:	Complete: Appendix D of SER; NUREG-1232, Vol. 4; Section 18.1, and Appendix L of SSER 6; Section 18.1 of SSER 5 and 15.
Implementation status:	Full implementation expected by August 1995.
NRC inspections:	Inspection Reports 50-390, 391/94-22 (April 28, 1994); audit reports in SSER 5 and 15.

(4) Environmental Qualification Program (TAC M63591; TI 2512/036)

Program review status: Complete: NUREG-1232, Vol. 4; Section 3.11 of SSER 15.

Implementation status: Full implementation expected by August 1995.

NRC inspections: Inspection Reports 50-390, 391/93-63 (October 18, 1993; 50-390, 391/94-28 (April 18, 1994); 50-390, 391/94-74 (January 13, 1995); 50-390, 391/95-15 (April 5, 1995); to come.

(5) Master Fuse List (TAC M76973; TI 2512/037)

Program review status: Complete: NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to O. D. Kingsley (TVA), February 6, 1991; letter, P. S. Tam (NRC) to TVA Senior Vice President, March 30, 1992 (Appendix U of SSER 9).

Implementation status: 100% (certified by letter, W. Museler (TVA) to NRC, April 2, 1993); staff concurrence in Inspection Report 50-390, 391/93-31 (May 6, 1993).

NRC inspections: Complete: Inspection Reports 50-390, 391/86-24 (February 12, 1987); 50-390, 391/92-05 (April 17, 1992); 50-390, 391/92-09 (June 29, 1992); 50-390, 391/92-27 (September 25, 1992); 50-390, 391/93-31 (May 6, 1993).

(6) Mechanical Equipment Qualification (TAC M76974; TI 2512/038)

Program review status: Complete: NUREG-1232, Vol. 4; Section 3.11 of SSER 15.

Implementation status: Full implementation expected by August 1995.

NRC inspections: Inspection Reports 50-390, 391/95-15 (April 5, 1995); to come.

(7) Microbiologically Induced Corrosion (TAC M63650; TI 2512/039)

Program review status: Complete: NUREG-1232, Vol. 4; Appendix Q of SSER 8; Appendix Q of SSER 10.

Implementation status: 100% (certified by letter, W. J. Museler (TVA) to NRC, August 31, 1993); staff concurrence in Inspection Report 50-390, 391/93-67 (November 1, 1993).

NRC inspections: Complete: Inspection Reports 50-390, 391/90-09 (June 22, 1990); 50-390, 391/90-13 (August 2, 1990); 50-390, 391/93-01 (February 25, 1993); 50-390, 391/93-09 (March 26, 1993); 50-390, 391/93-67 (November 1, 1993).

(8) Moderate Energy Line Break Flooding (TAC M63595; TI 2512/040)

Program review status:	Complete: NUREG-1232, Vol. 4; Section 3.6 of SSER 11.
Implementation status:	Full implementation expected by August 1995.
NRC inspections:	Inspection Reports 50-390, 391/93-85 (January 14, 1994); to come.

(9) Radiation Monitoring Program (TAC M76975; TI 2512/041)

Program review status:	Complete: NUREG-1232, Vol. 4; this program covers areas addressed in Chapter 12 of the SER and SSERs.
Implementation status:	Full implementation expected by August 1995.
NRC inspections:	Inspection Reports 50-390, 391/94-56 (October 6, 1994); to come.

(10) Soil Liquefaction (TAC M77548; TI 2512/042)

Program review status:	Complete: NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to TVA Senior Vice President, March 19, 1992; Section 2.5 of SSER 9.
Implementation status:	100% (certified by letter, W. J. Museler (TVA) to NRC, July 27, 1992); staff concurrence in SSER 11, Section 2.5.4.4.
NRC inspections:	Complete: Inspection Reports 50-390, 391/89-21 (May 10, 1990); 50-390, 391/89-03 (May 11, 1989); audit report by L. B. Marsh (NRC) (October 10, 1990); audit report, P. S. Tam (NRC) to D. A. Nauman (TVA), January 31, 1992; audit report, P. S. Tam (NRC) to M. O. Medford (TVA), May 26 and December 18, 1992; 50-390, 391/92-45 (February 17, 1993).

(11) Use-as-Is CAQs (TAC M77549; TI 2512/043)

Program review status:	Complete: NUREG-1232, Vol. 4.
Implementation status:	100% (certified by letter, W. J. Museler (TVA) to NRC, July 24, 1992); staff concurrence in Inspection Report 50-390, 391/93-10 (March 19, 1993).
NRC inspections:	Complete: Inspection Reports 50-390, 391/90-19 (October 15, 1990); 50-390, 391/91-08 (May 30, 1991); 50-390, 391/93-10 (March 19, 1993).

2 SITE CHARACTERISTICS

The staff has completed its review of Amendments 65 through 86 to the FSAR. On the basis of its review, the staff finds that the distances to the exclusion area and low population zone boundaries for the Watts Bar site (see SER Section 2.1.2) are still sufficient to reasonably assure that the calculated radiological consequences of postulated design-basis accidents do not exceed the dose guidelines in 10 CFR Part 100. Therefore, the staff's evaluation and conclusions in the SER are not changed and remain valid. The sections that follow supplement or revise the staff's previous evaluation in the SER.

The staff has performed radiological consequence assessments for design-basis accidents in Chapter 15 of this supplement using the revised atmospheric relative concentrations shown in Section 2.3.4 below.

The staff tracked its efforts by TACs M89446 and M89447.

2.3 Meteorology

By Amendments 65 through 86, the applicant revised the FSAR, refining the description of regional and local climatology and meteorology of the site. In Chapter 2 of the FSAR, the applicant reevaluated atmospheric dispersion at the site using 20 years of onsite meteorology data (January 1974 through December 1993). The applicant has also revised regional and local climatology information, including extremes of climate and severe weather, based on information accumulated since the SER was issued in June 1982.

2.3.1 Regional Climatology

In the SER, the staff described the general climate of the Great Tennessee Valley and of the Watts Bar site. These descriptions were based on climatological data from Chattanooga, Knoxville, Decatur, and Watts Bar Dam, in addition to available onsite data.

The severe-weather statistics for the region related to hail, high winds, thunderstorms, and ice storms that were presented in the SER still remain valid. However, the applicant has revised its tornado strike probability and recurrence interval (the SER stated this at about 1300 years). The applicant's current estimate of tornado strike probability, based on a longer period and smaller area, is 0.00015 per year (15 chances out of 100,000 of a tornado striking the Watts Bar site in any given year) with a recurrence interval of 6755 years. The staff independently estimated the tornado strike probability to be about 0.00018 per year (18 chances out of 100,000 of a tornado striking the Watts Bar site in any given year) with a recurrence interval of about 5400 years.

The staff's independent estimate of the tornado strike probability is based on (1) the methodology of WASH-1300 (U.S. Atomic Energy Commission, "Technical Basis for Interim Regional Tornado Criteria," 1974) as implemented in the TORNADO Computer Code (PNL-4483, "TORNADO, A Program To Compute Tornado Strike and Intensity Probabilities With Associated Wind Speeds and Pressure Drops at Nuclear Power Stations" (1982)), and (2) tornado data summarized in

NUREG/CR-4461, "Tornado Climatology of the Contiguous United States" (1986). The applicant's current estimate and the staff's estimate of tornado strike probability are lower than the estimate in the 1982 Watts Bar SER; therefore, the estimate in the SER is conservative and still valid as a design basis.

On the basis of this review, the staff reconfirms its conclusions in the 1982 SER that the applicant has considered appropriate regional meteorological conditions in the design and siting of this plant and, therefore, meets the requirements of 10 CFR 100.10 and 10 CFR Part 50, Appendix A, General Design Criteria (GDC) 2 and 4.

2.3.2 Local Meteorology

The applicant submitted onsite meteorological data covering the period from January 1974 through December 1993. Analysis of these data shows that the meteorological conditions at Watts Bar are consistent with conditions expected on the basis of the regional climatology. Winds tend to be light and flow up and down the Tennessee River Valley. The stable atmospheric conditions that occur at night are accompanied by light winds that are driven by local conditions rather than by the valley flow. Neutral atmosphere stability conditions may occur at any time and are prevalent during the transition period between day and night. During neutral conditions, the winds at the plant tend to be aligned with the prevailing valley flow.

Analysis of the data shows that extremely unstable conditions gave the highest average wind speeds during the 20-year period of onsite data collection. The applicant submitted information that shows the highest wind speeds during unstable conditions were associated with winds from the south-southwest. Such winds have the same frequency of occurrence as winds from any direction. This information also shows that the frequencies of calm winds in the 0.3-to-0.6-meter-per-second (0.6-to-1.4-miles-per-hour) windspeed class during extremely unstable atmospheric conditions (stability classes of A and B) are much lower than expected.

During a September 13-14, 1994, visit to the Watts Bar site, the staff requested additional information. In response, the applicant sent additional information by letters dated August 5, September 27, and November 4, 1994. The staff reviewed the additional meteorological data submitted by the applicant, examined an aerial photograph of the plant site (J.E. Jobst and R.A. Semmler, 1982, "An Aerial Radiological Survey of the Watts Bar Nuclear Plant and Surrounding Area" (EGG-1183-1842), EG&G Energy Measurement Group, Spring City, Tennessee), and considered the physical processes involved. The staff concludes that the association between the high average windspeeds and extremely unstable atmosphere conditions is probably caused by two factors:

(1) The general disruption of the atmosphere during unstable conditions prevents windspeeds from decreasing to the lowest speed classes. As a result, there are essentially no occurrences of low windspeed to reduce the average windspeeds for the extremely unstable classes.

(2) Temperature difference is related to the performance of the parameters used to approximate stability conditions. The temperature difference parameters perform satisfactorily under homogeneous atmosphere conditions. Under the conditions described above, a complex atmospheric vertical structure (multiple boundary layers) sets up, and the

temperature measurement points reflect significantly different conditions; consequently, the parameter does not perform well.

The shift in stability is not significant because it occurs under conditions associated with relatively good dispersion and occurs infrequently.

As discussed above, the staff has reviewed available information relative to local meteorological conditions of importance to the safe design and siting of this plant. On the basis of this review, the staff reconfirms its conclusions in the 1982 SER that the applicant has considered appropriate local meteorological conditions in the design and siting of this plant and, therefore, meets the requirements of 10 CFR 100.10 and 10 CFR Part 50, Appendix A, GDC 2.

2.3.4 Short-Term (Accident) Diffusion Estimates

Data from the applicant's meteorological system located at the Watts Bar site have been used to estimate atmospheric dispersion characteristic for the Watts Bar plant. The applicant has submitted meteorological data covering the 20-year period from January 1974 through December 1993. Data summaries for this period show a larger fraction of the calm condition (wind speeds below the anemometer threshold) and a lower annual average wind speed than seen in data used in the dispersion calculations presented in the SER.

The staff conducted an independent evaluation of the dispersion conditions using the 20-year meteorological data set and the method described in Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequences Assessments of Nuclear Power Plants." The evaluation assumed ground-level releases, a building cross-section area of 1800 square meters (20,000 square feet), and a terrain adjustment factor of 1.5. Neither deposition nor decay was considered. The result of the dispersion estimates for the exclusion area boundary (EAB) (1250 meters [0.77 mile]) and the outer radius of the low-population zone (LPZ) (4828 meters [3 miles]) to the southwest of the plant are shown in Table 2.3. The southwest sector was selected for the analysis because the applicant indicates that it is the sector with maximum normalized concentration values. The table also compares the staff's dispersion estimates with previously reported values.

The meteorological data for the longer period of record were used in the atmosphere dispersion calculation by the staff and by the applicant. These data provide more representative estimates of the meteorological conditions than the data from the 2-year period of record used in atmosphere dispersion calculation for the original SER.

2.3.5 Long-Term (Routine) Diffusion Estimates

This subject was evaluated in SSER 14.

Table 2.3 **Maximum-sector normalized concentrations at the boundaries of the EAB and LPZ.**

Boundary	Time Period	Atmospheric Relative Concentrations χ/Q (sec/m^3)		
		SSER 15	FSAR[1]	SER
Exclusion Area	0 to 2 hours	5.5×10^{-4}	6.0×10^{-4}	3.6×10^{-4}
Low Population Zone	0 to 8 hours	1.0×10^{-4}	6.8×10^{-5}	5.0×10^{-5}
	8 to 24 hours	6.0×10^{-5}	4.6×10^{-5}	3.3×10^{-5}
	1 to 4 days	2.6×10^{-5}	2.0×10^{-5}	1.3×10^{-5}
	4 to 30 days	8.0×10^{-6}	6.2×10^{-6}	3.7×10^{-6}

(1) FSAR Amendment No. 83

3 DESIGN CRITERIA - STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.9 Mechanical Systems and Compnents

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

3.9.3.3 Design and Installation of Pressure Relief Devices

Safety and Relief Valve Test Requirements

In the SER, the staff evaluated the applicant's partial response to Item II.D.1 in NUREG-0737, "Performance Testing of Relief and Safety Valves," and stated that it will report its findings on plant-specific testing in a supplement to the SER. By letters dated July 11, 1991; December 26, 1992; and July 19 and December 1, 1994, the applicant submitted information for staff review.

Appendix CC is a technical evaluation report (TER), prepared by Idaho National Engineering Laboratory (INEL) under contract with the NRC staff. The TER reports the results of the staff and INEL review of the applicant's submittals in response to Item II.D.1 (and subitems 1, 2, and 3). The staff endorses the findings contained in the TER, and concludes that the applicant has acceptably resolved this issue.

The staff tracked its efforts by TAC M79992.

3.11 Environmental Qualification of Mechanical and Electrical Equipment (Unit 1 Only)

This issue of environmental qualification (EQ) of mechanical and electrical equipment was not resolved in the SER, pending submittal by the applicant. Subsequent to issuance of the SER, the applicant submitted a number of documents for Unit 1. The staff reviewed those under TACs M63591 and M76974, and has delineated findings below.

The equipment used to perform a necessary safety function must be demonstrated capable of maintaining functional operability under all service conditions postulated to occur during its installed life, and for the time it is required to operate in response to any postulated accident conditions. This requirement is contained in General Design Criteria 1, 2, 4, and 23 of Appendix A to 10 CFR Part 50, and in Sections III and XI of Appendix B to 10 CFR Part 50. This requirement applies to safety-related equipment located both inside and outside containment. In 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important-to-Safety for Nuclear Power Plants," and NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," the staff gives more detailed requirements and guidance relating to the methods and procedures for demonstrating this capability. Regulatory Guide 1.89 ("Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants"), and NUREG-0588 supplement the Institute of Electrical and Electronics Engineers (IEEE) Standard 323-1974 ("IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations") which contains detailed information about the qualification of electrical equipment. For environmental qualification, equipment at Watts Bar is required to meet the Category II criteria in NUREG-0588.

3.11.1 Background

The staff issued NUREG-0588 in December 1979 to promote a more orderly and systematic implementation of EQ programs by industry and to provide guidance to the NRC staff for its use in ongoing licensing reviews. The positions contained in the report provide guidance on (1) how to establish environmental service conditions; (2) how to select methods considered appropriate for qualifying equipment in different areas of the plant; and (3) other specific topics, such as margin, aging, and documentation.

In February 1980, the staff requested that near-term operating license applicants review and evaluate the EQ documentation for each item of safety-related electric equipment, and identify the degree to which their qualification programs comply with the staff positions discussed in NUREG-0588. IE Bulletin 79-01B ("Environmental Qualification of Class 1E Equipment") and its supplements establish EQ requirements and guidance for operating reactors and license applicants.

The final rule on EQ of electric equipment important to safety for nuclear power plants became effective on February 22, 1983. This rule, Section 50.49 of 10 CFR Part 50, specifies the requirements for demonstrating the EQ of electrical equipment that is important to safety and is located in a harsh environment.

In June 1982, the staff issued the SER (NUREG-0847). In Section 3.11 of the SER, the staff cited the requirements of NUREG-0588 and stated that it would review information on EQ of electrical and mechanical equipment once the applicant submitted it.

The Watts Bar EQ Program has been under review by the staff since 1983, when the applicant first submitted information. Between February 14 and February 16, 1984, the NRC staff and its consultant (Idaho National Engineering Laboratories (INEL)), audited the EQ files for Watts Bar Unit 1. As documented in the audit report of March 14, 1984, the staff determined that the EQ files were incomplete.

The staff documented more information about the evolution of the Watts Bar EQ Program in "Safety Evaluation Report on Tennessee Valley Authority: Watts Bar Nuclear Performance Plan, Watts Bar Unit 1" (NUREG-1232, Vol. 4), issued in January 1990. In Section 3.3.4 of that report, the staff explains that in July and August 1985, the applicant conducted a management review of the EQ programs at Sequoyah, Browns Ferry, and Watts Bar. The applicant found that much of the qualification documentation was not fully auditable and, in some cases, the available documentation did not demonstrate full qualification. On the basis of this review, the applicant established EQ projects at the three plants with the responsibility for developing and implementing EQ programs at each site. These programs were to establish controls to ensure a consistent approach to EQ at all TVA sites.

After the Watts Bar Environmental Qualification Project was established, TVA submitted "Watts Bar Nuclear Plant Unit 1, Summary Status Update Report of

TVA's Compliance to 10 CFR 50.49 - Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants" to the NRC for review on September 30, 1986. The applicant supplemented this submittal by a letter to the NRC dated April 30, 1991. The staff and its contractor, INEL, reviewed these submittals and identified deficiencies and open items in the Watts Bar EQ Program. The staff met with the applicant on March 5, 1992 (summary dated March 13, 1992) and resolved some of the deficiencies and open items. The staff then issued a request for additional information on May 1, 1992. The applicant sent additional information in a letter to the NRC dated February 17, 1993. The staff planned an inspection of the Watts Bar Unit 1 EQ Program to resolve any remaining open items before fuel load.

Concurrent with the staff review, the applicant continued to update its EQ binders. The staff audited the applicant's qualification binders in March 1995. Although not all of the binders in the program were available for review, the staff found that those binders that were available during the audit had corrected the open items and deficiencies identified in the earlier reviews.

The staff performed an inspection between December 12 and 16, 1994, to review and assess the applicant's program for establishing EQ of electrical and mechanical equipment. The inspection examined the applicant's overall EQ organization and interfaces, EQ Program procedures, EQ Program documentation (e.g., 10 CFR 50.49 equipment/cable lists, EQ qualification files, environmental drawings, category and operating time calculations, and essentially mild calculations), EQ engineering support, EQ maintenance program, EQ training, and quality assurance/EQ interfaces. The staff documented this inspection in Inspection Report 50-390/94-74, dated January 13, 1995.

The scope of the inspection of February 27 through March 10, 1995, included evaluating the qualification criteria and the environments in which the equipment must function, assessing the qualification documentation, and examining the physical installation of the equipment. The principal area of review was the qualification of safety-related electrical equipment that must function to prevent or mitigate the consequences of a loss-of-coolant accident (LOCA) or a high-energy-line break (HELB) inside or outside of containment while subjected to the harsh environments associated with these accidents. While the inspection was focused on electric equipment in systems tested and turned over to the applicant's operating department as operating systems, the applicant's mechanical EQ Program was also reviewed. The results of this inspection are documented in Inspection Report 50-390/95-15, dated April 5, 1995.

This safety evaluation covers the applicant's EQ Program up to the March 1995 inspection. The scope of this report covers (1) the completeness of the list of systems and equipment to be qualified, (2) the criteria they must meet, (3) the environments in which they must function, and (4) the qualification documentation for the equipment.

3.11.2 Evaluation

To evaluate the Watts Bar EQ Program, the staff reviewed the applicant's submittals regarding EQ; examined electrical equipment on site; audited qualification documentation and environmental qualification documentation binders; and reviewed the acceptability of the components, qualification methods, and accident environments. The NUREG-0588 Category II criteria form the basis for determing the adequacy of the applicant's qualification program.

The staff's contractor (INEL) prepared a technical evaluation report; that report constitutes Appendix DD of this SSER.

3.11.2.1 Completeness of EQ List

In 10 CFR 50.49, the staff requires that applicants prepare a list of electrical equipment important to safety located in harsh environments. The list should be auditable and current. The equipment should be maintained on the list for the entire period of time during which the qualified piece of equipment is installed in the plant.

The applicant maintains such a list of qualified equipment, "lE Electrical Equipment Requiring Environmental Qualification Under 10 CFR 50.49," Volumes 1-3. These volumes are controlled documents; Revision 5, dated December 22, 1994, was the most recent revision available for staff review. Volume 1 contains lists of electrical equipment important to safety; Volumes 2 and 3 list the associated cables for these systems. Components and cables listed in these volumes are categorized by system. Qualification information about electrical splices for all systems appears in a generic section (System 510) in Volume 1.

In the inspection conducted December 12-16, 1994, the staff performed a programmatic review of Watts Bar EQ procedures. During this review, the staff examined the applicant's procedures, giving the format and content requirements for the EQ lists, in order to verify program acceptability, adequate program auditability, controls, and maintenance for the EQ of equipment in accordance with 10 CFR 50.49 and GDC 4 of Appendix A to 10 CFR Part 50.

During the inspection of February 28 — March 10, 1995, the staff reviewed the EQ lists to determine whether they were complete and accurate. At the time of the review, 16 systems in the component qualification list and 29 systems in the cable qualification list were complete and available for review. Seventy-five safety-related components located in harsh environments were selected using the Emergency Operating Instructions, system flow diagrams, and environmental drawings. The components were compared with the Watts Bar EQ list to verify completeness and accuracy.

Various instrumentation required by Regulatory Guide (RG) 1.97 was also selected from Watts Bar Procedure WB-DC-30-7 ("Post Accident Monitoring Instrumentation") and Calculation WBPEVAR8809048 ("PAM Instrumentation Evaluation and Verification Methodology, Standards and Guidelines") and verified to be on the EQ list and included in the EQ Program. In SSERs 9 and 14, the staff concluded that the applicant's RG 1.97 program was acceptable; the staff's evaluation of all identified deviations from RG 1.97 was reported in Section 7.5.2 of those SSERs.

The staff also performed walkdowns of two safety-related systems (residual heat removal and containment spray) and steam generator room 2. The walkdowns were performed to verify that the EQ list matched the installed configuration of the components in these systems and spaces, and to obtain cable and conduit numbers of EQ components for verification of the EQ cable list. Cables identified during the walkthrough were verified to be on the EQ cable list.

On the basis of the review of the Watts Bar component and cable EQ lists, the staff has determined that they are acceptable.

3.11.2.2 Qualification Methods

In NUREG-0588, the staff presents detailed procedures for qualifying safety-related electrical equipment in a harsh environment. The NUREG-0588 criteria apply to equipment that is important to safety as defined in 10 CFR 50.49. Type tests of identical equipment in a sequence consisting of thermal, radiation, and mechanical preaging; seismic and dynamic loading; and exposure to postulated LOCA/HELB conditions are the preferred method of qualification. The applicant extrapolated test data, using the Arrhenius methodology, to establish the qualified life preceding a LOCA/HELB harsh environment. Where a 40-year lifetime was not established, the Watts Bar EQ Program requires replacement before the end of the qualified life of the component. The staff has reviewed this approach and finds that the applicant's qualification methods are acceptable in that they comply with NUREG-0588.

3.11.2.2.1 Electrical Equipment in a Harsh Environment

During the inspection of December 12-16, 1994, the staff reviewed the following procedures related to the Watts Bar EQ Program:

- NP STD-6.5, "Electrical Equipment Environmental Qualification (EQ) Program Standard," Revision 3

- DS-M18.14.1, "Design Standard for Environmental Qualification of Electrical Equipment in Harsh Environments," Revision 0

- NEP-5.12, "Program Requirements for Equipment Qualification of Electrical Equipment in Harsh Environments," Revision 0

- SSP-6.05, "10 CFR 50.49 - Maintaining Electrical Equipment Environmental Qualification," Revision 2

- EAI-7.05, "Watts Bar 10 CFR 50.49 Program Requirements for Environmental Qualification of Electrical Equipment," Revision 6

- PAI-10.12, "QMDS Verification, Implementation and EQ Baseline Activities," Revision 2

Nuclear Power Standard 6.5 (NP STD-6.5), "Electrical Equipment Environmental Qualification (EQ) Program," delineates functional responsibilities and authorities of the Chief Engineer (Corporate Engineering), Site Engineering Manager, and Plant Manager for implementing the EQ Program. Corporate generic programmatic technical requirements are addressed in procedures NEP-5.12 and DS-M18.141, and these requirements are implemented at the site in SSP-6.05 and EAI-7.05. The staff reviewed these procedures and found that they describe in detail the Watts Bar EQ Program.

A major element of the EQ Program is the EQ binders, which contain all relevant information about the qualification of each equipment type. Equipment is identified in the EQ binder by plant component identifier, manufacturer, model number, location, elevation, procurement document number, safety function, mitigating accident, equipment category, and required operating time. The environmental qualification is evaluated in detail and documented. Data, calculations, and justification that establish a qualified life for the equipment and support qualification are contained in the EQ binder. Vendor qualification documents used to establish qualification such as test reports, analyses, and test plans, are also in the binder, as are requirements and schedules for EQ-related maintenance and equipment or component replacement. These requirements are essential to maintaining the EQ of the equipment on a continuing basis.

For Watts Bar Unit 1, the requirements of qualification under Category II of NUREG-0588 apply to equipment installed or in stock before February 22, 1983, except for replacement equipment procured between May 23, 1980, and February 22, 1983, which should have been procured to Category I qualification requirements. Replacement equipment installed on or after February 22, 1983, should comply with the requirements of 10 CFR 50.49 unless there are documented sound reasons to the contrary.

The staff has reviewed the applicant's qualification methods for electrical equipment important to safety and finds that they are in compliance with 10 CFR 50.49 and NUREG-0588 criteria, and are acceptable.

3.11.2.2.2 Safety-Related Mechanical Equipment in a Harsh Environment

The Mechanical Equipment Qualification (MEQ) Program is the method used by the applicant to demonstrate that active safety-related mechanical equipment conforms to the applicable requirements of 10 CFR Part 50 (Appendix A), GDC 1, "Quality Standards and Records"; GDC 4, "Environmental and Dynamic Effects Design Bases"; and 10 CFR Part 50 (Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"). The qualification methods defined in NUREG-0588 can also be applied to mechanical equipment. Procedure EAI-7.07, "Watts Bar Environmental Qualification Program for Mechanical Equipment in Harsh Environments," Revision 7, defines MEQ Program responsibilities and interfaces for implementing the MEQ Program at Watts Bar Unit 1. The MEQ Program is similar to the Electrical 10 CFR 50.49 EQ Program and has the following attributes:

- Provides for identification of safety-related mechanical equipment located in harsh environment areas, including required operating time.
- Provides for identification of nonmetallic subcomponents of this equipment.
- Provides for identification of the environmental conditions for which this equipment must be qualified. The environments defined in the electrical equipment program are also applicable to mechanical equipment.

- Provides for identification of non-metallic material capabilities.

- Documents the evaluation of environmental effects.

During the week of February 27, 1995, the staff inspected the qualification files for mechanical equipment in a harsh environment at the Watts Bar site. The applicant submitted a list of systems and equipment for audit. From this list, several packages were selected and audited. The audit was structured to identify potential weaknesses in the layout of the MEQ Program and included reviewing MEQ Program procedures and the list of systems and equipment submitted by the applicant, walking down equipment installed in the plant, and inspecting components stored in the warehouse for future use. This audit did not verify calculations or calculation methodology. On the basis of its audit of the applicant's MEQ Program, the staff found that the applicant has established and implemented a program for qualification of mechanical equipment. The staff concludes that the program and its implementation are acceptable.

3.11.2.3 Service Conditions

In NUREG-0588, the staff defines methods for determining the environmental conditions associated with loss-of-coolant accidents or high-energy-line breaks, inside or outside of containment. The review and evaluation of the adequacy of these environmental conditions are described below. The qualification documentation was reviewed to assure the qualification conditions envelop the conditions established by the applicant.

Watts Bar Nuclear Plant's equipment that is required to be environmentally qualified is identified by plant areas. These areas are shown on Environmental Data Drawings Series 47E235. These drawings identify both mild and harsh environments in the plant. The environmental parameters are given in tabular form for each area. These tables include temperature, pressure, relative humidity, radiation, chemical effects (containment spray), and submergence (flooding inside and outside containment) information for the areas for normal, abnormal, and accident conditions. The review and evaluation of these environmental conditions are described below.

3.11.2.3.1 Temperature, Pressure, and Humidity Conditions

The applicant determined the normal operating environments, the short-term abnormal environments, and the LOCA/HELB profiles used for equipment qualification. The 47E235 series of drawings contains these conditions and profiles. For example, the normal temperature range is from 60 °F (15.56 °C) to 104 °F (40 °C) (90 °F average) (32.22 °C average) in fan room 1, elevation 719 feet, 9 inches (the location of 1-LT-063-0181-E, a containment level transmitter). The worst-case abnormal temperature is 110 °F (43.33 °C) for up to 8 hours. The applicant used the normal and abnormal conditions and applicable margins to set the criteria for aging the test samples. The peak calculated temperature from a LOCA is 327 °F (163.39 °C) at the location of this transmitter. This temperature was exceeded in the LOCA simulation after accelerated aging of the equipment. The applicant's program requires aging and accident tests to envelop the required values or an analysis or evaluation to support the qualification in lieu of the enveloping test. The staff has reviewed this approach to qualification and determined that it is acceptable.

3.11.2.3.2 Submergence

The applicant evaluated the effects of flooding on equipment to ensure that safe shutdown can be achieved. The applicant's EQ Program states that equipment should be located above the maximum flood level. All electrical equipment examined was located above the potential flood levels identified in the 47E235 series of drawings. For example, the potential flood level in fan room 1 (the location of 1-LT-063-0181-E) is 717.9 feet. This potential flood level is 1.85 feet below the bottom of the transmitter. Thus, this transmitter will not become submerged. The staff finds that the applicant's approach to submergence complies with 10 CFR 50.49 and the NUREG-0588 criteria and is, therefore, acceptable.

3.11.2.3.3 Chemical Spray

The applicant evaluated the effects of chemical spray impingement on EQ equipment. For the equipment items that could be exposed to chemical spray, the documented testing includes the simulation of the chemical spray with a solution that encompassed the pH and buffering of the chemical spray solution. For equipment items that would not be exposed to chemical spray, no testing to simulate a chemical spray is required.

Chemical spray effects were included in the EQ testing of in-containment equipment. For example, in fan room 1 (the location of 1-LT-063-0181-E), the effect of chemical spray on the transmitter was tested for 24 hours. The results of that testing were extended to 30 days by analysis. All EQ binders examined either included testing to simulate the postulated chemical spray or testing in combination with analysis for equipment that would be exposed to chemical spray. The staff finds that the applicant's approach to simulating chemical spray complies with 10 CFR 50.49 and the NUREG-0588 criteria and is, therefore, acceptable.

3.11.2.3.4 Aging

The applicant's EQ Program considers the degrading influences of temperature, radiation, vibration, and mechanical stresses. The program requires establishment of a qualified life with maintenance and replacement schedules based on the aging evaluation.

The applicant used the Arrhenius method to establish the qualified life for each component. Aging conditions are more severe than the worst-case operating temperature plus additional margin. The Arrhenius equation is used to correlate the aging interval to a qualified life based on the activation energy of the most limiting component of the equipment. The applicant's approach to simulated equipment aging complies with 10 CFR 50.49 and the NUREG-0588 criteria and is, therefore, acceptable.

3.11.2.3.5 Radiation (Inside and Outside Containment)

The applicant's EQ Program procedures describe the approach to qualification of equipment for radiation exposure during normal operation and accident conditions. The applicant adhered to the methods of NUREG-0588. The applicant has determined the radiation levels postulated to exist following a LOCA. The accident radiation levels, which vary depending on location, are included in the 47E235 series drawings for both inside primary containment and in areas exposed to recirculating fluid lines outside of primary containment. The maximum total integrated radiation doses specified by the applicant are also location dependent. They are also included in the 47E235 series drawings for both inside primary containment and in areas exposed to recirculating fluid lines outside of primary containment.

In all cases examined, the total integrated radiation dose applied to the test specimen was greater than the total integrated radiation dose analyzed for that location plus margin. For example, ASCO Solenoid Valve Model NP831654E, 1-FSV-081-0012-A, had radiation aging simulation of 50 megarad gamma (not exceeding 1 megarad/hour) to simulate nonaccident radiation exposure. Later in the test sequence, 150 megarad gamma (not exceeding 1 megarad/hour) simulated the accident radiation exposure. The maximum calculated accident radiation exposure at the equipment location is 136 megarad gamma. Thus, the test included the expected radiation level plus margin.

All radiation testing examined encompassed the expected radiation levels at the location of the equipment. The applicant's approach to radiation exposure complies with 10 CFR 50.49 and the NUREG-0588 criteria and is, therefore, acceptable.

3.11.3 Environmental Qualification Inspection

An inspection was conducted of the applicant's qualification documentation and installed equipment between February 27 and March 10, 1995. The inspectors reviewed 32 equipment items and 24 associated cables and verified that the test data and analyses in the files supported the qualification status determined by the applicant.

The staff inspected the installed equipment during plant walkdowns to verify the manufacturer, model number, serial number, location, and proper installation consistent with the qualification documents, and to confirm that no damage to the equipment was evident. For the 32 pieces of electrical equipment and 24 associated cables, each of these attributes was verified and found consistent with the qualification documentation.

On the basis of the qualification binders and additional information supplied by the applicant, the staff determined that there is adequate documentation establishing the qualification of the inspected equipment as claimed in all audited cases. However, the scope of the staff's inspection of the applicant's EQ Program was limited by the number of systems for which EQ had been completed. Therefore, before fuel load, the applicant must confirm that the implementation of the EQ Program is complete. The staff will track such confirmation by TAC M63591.

3.11.4 Conclusion

The staff examined the Watts Bar Unit 1 program for the environmental qualification of safety-related equipment. This review included the environmental conditions resulting from design-basis accidents, the methods used for qualification, and the documentation for specific equipment items. On the basis of the results of this review, the staff concludes that the applicant's environmental qualification program conforms to the requirements of 10 CFR 50.49; the relevant parts of General Design Criteria 1, 2, 4, and 23 of Appendix A to 10 CFR Part 50; Sections III and XI of Appendix B to 10 CFR Part 50; and the criteria specified in NUREG-0588.

Both the Electrical and Mechanical EQ programs are being implemented in accordance with program controls and procedures, and the completed EQ documentation files are acceptable. Quality assurance assessments have been effective in identifying problems with the EQ Program implementation, and those identified EQ problems are adequately addressed by the applicant's corrective action programs. The staff's acceptance of the applicant's EQ Program is subject to confirmation by the applicant that the implementation of the EQ Program is complete. The staff will track such confirmation by TAC M63591.

4 REACTOR

4.3 Nuclear Design

4.3.2 Design Description

4.3.2.7 Vessel Irradiation

The following supersedes the evaluation in the SER. The staff tracked its efforts by TACs M77896, M85037, and M85038:

As a result of the revised 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock (PTS) Events," the staff re-evaluated the applicant's neutron flux and fluence calculation, submitted by letter dated July 17, 1992. The estimated peak neutron fluence at the inner surface for the end of life (expiration of operating license) is taken to be 3.18×10^{19} neutrons/cm^2 (energy greater than 1.0 MeV). The accompanying estimated neutron flux is 3.15×10^{10} neutrons/cm^2/second. Based on a 40-year design life and an 80-percent capacity, end of life is taken to be 32 effective full power years. This information was subsequently incorporated into FSAR Section 5.2.4.3 by Amendment 72.

In Section 5.3.1 of SSER 11, the staff completed its review of the updated flux and fluence calculation which led to these revised values and found it acceptable. That evaluation is incorporated by reference.

5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.2 Overpressure Protection

In the SER, the staff stated that "Item II.D.1 of NUREG-0737 requires performance testing of relief and safety valves (discussed in Section 3.9.3.3) and Item II.D.3 (Indication of Relief and Safety Valve Position) is discussed in Section 7.8.1. Subject to the resolution of the above issues by the applicant, the staff concludes that the overpressure protection provided for Watts Bar at hot operating conditions will comply with the guidelines of SRP Section 5.2.2 and the requirements of GDC 15."

Item II.D.3 was found acceptable in Section 7.8.1 of the SER. Item II.D.1 (and subitems 1, 2, and 3) is resolved in Section 3.9.3.3 of this SSER.

5.2.4 Inservice Inspection and Testing

In SSERs 10 and 12, the staff approved the applicant's preservice inspection program, as updated to Revision 23, and relief requests. By letter dated April 13, 1995, the applicant submitted Revisions 24 and 25.

These revisions contain only administrative and procedural changes, and clarifications of the preservice inspection program. There were no new or revised requests for relief. The staff found no deviations from regulatory requirements or commitments and, therefore, concludes that the changes to the preservice inspection program as conveyed in Revisions 24 and 25 are acceptable. The staff tracked its efforts by TAC M92162.

6 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.1 Containment Functional Design

The following section conveys the staff's evaluation of the containment barrier seals, and associated surveillance requirements. The issue arose as a result of the staff's review of the draft Watts Bar Unit 1 Technical Specifications. The staff tracked its efforts by TAC M76742.

6.2.1.1 Containment Structure

Section 3.8.3 of the FSAR describes the seal installed across the gap between the inside surface of the steel containment building and the concrete divider barrier structures within the containment. This seal forms part of the divider barrier between the upper and lower compartments. It is located along the bottom of the concrete floor under the ice condenser, between the ends of the ice condenser and the refueling canal structure, and along the vertical sides of the refueling canal structure. The areas in which the seals are located are missile protected. Under design-basis-accident/loss-of-coolant-accident (DBA/LOCA) conditions, the seals would be exposed to a severe environment.

The seal was constructed from flat strips of material consisting of two plies of dacron fabric impregnated with a terpolymer of ethylene propylene diene (Class M of ASTM D1418). The flat strips were folded longitudinally and the edges butted and sewn ("pigtailed") to create individual sections in the form of a "tube." During installation, the individual sections were field-spliced in place with vulcanized joints to form one long continuous seal. In order to make the field splices, the pigtailing was removed in area of the overlap of the sections being joined. During installation, metal bars of 3/8-inch diameter were inserted into the seal for use in attachment to the containment structures with bolted clamps. Slack was provided to allow for seismic movement between the containment shell and interior structures forming part of the divider barrier. Room-temperature-vulcanized (RTV) sealant was applied to the exposed area where the seal contacts the structure. The seal vendor is the Presray Corp. Presray is a major supplier of seals used in the industry for airlock doors, refueling cavity seals, and such. Figure 6.1 depicts a typical cross-section of the seal in an unspliced location. The seal used at Watts Bar is the same as the seal used at Sequoyah. The Sequoyah and Watts Bar seals are of a different design from those of other ice condenser facilities; other facilities use a thick, solid seal material.

The divider barrier forms part of the boundary between the upper and lower compartments of the primary containment. During the blowdown phase of a LOCA, a significant pressure difference may exist between these upper and lower compartments. If the divider barrier leaks, the resultant ice condenser bypass leakage could cause the analyzed peak accident containment pressure to be exceeded. During the post-blowdown phase of an accident, the seal is subjected only to the relatively low differential pressure resulting from operation of the upper containment air return fans; however, its integrity

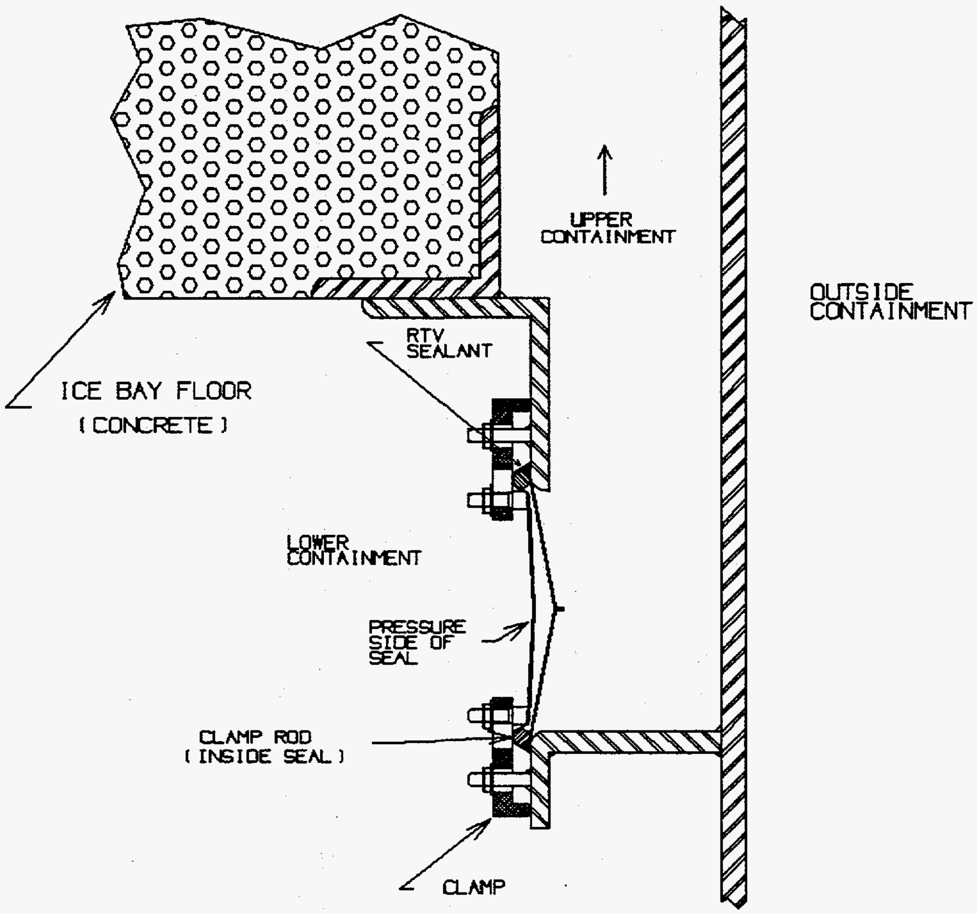

Figure 6.1 Divider Barrier Seal

continues to be important for proper long-term ice condenser operation. The applicant reviewed the potential effects of additional LOCA steam bypass due to divider barrier seal leakage, and found that the effects of leakage due to small cracks would be insignificant. The staff did not review the applicant's calculation; however, the staff noted that other analyses have shown that ice condenser containments can tolerate a relatively high amount of steam bypass (e.g., "The Effect of Steam Bypass on Ice Condenser Containment Pressurization," G. Pertmer et al., MDNE-82-111, Department of Chemical and Nuclear Engineering, University of Maryland, July 1982).

The original qualification testing of divider barrier seal materials establishes their qualification for accident conditions but not with consideration of potential degradation due to long-term thermal and radiation aging. Since resilient materials are subject to deterioration due to heat and radiation aging and mechanical damage, surveillance tests are imposed by technical specifications to ensure that the condition of the seals is monitored. The standard surveillance requirements (see NUREG-1431, "Standard Technical Specifications") prescribe periodic visual inspections and tensile strength tests to be conducted periodically on seal specimens. In a letter dated May 8, 1980, TVA described the originally planned program for preoperational and periodic seal tests at Sequoyah and Watts Bar. The final Sequoyah technical specification included surveillance tests requiring that visual and tensile tests be performed at 18-month intervals, consistent with the Standard Technical Specifications. The divider barrier seal does not have provisions for inplace testing.

In 1991, an inspector noted water draining from the seal during a period when the ice bays were being thawed because of impending construction/licensing delays. Inspection of the seal revealed compression marks and splits of 2 inches in length under the holddown clamps. The damage was attributed to overtorquing of the holddown bolts. As a result of this discovery, the applicant undertook an effort to assess the effects of compression marks and splits of up to 2 inches in length. At the same time, the applicant undertook to qualify the seal for a 40-year life. Sections of the damaged seal material were removed, cut into test specimens, and subjected to accelerated thermal and radiation aging conditions equivalent to a DBA at 40 years of life (i.e., 3.0E7 rads integrated radiation level and accelerated heat equivalent to 120°F for 40 years), and then tested under simulated DBA-LOCA pressure and temperature conditions. Finally, they were burst tested. The pigtailing was removed from some of the test specimens for these tests. The staff reviewed "Divider Barrier Seal LOCA and Burst Report," Presray Report No. TR17361-1, while on site on February 23, 1995.

The test findings indicated that the basic seal material is acceptable for a 40-year life. However, the findings also indicated that (1) the seals will develop small leakage paths at the split locations (but not at the compression mark locations) and (2) the splits will not propagate unless there is a greater than 1/8-inch split under a clamp which is fastening a part of the seal not having the pigtail stitching. When the pigtail stitching is removed, the strength of the seal is reduced considerably. Locations not having intact pigtail stitching are

- vulcanized field joints from the original installation

- splices where replacement seal material was added using cold-bond joints to replace the test sections that were removed

The applicant has stated that the seal under the clamps near the original vulcanized field splice joints will be visually inspected, and damaged areas will be repaired using the cold-bond procedure.

The sections of the installed seal that were removed for testing were replaced with sections made from identical seal material. However, the new splices used to install the replacement seal sections were made using a cold-bond adhesive, in conjunction with bolted fasteners, as it was not physically possible to use the original vulcanization process. (Any areas of the seal found to be damaged in the future will be replaced in a similar manner.) Because the pressure testing described in the previously mentioned Presray report was not performed on specimens having cold-bond spliced joints, but only on aged specimens of the damaged vulcanized sections of the seal, additional testing was necessary to demonstrate the LOCA integrity of the cold-bond spliced joints. That testing is described in Presray Report TR16550-1, "Divider Barrier Seal Splice Procedure Test Report." The staff reviewed the report while on site on February 23, 1995. The seal material used in the cold-bond adhesive testing had been heat-aged before splicing; however, the actual spliced joint had not, as it was not possible to accomplish accelerated aging of the materials simultaneously due to their different aging characteristics. The cold-splice testing indicated that spliced joints made of aged seal material but unaged adhesive will withstand LOCA conditions. The surveillance testing program will be relied upon to monitor the condition of the adhesive, and to ensure that the joints will be repaired when the adhesive has aged.

In view of the fact that additional testing has shown that the vulcanized, pigtailed seal areas are suitable for a 40-year life and that areas using the cold-bond adhesive process must be periodically tested for deterioration, the applicant has proposed unique surveillance requirements for Watts Bar. These include deletion of the standard tensile strength tests and supplemental periodic testing of the cold-bond adhesive repairs.

The standard periodic visual inspection surveillance would be retained. This test monitors the conditions of the visible portions of the seal and the RTV compound applied to the seal. However, the applicant proposes not to include periodic tensile (burst) strength testing. This is based on the laboratory testing described above which indicates that the material retains sufficient strength to perform its safety function after 40 equivalent years of thermal and radiation aging.

The cold-splice adhesive used in the installation of the material that replaces the removed specimens has been found to provide sufficient strength for LOCA integrity in the tested (unaged) conditions, but to be vulnerable to heat aging. As indicated above, its condition must, therefore, be monitored through additional unique periodic surveillance testing. Peel tests would be performed on test specimens at 18-month or 36-month intervals, the interval for each joint location being dependent on the initial peel test result for that location. The peel test procedure and recommended test interrvals are described in Presray Report No. PR16550, "Procedure for Cold Bonding Divider Barrier Seal Splice Joints." The staff reviewed the report while on site on February 23, 1995. The test specimens are not part of the actual divider barrier seal. They are specimens of material removed during the repair and are located near the seal at each splice joint.

The staff reviewed the information submitted by the applicant and concluded that a revised divider barrier seal surveillance program is appropriate for Watts Bar Unit 1. The tensile testing may be deleted based on the additional testing that demonstrates that the seal material, including the original vulcanized joints, is qualified for a 40-year life and DBA-LOCA. Additional peel tests on test coupons will be performed periodically to monitor the condition of the adhesive used in the cold-bond/bolted splices. The test frequency for the peel tests will be as recommended by the vendor.

6.6 Inservice Inspection of Class 2 and 3 Components

In SSERs 10 and 12, the staff approved the applicant's preservice inspection program, as updated to Revision 23, and relief requests. By letter dated April 13, 1995, the applicant submitted Revisions 24 and 25.

These revisions contain only administrative and procedural changes, and clarifications of the preservice inspection program. There were no new or revised requests for relief. The staff found no deviations from regulatory requirements or commitments and, therefore, concludes that the changes to the preservice inspection program as conveyed in Revisions 24 and 25 are acceptable. The staff tracked its efforts by TAC M92162.

7 INSTRUMENTATION AND CONTROLS

7.1 Introduction

7.1.3 Design Criteria

7.1.3.1 Safety System Setpoint Methodology

In the SER, the staff reported that it had reviewed the applicant's setpoint methodology and found it acceptable. By letter dated July 29, 1994, the applicant submitted Topical Report WCAP-12096, Revision 6, "Westinghouse Setpoint Methodology for Protection System Watts Bar (WBN) Units 1 and 2, Eagle 21 Version" (proprietary). The applicant submitted the topical report to complete a commitment made in a letter dated July 9, 1991, to revalidate the reactor coolant system flow measurement uncertainty. Westinghouse Electric Corporation prepared the topical report for the applicant to describe the methodology that was used to determine the instrument setpoints associated with the reactor protection system and the engineered safety features actuation system.

The setpoint methodology was developed by Westinghouse in response to a series of questions raised by the staff in March 1977 to several utilities concerning the technique used in determining the overall allowance for each setpoint. Westinghouse developed the methodology by using a basic underlying assumption that several of the error components and their parameters are independent. The use of this assumption changed the summation technique used for allowance value calculation from arithmetic summation to the square root of the sum of the squares. Compared to the strictly arithmetic summation, the statistical summation allows a larger margin in the total allowance. The staff has accepted this approach.

The staff reviewed (1) WCAP-12096, Revision 6, and (2) WCAP-11239, "Westinghouse Setpoint Methodology for Protection Systems, Sequoyah Units 1 and 2 Eagle 21 Version." The staff reviewed these reports according to the guidelines in Regulatory Guide 1.105, "Instrument Setpoints for Safety-Related Systems," and ISA S67.04, "Setpoints for Nuclear Safety-Related Instrumentation Used in Nuclear Power Plants." Based on the review of these documents, the staff concludes that the WBN setpoint methodology is in compliance with ISA SS67.04 and Regulatory Guide 1.105.

The staff review included a telephone conference with the applicant and Westinghouse representatives on November 22, 1994, during which the following points were clarified:

(1) The use of the same setpoint methodology at other plants;

(2) Comparison of Watts Bar's input values, such as rack measurement and test equipment accuracy, rack comparator setting accuracy, and other parameters shown in Table 3, to that of Sequoyah Nuclear Plant;

(3) An explanation of how many bits are used to represent the analogue input value when analogue input is converted to digital representation.

The applicant's setpoint methodology is the same as that used in other plants such as Sequoyah, and Diablo Canyon (both these plants have a system design similar to that of Watts Bar.) The NRC has previously approved the use of this setpoint methodology in those plants.

Watts Bar's parameter values used in the allowance calculation shown in Table 3 of WCAP-12096, Revision 6, are the same as other plants where Watt Bar's hardware, or test and measurement equipment are the same. Differences in hardware, or test and measurement equipment from other plants account for differences in parameter values.

The Eagle-21 system uses a 12-bit analogue-to-digital converter. For the allowance calculation, Westinghouse used a 2-percent error value, which allows errors in the last two bits of the converter.

The staff concludes that Watt Bar's setpoint methodology is acceptable based on (1) previous acceptance of Westinghouse's setpoint methodology used at other plants, (2) the similarity between the Watts Bar design and previously approved designs, such as Sequoyah, and (3) the Watts Bar setpoint methodology is in compliance with Regulatory Guide 1.105 and ISA S67.04.

This effort was tracked by TAC M89390.

7.2 Reactor Trip System

In SSER 13, the staff identified two open issues related to the Unit 1 Eagle-21 system. The issues concern the trip time delay (TTD) postmodification test of the reactor protection system (RPS), and electromagnetic interference and radio frequency interference (EMI/RFI) environmental qualification of Eagle-21 equipment in the RPS. For TTD postmodification testing, the staff reviewed the site acceptance test results to verify that the TTD design is properly implemented into the RPS. For EMI/RFI qualification, the staff reviewed the site survey results and analysis in a report entitled "Analysis for EMI/RFI Mapping of Auxiliary Electric Equipment Room for Tennessee Valley Authority's Watts Bar Nuclear Plant Unit 1," which Tennessee Valley Authority (TVA) submitted by letter dated August 12, 1994. Details are in the following sections.

As stated in SSER 13, the evaluation in the following sections was tracked by TAC M81063.

7.2.1 System Description

7.2.1.1 Updated Protection Features

Steam Generator Low-Low Trip Time Delay

In SSER 13, the staff reported that the applicant has committed to submit the set point methodology document. The applicant submitted that document by letter dated July 29, 1994; Section 7.1.3.1 (above) delineates staff's evaluation of that submittal.

The staff reviewed the site acceptance test results for the Watts Bar Nuclear (WBN) Plant Eagle-21 RPS, and participated in a meeting with TVA representatives to discuss certain steam generator level TTD issues. The review was performed on site on January 17, 18, and 19, 1995, as part of the preoperational test inspection.

The applicant tested the TTD by checking the trip time delays for the single and multiple steam generator operating cases at 0-percent power and 50-percent power. The applicant indicated that the method and components, which include software, used to calculate the time delay for power between 0 percent and 50 percent are the same. Therefore, checking two time delays is sufficient to verify the TTD design. The applicant also stated that the staff has previously accepted the TTD design at the Sequoyah Nuclear Plant, which is similar to the Watts Bar design. The TTD software was validated as part of the overall Eagle-21 software validation which was previously reviewed and accepted by the staff. Finally, the Watts Bar Eagle-21 site acceptance test results indicated that the TTD was functioning correctly. The staff reviewed the applicant's site acceptance test results and found them acceptable. The results of the staff's review and acceptance of the test results were reported in Inspection Report 50-390/95-11, dated March 27, 1995. On this basis, the staff concludes that this open issue is closed.

7.2.1.2 Watts Bar—Specific Issues

EMI/RFI Concerns

In response to the the staff's EMI/RFI concerns, Westinghouse performed an additional site survey and an analysis for the applicant to demonstrate that EMI/RFI at Watts Bar Unit 1 does not impair the operation of the Eagle-21 equipment. Westinghouse documented the site survey data and analysis in a report, "Analysis for EMI/RFI Mapping of Auxiliary Electric Equipment Room for Tennessee Valley Authority's Watts Bar Nuclear Plant Unit 1," which the applicant submitted on August 12, 1994. On the basis of the site survey, test results, and analysis, the applicant and Westinghouse concluded that the installed Eagle-21 system in Unit 1 will not be impaired by EMI/RFI.

The staff reviewed the Westinghouse report to ensure that the survey data are enveloped by the susceptibility threshold levels previously established by Westinghouse and accepted by the staff. In addition, the staff spoke twice with TVA representatives by phone on December 19, 1994 and January 18, 1995, to discuss certain EMI/RFI issues raised in the report. As a result of these calls, the applicant responded by letter on February 16, 1995.

Initial Test

Westinghouse performed EMI/RFI tests on the Eagle-21 equipment in an echoic chamber as described in Topical Report WCAP-11733, "Noise, Fault, Surge and Radio Frequency Interference Test Report for Westinghouse Eagle-21 Process Protection Upgrade System," which was previously accepted by the staff during the Zion Station Eagle-21 replacement review. Westinghouse stated that the tests were conducted in accordance with Scientific Apparatus Manufacturers Association (SAMA) Standard PMC 33.1-1978, "Electromagnetic Susceptibility of Process Control Instrumentation," with field strengths of 3 V/m and 10 V/m over a frequency range of 20 MHz to 1 GHz. Two types of tests were performed. The first test was a modulation test which consists of a sweep of the signal generator over the frequency range for multiple data points. The second test was a keying test to simulate the keying of a transmitter.

The report submitted on August 12, 1994, stated that the configuration of the Eagle-21 system used for the test described in WCAP-11733 is similar to the Eagle-21 system configuration installed at Unit 1.

Site Survey

The referenced report contains the site survey data that characterize the EMI/RFI environment in the auxiliary electric equipment room in the vicinity of the Eagle-21 process protection equipment. The following surveys were performed in the auxiliary electric equipment room by Westinghouse for TVA:

- CE01 - Conducted Emissions Test, Power and Signal Leads, Common and Differential Mode, 30 Hz to 15 KHz
- CE03 - Conducted Emissions, Power and Signal Leads, Common and Differential Mode, 15 KHz to 50 MHz
- CE07 - Conducted Emissions, Power and Signal Leads, Switching Transients, Time Domain, Common and Differential Mode
- REXX - Radiated Emissions, DC Magnetic Field,
- RE01 - Radiated Emissions, 30 Hz to 50 KHz
- RE02 - Radiated Emissions, 14 KHz to 1 GHz
- RE02.1 - Radiated Emissions, Hand Held Radio Profile

The report submitted on August 12, 1994, stated that these site surveys were performed during a hot functional test. The applicant pointed out that the other plants performed the EMI/RFI site survey measurements of their equipment room with old protection system racks before installing the Eagle-21 system. The Unit 1 EMI/RFI site survey measurements were performed subsequent to installing the Eagle-21 system, and the applicant, therefore, believed that the Unit 1 site survey data were more representative of actual operating environments than the data collected for other plants.

Analysis

Westinghouse performed an analysis by comparing the onsite survey measurements with the Eagle-21 equipment factory test results and susceptibility threshold levels previously used at Zion. The staff previously accepted the use of these threshold levels for the Eagle-21 equipment EMI/RFI environment analysis. The analysis showed that all surveyed data were enveloped by the Eagle-21 equipment EMI/RFI threshold levels with at least a 6-dB margin as specified in the Westinghouse acceptance criteria, except for the CE03 measurement at 1.8 MHz.

The survey data showed that a 152-dBμA transient, which is about 5 dB below the susceptibility threshold level, occurred during the CE03 test at 1.8 MHz. The transient occurred only once on the Rack #5 power lead at 3.5 minutes after the start of the survey measurement. Westinghouse stated that although the origin of the transient was not determined, the transient did not appear on any of the radiated emission tests or CE03 signal cable tests. In addition, the report submitted on August 12, 1994, states that the threshold susceptibility levels for the Eagle-21 system are determined by assuming the worst design. However, the actual Eagle-21 equipment has EMI/RFI-eliminating design features, such as low-pass filters, which would filter out high-frequency noises. Therefore, the Unit 1 Eagle-21 equipment can meet the 6-dB margin criterion because the additional 1-dB margin (between measured 5 dB and the specified 6 dB) can be obtained from its EMI/RFI eliminating design features.

Conclusion

The applicant also restated its commitment that administrative controls at Watts Bar will prohibit the use of radios and portable telephones in the vicinity of the Eagle-21 equipment. This will preclude the majority of external EMI/RFI sources as a potential cause of improper operation of Eagle-21.

On the basis of (1) the staff's review of the survey report, (2) clarification of the survey analysis report transmitted by the applicant's letter of February 16, 1995, (3) the staff's previous acceptance of Westinghouse's threshold levels for the Eagle-21 equipment, (4) the applicant's commitments to maintain the prohibition of radios and portable telephones near the Eagle-21 equipment, and (5) the integrity of the Eagle-21 system, the staff concludes that the Eagle-21 EMI/RFI issue is resolved.

7.5 Safety-Related Display Information

7.5.2 Post-Accident Monitoring System

In SSER 9, the staff evaluated the post-accident monitoring system and concluded that the applicant either conforms to or has adequately justified deviations from the guidance of Regulatory Guide (RG) 1.97, Revision 2, for each post-accident monitoring variable. In SSER 14, the staff evaluated additional deviations from RG 1.97, Revision 2, concerning post-accident monitoring instrumentation in response to the applicant's May 9, 1994, letter. By letter dated April 21, 1995, the applicant identified further deviations from RG 1.97, Revision 2. The staff has reviewed these deviations and concludes that the applicant either conforms to, or has adequately justified the deviations from the guidance of RG 1.97, Revision 2, as follows:

(1) RG 1.97, Revision 2, recommends that the position of containment isolation valves be monitored. The applicant deviates from this recommendation in that the position of certain relief valves designated as containment isolation valves for the containment spray system, chemical and volume control system, safety injection and residual heat removal (RHR) system are not monitored. Containment spray system valves 1-RFV-72-508 and 1-RFV-72-509; chemical volume and control system valves 1-RFV-62-505, 1-RFV-62-1220, 1-RFV-62-1221, and 1-RFV-62-1222; and safety injection and RHR system valves 1-RFV-63-511, 1-RFV-63-534, 1-RFV-63-535, 1-RFV-63-536, 1-RFV-63-626, 1-RFV-63-627, 1-RFV-63-637, and 1-RFV-63-835 are closed during normal operation and remain closed during and after an accident. Therefore, the position of these valves does not need to be monitored and this deviation is acceptable.

(2) RG 1.97, Revision 2, recommends a range of 1.0E-6 to 1.0E+5 microcuries per cubic centimeter (μCi/cc) to monitor noble gas releases through the condenser air removal system exhaust. By letter dated May 9, 1994, the

applicant proposed a range of 3.2E-7 to 3.5E+3 μCi/cc to monitor the condenser pump vacuum exhaust vent radiation level. The staff approved this range in SSER 14. In its April 21, 1995, submittal, the applicant proposed a new range of 4.0E-7 to 2.4E+3 μCi/cc to reflect revisions in supporting calculations.

The applicant stated that the only credible accident monitored by this variable is a steam generator tube rupture (SGTR). The Standard Review Plan (NUREG-0800) recommends that the tube rupture accident be analyzed using the highest isotope concentrations allowed by the plant's Technical Specifications (TSs). The highest concentration of mixed noble gas isotopes that can be present under the TS limit is 1.45E+3 μCi/cc as determined in TVA calculation WBNAPS3-048. This value is within the proposed range for the condenser vacuum pump exhaust radiation monitor under expected design-bases conditions. On this basis, this deviation is acceptable.

(3) In its April 21, 1995, submittal, the applicant revised the range of the essential raw cooling water (ERCW) radiation monitors. The applicant revised the lower value from 1.93E-4 to 3.3E-4 μCi/cc and the upper value from 1.93E-2 to 1.65E-2 μCi/cc.

The applicant stated that the ranges were revised as a result of 10 CFR Part 20 revisions. Revisions to 10 CFR Part 20 caused the upper value of the required measurement range to decrease slightly. Since the lower range value is calculated based on the upper range value, the resultant required lower range value became lower. On the basis of the applicant's recalculation of the required monitoring range, this deviation is acceptable.

(4) RG 1.97, Revision 2, recommends the use of permanently installed monitors to measure radiation exposure rates. The staff previously accepted the use of portable monitors in SSER 9. In its April 21, 1995, submittal, the applicant made a clerical change to the table of post-accident monitoring variables to reflect the applicant's use of portable monitors. This revision is acceptable.

The staff tracked its efforts by TAC M92195.

8 ELECTRICAL POWER SYSTEMS

The staff tracked its efforts on electrical power systems by TACs M89109 and M89110.

8.2 Offsite Electric Power Systems

8.2.2 Compliance With GDC 17

8.2.2.3 Compliance With GDC 17 for the Duration of Offsite System Contingencies

The material that follows supplements the information in SSER 13.

In SSER 13, the staff stated that the applicant would manually enable an automatic load shedding scheme for balance-of-plant (BOP) loads during offsite power system contingencies. This fulfilled the requirements of GDC 17 by providing adequate capacity and capability and was acceptable.

In Inspection Report 50-390, 391/95-08, dated March 28, 1995, the inspector stated that the applicant's procedure for grid contingencies (when any one of certain transmission lines is out of service) would require entry into a limiting condition for operation (LCO) for one offsite circuit inoperable. The inspector found this reasonable because one of the transmission lines in the grid being out of service would mean that the 161-kV grid could not supply adequate voltage for a design-basis accident simultaneously with a fault in the grid.

In a followup conference call on April 26, 1995, the applicant agreed with the information in the inspection report and stated specifically that, with one of the transmission lines out of service, a fault in the 161-kV switchyard (in addition to the grid contingence) could lead to loss of both preferred offsite sources due to breaker action and breakup of the switchyard into sections. Therefore, when one of the transmission lines is out of service, entering the LCO for one offsite circuit inoperable (the specific offsite circuit to be determined by conditions such as switchyard alignment) was determined to be appropriate. The applicant also reiterated that no credit is taken for the automatic load shedding scheme during grid contingencies as stated on Page 8.2-15 of the plant's FSAR.

This satisfies the requirements of GDC 17 and is acceptable. The staff has no more concern in this regard.

8.3 Onsite Electric Power System

8.3.3 Common Electrical Features and Requirements

8.3.3.1 Compliance With GDCs 2 and 4

8.3.3.1.4 Use of Waterproof Splices in Potentially Submersible Sections of Underground Duct Runs

The material that follows supplements the information in SSER 14.

In SSER 14, the staff stated that an NRC resident inspector at Watts Bar had indicated that numerous cable splices had been used throughout the emergency diesel generator output cable runs. These splices were located in cable trays and in manholes. The staff found this unacceptable because of the relative importance of these cables.

Subsequently, in Inspection Report 50-390, 391/94-72, dated November 10, 1994, the staff raised additional issues pertaining to splices in these cables. Among those issues was the use of splices rated for 600 V in 6900-V applications. During a staff visit to the site on November 1, 1994, other issues pertaining to inadequate splice installation (e.g., insufficient number of crimps, flash points not removed, inadequate crimp overlap, and use of wrong crimping tools) were discussed with the resident inspectors and the applicant. All these issues related to inadequate splices at Watts Bar were further discussed in a meeting with the applicant on November 3, 1994. At that meeting, TVA gave the bases for a determination that safety-related splices at the plant are adequate. Nevertheless, the staff expressed concerns related to the adequacy of splices and asked the applicant to submit a formal, written response to those concerns.

The applicant submitted a formal response to the staff's concerns in a letter dated November 18, 1994. After performing visual inspections, X-ray tests, and personnel interviews, the applicant concluded that all Thomas & Betts connectors (except for the 15-kV type, which have the most complex installation procedure), Burndy connectors, and Penn Union connectors (pending successful commercial dedication) were installed correctly and, if insulated properly, would be suitable for 6900-kV use. On the basis of some testing (secureness, static heat rise, and pullout) to Underwriter Laboratories Standard for Safety UL 486A, visual inspections, and load testing, the applicant concluded that Thomas & Betts 15-kV connectors with the worst-case crimping configuration were suitable for interim use and committed to perform additional testing to support long-term suitability. Also, the applicant performed thermography inspections on some of the splices in the emergency diesel generator output cables while they were carrying loads. As documented in Inspection Report 50-390, 391/94-82, dated January 13, 1995, several of these inspections were witnessed by a resident inspector and preliminary results indicated that the observed temperatures at the splices were relatively low.

In a January 5, 1995, letter, the applicant reiterated much of the information from the November 18, 1994, letter and stated that the Penn Union connectors had received successful commercial dedication for use at Watts Bar. Additionally, the applicant stated that the Thomas & Betts 15-kV connectors with the worst-case installation were (1) further tested to UL 486A, as documented in TVA Central Services Laboratory Report 95-0248; (2) had acceptable test results; and (3) were suitable for long-term use at Watts Bar. Also, in response to a staff concern, the applicant stated that measured voltage drops across splices in the emergency diesel generator output cables obtained during testing indicated that the splices presented no additional impedance beyond values already used in voltage analyses and that there would be no adverse operating temperatures in parallel conductors if a splice should fail open while carrying required load current.

The applicant's resolutions of the staff's concerns about numerous and improper splices have been supported by the connector manufacturers (Thomas & Betts and Burndy) and by the manufacturer of the insulating material (Raychem). The staff finds that the applicant has adequately justified the acceptability of the installed splices at Watts Bar, and the issues are resolved.

8.3.3.1.5 Dow Corning RTV-3140 Used To Repair Damaged Kapton-Insulated Conductors

In Inspection Report 50-390, 391/94-61, dated October 12, 1994, the staff discussed inadequate corrective actions for the repair of damaged Kapton-insulated conductors located at containment electrical penetrations. One repair method that the applicant used involved the application of Dow Corning RTV-3140 silicone over damaged areas. Because this repair method has not been used previously at any other nuclear power plant in the country, the staff asked the applicant to submit additional justification for this method.

The staff discussed this method of repair with the applicant during a November 3, 1994, meeting (summary dated November 16, 1994). The applicant's December 6, 1994, letter gives the technical basis for the use of RTV-3140 for repairs of this nature and mentions that tests (including wet electrical testing after exposure to thermal cycling, thermal aging, and radiation) would be performed to confirm the capability of this material. In a February 10, 1995, letter, the applicant stated that even though the testing had been completed, a decision had been made to limit the use of this repair method to only non-Class 1E penetration pigtails with damage found no more than two inches from the penetration seal.

This use of RTV-3140 for repair of damage to Kapton-insulated conductors is acceptable for the limited use described.

8.3.3.5 Compliance With GDC 18

8.3.3.5.1 Compliance With Regulatory Guides 1.108 and 1.118

The material that follows replaces the entire discussion under Subsection (1) in SSER 14 and supplements Subsection (3) in SSER 14. Also, the material does not change the staff's original conclusions.

(1) Class 1E Standby Power System Testing

In Section 8.1.5.3 of the FSAR, as updated by Amendment 86, the applicant indicated that the Watts Bar electrical system design does not fully comply with Position C.2.2.6 of RG 1.9 (Revision 3). Position C.2.2.6 of RG 1.9 (Revision 3) recommends that a combined safety injection actuation signal (SIAS) and loss of offsite power (LOOP) test be performed (as part of preoperational and periodic testing programs) to demonstrate that the emergency diesel generator can satisfactorily respond to a LOOP in conjunction with a SIAS in whatever sequence they might occur (e.g., loss-of-coolant accident (LOCA) followed by delayed LOOP or LOOP followed by LOCA). In clarification, the applicant stated:

> The design basis at WBN is a simultaneous LOOP/LOCA, not LOOP followed by LOCA. Although there are some design features to meet the effects of LOOP followed by LOCA, there is no analysis to demonstrate the design will meet the DG voltage and frequency requirements.

On the basis of this clarification, the staff understood that an actual simulated LOOP followed by a LOCA test would not be performed. In place of an actual simulation, the staff understood that the operability of the additional design features installed to meet the effect of LOOP before a LOCA would be demonstrated by a number of different tests.

The following features will be tested as required by the Watts Bar Technical Specifications:

- A simultaneous LOOP/LOCA event will be demonstrated by simulating an actual LOOP and LOCA signal.
- The capability of the diesel generator to start and operate at no load will be demonstrated by test.
- With the standby diesel generator operating at no load, Class 1E buses are deenergized, loads are shed from the buses, and the standby diesel generator energizes permanently connected loads.

The following features will be tested individually, circuit by circuit, during component testing preceding plant preoperational testing in a nonintegrated manner:

- After a LOOP followed by a delayed LOCA, loads already sequentially connected to Class 1E buses which are not required for an accident are disconnected.
- After a LOOP followed by a delayed LOCA, loads already sequentially connected to the Class 1E buses which are required for an accident remain connected.
- After a LOOP followed by a delayed LOCA, loads awaiting sequential loading that are not required for an accident will not be connected.
- After a LOOP followed by a delayed LOCA, loads awaiting sequential loading that are required for an accident are either sequentially loaded as a result of the nonaccident loading sequence or have their sequential timers reset to time zero from which they are then sequentially loaded in accordance with the accident sequence.

The following will be tested as part of the Watts Bar preoperational test program:

- A random load test involving the simultaneous starting of a containment spray pump (the largest accident load) and a fire pump (the largest non-accident load) will be performed to demonstrate that a partially loaded diesel generator is capable of supplying additional accident loads (albeit not the worst-case loading that may occur due to automatic load sequencing during a LOOP followed by a delayed LOCA which could involve nearly simultaneous starting of two accident loads).

Criteria III and XI of 10 CFR Part 50, Appendix B, require that (1) measures be provided for verifying or checking the adequacy of design by design reviews, by the use of alternative or simplified calculational methods, or by the performance of a suitable testing program; and (2) a test program be established to ensure that systems and components perform satisfactorily and that the test program includes operational tests during nuclear power plant operation.

The staff concludes that tests performed before preoperational testing and a preoperational and periodic test program which includes the testing described above will provide some assurance of the capability of the additional design features installed to meet the effect of LOOP before a LOCA. On the basis of this information and the commitment to comply with the recommendations of Regulatory Guide 1.118 (Revision 2) and IEEE Standard 338-1977 documented in FSAR Section 8.1.5.2, the staff concludes that the additional design features installed to meet the effect of LOOP before a LOCA will be appropriately tested as part of the applicant's test programs and are, therefore, acceptable.

In addition, the staff has initiated a generic issue involving the capacity and capability of safety systems to respond to a nonsimultaneous LOOP/LOCA event including a LOCA with a delayed LOOP or a LOOP with a delayed LOCA. Any requirements resulting from this generic issue would be applicable to Watts Bar as well as other plants.

(3) <u>Non-Class 1E Circuitry Used for Transmitting Signals Needed for Starting Diesels Generators</u>

In SSER 14, the staff stated that the circuitry used to supply the emergency start signal as a result of an accident was non-Class 1E for the two diesel generators on the unit that is not experiencing an accident and Class 1E for the two diesel generators on the unit experiencing an accident. While this statement is true, it should be noted that specifically a safety injection signal in one train of one unit will start its corresponding diesel generator via Class 1E circuitry and also start all four diesel generators via non-Class 1E circuitry, as shown in Figures 8.3-29B through 29E of the FSAR.

9 AUXILIARY SYSTEMS

9.1 Fuel Storage Facility

9.1.2 Spent Fuel Storage

In a letter sent to NRC on April 21, 1995, the applicant submitted proposed FSAR change pages, updating the criticality analysis for the Watts Bar spent fuel storage racks. The staff reviewed the proposed resolution to the Boraflex degradation issue and finds it acceptable. This effort was tracked by TAC M92159.

As stated in the SER, Boraflex is used as a neutron absorber in the cell walls of the spent fuel storage racks to reduce the reactivity of the racks. Experimental data from test programs and measurements performed at various boiling-water reactor and pressurized-water reactor spent fuel storage pools have indicated that when Boraflex is exposed to gamma radiation, the material may shrink by as much as 3 or 4 percent. Shrinkage saturates at an integrated gamma exposure of about 1 to 2 x 10^{10} rad. Data from laboratory tests and spent fuel pool silica measurements have also identified a second factor that could impact storage rack service life, that is, the potential gradual release of silica from Boraflex following gamma irradiation and long-term exposure to the wet pool environment. The staff issued Information Notices 87-43 and 93-70 describing the industry problems associated with Boraflex.

Because of these problems, the applicant has reevaluated the adequacy of the spent fuel storage racks to prevent criticality for long-term storage of irradiated fuel assemblies. Criticality analysis by the applicant using NRC-accepted methods shows that fuel initially enriched to up to 3.5 weight percent U-235 will limit the effective multiplication factor (k_{eff}) of the racks to 0.95 or less if arranged in a checkerboard pattern (i.e., no face-adjacent fuel), even without credit for the soluble boron in the pool water or the Boraflex in the cells. Therefore, the applicant has imposed temporary administrative restrictions to ensure the storage of fuel in a fuel/water checkerboard pattern until future Boraflex surveillance data in the pool are available. Adjustments to these storage restrictions will be reevaluated at that time. The staff concludes that these storage restrictions are sufficient to ensure that the spent fuel rack k_{eff} will not exceed 0.95 with 3.5 weight percent U-235 fuel, including all uncertainties, if fully flooded by unborated water and assuming no credit for Boraflex.

9.1.3 Spent Fuel Pool Cooling and Cleanup System

By letter dated March 8, 1995, the staff responded to a submittal, filed as a petition pursuant to 10 CFR 2.206, regarding spent fuel pool storage safety issues. The submittal, filed on November 28, 1994, by David A. Lochbaum and Donald C. Prevatte, requested that the staff perform specific analyses related to spent fuel pool cooling when reviewing certain licensing actions. The petition followed a notification pursuant to 10 CFR Part 21, dated November 27, 1992, on the same subject. On October 7, 1993, the staff issued Information Notice 93-83, "Potential Loss of Spent Fuel Pool Cooling Following a Loss of Coolant Accident (LOCA)."

In its March 8, 1995, letter (W. Russell to D. A. Lochbaum and D. C. Prevatte responding to 2.206 submittal), the staff discussed the extent to which it would treat spent fuel pool cooling issues in reviewing applications under the affected licensing process. With regard to issuance of operating licenses pursuant to 10 CFR 50.57, the staff stated that the Watts Bar Nuclear Plant, Unit 1, is the only application for an operating license currently under review. The staff issued the SER in June 1982, concluding that the spent fuel pool cooling and cleanup system design was in compliance with all applicable design criteria and was, therefore, acceptable. As a result of the March 8, 1995, submittal, the staff reevaluated the basis for the SER conclusions. The staff's efforts were tracked by TAC M91521.

The staff performed a detailed evaluation of spent fuel pool cooling capability based on a Final Safety Analysis Report review, and an onsite review conducted on March 7, 1995. The staff determined that the design of the spent fuel pool cooling system would maintain cooling through the initiating events postulated in the 10 CFR Part 21 report. Specifically, components necessary to maintain spent fuel cooling are restarted on restoration of power following a low-voltage relay actuation, are maintained running after a safety injection signal, and are designed to remain functional following a safe-shutdown earthquake. Although the rate of flow to the spent fuel pool cooling system heat exchangers from the component cooling system (CCS) may be reduced following a safety injection signal, the applicant had evaluated and verified compliance with Item II.B.2 of NUREG-0737 for access to the auxiliary building to manually realign the CCS for spent fuel pool cooling. The applicant stated that this evaluation considered operator actions necessary to align the alternate spent fuel pool cooling pump for operation.

The staff reevaluated the spent fuel pool cooling capability at Watts Bar considering the identified issues, and determined that the applicant has demonstrated an acceptable capability to maintain or recover spent fuel pool cooling following design-basis events with the potential to interrupt spent fuel pool cooling. Therefore, the staff concludes that the spent fuel pool cooling system satisfies the requirements of General Design Criterion 44 with regard to transferring heat from the spent fuel to an ultimate heat sink under normal operating and accident conditions.

13 CONDUCT OF OPERATIONS

13.6 Physical Security Plan

In the SER, the staff stated that it reviewed three of the applicant's documents in accordance with Section 13.6, "Physical Security," of the July 1981 edition of the Standard Review Plan. The staff identified certain open issues (Outstanding Issue 15).

Subsequently, the applicant submitted the following security plans, superseding previous submittals:

- "Watts Bar Nuclear Plant Security Plan," dated March 1, 1994 (submittal letter dated March 15, 1994), and letters dated August 2, October 21, and December 20, 1994, to correct certain pages

- "Watts Bar Nuclear Plant Personnel Training and Qualification Plan," dated April 1, 1994 (submittal letter dated April 21, 1994)

- "Watts Bar Nuclear Plant Safeguards Contingency Plan," dated October 1, 1990 (submittal letter dated November 5, 1990)

The following safety evaluation summarizes how the applicant has provided for meeting the requirements of 10 CFR Part 73. The staff's safety evaluation is composed of a basic analysis that is available for public review (following) and a protected appendix, which contains safeguards information under 10 CFR 73.21, that is not available for public disclosure.

The staff will impose a license condition in the Watts Bar operating license, to read as follows:

> The licensee shall fully implement and maintain in effect all provisions of the physical security, personnel training and qualification, and safeguards contingency plans previously approved by the Commission, and all amendments and revisions to such plans made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain safeguards information protected under 10 CFR 73.21, are entitled "Watts Bar Nuclear Plant Physical Security Plan," with revisions submitted through March 1, 1994; "Watts Bar Nuclear Plant Personnel Training and Qualification Plan," with revisions submitted through April 1, 1994; and "Watts Bar Nuclear Plant Safeguards Contingency Plan," with revisions submitted through October 1, 1990.

After reviewing these documents and visiting the site, the staff has concluded that the protection provided by the applicant against radiological sabotage at the Watts Bar Nuclear Plant conforms to the requirements of 10 CFR Part 73. Accordingly, the protection provided will ensure that the health and safety of the public will not be endangered. Since the applicant's submittals listed above supersede previous submittals, and the staff finds them acceptable (see following sections), Outstanding Issue 15 is considered resolved. The staff tracked its efforts by TACs M63657, M83973, M90696, and M90729.

13.6.1 Physical Security Organization

To satisfy the requirements of 10 CFR 73.55(b), the applicant has provided a physical security organization that includes a site security shift supervisor who is on site at all times with the authority to direct the physical protection activities. To implement the commitments made in the physical security plan, personnel training and qualification plan, and the safeguards contingency plan, the applicant prepared written security procedures, specifying the duties of the security organization members. The procedures are available to the staff for inspection.

The training program and critical security tasks and duties for the security organization personnel are defined in the "Watts Bar Nuclear Plant Personnel Training and Qualification Plan," which conforms to the requirements of 10 CFR Part 73, Appendix B, for the training, equipping, and qualification of the security organization members. The physical security plan and the training program provide commitments that preclude the assignment of any individual to a security-related duty or task before the individual has been trained, equipped, and qualified to perform the assigned duty in accordance with the approved personnel training and qualification plan.

13.6.2 Physical Barriers

In conforming to the requirements of 10 CFR 73.55(c), the applicant has provided a protected area barrier which conforms to the standards as defined in 73.2, definitions for "Physical Barriers." The applicant provided an isolation zone of at least 20 feet, to permit observation of activities along the barrier, on both sides of the barrier with the exception of the locations listed in the appendix. The staff has reviewed those locations and determined that the security measures in place are satisfactory and continue to conform to the requirements of 10 CFR 73.55(c).

Illumination of 0.2 foot-candle is maintained for the isolation zones and exterior areas within the protected areas. In areas in which illumination of 0.2 foot-candle cannot be maintained, special procedures are applied as described in the appendix.

The protected area is patrolled at random intervals to detect the presence of unauthorized persons, vehicles, and materials.

13.6.2.1 Identification of Vital Areas

The appendix identifies those areas determined to be vital for protection purposes. Vital equipment is located within vital areas which are located within the protected area, and which require passage through at least two barriers, as defined in 10 CFR 73.2, definitions for "Physical Barriers," with certain exceptions, to gain access to vital equipment. The staff has reviewed those exceptions and has determined that the barriers are sufficiently substantial to meet the intent of the two-barrier requirement.

Except for the exceptions noted in the appendix, vital area barriers are separated from the protected area barrier. The control room has bullet-resistant walls, doors, ceilings, and floors. On the basis of these findings as stated in the appendix, the staff has concluded that the applicant's program for identification and protection of vital equipment satisfies the regulatory intent.

13.6.3 Access Requirements

In accordance with 10 CFR 73.55(d), all points of personnel and vehicle access to the protected area are controlled. The individual responsible for controlling the final point of access into the protected area (PA) through the access control portals (ACPs) is stationed in a bullet-resistant structure. Personnel access points within the ACP that are not controlled from a bullet-resistant structure shall be manned by an armed member of the security force (MSF) and observed by closed-circuit television (CCTV) when open. Access into the PA, other than the ACP, that is not controlled from within a bullet-resistant structure will be controlled by a minimum of two MSFs, one of whom will be armed and capable of continuous communications with the central alarm station (CAS) and secondary alarm station (SAS). As part of the access control program, vehicles (except under emergency conditions), personnel, packages, and materials entering the PA are searched for explosives, firearms, and incendiary devices by electronic search equipment or physical search or both.

Except for TVA-designated vehicles, vehicles admitted to the PA, are controlled by escorts. TVA-designated vehicles are limited to those essential for plant operations. Positive control over these vehicles is maintained by personnel authorized to use the vehicles, or the vehicles are immobilized to prevent their use by unauthorized persons.

A photobadge/hand-geometry biometric system, utilizing encoded information, identifies individuals who are authorized unescorted access to protected and vital areas and is used to control access. Individuals not authorized unescorted access are issued badges that have no photo; these indicate that an escort is required. Access authorizations are limited to those people who have a need for access to perform their duties. (Details of the staff's review may be found in exemption and safety evaluation, 59 *Federal Register* 66060, dated December 22, 1994.)

Unoccupied vital areas are locked and alarmed. Anytime frequent access is permitted to the containment, access shall be positively controlled by an MSF to ensure that only authorized individuals are permitted to enter. In addition, all doors and personnel/equipment hatches into the containment are locked and alarmed. Keys, locks, combinations, and related equipment are changed annually. When an individual's access authorization has been terminated due to lack of reliability or trustworthiness, or for poor work performance, the keys, locks, combinations, and related equipment to which that person had access are changed.

13.6.4 Detection Aids

In conformance to the requirements of 10 CFR 73.55(e), the applicant has installed intrusion detection systems at the PA barrier, at entrances to vital areas, and at all emergency exits. Alarms from the intrusion detection system annunciate within the continuously manned CAS located in the PA, and within an SAS also located in the PA. The CAS is constructed so that the walls, floors, ceilings, and doors are bullet resistant. The alarm stations are located and designed in such a manner that a single act cannot interdict the capability of calling for assistance or responding to alarms. The CAS has no other functions or duties that would interfere with its alarm response function.

The intrusion detection system transmission lines and associated alarms annunciation hardware are line-supervised and tamper-indicating. Alarm annunciators indicate the type of alarm and its location when activated. An automatic indication of when the alarm system is on standby power is provided in the central alarm station.

13.6.5 Communications

As required in 10 CFR 73.55(f), the applicant has provided for the capability of continuous communications between CAS and SAS operators and the MSFs through the use of a conventional telephone system and a security radio system. In addition, direct communication with the local law enforcement authorities is maintained through the use of a conventional telephone system and a two-way FM radio link.

All non-portable communication equipment has backup power from diesel generators.

13.6.6 Test and Maintenance Requirements

In complying with the requirements of 10 CFR 73.55(g), the applicant has set up a program for testing and maintaining all intrusion alarms, emergency alarms, communication equipment, physical barriers, and other security-related devices or equipment. Equipment or devices that fail to meet the design performance criteria or have otherwise failed to operate will be compensated for by appropriate compensatory measures as defined in the Watts Bar Nuclear Plant Physical Security Plan and in site procedures. The compensatory measures defined in these plans will ensure that the effectiveness of the security system is not reduced by failures or other contingencies affecting the operation of the security-related equipment or structures.

Intrusion detection systems are tested for proper performance at the beginning and end of any period in which they are used for security. Such testing will be conducted at least once every seven days.

Systems for onsite communications are tested at the beginning of each security shift. Offsite communications are tested at least once each day.

The Manager, Quality Assurance (QA) directs and controls audits of the security program. This position is independent of site security management and supervision. The audits, focusing on the effectiveness of the physical protection provided by the onsite security organization in implementing the approved security program plans, include, but are not limited to, a review of the security procedures and practices, system testing and maintenance programs, and local law enforcement assistance agreements. The results of the audit are documented and reported to the site Vice President and the protective services manager for review and necessary action.

13.6.7 Response Requirements

In compliance with the requirements of 10 CFR 73.55(h), the applicant will have armed responders immediately available for response duties on all shifts consistent with the requirements of the regulations. In addition, the applicant has established and documented liaison with local law enforcement authorities to gain additional response support in the event of security events.

The applicant's safeguards contingency plan for dealing with thefts, threats, and radiological sabotage events conforms to the requirements of 10 CFR Part 73, Appendix C. The plan identifies appropriate security events which could initiate a radiological sabotage event and identifies the applicant's pre-planning, response resources, safeguards contingency participants and coordination activities for each identified event. Through this plan, upon the detection of an abnormal presence or abnormal activities within the protected or vital areas, response activities using the available resources would be initiated. The response activities and objectives include the neutralization of the existing threat by requiring the respondents to use force sufficient to counter the force directed at them, including the use of deadly force when they have a reasonable belief that it is necessary in self defense or in the defense of others.

13.6.8 Personnel Reliability

In conforming to the requirements of 10 CFR 73.55(a) to protect against the design-basis threat as stated in 10 CFR 73.1(a)(1)(ii), the applicant has provided for an employee screening program. Personnel who successfully complete the personnel reliability program may be granted unescorted access to protected and vital areas at the Watts Bar site. All other personnel requiring access to the site are escorted by persons authorized and trained for escort duties and who have successfully completed the employee screening program.

The personnel reliability program conforms to all elements of Regulatory Guide 5.66 (June 1991), which satisfies the requirements of 10 CFR 73.56.

13.6.9 Land Vehicle Bomb Control Program

In accordance with the provisions of 10 CFR 73.5, "Specific Exemptions," the applicant requested an exemption from the schedule portion of 10 CFR 73.55(c)(10) that requires a license applicant whose application was submitted before August 31, 1994, to incorporate a land vehicle bomb control program into the site physical security plan and implement it by the date of receipt of the operating license. Since Watts Bar Unit 1 will seek to obtain an operating license ahead of the schedule by which operating power reactors are required to fully implement the vehicle control measures, the applicant requested, by letter dated November 30, 1994, that Watts Bar be granted the same implementation period (February 27, 1996) provided to operating reactor licensees to implement the land vehicle bomb control program. By letter dated February 27, 1995, the applicant submitted a summary description of the proposed vehicle control measures, as required by 10 CFR 73.55(c)(7) and (8).

The Commission extended the implementation schedule for operating plants to 18 months from the effective date of the rule, given that it involves a new program for power reactor sites, some procurement problems may arise, and scheduling problems may occur. Under the present rule and current licensing schedule, the applicant would be required to implement the rule (in approximately 8 to 10 months depending on the actual date of license issuance) ahead of operating power reactors.

Watts Bar is in the final stages of closing out numerous corrective action programs and completing final construction and testing activities which involve the dedication of considerable material and manpower resources. On the basis of these considerations and because Watts Bar has in place operating reactor interim measures described in Generic Letter 89-07, "Power Reactor Safeguards Contingency Planning for Surface Vehicle Bombs," granting to Watts Bar the same amount of time to implement the land vehicle bomb control program as operating power reactors will not endanger life or property or the common defense and security. This exemption will be incorporated in the operating license.

15 ACCIDENT ANALYSIS

15.3 Limiting Accidents

15.3.1 Loss-of-Coolant Accidents

In SSER 12, the staff evaluated a reanalysis of the small-break loss-of-coolant accident (SBLOCA), and accepted the revised calculated peak cladding temperature as 2089 °F (1143 °C). By letter dated February 16, 1995, the applicant submitted another reanalyses for SBLOCA and other postulated events. The applicant has included the changes in FSAR Amendment 89. The staff tracked its efforts by TAC M89427.

In a letter dated April 23, 1994, the applicant committed to reanalyze the SBLOCA event before loading fuel in Unit 1. Westinghouse Electric Corporation performed this analysis in June 1994. The SBLOCA reanalysis incorporated various emergency core cooling system (ECCS) evaluation model changes that were described in the letter of April 23, 1994, and which, cumulatively, exceeded the threshold for a "significant" change in the model as defined by 10 CFR 50.46.

In addition to incorporating the ECCS model changes, the applicant's latest SBLOCA reanalysis relaxed the setpoint tolerance for the main steam safety valves and the pressurizer safety valves from plus or minus 1 percent to plus or minus 3 percent.

In FSAR Table 15.1-3, "Trip Points and Time Delays to Trip Assumed in Accident Analyses," the low pressurizer pressure limiting trip point was changed from 1845 psig to 1910 psig. The new, higher trip point still provides margin between the analysis and the safety limit.

The applicant stated that the reanalysis used ECCS flow rates that were based on a detailed technical evaluation of as-installed ECCS pump flow characteristics and plant-specific piping arrangements, as confirmed by measured pump performance.

The SBLOCA was reanalyzed for break sizes of 3, 4, and 6 inches. In FSAR Chapter 15.3.1.2, the applicant listed the use of the following approved codes for the reanalysis: NOTRUMP and LOCTA-IV. The applicant also indicated that the analysis was performed with the approved Westinghouse evaluation model. The following results were reported in FSAR Section 15.3.1.4:

(1) The calculated peak fuel element cladding temperature of 1452 °F (788.89 °C) provides margin to the limit of 2200 °F (1204.44 °C), based on a Fq value of 2.40.

(2) The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1 percent of the total amount of Zircaloy in the reactor.

(3) The cladding temperature transient is terminated at a time when the core geometry is still amenable to cooling. The oxidation limit of 17 percent of the cladding thickness is not exceeded during or after quenching.

(4) The temperature is reduced, and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

These conclusions for the SBLOCA analyses are acceptable as approved codes (listed in Table 15.1-2 of the FSAR) were used in the analyses and the results of the SBLOCA reanalysis demonstrated that Watts Bar satisfies the ECCS acceptance criteria of 10 CFR 50.46. The reanalysis determined that a 4-inch break size is limiting for an SBLOCA event, and the resulting peak cladding temperature for this limiting break is 1452 °F.

In addition to the SBLOCA reanalysis, the relaxation of the setpoint tolerance for the main steam safety valves and the pressurizer safety valves affected the four FSAR Chapter 15 accident analyses:

- loss of electrical load and/or turbine trip (FSAR Section 15.2)

- loss of normal feedwater (FSAR Section 15.2.8)

- major rupture of a main feedwater pipe (FSAR Section 15.4)

- single reactor coolant pump locked rotor (FSAR Section 15.4)

Each of these four postulated accidents was reanalyzed by Westinghouse for relaxation of the setpoint tolerance for the main steam safety valves and the pressurizer safety valves from plus or minus 1 percent to plus or minus 3 percent. The results were found to be acceptable and did not change the previously acceptable results. Approved codes (listed in Table 15.2 of the FSAR) were used. The results of the reanalyses of the accident transients were small changes from the previously acceptable analyses and did not exceed any safety limits, including those for departure from nucleate boiling (DNB) and pressure.

Changes were also made in FSAR Chapter 6 to note the change in the approximate time required before ECCS switchover from cold-leg recirculation to hot-leg recirculation (i.e., 12 hours instead of 15 hours). This change in switchover timing resulted from using the revised ECCS flow rates described above for the SBLOCA reanalysis.

15.4 Radiological Consequences of Accidents

The staff revised short-term atmospheric relative concentration values (χ/Qs) as shown in Section 2.3.4 of this supplement. This revision resulted in higher calculated offsite radiological consequences than those in Table 15.1 of SSER 4. The revised radiological consequences are in revised Table 15.1 of this supplement. The χ/Q values in Tables 15.2 through 15.6 of the SER are also corrected with these revised values; the revised values are indicated with vertical bars. All other estimations in these tables were unchanged from the SER (for Tables 15.2 through 15.5) and from SSER 4 (for Table 15.6).

On the basis of its review of the FSAR up to Amendment 88), the staff finds that the distances to the exclusion area and low population zone boundaries for the Watts Bar site are still sufficient to give reasonable assurance that the calculated radiological consequences of a postulated design-basis accident do not exceed the dose guidelines in 10 CFR Part 100. Therefore, the staff's evaluation and conclusions issued in the Watts Bar Safety Evaluation Report (NUREG-0847) in June 1982 remain valid.

The staff tracked its efforts by TACs M89446 and M89447.

Table 15.1 Radiological consequences of design-basis accidents (revised)

Postulated accident	Exclusion area boundary, rems (sievert) Thyroid	Exclusion area boundary, rems (sievert) Whole body	Low population zone, rems (sievert) Thyroid	Low population zone, rems (sievert) Whole body
Loss-of-coolant accident				
Containment leakage				
0-2 hr	4.7 (0.05)	2.8 (0.03)	1.6 (0.02)	0.9 (0.009)
2-8 hr	-	-	1.1 (0.01)	0.4 (0.004)
8-24 hr	-	-	0.8 (0.01)	0.3 (0.003)
24-96 hr	-	-	3.8 (0.04)	0.3 (0.003)
96-720 hr			3.1 (0.03)	0.1 (0.001)
Total containment leakage	4.7 (0.05)	2.8 (0.03)	10.4 (0.11)	2.0 (0.01)
ECCS component leakage	8.3 (0.08)	0.01 (0.0001)	33.0 (0.33)	0.04 (0.001)
TOTAL LOCA	13.0 (0.13)	2.8 (0.03)	43.4 (0.44)	2.0 (0.01)
Main steamline break outside secondary containment				
Long-term operation case (Case 2)	11.0 (0.11)	<0.1	11.2 (0.1)	<0.1
Short-term operation case (Case 3)	13.6 (0.14)	<0.1	13.7 (0.14)	<0.1
Control rod ejection accident				
In containment leakage pathway	53.5 (0.54)	<0.9	84.0 (0.8)	0.4 (0.004)
In secondary system release pathway	18.3 (0.18)	<1.0	6.0 (0.06)	<0.1
Fuel-handling accident				
In fuel-handling area	1.5 (0.02)	<1.0	0.2 (0.002)	<0.1
Inside primary containment	39 (0.39)	0.6 (0.01)	2.8 (0.028)	<0.1
Small-line failures outside containment	26.0 (0.26)	<0.1	4.6 (0.046)	<0.1
Steam generator tube rupture				
(1) (DEI-131 at 60 μCi/gram)	111.5 (1.12)	<0.1	24.0 (0.24)	<0.1
(2) (DEI-131 at 1 μCi/gram)	19.9 (0.2)	<0.1	6.0 (0.06)	<0.1

Note: DEI = dose equivalent iodine-131.

Table 15.2 Assumptions used for calculating the radiological consequences following a postulated loss-of-coolant accident

Item		Assumption
Power level (MWt)		3592
Operating time (yr)		3
Fractions of core inventory available for leakage (%)		
Iodines		25
Noble gases		100
Initial iodine composition in containment (%)		
Elemental		91
Organic		4
Particulate		5
Primary containment volumes (ft^3)		
Upper compartment		6.51E5
Lower compartment (including ice condenser)		5.85E5
Shield building annulus volume (ft^3)		3.75E5
Mixing fraction in annulus (%)		50
Annulus ventilation flow distribution (ft^3)		
Time step	Recirculation flow (ft^3/min)	Exhaust flow (ft^3/min)
0-30 sec	0	0
30-180 sec	770	3230
180-360 sec	1700	2300
360-600 sec	2500	1500
600-1200 sec	3340	660
1200-1800 sec	3810	190
1800 sec-30 day	3900	100
Filter efficiencies (%)		
Elemental iodine		99
Organic iodine		95
Particulate iodine		99
Ice condenser removal efficiency (%)		
Elemental iodine		30
Flowrate through ice condenser (ft^3)		40,000
Period of ice condenser effectiveness (min)		10-60
Primary containment leak rates (%)		
0-24 hr		0.25
24 hr-30 days		0.125
Bypass leakage fraction (%)		0
Minimum exclusion area boundary distance (m)		1250
Low population zone distance (m)		4828
Atmospheric diffusion (χ/Q) values (sec/m^3)		
0-2 hr at 1250 m		5.5E-4
0-8 hr at 4828 m		1.0E-4
8-24 hr at 4828 m		6.0E-5
1-4 days at 4828 m		2.6E-5
4-30 days at 4828 m		8.0E-6

Table 15.3 Assumptions used for calculating the radiological consequences following a postulated main steamline break accident outside containment

Item	Assumption
Power level (MWt)	3592
Preaccident dose-equivalent I-131 in primary coolant (Case 2) (μCi/g)	1.0
Preaccident dose-equivalent I-131 in primary coolant (Case 3) (μCi/g)	60.0
Primary-to-secondary leak rate, as limited by Technical Specifications (gpm)	1
Amount of the 1-gpm leak that occurs in the affected steam generator	All
Amount of the iodine transported to the shell side of the steam generator by the leakage lost to the environment without decay	All
Minimum exclusion area boundary distance (m)	1250
Low population zone distance (m)	4828
Iodine release rate from fuel increases by a factor of 500 as a result of the accident (Case 2) χ/Q *values* (sec/m^3)	
0-2 hr at 1250 m	5.5E-4
0-8 hr at 4828 m	1.0E-4

Table 15.4 Assumptions used for calculating the radiological consequences following a postulated steam generator tube rupture accident

Item	Assumption
Power level (MWt)	3592 MWt
Preaccident dose equivalent I-131 in primary coolant (μCi/gm) (two cases analyzed)	1 60
Initial secondary coolant activity (μCi/g DEI-131)	0.1
Primary-to-secondary leak rate (to unaffected steam generator)	1
Isolation of affected steam generator (min)	30
Minimum exclusion area boundary distance (m)	1250
Low population zone distance (m)	4828
Iodine release rate from fuel increases by a factor of 500 at reactor trip for iodine spiking case χ/Q values (sec/m^3)	
0-2 hr at 1250 m	5.5E-4
0-8 hr at 4828 m	1.0E-4

Table 15.5 Assumptions used for calculating the radiological consequences following a postulated control rod ejection accident

Item	Assumption
Power level (MWt)	3592
Volume of primary coolant (ft^3)	11,790
Primary-to-secondary leak rate as limited by Technical Specifications (gpm)	1
10% of the fuel rods experience cladding failure, releasing all their gap radioactivity. The released activity is mixed immediately with the primary coolant	-
0.25% of the fuel rods experience fuel melting and all released activity is mixed immediately with the primary coolant	-
A fraction of the iodine transported to the shell side of steam generators is lost to the environment	-
10% of the iodine transported to and mixed with the secondary coolant is lost during the course of the accident	-
Primary system depressurized in about 1000 secs, terminating primary-to-secondary leak	-
For the containment pathway, 50% of the iodine released into the containment is plated out instantaneously	-
Primary containment leak rate per day (containment leakage pathway) (%)	0.25
With loss of offsite power, the steam releases from the loss of ac power analysis is used to supplement the primary to secondary leakage values	-
Minimum exclusion area boundary distance (m)	1250
Low population zone distance (m)	4828
Iodine concentration in the secondary coolant (μCi/g DEI-131)	0.1
χ/Q values (sec/m^3)	
0-2 hr at 1250 m	5.5E-4
0-8 hr at 4828 m	1.0E-4

Table 15.6 Assumptions used for calculating the radiological consequences following a postulated fuel-handling accident

Item	Assumption
Power level (MWt)	3592
Number of fuel rods damaged	264
Total number of fuel rods in core	50,952
Radial peaking factor of damaged rods	1.65
Shutdown time (hr)	100
Inventory released from damaged rods (iodines and noble gases) (%)	10
Pool decontamination factors	
Iodines	100
Noble gases	1
Iodine fractions released from pool (%)	
Elemental	75
Organic	25
Iodine removal efficiencies for ABGTS (spent fuel pool area) (%)	
Elemental	99
Organic	99
Particulate	99
Iodine removal efficiencies for reactor building purge system (%)	
Elemental	90
Organic	30
Particulate	90
Minimum exclusion area boundary distance (m)	1250
Low population zone distance (m)	4828
Atmospheric diffusion (χ/Q) values (sec/m^3)	
0-2 hr at 1250 m	5.5E-4
0-8 hr at 4828 m	1.0E-4

17 QUALITY ASSURANCE

In SSERs 10 and 13, the staff incorporated by reference its evaluation of the revisions TVA submitted after SSER 5 was published. Since SSER 13 was issued, TVA has submitted the following additional revisions:

(1) Letter, Bruce S. Schofield (TVA) to NRC, dated January 7, 1994 (Revision 4 of the program), NRC acceptance per 10 CFR 50.54(a)(3)(iv).

(2) Letter, Bruce S. Schofield (TVA) to NRC, dated January 12, 1994, providing supplemental/revised information on Revision 4.

(3) Letter, Bruce S. Schofield (TVA) to NRC, dated February 25, 1994, providing supplemental/revised information on Revision 4.

(4) Letter, M. O. Sanford (TVA) to NRC, dated March 14, 1995 (Revision 5 of the program)--NRC review and acceptance in letter, Albert F. Gibson (NRC) to Oliver D. Kingsley (TVA), April 14, 1995.

The staff tracked its efforts by TAC M76972.

18 HUMAN FACTORS ENGINEERING

18.1 Detailed Control Room Design Review

In the SER, SSER 5, and SSER 6, the staff reported its evaluation of the Watts Bar Unit 1 detailed control room design review (DCRDR). In SSER 6, the staff also delineated findings of an onsite audit it conducted in 1990. The staff concluded in SSER 6 that the DCRDR conducted at Watts Bar Unit 1 satisfied the DCRDR programmatic requirements of Supplement 1 to NUREG-0737. The staff stated that before startup, it planned to confirm by audit that the control room improvements that TVA had committed to complete before fuel loading were completely and properly implemented.

In addition, the DCRDR was identified as a special program under TVA's Watts Bar Nuclear Performance Plan, which the staff evaluated in NUREG-1232, Volume 4, "Safety Evaluation Report on Tennessee Valley Authority, Watts Bar Nuclear Plant Nuclear Performance Plan." This evaluation picks up where NUREG-1232, Volume 4, leaves off.

During March 28-30, 1995, the staff conducted a final onsite audit of the DCRDR at Watts Bar Unit 1. In the following sections, the staff reports its findings and conclusions about the Watts Bar Unit 1 DCRDR. The staff tracked its efforts by TAC M63655.

18.1.2 Evaluation

The March 28-30, 1995, onsite audit methodology consisted of (1) evaluating corrective actions for safety-significant human engineering discrepancies (HEDs), (2) assessing the overall control room design using applicable human factors engineering guidelines and the guidance in NUREG-0800, (3) interviewing operations and quality assurance personnel, and (4) discussing with the applicant, its DCRDR submittal of March 7, 1995.

The applicant defines HED categories as follows:

- Category I - errors resulting from HEDs directly challenge or cause a loss of a critical safety function (CSF)
- Category II - errors resulting from HEDs reduce or cause the loss of resource(s) needed to maintain a CSF
- Category III - errors resulting from HEDs adversely affect normal operation or have the potential to affect CSF resource(s)
- Category IV - errors resulting from HEDs have no safety-significant effect on plant operations

The applicant used three progressive levels of evaluation to determine category (i.e., likelihood that the HED will cause an error; results of the error, if uncorrected, on operating performance degradation; and effect of the error on maintenance and/or restoration of a CSF) plus a separate evaluation of safety significance.

During the March 28-30, 1995, onsite audit, the staff evaluated the corrective actions for 35 HEDs categorized as safety significant that were either completed (29 of 35) or scheduled to be completed (6 of 35). The applicant stated in the summary report dated October 2, 1987, as supplemented by letters dated February 23, 1990; March 28, 1990; and May 16, 1995, that the corrective actions for HEDs 19, 93, 119, 151, and 157 would be fully implemented before fuel loading, and that HED 15 would be fully implemented 120 days after fuel loading. Table 18.1 summarizes the six safety-significant HEDs (numbers 15, 19, 93, 119, 151, and 157) for which corrective actions had not been fully implemented at the time of the March 28-30, 1995, onsite audit.

Table 18.1 Six safety-significant HEDs for which corrective actions had not been fully implemented by the March 28-30, 1995, audit

HED	Description/Corrective Action	Open	Closed
15 Cat. 2	The acoustic level (noise) in the main control room (MCR) is too high and interferes with operator communication. Ongoing construction of Unit 2 interferes with operations at Unit 1. Some alarm horns are too loud, specifically those on the radiation monitors and the fire alarm panel. Vacuum cleaners used for cleaning the MCR are too noisy. <u>Corrective actions</u>: The radiation monitor was relocated to an area behind the MCR panels muffling the sound from the radiation monitors. The fire alarm console and its associated alarm system has been replaced. The MCR annunciator system has been replaced with a new system. Carpet has been installed in the MCR. MCR and auxiliary control room alarm horn acoustic levels relative to ambient noise levels will be adjusted. Quieter vacuum cleaners will be used. The use of vacuum cleaners will be limited. In accordance with Site Standard Practice 12.01, the assistant shift operating supervisor will terminate any activity in the MCR that interferes with operations. Unit 2 construction activities have been terminated. The NRC will review the results of the sound/noise survey, discussed below, which will be completed by TVA 120 days after fuel load, in relation to NUREG-0700 Guideline 6.1.5.5, "Auditory Environment."	X	

Table 18.1 (Continued)

HED	Description/Corrective Action	Open	Closed
19 Cat. 3	The storage of spare parts and expendable items, especially spare bulbs and fuses, is inadequate in the MCR. Supplies are not inventoried regularly. A cabinet is not available for spare parts and supplies. Corrective actions: Cabinets have been designated for spare parts and supplies in the MCR. Supplies have been categorized and organized so inventories can be readily checked. An operations procedure is being developed for checking and reordering supplies. Plant Administrative Instruction 10.10 has been issued to address control of fuses. The corrective actions are essentially complete, except for the administrative actions remaining to close this HED. The NRC will review the administrative aspects of closing this HED, per Administrative Instruction 1.89, "Closing Out Control Room Human Engineering Concerns and Discrepancies," and the completion of the operations procedure for checking and reordering supplies.	X	
93 Cat. 3	Recorder 1-RR-90-1 has problems, including those pertaining to scale compatibility and transformation factors and readability. Corrective actions: The analog recorder will be replaced with a digital recorder that has better human factors characteristics in relation to the identified problems. The NRC will verify that the digital recorder was installed.	X	
119 Cat. 3	Multipoint recorders are slow and do not show clearly which point is printing. Operators would like digital recorders. Recorders 1-ZR-412 and 0-RR-90-12A will be replaced with a digital-type recorder. The NRC will verify that the digital recorders were installed.	X	

Table 18.1 (Continued)

HED	Description/Corrective Action	Open	Closed
151 Cat. 1	Eberline System problems (e.g., problems related to reliability, accuracy, information inputs and processing capabilities, and documentation/procedure adequacy) detract from the usability of the system. <u>Corrective actions</u>: The Eberline System hardware will be upgraded to improve operability. Some of the input monitors will be replaced and will not be included as part of the Eberline System operator interface. A data link will be provided to the emergency response facilities data system (ERFDS). The ERFDS will be used as the primary operator interface. The NRC will verify that special displays for post-accident monitoring are added to the ERFDS and that tags have been placed on the Eberline System directing operators to the ERFDS as the primary interface.	X	
157 Cat. 2	Improvements are needed in the layout of the condensate and feedwater system components on panels M-2, M-3, and M-4. <u>Corrective actions</u>: Improvements in the panel layouts included the rearrangement of components, application of demarcation for groups and subgroups, extensive use of hierarchial labeling, improved component nameplates, and the addition of operator aid placards. The guardrail around the front edge of the benchboard panels has been replaced, and the panels have been repainted. Note: The panel arrangements derived from this study have been installed on the simulator for about three years. Feedback from the Watts Bar operator training section and operators has been positive. The NRC will verify that design change number W-30427 in relation to the layout of the auxiliary feedwater pump control switches was completed and that the nameplates associated with HED number 9 were installed.	X	

During the onsite audit, the applicant reiterated its previous commitment (September 6, 1984, as supplemented by a letter dated May 16, 1995) to conduct, and resolve discrepancies for a final sound/noise survey based on applicable NUREG-0700 guidelines (e.g., background noise levels should not exceed 65 decibels) that would be conducted 120 days after fuel loading. A final sound/noise survey was planned after fuel loading in order to take into account such contributors as noise from the main turbine deck. The verification of corrective actions for the six subject HEDs, including the acceptability of the sound/noise survey results, will be reviewed by Region II personnel. The staff finds that the applicant's proposed corrective actions, commitments, and schedules pertaining to HED numbers 15, 19, 93, 119, 151, and 157 are satisfactory. During the March 28-30, 1995, onsite audit, the staff evaluated the completed corrective actions for the 29 safety-significant HEDs summarized in Table 18.2 below. Operator interviews indicated that operators were positive concerning the Watts Bar Unit 1 DCRDR design (e.g., panel layouts). The staff finds that the applicant's corrective actions and justifications for the HEDs detailed in Table 18.2 are satisfactory.

Table 18.2 Twenty-nine safety-significant HEDs for which corrective actions have been completed

HED	Description/Corrective Action	Open	Closed
8 Cat. 2	Replacing fuses on the 125-V vital battery boards can result in electrical shock as the individual must place hand and head between live fuse arrays to read labels and change fuses. <u>Corrective actions:</u> Permanent caution signs have been placed on the door along with a fuse locator diagram.		X
43 Cat. 3	There are numerous multiple input annunciators in the control room. <u>Corrective actions:</u> Reflash will be provided to the appropriate windows.		X
56 Cat. 3	There is no alarm for high seal water flow to the reactor coolant pumps. <u>Corrective actions:</u> This flow is set under specified conditions and checked to comply with technical specifications by use of surveillance instructions. No corrective action is planned.		X
62 Cat. 3	Several annunciator alarms for shared equipment are not duplicated in the Unit 2 control room. <u>Corrective actions:</u> No corrective actions will be taken in the Unit 2 control room before Unit 1 fuel loading. This item will be addressed in the Unit 2 control room design review program.		X

Table 18.2 (Continued)

HED	Description/Corrective Action	Open	Closed
76 Cat. 4	Trip signal is needed for the centrifugal charging pumps (CCPs) when the residual heat removal (RHR) pumps are not working (after station blackout, transfer to containment sump, safety injection (SI) reset, loss-of-coolant accident). The RHR pumps do not automatically load as the CCPs do. <u>Corrective actions:</u> An automatic CCP trip could be detrimental to SI system reliability. This concern is covered during simulator training, requalification and AOI [abnormal operating instruction] and EI [electrical and instrumentation] monthly review. Procedures address this concern.		X
82 Cat. 4	Setpoint adjustments on controllers can be changed accidently by brushing up against the setpoint controls. <u>Corrective actions:</u> No correction is needed. In the horseshoe, guardrails are provided. In addition, red carpet warns personnel that they are too close to the control panel.		X
87 Cat. 4	The process radiation monitoring system functional test selector switches could be accidently activated. They are so low on panel M-12, they could be kicked. Their pulled position makes this more likely. <u>Corrective actions:</u> The switch function in the pull position is to disable automatic functions for the selected monitor. This position is used for testing purposes. TVA's evaluation team concluded it is not credible for these switches to be kicked or bumped into the pull position.		X
91 Cat. 3	Scale/mathematical conversion is required to relate controller setpoint to associated parameter displays. <u>Corrective actions:</u> A placard with required setpoints and corresponding controller percentages has been placed next to the controllers that have or require a specific setpoint that needs to be controlled.		X

Table 18.2 (Continued)

HED	Description/Corrective Action	Open	Closed
92 Cat. 3	Indicators or status lights are needed to show which station air compressor is running and controls are needed to start/stop each of the station air compressors. <u>Corrective actions:</u> System is designed to run automatically. Indications have been added to main control room panel M-15 to provide status of pressure in the A and B auxiliary control air receiver tanks.		X
99 Cat. 1	There is no narrow range containment pressure indication in the horseshoe. <u>Corrective actions:</u> A pressure differential indicator recorder has been added to panel M-6.		X
103 Cat. 4	General Electric controllers have moving-scale, fixed-pointer indicators. <u>Corrective actions:</u> The operators are familiar with this type of controller used throughout the industry. It presents no problem or point of confusion in operation. No corrective action is planned. This was rated Category 4, not safety significant.		X
107 Cat. 3	The boron injection tank (BIT) flow indicator has a square root scale. It is difficult to verify flow through the BIT when reactor coolant system pressure is maintained. Emergency instructions require BIT flow to be verified. Operators are not able to read the lower portion of the scale. <u>Corrective actions:</u> The square root scale has been changed to a linear scale with a range of 0 to 1000 gpm.		X
110 Cat. 4	Incore thermocouple indicator ITI-94-A20 has a moving-scale fixed-pointer meter which is not recommended by NUREG-0700 Guideline 6.5.2.5. The scale is also labeled upside down. <u>Corrective actions:</u> The installation of the reactor vessel level instrumentation system with incore thermocouple readout capabilities resolves this HED. The existing indicators were removed.		X

Table 18.2 (Continued)

HED	Description/Corrective Action	Open	Closed
132 Cat. 3	The failure mode for the percent flux differential indicators is not apparent. They fail midscale at the zero graduation mark. <u>Corrective actions:</u> Four independent channels monitor flux differential. A channel reading that differs from the others is easily detected.		X
153 Cat. 1	The pyrotronics system has individual fire detector alarm lights on each of the MCR panels. The power supply for these lights does not have capacity for more than one light. <u>Corrective actions:</u> The remote lights have been removed from the control room panels.		X
159 Cat. 1	The feedwater indication reset pushbuttons are presently located in the auxiliary instrument room. They should be moved to the MCR. <u>Corrective actions:</u> These pushbuttons have been moved to panel M-3 in the MCR.		X
160 Cat. 3	Improvements are needed in the layout of components in panels M-4 and M-5. <u>Corrective actions:</u> Layout has been improved. Changes have been defined with operations personnel input.		X
162 Cat. 3	There is no status indication for the cold overpressurization mitigation system. <u>Corrective actions:</u> Status lights have been installed on panels HS-68-334AD and HS-68-340AD to indicate when the system is armed.		X
163 Cat. 2	Panel layout for the emergency core cooling system needs to be changed to improve operability. <u>Corrective actions:</u> Components related to the emergency core cooling system on panel M-6 have been relocated or removed to achieve functional grouping of controls and displays. Labeling and demarcation have been improved.		X
167 Cat. 2	Improvements are needed in the layout of panel M-9. <u>Corrective actions:</u> A better layout design using functional grouping and mimics has been implemented.		X

Table 18.2 (Continued)

HED	Description/Corrective Action	Open	Closed
176 Cat. 2	Pressure is not indicated on panel M-27B for the annulus vacuum control portion of the emergency gas treatment system. This indication is needed to support an existing technical specification requirement. <u>Corrective actions:</u> Differential pressure indication for the annulus vacuum fan has been installed on panel M-27B.		X
181 Cat. 3	Panel L-10 components are not arranged in logical order and are not in the same relative positions as similar components in the MCR. <u>Corrective actions:</u> Panel layout has been rearranged in a more logical and consistent order.		X
192 Cat. 3	The auxiliary feedwater level controllers and the level program can be changed, and the level control handswitches can be disabled from the auxiliary control room (ACR). <u>Corrective actions:</u> A caution label has been added on ACR controllers LIC-3-148B, -156B, -164B, and -171B, requiring that the MCR operator be notified before setpoints are adjusted.		X
193 Cat. 3	It is difficult to distinguish an unilluminated rod bottom light. <u>Corrective actions:</u> The area around each light has been changed to white to increase the contrast.		X
199 Cat. 4	Certain valves could be opened when phase A isolation has not been reset. <u>Corrective actions:</u> An evaluation of air-operated valves (using schematic drawings) showed that it would take a deliberate act for the operator to open one of these valves. Once the switch was released, the valve would close.		X
200 Cat. 1	There are no phase B isolation status lights in the MCR. <u>Corrective actions:</u> The monitor light panels displaying the phase B isolation status lights have been replaced with a master isolation status panel and containment isolation status panel.		X

Table 18.2 (Continued)

HED	Description/Corrective Action	Open	Closed
202 Cat. 1	Modifications do not contain written functional descriptions. As a result, operations training and procedures personnel are not fully aware of the implementation of the modification. <u>Corrective actions:</u> Modification packages have been changed to include a functional statement identifying any operational impacts of the change.		X
209 Cat. 3	Main feedwater bypass flow is not indicated in the MCR. <u>Corrective actions:</u> Bypass flow is now indicated on panel 1-M-4 for each of the steam generator loops.		X
218 Cat. 4	Excess letdown flow is not indicated in the MCR. <u>Corrective actions:</u> No corrective action is planned. Excess letdown is designed to let down the amount of reactor coolant pump seal supply flow that goes into the RCS (about 20 gpm). Since temperature limits the amount of flow through the heat exchanger, a flow indicator is not needed.		X

18.1.3 Conclusions

The staff concludes that the DCRDR program implemented at Watts Bar Unit 1 conforms to the DCRDR requirements of Supplement 1 to NUREG-0737. Therefore, the proposed license condition (see the SER) is no longer needed. In addition, the staff considers that the special DCRDR program has been effectively implemented.

18.2 <u>Safety Parameter Display System</u>

In the SER, SSER 5, and SSER 6, the staff reported its evaluation of the Watts Bar Unit 1 safety parameter display system (SPDS). In SSER 6, the staff also delineated findings of an onsite audit that the staff conducted in 1990. The staff concluded in SSER 6 that the Watts Bar Unit 1 SPDS was in the design/development phase and should conform to the SPDS requirements of Supplement 1 to NUREG-0737. In SSER 6, the staff stated that it planned to confirm by audit that the SPDS upgrades that the applicant had committed to complete before fuel loading were completely and properly implemented.

The following sections delineate the staff's findings and conclusions during an onsite audit at Watts Bar Unit 1, March 28-30, 1995. The staff tracked its efforts by TAC M73723.

18.2.2 Evaluation

The March 28-39, 1995, onsite audit methodology consisted of (1) evaluating the SPDS implementation using the guidance in NUREG-1342, (2) assessing the overall SPDS design using applicable human factors engineering guidelines and the guidance in NUREG-0800, (3) interviewing operations and quality assurance personnel, and (4) discussing with the applicant, its recent SPDS submittal of February 22, 1995.

In its letter of February 22, 1995, the applicant stated with regard to SPDS upgrades:

> The SPDS hardware and software upgrades include replacement of the analog multiplexers, use of a Digital Equipment Corporation (DEC) 1184 computer for processing SPDS inputs and algorithms, use of a DEC VAX for the man-machine interface, replacement of the cathode ray tube (CRT) monitors and keyboards with 14 inch touch-sensitive CRT's and a keyboard for each console, and use of new software for the man-machine interface. The software is based on the Science Applications International Corporation SPDS system that TVA implemented at Browns Ferry Nuclear Plant. The upgraded SPDS offers enhancements over the previous system including improved user selection displays using touch-sensitive CRT screens or using special keys on a keyboard. Other improvements include faster response [time] to user requests and an improved control room layout.

In its letter of February 22, 1995, the applicant also clarified previous SPDS commitments as follows:

- Critical safety function status indication is now located in the upper right-hand corner of all displays that can appear on the SPDS consoles.
- Critical safety function status boxes are provided that change color in accordance with the emergency operating instruction status tree alarms.
- A lighter blue for poor/bad data is used instead of a dark blue to provide a greater contrast with the display background.
- The SPDS status box is normally green (no alarm) and changes to a solid color in accordance with the level of the status tree alarm (red - highest, orange - middle, yellow - lowest).

The staff observed the corrective actions associated with these commitments and others discussed in the applicant's letter, and finds that the SPDS implementation with regard to the subject commitments is satisfactory.

The onsite audit consisted of an evaluation of the human factors aspects of the SPDS in the Unit 1 control room. The staff identified two examples of inconsistent use of nomenclature. One example was the use of "CNTMT" on one display screen and "CNMT" on another display screen. Another example was the use of "F°" and "°F" on the same display screen and "DEGF" on a different display screen. These examples were inconsistent with NUREG-0700 Guideline 6.5.1.4.e, which states: "CONSISTENCY WITH PROCEDURES - The printed message should use the same terms as the procedures in display identification, parameter identification, and units displayed." The applicant acknowledged its intent to correct these inconsistencies through an existing commitment to verify display formats against operating procedures that is tracked via NRC commitment Item Number 830138016 in the applicant's "Tracking and Reporting of Open Items System." The staff notes that this commitment satisfactorily resolves this issue.

In response to the 1990 SPDS onsite audit results, the applicant committed to document its process for ensuring adequate operator review of and input into the SPDS design. In its letter of February 22, 1995, the applicant stated that control room operators had provided input on the SPDS design as part of the SPDS Users Group effort. It also stated that users of the SPDS could document problems and/or request changes to SPDS software (subject to approval) under the process described in Plant Administrative Instruction 1.05, "Software Development and Maintenance for Plant Process Monitoring and Control Computers." Operator interview results indicated that operators liked the current SPDS design (e.g., touch screen capability, response time, and size of monitor). The applicant's response satisfactorily resolves this issue.

The staff discussed with site personnel personnel two SPDS enhancements. One enhancement would allow storage of the SPDS keyboard under the counter of the SPDS monitor. The applicant also indicated that consideration was being given to redesigning the SPDS keyboard to make it smaller, thus making it easier to store under the counter. Another enhancement was a flashing message (i.e., "TIME IS NOT UPDATING") and a blank screen to show that the SPDS was not functional. The current design flashes "TIME IS NOT UPDATING" and at the same time displays the most recent data on the screen when the SPDS is not functional. Since the applicant was still evaluating these enhancements, it had not yet reached the commitment stage.

In the letter of February 22, 1995, the applicant reiterated its commitment to have a "functional SPDS" installed before fuel loading. A "functional SPDS" provisionally satisfies the SPDS requirements of Supplement 1 to NUREG-0737. In its letter of July 11, 1989, the applicant had committed to complete the following activities during the first operating cycle: (1) document SPDS availability, (2) resolve operator comments, and (3) verify SPDS displayed data with main control room indications. After these activities were completed, the applicant further stated that the SPDS would be declared "operational" before restart from the first refueling outage. In addition, it has committed to provide a supplemental response to Generic Letter 89-06, within 2 months after the Unit 1 SPDS is declared "operational," addressing certification of compliance with the SPDS requirements of Supplement 1 to NUREG-0737. The applicant's commitments and schedules are satisfactory.

During the onsite audit, the staff determined that the applicant must complete several activities before it can declare the SPDS "functional." It must close all SPDS-related design changes, issue software documentation, and complete SPDS testing. The SPDS design changes include verifying SPDS setpoints, logic flows, and display formats against operating procedures, control room instrumentation, and system/sensor characteristics. The SPDS software documentation will be issued in accordance with Watts Bar's Site Standard Practice 2.12, "Control and Use of Computer Software." The remaining SPDS testing includes power tests, software verification and validation tests, and point tests (i.e., inputs and outputs). After the remaining SPDS activities are completed and issues that arise are resolved, the applicant has committed to send a letter to the NRC once the SPDS is declared "functional" before fuel loading. TVA's commitments and schedules are satisfactory.

18.2.3 Conclusion

On the basis of the applicant's SPDS commitments and the audit results, the staff concludes that the Watts Bar Unit 1 SPDS has satisfied or will satisfy all of the eight SPDS requirements of Supplement 1 to NUREG-0737.

APPENDIX A

CHRONOLOGY OF RADIOLOGICAL REVIEW OF WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2, OPERATING LICENSE REVIEW

The following is a list of documents; most of them are referenced in this SSER. In no way is this an exhaustive list of all correspondence exchanged between the staff and the applicant during this period. The reader may obtain an exhaustive list through the NRC document control system (NUDOCS), the Public Document Room, or the Local Public Document Room.

NRC Letters and Summaries

September 2, 1994	Letter, S. F. Newberry to O. D. Kingsley (TVA), requesting additional information on severe accident mitigation design alternatives.
September 15, 1994	Summary, P. S. Tam, of August 30, 1994, meeting, on use of Thermo-Lag fire retardant material.
September 16, 1994	Letter, S. F. Newberry to O. D. Kingsley (TVA), requesting additional information on potential environmental impacts.
September 20, 1994	Letter, W. T. Russell to J. Proffitt, discussing public participation in NRC meetings.
September 27, 1994	Letter, S. F. Newberry to O. D. Kingsley (TVA), requesting additional information on severe accident mitigation design alternatives.
October 5, 1994	Letter, P. S. Tam to O. D. Kingsley (TVA), transmitting staff evaluation on individual plant examination for severe accidents.
October 11, 1994	Letter, P. S. Tam to O. D. Kingsley (TVA), requesting additional information on FSAR Chapter 3.
October 11, 1994	Summary, L. A. Dudes, of October 4, 1994, meeting on technical specifications.
October 17, 1994	Letter, S. F. Newberry to O. D. Kingsley (TVA), requesting additional information on severe accident mitigation design alternatives.
October 20, 1994	Letter, P. S. Tam to O. D. Kingsley (TVA), finding Watts Bar's response to Bulletin 90-01 (on fill oil in Rosemount transmitters) acceptable.

October 21, 1994	Letter, P. S. Tam to O. D. Kingsley (TVA), transmitting request for additional information on the applicant's response to Bulletin 88-08, "Thermal Stress in Piping Connected to Reactor Coolant System."
October 24, 1994	Letter, S. F. Newberry to O. D. Kingsley (TVA), requesting additional information on potential environmental impacts.
October 31, 1994	Letter, P. S. Tam to O. D. Kingsley (TVA), transmitting copy of environmental assessment and finding of no significant impact regarding the applicant's request to extend the expiration date of Unit 1 construction permit.
November 1, 1994	Summary, P. S. Tam, of October 19, 1994, management meeting on licensing status.
November 8, 1994	Letter, P. S. Tam to O. D. Kingsley (TVA), transmitting order to extend expiration date of Unit 1 construction permit.
November 9, 1994	Summary, P. S. Tam, of November 4, 1994, meeting on licensing status.
November 16, 1994	Summary, P. S. Tam, of November 3, 1994, meeting on splices on electrical cables.
November 17, 1994	Letter, P. S. Tam to O. D. Kingsley (TVA), stating the staff's position on three issues in FSAR Chapter 14.
November 22, 1994	Letter, F. J. Hebdon to O. D. Kingsley (TVA), transmitting copy of environmental assessment and finding of no significant impact on the applicant's application for exemption from requirements of 10 CFR 73.55.
November 22, 1994	Summary, P. S. Tam, of November 2-8, 1994, meeting on technical specifications.
December 7, 1994	Summary, P. S. Tam, of November 29-December 2, 1994, meeting on technical specifications.
December 9, 1994	Letter, F. J. Hebdon to O. D. Kingsley (TVA), accepting the applicant's response to Generic Letter 93-04.
December 13, 1994	Summary, P. S. Tam, of November 29, 1994, meeting regarding construction and licensing issues.
December 15, 1994	Letter, F. J. Hebdon to O. D. Kingsley (TVA), transmitting exemption from certain requirements of 10 CFR 73.55.

December 15, 1994	Letter, F. J. Hebdon to O. D. Kingsley (TVA), transmitting copies of Supplement 14 of the Watts Bar Safety Evaluation Report, NUREG-0847.
December 20, 1994	Summary, P. S. Tam, of December 14, 1994, meeting on licensing status.
December 21, 1994	Summary, P. S. Tam, of December 14, 1994, meeting on fire protection.
December 23, 1994	Letter, R. P. Zimmerman to O. D. Kingsley (TVA), requesting, under 10 CFR 50.54(f), additional information on Thermo-Lag fire barriers.
December 28, 1994	Letter, S. A. Varga to O. D. Kingsley (TVA), transmitting Operating Readiness Assessment Team inspection report, IR 50-390/94-202.
January 18, 1995	Letter, F. J. Hebdon to O. D. Kingsley (TVA), transmitting updated proof-and-review version of the Watts Bar Unit 1 Technical Specifications, and requesting additional information.
February 1, 1995	Letter, W. T. Russell to O. D. Kingsley (TVA), transmitting summary of management meeting held on January 12, 1995.
February 6, 1995	Letter, P. S. Tam to O. D. Kingsley (TVA), accepting Revision 6 of the corrective action program on replacement items.
February 9, 1995	Letter, S. F. Newberry to O. D. Kingsley (TVA), requesting additional information on severe accident mitigation design alternatives.
February 10, 1995	Summary, P. S. Tam, of February 8, 1995, meeting on licensing status.
February 14, 1995	Letter, S. F. Newberry to O. D. Kingsley (TVA), requesting additional information on the Watts Bar Environmental Report.
February 22, 1995	Letter, P. S. Tam to O. D. Kingsley (TVA), acknowledging receipt of several letters on TMI Item II.B.2, and stating that the information has been found acceptable in Supplement 14 of the Watts Bar Safety Evaluation Report.
February 23, 1995	Letter, F. J. Hebdon to O. D. Kingsley (TVA), transmitting copy of contractor report assessing status of all pertinent unresolved safety issues, generic safety issues, TMI issues, and other multiplant issues. The letter also asked the applicant to submit its understanding of the status of these issues.

February 27, 1995	Letter, P. S. Tam to O. D. Kingsley (TVA), informing licensee of an upcoming audit of the safety parameter display system and detailed control room design review.

TVA Letters

May 8, 1980	Letter, L. M. Mills to NRC, transmitting information on containment barrier seals for Sequoyah Nuclear Plant.
August 5, 1994	Letter, D. E. Nunn to NRC, transmitting response to request for additional information relating to Final Environmental Statement.
August 12, 1994	Letter, D. E. Nunn to NRC, submitting information on electromagnetic interference and radiofrequency interference for the Eagle-21 process protection system.
September 8, 1994	Letter, D. E. Nunn to NRC, submitting additional information on certification of the Watts Bar Unit 1 Technical Specifications.
September 27, 1994	Letter, D. E. Nunn to NRC, transmitting response to request for additional information relating to environmental statement.
October 7, 1994	Letter, D. E. Nunn to NRC, submitting additional information on severe accident mitigation design alternatives.
October 21, 1994	Letter, D. E. Nunn to NRC, submitting supplemental information on the physical security plan.
October 21, 1994	Letter, D. E. Nunn to NRC, submitting additional information on post-fire safe shutdown capability.
October 24, 1994	Letter, M. O. Medford to NRC, requesting exemption from 10 CFR 73.55 to permit use of hand geometry biometrics for access control.
October 27, 1994	Letter, D. E. Nunn to NRC, submitting additional information on severe accident mitigation design alternatives.
November 4, 1994	Letter, D. E. Nunn to NRC, transmitting response to request for additional information relating to environmental statement.
November 4, 1994	Letter, D. E. Nunn to NRC, submitting additional information on safety and relief valves.
November 11, 1994	Letter, D. E. Nunn to NRC, submitting additional information on seismic qualification of Thermo-Lag fire barriers.

November 15, 1994	Letter, D. E. Nunn to NRC, submitting additional information on environmental issues.
November 18, 1994	Letter, D. E. Nunn to NRC, submitting additional information on post-fire safe shutdown capability.
November 18, 1994	Letter, D. E. Nunn to NRC, submitting latest revision of Fire Protection Report.
November 30, 1994	Letter, D. E. Nunn to NRC, requesting exemption from the schedular requirement to implement 10 CFR 73.55(c)(10) on vehicular bombs.
December 1, 1994	Letter, D. E. Nunn to NRC, submitting additional information on testing of safety and relief valves.
December 6, 1994	Letter, D. E. Nunn to NRC, submitting additional information on repair of Kapton insulation damage.
December 16, 1994	Letter, P. P. Carier to NRC, transmitting revised topical report on TVA organization.
December 20, 1994	Letter, D. E. Nunn to NRC, transmitting corrected pages to Physical Security Plan.
December 23, 1994	Letter, D. E. Nunn to NRC, submitting additional information on use of Thermo-Lag fire retardant materials.
December 23, 1994	Letter, D. E. Nunn to NRC, submitting information on pressure temperature limits.
December 23, 1994	Letter, D. E. Nunn to NRC, submitting additional information on Thermo-Lag testing.
January 5, 1995	Letter, D. E. Nunn to NRC, transmitting additional information on FSAR Chapter 14.
January 11, 1995	Letter, D. E. Nunn to NRC, transmitting additional information on electrical cable separation.
January 20, 1995	Letter, D. E. Nunn to NRC, transmitting Revision 6 of the corrective action plan on replacement items program.
January 27, 1995	Letter, D. E. Nunn to NRC, submitting response to questions raised by the staff in the December 14, 1994, fire protection meeting.
February 3, 1995	Letter, D. E. Nunn to NRC, submitting additional information on FSAR Chapter 3.
February 10, 1995	Letter, D. E. Nunn to NRC, revising the applicant's position on repair of Kapton insulation damage.

February 16, 1995	Letter, D. E. Nunn to NRC, submitting testing results for Eagle-21 process protection system.
February 16, 1995	Letter, D. E. Nunn to NRC, submitting analysis on latest small-break LOCAs and other postulated accidents.
February 17, 1995	Letter, D. E. Nunn to NRC, submitting additional information on FSAR Chapter 11.
February 22, 1995	Letter, D. E. Nunn to NRC, submitting updated information on the safety parameter display system.
February 27, 1995	Letter, O. J. Zeringue to NRC, transmitting summary description of the proposed vehicle control measures per 10 CFR 73.55(c)(7) and (8).
April 13, 1995	Letter, R. R. Baron, to NRC, submitting Revisions 24 and 25 to applicant's preservice inspection program.
April 21, 1995	Letter, R. R. Baron, to NRC, submitting revised information regarding compliance with Regulatory Guide 1.97.

APPENDIX C

UNRESOLVED SAFETY ISSUES*

* Appendix C first appeared in the SER; this is a supplement.

C.3 Unresolved Safety Issues

In this section of Appendix C of the SER, the staff stated that a number of unresolved safety issues are applicable to Watts Bar. The following is the current status of those safety issues, and supplements Appendix C of the SER.

This effort was tracked by TAC M90068. Detailed status information may be obtained by accessing public documents filed under TAC M90068.

A-1 Water Hammer

This issue was specifically resolved for Watts Bar in Appendix C to the SER, and was generically resolved in March 1984 with issuance of NUREG-0927, "Evaluation of Water Hammer Occurrence in Nuclear Power Plants--Technical Findings Relevant to Unresolved Safety Issue A-1." The Standard Review Plan was revised to ensure that future plants (those that request construction permits after 1984) would incorporate design improvements. Thus the staff's evaluation in Appendix C of the SER still stands with no need for additional action. This issue is closed for Watts Bar.

A-2 Asymmetric Blowdown Loads on PWR Primary Systems

Generic resolution of this issue led to revision of 10 CFR 50, Appendix A, General Design Criterion 4. As a result, upon the applicant's requests, the staff approved leak-before-break analyses applied to the reactor coolant system primary loop piping (see SSER 5, Appendix J) and the pressurizer surge line (see SSER 12, Section 3.6.3). The applicant has accordingly revised the FSAR. This issue is resolved for Watts Bar.

A-3 Westinghouse Steam Generator Tube Integrity

In September 1988, the staff issued NUREG-0844, concluding that risk from steam generator tube rupture events is not a significant contributor to total risk at a given site, nor to the total risk to which the general public is routinely exposed. This message was affirmed in SECY-88-272, "Technical Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity." No plant-specific action is needed of Watts Bar, and the issue is considered complete.

A-9 Anticipated Transients Without Scram (ATWS)

This issue was resolved for Watts Bar (see SSER 9, Appendix W, "Safety Evaluation Report, Watts Bar Nuclear Plant, Units 1 and 2, Compliance With ATWS Rule, 10 CFR 50.62"). There is currently no existing guidance on ATWS equipment technical specifications. When the guidance is developed, the Watts Bar Technical Specifications will be modified accordingly.

A-11 Reactor Vessel Materials Toughness

The staff issued a safety evaluation on equivalent margin analysis for Unit 1 in SSER 14, Section 5.3.1.1.1. In that safety evaluation, the staff found

Watts Bar Unit 1's equivalent margin analysis acceptable. This issue is thus closed for Watts Bar.

A-17 Systems Interaction in Nuclear Plants

Generic Letter 89-18, dated September 6, 1989, resolved this issue and required no action of the addressees. On such basis, this issue is considered resolved for Watts Bar.

A-24 Qualification of Class 1E Safety-Related Electrical Equipment

The staff completed review of Watts Bar Unit 1's equipment environmental qualification program (see Section 3.11 of SSER 15). Thus, this issue is resolved for Watts Bar Unit 1.

A-26 Reactor Vessel Pressure Transient Protection

Resolution of this issue was documented in Section 5.2.2, "Overpressure Protection", Section 7.6.5, "Overpressure Protection During Low Temperature Operation", of the SER, and Section 8.3.3.4, "Compliance With NUREG-0737 Items", of SSER 7. In addition, the staff found the applicant's response to Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," acceptable (letter, S.C. Black to O. D. Kingsley, June 29, 1989). A related issue, pressurized thermal shock, was also fully resolved (see A-49 below). On the basis of all these completed actions, this issue is complete for Watts Bar.

A-31 Residual Heat Removal Shutdown Requirements

Resolution of this issue was documented in Section 5.4.3, "Residual Heat Removal System," of the SER and SSER 5. On such basis, this issue is resolved for Watts Bar.

A-36 Control of Heavy Loads Near Spent Fuel Pool

This issue was resolved for Watts Bar in SSER 13, Section 9.1.4, "Fuel Handling System."

A-40 Seismic Design Criteria

This issue was resolved for Watts Bar in SSER 7, Section 3.7.3, "Seismic Subsystem Analysis."

A-44 Station Blackout

Resolution of this issue led to revision of 10 CFR 50.63, "Loss of All Alternating Power." The applicant submittal proposed actions to meet the regulation, and this issue was resolved for Watts Bar by letters, P. S. Tam to M. O. Medford, September 9 and March 18, 1993.

A-45 Shutdown Decay Heat Removal Requirements

SECY-88-260, "Shutdown Decay Heat Removal Requirements (USI A-45)" stated that resolution of this issue would be through plant-specific analyses under the Individual Plant Evaluation program (IPE). Generic Letter 88-20 was issued to request plant-specific actions on IPE. IPE was resolved for Watts Bar by letter, P. S. Tam to O. D. Kingsley, October 5, 1994.

A-47 Safety Implications of Control Systems

Generic Letter 89-19 communicated required plant-specific actions to resolve this issue. The staff accepted the applicant's proposal by letter, P. S. Tam to O. D. Kingsley, October 24, 1990. The letter stated that appropriate requirements will be incorporated into the Watts Bar Technical Specifications. The Watts Bar requirements are specified in the Watts Bar Unit 1 Technical Specifications, and Technical Requirements Manual (TRM), in accordance with the newest Standard Technical Specifications. The current draft TRM contains requirements regarding isolation devices (Section 3.8.1). On the basis of the October 24, 1990, letter and the requirements in the draft Watts Bar Unit 1 TRM, this issue is resolved for Watts Bar Unit 1.

A-48 Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

Generic resolution of this issue was effected by issuance of the final rule, 10 CFR 50.44. This issue was resolved for Watts Bar in Section 6.2.5, "Combustible Gas Control Systems," of SSER 8, SSER 4, and the SER. The evaluation in SSER 8 specifically addressed compliance with 10 CFR 50.44.

A-49 Pressurized Thermal Shock

Generic resolution of this issue was effected by issuance of the final rule, 10 CFR 50.61; Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials"; and Generic Letters 88-11 and 92-01. The issue was resolved for Watts Bar by letter, S. C. Black to O. D. Kingsley, June 29, 1989 (regarding Generic Letter 88-11); Section 5.3.1, "Reactor Vessel Materials", of SSER 11; Section 5.3.1 of SSER 14 (regarding 10 CFR 50 Appendix G and Generic Letter 92-01).

APPENDIX E

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APPENDIX CC

TECHNICAL EVALUATION REPORT:
TMI ACTION II.D.1, RELIEF AND SAFETY VALVE TESTING

TECHNICAL EVALUATION REPORT
TMI ACTION--NUREG-0737 (II.D.1)

RELIEF AND SAFETY VALVE TESTING
WATTS BAR NUCLEAR PLANT - UNITS 1 AND 2
DOCKET NOs. 50-390 AND 50-391

1. INTRODUCTION

1.1 Background

In the past, safety and relief valves installed in the primary coolant system of light water reactors have performed improperly. There were instances of valves opening below setpoint, valves opening above setpoint, and valves failing to open or reseat. From the past instances of improper valve performance, it is not known whether they occurred because of limited valve qualification or because of a basic unreliability in the valve design. It is known that the failure of a power-operated relief valve (PORV) to reseat was a significant contributor to the Three Mile Island, Unit 2, sequence of events. These facts led the task force that prepared NUREG-0578 (Reference 1) and, subsequently, NUREG-0737 (Reference 2) to recommend development and execution of programs to: (a) reexamine the functional performance capabilities of pressurized water reactor (PWR) safety, relief, and block valves and (b) verify the integrity of the pressurizer safety and relief valve piping systems for normal, transient, and accident conditions. The task force deemed these programs necessary to reconfirm that Licensees and Applicants indeed satisfied General Design Criteria 14, 15, and 30 of 10 CFR 50, Appendix A, for the subject equipment.

1.2 General Design Criteria and NUREG Requirements

General Design Criteria 14, 15, and 30 require: (a) the reactor primary coolant pressure boundary be designed, fabricated, and tested so as to have an extremely low probability of abnormal leakage; (b) the reactor coolant system (RCS) and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions are not exceeded during normal operation or anticipated operational occurrences; and (c) the components that are part of the reactor coolant pressure boundary be constructed to the highest quality standards practical.

To reconfirm the integrity of overpressure protection systems and thereby assure compliance to the General Design Criteria, the Nuclear Regulatory Commission (NRC), Division of Licensing, Office of Nuclear Reactor Regulation, issued the NUREG-0578 position as a requirement in a letter dated September 13, 1979, to all operating nuclear power plants. The NRC incorporated this requirement as Item II.D.1 of NUREG-0737, and they issued NUREG-0737 for implementation on October 31, 1980. As stated in the NUREG reports, each PWR Licensee or Applicant shall:

1. Conduct testing to qualify RCS relief and safety valves under expected operating conditions for design basis transients and accidents.

2. Determine valve expected operating conditions through the use of

analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Rev. 2.

3. Choose the single failures such that the dynamic forces on the safety and relief valves are maximized.

4. Use the highest test pressures predicted by conventional safety analysis procedures.

5. Include in the relief and safety valve qualification program the qualification of the associated control circuitry.

6. Provide test data for NRC staff review and evaluation, including criteria for success or failure of valves tested.

7. Submit a correlation, or other evidence, to substantiate the valves tested in a generic test program demonstrate the functionability of as-installed primary relief and safety valves. This correlation must show the test conditions used are equivalent to expected operating and accident conditions as prescribed in the Final Safety Analysis Report (FSAR). The effect of as-built relief and safety valve discharge piping on valve operability must also be considered.

8. Qualify the plant specific safety and relief valve piping and supports by comparing to test data and/or performing appropriate analyses.

2. PWR OWNER'S GROUP RELIEF AND SAFETY VALVE PROGRAM

In response to the NUREG requirements previously listed, a group of utilities with PWRs requested the assistance of the Electric Power Research Institute (EPRI) in developing and implementing a generic test program. The test program covered pressurizer PORV block valves, PORVs, safety valves, and associated piping systems. The Tennessee Valley Authority (TVA), the owner of the Watts Bar Nuclear Plant (WBN), Units 1 and 2, was one of the utilities sponsoring the EPRI Safety and Relief Valve Test Program. The participating utilities transmitted the reports containing the results of the program to the NRC in Reference 3. Lockheed Idaho Technologies Company discusses the applicability of those reports below.

The Electric Power Research Institute developed a plan (Reference 4) for testing PWR safety and relief valves under conditions that bounded actual plant operating conditions. Through the valve manufacturers, EPRI identified the valves used in the overpressure protection systems of the participating utilities and selected representative valves for testing. The valves included enough of the variable characteristics so that their testing would adequately demonstrate the performance of the valves used by utilities (Reference 5). Through the nuclear steam supply system vendors, EPRI evaluated the FSARs of the participating utilities. They then developed a test matrix that bounded the plant transients requiring overpressure protection (Reference 6).

The utilities that participated in the EPRI Safety and Relief Valve Test Program also obtained information regarding the performance of PORV block valves (Reference 7). The Electric Power Research Institute developed a list of block valves used or intended for use in participating PWR plants. They then selected for testing seven block valves to represent the block valves used in PWR plants. Westinghouse Electro-Mechanical Division (WEMD) performed additional tests on valve models they manufacture (Reference 8).

The Electric Power Research Institute contracted with Westinghouse Corporation to produce a report on the inlet fluid conditions for pressurizer safety and relief valves in Westinghouse designed plants (Reference 9). Because Westinghouse designed WBN, Units 1 and 2, that report is relevant to this evaluation.

The Electric Power Research Institute sponsored several test series. They tested PORVs and PORV block valves at the Duke Power Company Marshall Steam Station located in Terrell, North Carolina. At the Marshall Station, EPRI conducted steam tests only. Therefore, EPRI tested block valves at Marshall only for full flow, full pressure steam conditions. However, WEMD performed water flow tests on four block valve models they manufacture. The Electric Power Research Institute conducted additional PORV tests at the Wyle Laboratories Test Facility located in Norco, California. They tested safety valves at the Combustion Engineering Company Kressinger Development Laboratory located in Windsor, Connecticut. In Reference 10, EPRI reported the relief and safety valve test results; References 7 and 8 contain the block valve test results.

The EPRI test program's primary objective was to test the various types of primary system safety valves used in PWRs for the full range of expected inlet conditions. Based on analyses, EPRI limited the conditions selected for testing to steam, subcooled water, and steam to water transition. Additional objectives were to: (a) obtain valve capacity data, (b) assess hydraulic and structural effects of associated piping on valve operability, and (c) obtain piping response data for verifying analytical piping models.

The Electric Power Research Institute did not design the test program to provide information on valve reliability. The EPRI program plan (Reference 4) states, "During the course of the specified tests, each valve will be subjected to a number of operational cycles. However, it should be noted that the test program, to be completed by July, 1981, is not intended to provide valve lifetime, cyclic fatigue or statistical reliability data."

Reference 11 contains the NRC staff approval of the EPRI test program. In Reference 11, the staff concluded the EPRI program produced enough generic safety valve and PORV performance information to enable utilities to comply with the plant specific information requirements in NUREG-0737, Item II.D.1. Transmittal of the test results meets Item 6 (provide test data to the NRC) of Section 1.2 in this report.

3. PLANT SPECIFIC SUBMITTAL

The TVA submitted their WBN, Units 1 and 2, evaluation report on the pressurizer safety valves, PORVs, PORV block valves, and a summary of the piping analysis on July 22, 1983 (Reference 12). The TVA submitted additional information on July 11, 1991 (Reference 13). The NRC transmitted requests for additional information to TVA on October 10, 1991, (Reference 14) and April 15, 1993, (Reference 15) to which the Applicant responded on December

26, 1992, July 19, 1994, and December 1, 1994, (References 16, 17, and 18).

4. REVIEW AND EVALUATION

4.1 Valves Tested

The overpressure protection systems at WBN, Units 1 and 2, each use three safety valves, two PORVs, and two PORV block valves. The safety valves are Crosby Model HB-BP-86 6M6 valves with steam internals. The PORVs are Target Rock Model No. 82UU-001 3-inch (in.) by 3-in. solenoid operated valves. Neither the safety valves nor the PORVs have water seals upstream of the valves. The block valves are Westinghouse Model 3GM88 motor-operated gate valves with Limitorque SB-00-15 motor operators.

The EPRI program tested the safety valve used at WBN, Units 1 and 2, the Crosby Model HB-BP-86 6M6 valve. At WBN, Units 1 and 2, TVA mounted the safety valves on loop seal piping, but they modified the piping to be self-draining to prevent formation of a water seal at the valve inlets. The plant valves have steam internals. During testing, EPRI tested the valve on a long inlet piping configuration with and without a loop seal, and those configurations bound the WBN, Units 1 and 2, installation. The test valve had loop seal internals. Only the material used in the valve seats differs from the steam internals, and this does not affect valve operability within the limited number of cycles in the test program. In Reference 16, TVA stated the ring settings for the Crosby 6M6 valves at WBN, Units 1 and 2, were factory set ring settings. Therefore, TVA can use the results from the EPRI tests with factory ring settings to demonstrate operability of the plant valves.

The Target Rock PORV used at WBN, Units 1 and 2, has the same general size, configuration, and principle of operation as the valve tested by EPRI. However, the plant and test valves have differences that both do and do not affect valve operability (References 16 and 17). The differences include inlet, outlet, bonnet, fixed core, valve body through bore, and magnetic sleeve changes. Target Rock changed the main disc, pilot disc, pilot seat insert, and sleeve materials and increased the main disc lift. They added expander rings under the piston rings in the plant valves to compensate for a loss of initial piston ring tension in high differential pressure service conditions. Target Rock also transferred guidance of the plunger and moveable core from the bonnet wall to the inner disc rod to compensate for scratches observed in the EPRI tests on the plunger, moveable core, and bonnet wall. Finally, they increased the nominal power input of the solenoid from 120 to 163 watts. Tennessee Valley Authority stated Target Rock based these changes on field experience and the EPRI tests. They stated these changes should improve the operability and performance characteristics of the plant valves and increase their flow capacity relative to the test valve. Therefore, Lockheed Idaho considers the test valve representative of the plant valves.

The block valves used in WBN, Units 1 and 2, are the same design as one of the valves tested in the EPRI test program, the Westinghouse 3GM88 block valve. The Electric Power Research Institute tested the valve in a horizontal configuration, and TVA installed the plant valves in the same position (Reference 16). The plant valves have Limitorque SB-00-15 motor operators; the test valve used the same Limitorque operator. During EPRI testing, the 3GM88 block valve fully closed only when EPRI set the operator to produce

182 foot-pounds (ft-lb) of torque. In Reference 13, TVA stated they changed the Unit 1 block valve operators to limit-control closure rather than torque-control closure to ensure complete valve closure. In Reference 17, TVA stated the 3GM88 operator supplied by Westinghouse is capable of providing 222 ft-lb of torque with 87% voltage based on the nameplate motor start torque and a run/pullout efficiency of 60%. This exceeds the 182 ft-lb shown by EPRI to be adequate to open/close the 3GM88 block valve. It also exceeds the torque TVA calculated was required, 137 ft-lb, to operate the block valve under worst case conditions (Reference 17). For Unit 2, TVA committed to modifying the block valve operators before fuel load. Based on this information, the test valve represents the plant valves.

Based on the above, Lockheed Idaho considers the test valves applicable to the WBN, Units 1 and 2, valves and to have fulfilled the requirements of Items 1 and 7 of Section 1.2 in this report regarding applicability of the test valves.

4.2 Test Conditions

As stated previously, Westinghouse designed WBN, Units 1 and 2. Reference 9 contains the valve inlet fluid conditions that bound the overpressure transients for Westinghouse-designed PWRs. In Reference 16, TVA stated they verified the inlet conditions in Reference 9 are still applicable to WBN, Units 1 and 2, except for the main feedwater line break (FWLB); that event was reanalyzed in 1991.

In Reference 17, TVA stated the new FWLB inlet conditions from the 1991 analysis are bounded by those in Reference 9, and that the differences between the two sets of inlet conditions do not affect the applicability of the EPRI tests to WBN, Units 1 and 2. Lockheed Idaho disagrees with TVA's statement in regard to the range of liquid temperatures at the valve inlet. The 1991 FWLB analysis for WBN, Units 1 and 2, resulted in temperature ranges of 603.7 to 624.6°F if the PORVs opened and 607.2 to 629.1°F if only the safety valves opened. The Reference 9 liquid temperature range was given as 630.8 to 637.0°F. Lockheed Idaho considers the 1991 results more limiting because of the colder liquid temperatures calculated in that analysis. Colder liquid temperatures were more challenging in the EPRI tests because valve operability problems generally occurred in the tests with colder water. Therefore, Lockheed Idaho will use the liquid temperature ranges from the 1991 FWLB reanalysis in the valve operability review.

The events considered in Reference 9 include FSAR, extended high pressure injection, and low temperature overpressurization events. The conditions applicable to WBN, Units 1 and 2, are those for four-loop plants. The following sections discuss the bounding inlet conditions for each of these events and the applicable EPRI tests.

4.2.1 FSAR Steam Discharge Transients

When the safety valves open alone, the limiting steam discharge conditions are peak pressurizer pressure, 2555 pounds per square inch absolute (psia), and maximum pressurization rate, 144 pounds per square inch/second (psi/s). The peak pressure is from the loss-of-load event, and the peak pressurization rate is from the locked rotor event. The maximum expected backpressure is 550 psia (Reference 16).

In the test program, EPRI completed two steam tests on the Crosby 6M6 safety valve with a long inlet configuration and a drained loop seal. One of these tests (Test 1411) applies to the Crosby valves at WBN, Units 1 and 2, because the ring settings in this test (-77, -18) are representative of typical PWR plant ring settings. The ring settings represent the upper and lower ring positions measured from the level position referenced to the bottom of the disc ring. In Reference 16, TVA stated the ring settings used at WBN, Units 1 and 2, are -81 to -129 (upper ring) and -18 (lower ring), and these ring settings were determined by the valve manufacturer. Therefore, Lockheed Idaho considers the plant and test valve ring settings comparable.

In the applicable test, the valve opening pressure was 2410 psia, the pop pressure was 2420 psia, and the peak tank pressure reached 2664 psia. The pressurization rate was 300 psi/s and the peak backpressure was 245 psia. The test inlet fluid conditions for this steam test, except for the backpressure, are representative of the expected conditions for steam discharge.

To assess Crosby 6M6 valve performance with high backpressure, Lockheed Idaho will use Test 929, a cold loop seal/steam test. The peak backpressure in this test, 710 psia, develops after loop seal discharge, and full steam flow is established. Other conditions for this test, peak tank pressure, 2726 psia, and pressurization rate, 319 psi/s, also bound the plant inlet conditions.

For FSAR transients resulting in steam discharge, the PORVs will open at a pressure somewhat above the opening setpoint of 2350 psia. The maximum pressurizer pressure is 2532 psia (loss-of-load) and maximum pressurization rate is 130 psi/s (locked rotor). These results assume both the safety and relief valves open. The plant backpressure for the PORVs is 550 psia (Reference 16).

During testing, EPRI completed 15 steam tests on the Target Rock test PORV. In the steam tests, the maximum pressure at valve opening ranged from 2425 to 2521 psia. The maximum backpressure for these tests ranged from 155 to 520 psia. In these tests, the peak test pressure and backpressure are slightly lower than those expected at the plant. However, Lockheed Idaho considers the EPRI test conditions in the PORV steam tests adequate to represent FSAR steam transients. In addition, TVA stated (Reference 16) that they contracted with Target Rock to provide additional PORV tests with backpressure variations from 415 to 2350 psia.

4.2.2 <u>FSAR Water Discharge Transients</u>

The limiting FSAR transient, with respect to water flow through the safety valves and PORVs, is the feedwater line break (FWLB). The Westinghouse inlet conditions report (Reference 9) provided WBN, Unit 1, inlet conditions for the FWLB transient. These conditions also apply to WBN, Unit 2 (Reference 16). As discussed above, however, TVA reanalyzed the FWLB accident. In Reference 17, TVA stated the new FWLB inlet conditions are bounded by those from the Westinghouse valve inlet conditions report (Reference 9). From Reference 9 assuming the PORVs do not open, the safety valve inlet conditions include maximum pressurizer pressure, 2575 psia;

maximum liquid surge rate, 430.4 gallons per minute (gpm); and maximum pressurization rate, 1.6 psi/s. The liquid temperature range was taken from the 1991 analysis as provided by TVA in Reference 16; the liquid temperatures range from 607.2 to 629.1°F.

If the PORVs open, they will see conditions similar to the safety valves as discussed above. The PORV valve inlet conditions include maximum pressurizer pressure, 2575 psia; maximum liquid surge rate, 430.4 gpm; maximum pressurization rate, 1.6 psi/s; and liquid temperatures ranging from 603.7 to 624.6°F. The liquid temperature range was provided by TVA in Reference 16.

With the Crosby 6M6 test valve, EPRI performed one transition test (931a) with ring settings applicable to those at the plant. This test included a loop seal upstream of the valve. However, with respect to valve operability, Lockheed Idaho can use this test to evaluate the plant valves without loop seals. The peak pressure and pressurization rate in the test were 2578 psia and 2.5 psi/s. The tank water temperature was 641°F. After the valve closed, the system repressurized, and the valve cycled on 635°F water (Test 931b). In addition, one water test (932) was run with ring settings applicable to the plant valves. The peak pressure was 2520 psia, and the pressurization rate was 3.0 psi/s. The water temperature at the valve inlet at the start of the test was 463°F, and the tank water temperature was 515°F. These conditions bound those at the plant.

For the Target Rock PORV, EPRI completed one transition test and four high pressure water tests. In the transition test, the peak pressure was 2500 psia, and the water temperature was 656°F. In the water tests, the pressure ranged from 2490 to 2536 psia, and water temperatures ranged from 451 to 648°F. The peak pressures in the tests discussed above are lower than the 2575 psia expected in a FWLB. However, Reference 6 stated the inlet pressure will affect PORV performance only during valve opening and closing. Reference 6 also noted the Target Rock PORV opens quickly (less than 0.7 second). For the FWLB at WBN, Units 1 and 2, the calculated pressurization rate is 1.6 psi/s. At this pressurization rate, the plant PORVs will be fully open before the inlet pressure exceeds the test pressure. Therefore, Lockheed Idaho considers testing of the PORV at 2490 to 2536 psia adequate. Lockheed Idaho also considers all the above conditions representative of those expected for the plant PORVs.

4.2.3 Extended High Pressure Injection Events

The limiting extended High Pressure Injection (HPI) event is a spurious activation of the safety injection system at power. For four-loop plants, this event challenges both the safety valves and PORVs. The PORVs and safety valves open on steam, but liquid discharge would not occur until the pressurizer became water solid. According to References 9 and 12, this would not occur for at least 20 minutes into the event, and this allows ample time for operator action. Thus, Lockheed Idaho disregarded the potential for liquid discharge in extended HPI events.

4.2.4 Low Temperature Overpressurization Events

At WBN, Units 1 and 2, TVA uses the PORVs for protection from low temperature overpressurization (LTOP) events. The fluid conditions for these events can vary between steam and subcooled water because of administrative requirements for maintaining a pressurizer steam bubble during low temperature operations. In Reference 16, TVA provided the plant specific range of potential fluid conditions for LTOP events. The current LTOP control system varies the PORV setpoint from 485 to 2365 psia as the RCS temperature ranges from 70 to 450°F. In Reference 16, TVA stated a new LTOP control system is being implemented that will slightly increase the PORV setpoints so that they vary from 500 to 2365 psia as the RCS temperature ranges from 70 to 450°F.

In addition to the various high pressure tests previously mentioned, EPRI performed two low pressure water tests on the Target Rock PORV. The test pressures were 690 and 715 psia, while the valve inlet temperatures were 114 and 447°F. These test conditions, together with the high pressure test conditions, adequately represent the expected LTOP inlet conditions at WBN, Units 1 and 2.

4.2.5 Block Valve Inlet Conditions

The block valves operate over a range of fluid conditions (steam, steam-to-water, and water) similar to those of the relief valves. However, EPRI tested the block valves only under full pressure steam conditions (to 2420 psia). For Westinghouse-manufactured valves, WEMD performed additional water flow tests. The WEMD test conditions ranged from 60 to 600 gpm water flow and 1500 to 2600 psi differential pressure. Based on Reference 8, Westinghouse found four things concerning block valves with similar internal materials. Under full pressure steam conditions, Westinghouse found the required torque to open or close the valve:

1. Depends almost entirely on the differential pressure across the valve disc.
2. Is rather insensitive to momentum loading.
3. Is nearly the same for water or steam.
4. Is nearly independent of the flow.

Thus, full pressure steam tests are adequate to show valve operability for steam and water conditions.

4.2.6 Other Transients

Two transient conditions not part of the design basis are feed and bleed decay heat removal and anticipated transients without scram. This review did not consider the response of the overpressure protection system to these two transient conditions. Neither the Applicant nor the NRC have evaluated the performance of the system for these events.

4.2.7 Inlet Conditions Summary

The presentation above demonstrates that the test conditions bounded the conditions for the plant valves, and it verifies TVA met Items 2 and 4 of Section 1.2 in this report. That is, TVA determined the conditions for the operational occurrences and chose the highest predicted pressures for the tests. They also met the portion of Item 7 that requires showing test conditions are equivalent to those prescribed in the FSAR.

4.3 Operability

The safety valves and PORVs operate over a range of full pressure steam, steam-to-water transition, and subcooled water fluid conditions, and EPRI tested the valves for the required range of conditions. The block valves also operate over a range of steam and liquid flow conditions. The Electric Power Research Institute tested the block valves with full pressure steam; those test results also apply to liquid flow.

4.3.1 Safety Valves

In the one applicable steam test (1411), the safety valve opened at 2410 psia (-3.6% of the setpoint) and operated stably. The valve achieved 107% of rated steam flow at 3% accumulation and 92% of rated lift, and it closed with 8.2% blowdown. Test 929 was the loop seal test used to bound valve performance with high backpressures. In the test, the valve was stable on steam and achieved 111% of rated flow at 3% accumulation and 93% of rated lift. The valve closed with 5.1% blowdown. Thus, in the applicable tests, the valve performed its safety function of opening, relieving pressure, and closing.

A FWLB can result in high pressure and temperature liquid discharge through the safety valves. A loop seal/transition test (931a) and two water discharge test (931b and 932) bound the expected behavior of the plant valves. In Test 931a, the valve opened at 2570 psia (+2.8% of the setpoint), fluttered or chattered during loop seal discharge, stabilized during steam and water discharge, and closed with 12.7% blowdown. At 2415 psia with 641°F water, the valve passed 2355 gpm of liquid with the valve at 56% of rated lift. In Test 931b, the valve opened on 635°F water within -1% of the setpoint, chattered during opening, stabilized, and closed with 4.8% blowdown. The operators did not record the liquid flow rate in Test 931b. In Test 932, the valve opened and immediately began to chatter. The operators manually terminated the test by opening the valve to stop the chatter. Because the pressurizer safety valves are designed for steam relief, valve chatter when passing highly subcooled water is not unexpected. The temperatures expected in a FWLB at WBN, Units 1 and 2, (607 to 629°F) fall between the available test data at 635 and 463/515°F. However, based on engineering judgement, the WBN, Unit 1 and 2, FWLB temperatures are close enough to the hot water EPRI tests to conclude the plant safety valves will operate satisfactorily during a FWLB.

The largest bending moment EPRI induced on the discharge flange of the Crosby 6M6 test valve was 298,750 inch-pounds (in-lb) (Test 908). Application of this bending moment did not affect valve performance. This applied moment exceeds the maximum estimated bending moment of 90,984 in-lb for WBN, Unit 1, valves. The plant value is based on the absolute sum of the maximum, faulted, local M_y and M_z moments provided by TVA in Reference 16. In Reference 16, TVA committed to calculating a similar bending moment for Unit 2 during the reanalysis of the Unit 2 pipe support loads prior to Unit 2 fuel load. Therefore, the bending moments imposed during discharge transients will not affect Unit 1 valve performance.

As stated earlier, the maximum observed blowdown in the applicable EPRI tests was 12.7%, and this exceeds the design value of 5%. Thus, TVA must demonstrate that extended blowdown will not impact plant safety and valve operability. They provided this information in Reference 16. Tennessee Valley Authority stated Westinghouse evaluated the effect of 13% safety valve blowdown on various accident analyses. The Westinghouse evaluation found:

1. Extended safety valve blowdown of up to 13% will not cause the pressurizer to fill in any licensing basis event where the pressurizer does not already become water solid.

2. Extended safety valve blowdown of up to 13% will not challenge any safety systems that were not previously challenged in the licensing basis safety analyses.

3. Extended safety valve blowdown of up to 13% will not cause voiding of the primary system in any licensing basis event.

Therefore, the extended blowdown observed in the EPRI tests does not impact plant safety or valve operability.

For the steam test to adequately demonstrate safety valve stability, the test inlet piping pressure drop should exceed the plant inlet piping pressure drop. In Reference 16, TVA provided the WBN, Units 1 and 2, calculated values. The pressure drops calculated for the plant safety valves were 256.51 and 152.73 pounds per square inch differential (psid) for opening and closing, respectively. The corresponding pressure drops for the test valve on the loop seal configuration were 263 psid on opening and 181 psid on closing. Therefore, the plant valves should be as stable as the test valves.

4.3.2 Power-Operated Relief Valves

For all applicable tests on the Target Rock PORV (non-loop seal tests), the valve opened and closed on demand. Total valve opening times were less than 0.66 second (s) and closing times were less than 0.69 s. The Electric Power Research Institute inspected the valve after the completion of testing. Based on the limited number of cycles in the test program, EPRI observed no damage that would affect future valve performance.

Reference 6 indicated the Target Rock PORV is susceptible to backpressure effects because it is a pilot valve design. In Reference 16, TVA argued the pilot valve design of the Target Rock PORV is different from other pilot valve operated PORVs because the pilot valve is internal to the valve. The main and pilot discs are mechanically linked together so the solenoid force applied to the pilot disc assists in opening the main disc. At zero differential pressure across the PORV, the solenoid force alone is sufficient to lift the main disc. Further, TVA stated Target Rock personnel indicated increased backpressure makes it easier to open or close the valve, but does not affect the ability of the valve to remain open. Therefore, TVA concluded the backpressure would only affect flow through the valve and the valve opening and closing times. Based on the information from TVA, Lockheed Idaho agrees with this conclusion.

In Reference 16, TVA also discussed specific test data to show the Target Rock PORV is not affected by backpressure. The EPRI test valve was subjected to a 520 psia backpressure in one test, and this backpressure is close to the 550 psia backpressure expected at the plant. The valve fully opened and closed on demand in this test. In addition, TVA stated they contracted with Target Rock to conduct additional tests on one of the PORVs from WBN, Units 1 and 2. Backpressures in these tests ranged from 415 to 2350 psia, and the PORV operated normally.

During EPRI testing, the operators induced a bending moment of 32,900 in-lb on the Target Rock test PORV. In Reference 17, TVA discussed the results of an evaluation performed by Target Rock on the differences between the EPRI test valve and the Target Rock PORVs installed at the plant. The evaluation concluded that, because of the differences in valve design, the potential for valve binding was greater in the EPRI test valve relative to the valves installed at WBN, Units 1 and 2. Application of the test bending moment did not affect test valve performance. The maximum calculated bending moment for the WBN, Unit 1, valves is 28,596 in-lb, and this is less than the bending moment applied to the test valve. The plant value is based on the absolute sum of the maximum, faulted, local M_y and M_z moments provided by TVA in Reference 16. In Reference 16, TVA committed to calculating a similar bending moment for Unit 2 during the reanalysis of the Unit 2 pipe support loads prior to Unit 2 fuel load. Based on the above, Lockheed Idaho expects the Unit 1 Target Rock PORVs to operate with the maximum expected bending moment at the plant.

Based on valve performance during testing and other information provided, the Applicant verified the PORVs operated properly under expected fluid transient conditions.

4.3.3 PORV Control Circuit Qualification

NUREG-0737, Item II.D.1, requires the qualification of the PORVs and their associated control circuitry for design basis accidents and transients. The EPRI test program included the PORV control circuitry attached directly to the valve (Reference 19). It did not include the circuits away from the valve such as pressure sensing devices, cables, transmitters, etc. The individual utilities still need to meet the NUREG-0737, Item II.D.1, requirements for the circuits away from the valve. Based on Reference 11, the NRC concluded Applicants could meet the NUREG environmental qualification requirement for those circuits by including them in their 10 CFR 50.49 program. However, TVA stated (Reference 16) the PORV control circuitry at WBN, Units 1 and 2, contains non-environmentally qualified inputs.

In question 12, Reference 14, the NRC gave TVA several alternatives if the PORV control circuitry is not included in the 10 CFR 50.49 program. In Reference 16, TVA discussed how they met one of those alternatives. The NRC alternative stated:

> The PORVs are not required to perform a safety function to mitigate the effects of any design basis event in a harsh environment and failure in a harsh environment will not adversely impact safety functions or mislead the operators (PORVs will not experience any spurious actuations and, if emergency operating procedures do not specifically prohibit use of PORVs in accident mitigation, it must be ascertained the operators can close the PORVs under harsh environment conditions).

In Reference 16, TVA stated they do not require the PORVs to perform a safety function to mitigate the effects of a design basis event in a harsh environment and failure in a harsh environment will not adversely affect safety functions or mislead the operators. They noted no credit is taken for PORV operation to mitigate the effects of an accident, except for high-point venting of the RCS. This venting is accomplished by remote-manual control of the environmentally qualified PORVs using only environmentally qualified portions of the control circuitry.

Lockheed Idaho notes the NRC alternative requires that the PORVs not experience any spurious actuations. In Reference 16, TVA did not state the PORVs will not experience any spurious actuations. However, TVA did say that, if the PORV opens spuriously due to an environmentally induced failure of one of the non-qualified inputs to the PORV control circuitry, the operators can still close the PORV by remote-manual control using the qualified control circuits discussed above. If a postulated single-failure is assumed to prevent remote-manual closure, TVA stated the operators can use the environmentally qualified block valve and environmentally qualified block valve control circuitry to isolate the PORV. The plant post-accident monitoring system provided positive indication of both PORV and block valve position. Although this does not meet the NRC requirement regarding spurious PORV activation directly, Lockheed Idaho considers that TVA meets the intent of the NRC requirement by having a single-failure proof means of closing and/or isolating the PORVs in a harsh environment should a spurious activation occur.

Therefore, Lockheed Idaho concludes TVA meets the environmental qualification requirements for the control circuitry.

With respect to the qualification of the control circuits during normal operation, TVA stated they submitted Unit 1 technical specifications to the NRC that include surveillance requirements to ensure the operability of the PORVs, block valves, and their associated control circuits. In Reference 16, TVA also stated they had committed to follow the recommendations in Generic Letter (GL) 89-10 for safety-related motor-operated valve testing and surveillance and to implement the improvements identified in GL 90-06 for PORV and block valve reliability (References 20 and 21). This meets the requirements to qualify the control circuits for normal operation.

4.3.4 PORV Block Valves

The PORV block valve must close over a range of steam and water conditions. As described in Section 4.2 of this report, high pressure steam tests adequately bound operation over the full range of inlet conditions. As described in Section 4.1 of this report, the tests conducted with the 3GM88 valve and SB-00-15 operator demonstrate plant valve operability. This is because TVA modified the Unit 1 block valve operator to close on limit, and, in this mode of operation, the torque provided by the operator is greater than that used in the EPRI tests. In Reference 17, TVA noted the Unit 2 block valve operator will be modified in the same way as the Unit 1 operator prior to fuel load. The valve tested opened and closed successfully with the test valve operator set to produce 182 ft-lb of torque (Reference 7). Therefore, the tests demonstrated acceptable valve operation. In addition, TVA stated (Reference 17) the block valve operating requirements and capabilities are validated by dynamic testing that is part of the WBN, Units 1 and 2, GL 89-10 test program. Including the PORV block valves in the GL 89-10 test program provides additional assurance the block valves will operate acceptably.

4.3.5 Operability Summary

The facts presented above demonstrate TVA met Item 1 (conducting tests to qualify the valve) and met Item 7 (considering the affects of discharge piping on operability) of Section 1.2 in this report. Meeting the NRC alternative to qualifying the control circuits under 10 CFR 50.49 and committing to meet the requirements of GL 90-06 adequately satisfies Item 5 of Section 1.2 in this report regarding the PORV control circuitry.

4.4 Piping and Support Evaluation

This evaluation covers the piping and supports extending from the pressurizer nozzle to the pressurizer relief tank. The Applicant designed the piping for deadweight, internal pressure, thermal expansion, earthquake, and safety and relief valve discharge conditions. This section discusses the calculation of the hydraulic force time histories due to valve discharge, the structural analysis methods, and the load combinations and stress evaluation. This evaluation is for WBN, Unit 1. In References 13 and 16, TVA stated they plan to redo the thermal-hydraulic and structural analyses for the piping and supports for Unit 2 as part of a hanger and analysis update program. This will be completed prior to fuel load for Unit 2.

4.4.1 Thermal-Hydraulic Analysis

The TVA used pressurizer fluid conditions in the thermal-hydraulic analysis such that the calculated pipe discharge forces would bound the forces for any of the FSAR, HPI, and low temperature overpressurization events, including the single failure that would maximize the forces on the valve.

They analyzed the safety valve and PORV discharge transients in six separate cases (Reference 16). These cases included: (a) the three safety valves open and close and the relief valves remain closed, (b) the two relief valves open and close and the safety valves remain closed, and (c) the two relief valves open and close during the LTOP mode of operation. Lockheed Idaho considers this approach acceptable because the safety valves and PORVs have different setpoints. Therefore, they will not lift simultaneously.

A valve operating condition that is more likely to occur would be a PORV discharge followed by a safety valve discharge. Because the PORVs have a lower setpoint, they would open first. In this case the PORV piping loads would be the same as those calculated from case b above. This scenario, however, reduces the safety valve piping loads due to the backpressure buildup in the discharge piping resulting from the PORV discharge; therefore, TVA need not analyze this condition.

Because there are no water seals upstream of the safety valves, the steam discharge condition would generate the highest loads on the safety valve piping system. The analyzed safety valve steam discharge cases adequately represent the conditions expected for the safety valve piping system as discussed below. Similarly, the PORV discharge cases adequately represent the conditions expected for the PORV piping system. Also, TVA stated in Reference 18 that valve opening on water is not calculated for FSAR transients. Lockheed Idaho notes that valve inlet conditions for NUREG-0737, Item II.D.1, were to be based on FSAR transients. Therefore, Lockheed Idaho considers the selection of these cases adequate to represent the limiting conditions for the piping load evaluation.

For the safety valve opening case, TVA assumed the safety valves opened at 2575 psia, passed saturated steam at 673°F, and had a pressurization rate of 54 psi/s. The maximum pressure was 2748 psia. The safety closing case assumed the safety valves closed at 2374.7 psia on saturated steam at 673°F. When the PORVs passed steam, TVA assumed they opened at 2420 psia, passed steam at 663°F, and had a pressurization rate of 54 psi/s. The maximum pressure was 2525 psia. The PORV closing case assumed the PORVs closed at 2400 psia on saturated steam at 663°F. For PORV water discharge, TVA assumed LTOP type conditions. For PORV closing, TVA assumed a pressure of 850 psi and a water temperature of 380°F. For PORV opening, the assumed conditions were pressure of 605 psia and 70°F water.

The pressurization rate used in the thermal-hydraulic analyses, 54 psi/s, is less than the 144 psi/s discussed in the Westinghouse valve inlet conditions report. In Reference 17, TVA responded to a question on the adequacy of the pressurization rate used for WBN, Units 1 and 2. TVA noted the peak loads were calculated to occur within 0.28 s of the valve opening. Therefore, use of a 144 psi/s pressurization rate would result in a maximum pressure increase of 25 psi. This is 1% of the pressurizer safety valve opening pressure of 2500 psia. For the other cases analyzed, the percent increase in pressure was less than 1% of the valve opening or closing pressure for the particular case analyzed because of shorter times to the peak calculated loads. Therefore, Lockheed Idaho considers use of the 54 psi/s pressurization rate adequate for the thermal-hydraulic analysis.

The TVA performed the thermal-hydraulic analysis using the WATHAM and STEHAM computer programs. In Reference 16, TVA stated Stone & Webster Engineering Corporation (SWEC) used these programs to analyze steam discharge (STEHAM) and water discharge (WATHAM). Lockheed Idaho reviewed STEHAM in other utilities' submittals (Reference 22) and determined it was adequately qualified by SWEC for valve discharge thermal-hydraulic analyses. In Reference 17, TVA provided information on the qualification of WATHAM that showed the program was also adequate to calculate valve thermal-hydraulic analyses.

Lockheed Idaho reviewed the key input parameters and assumptions made in the thermal-hydraulic analysis, such as the valve opening time, time step size, valve flow rates, etc. The valve opening time for the safety valves was 0.008 s on steam. This time adequately represents that measured in the EPRI tests for steam inlet conditions (valve opening time in the applicable steam test was 0.026 s). The valve flow area used in the safety valve discharge analysis produced a flow corresponding to greater than 111% of the rated flow. This is adequate for the Crosby valves used at the plant which passed 111% or less of rated flow in the EPRI tests. The TVA assumed the PORVs opened in 0.006 s for steam and water discharge. The measured opening times for the Target Rock PORV in the EPRI tests were 0.2 s on water and 0.66 s on steam. The use of faster times is conservative due to the larger acoustic wave generated by the faster valve opening time. The flow rate used in the analysis for the PORVs was 111% of the valve rated flow (233,333 pounds mass/hour). In Reference 18, TVA noted that adjusting the EPRI measured flow for the EPRI test valve to the larger orifice used in the WBN, Unit 1 and 2, PORVs results in a flow range of 226,930 to 240,720 pounds mass/hour. Therefore, Lockheed Idaho considers the analysis value representative of that expected at the plant. The time step control resulted in time steps between 0.0006 to 0.001 s, and this is adequate based on the code verification problems. Therefore, Lockheed Idaho considers the thermal-hydraulic analysis adequate for predicting the safety valve and PORV discharge loads.

4.4.2 Stress Analysis

The TVA calculated the dynamic structural responses of the piping system to safety valve/PORV discharge transients using the TPIPE computer program. Based on Reference 16, TVA described TPIPE and its qualification against other computer programs used throughout the industry in Section 3.9.1.2.1 of the FSAR. Lockheed Idaho concluded NRC acceptance of TPIPE in the FSAR indicates the adequacy of the program.

The TVA calculated the piping system response using the direct integration method. In Reference 16, TVA supplied information on the important structural analysis parameters of time step, cutoff frequency, damping, and mass point spacing. The time step chosen for the structural analysis was 0.00025 s. This time step is small enough to accurately represent the wave functions for the cutoff frequency selected, 500 Hz. The TVA modeled damping at or below the values given in Regulatory Guide 1.61 for the frequencies to be analyzed. Mass point spacing for the major pipe sizes was less than 3.5, 5.0, and 8.0 ft for 3, 6, and 8 inch piping. Lockheed Idaho considers these structural analysis parameters adequate.

The TVA took the load combinations for the piping and supports from TVA Design Criteria WB-DC-40-31.7. In Reference 16, TVA stated they based the load combinations on FSAR commitments, and this is adequate. For the stress limits, TVA based the upstream piping stress limits on ASME Class 1 requirements and the downstream piping stress limits on ASME Class 2 requirements (Reference 17). The ASME Section III Code version was the 1971 Edition with Addenda through Summer 1973. They took the allowable stress limits for the upstream and downstream piping supports from TVA Design Criteria WB-DC-40-31.9, and this is consistent with the plant FSAR (Reference 17). The requirements of the AISC code, 7th Edition, with Supplements 1, 2, and 3, and/or later editions and manufacturer allowable loads were used to develop TVA Design Criteria WB-DC-40-31.9.

The piping stress summaries provided by TVA (Reference 16) compare the highest stresses in the piping with the applicable stress limits in the form of ratios (calculated over allowable). Lockheed Idaho reviewed the piping stress results and found all stresses within the applicable stress limits. Lockheed Idaho reviewed a similar comparison in Reference 16 for the pipe supports, and all supports met the applicable requirements. In Reference 16, TVA stated they calculated the stresses and loads on the pressurizer safety valve and PORV nozzles. Westinghouse and TVA personnel reviewed these stresses and loads and found them acceptable. Finally, TVA indicated they had to modify the piping and support configuration during the course of the analysis to meet the piping design criteria. They committed to complete the Unit 1 modifications needed to bring the actual plant configuration into agreement with the final system analyzed and the stress analysis results discussed above prior to Unit 1 fuel load (Reference 16).

4.4.3 Piping and Support Summary

The applicant met Item 3 of Section 1.2 in this report by selecting bounding cases for the piping evaluation. Based on the piping and support stress analysis TVA provided, they also met Item 8.

5. EVALUATION SUMMARY

The Applicant for WBN, Units 1 and 2, provided an acceptable response to the requirements of NUREG-0737, Item II.D.1. Therefore, the Applicant reconfirmed General Design Criteria 14, 15, and 30 of Appendix A to 10 CFR 50 were met with regard to the safety valves, PORVs, and block valves. The discussion below provides the rationale for this conclusion.

The Applicant participated in the development and execution of an acceptable test program. The program would qualify the operability of prototypical valves and demonstrate their operation would not invalidate the integrity of the associated equipment and piping. The Electric Power Research Institute successfully completed the subsequent tests under operating conditions that by analysis bounded the most probable maximum forces expected from anticipated operational occurrences and design basis events. The generic test results and piping analyses showed that the tested valves functioned correctly and safely for all steam and water discharge events specified in the test program that are applicable to WBN, Units 1 and 2. Also, the pressure boundary component design criteria were not exceeded. Analysis and review of the test results and the Applicant's justifications indicated direct applicability of the prototypical valve and valve performance to the in-plant valves and systems covered by the generic test program. The Applicant's analysis showed the plant specific piping was acceptable.

Thus, the Applicant met the requirements of Item II.D.1 of NUREG-0737 (Items 1-8 of Section 1.2 in this report). Therefore, the Applicant demonstrated by testing and analysis for the subject equipment that: (a) the reactor primary coolant pressure boundary will have a low probability of abnormal leakage (General Design Criterion No. 14), (b) the reactor primary coolant pressure boundary and its associated components (piping, valves, and supports) were designed with sufficient margin such that design conditions are not exceeded during relief/safety valve events (General Design Criterion No. 15), and (c) the valves and associated components were constructed in accordance with high quality standards (General Design Criterion No. 30).

Lockheed Idaho performed this review for both WBN, Units 1 and 2. However, the applicability of this review to Unit 2 depends on the Applicant verifying that the Unit 2 as-built system conforms to the Unit 1 design reviewed in this report.

6. REFERENCES

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2. USNRC, Division of Licensing, Clarification of TMI Action Plan Requirements, NUREG-0737, November 1980.

3. Letter, D. P. Hoffman, Consumers Power Co. to H. Denton, NRC, "Transmittal of PWR Safety and Relief Valve Test Program Reports," September 30, 1982.

4. Electric Power Research Institute, Program Plan for Performance Testing of PWR Safety and Relief Valves, July 1980.

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13. Letter from E. G. Wallace, TVA, to USNRC Document Control Desk, "In the Matter of the Application of the Tennessee Valley Authority, Docket Nos. 50-390/50-391, Watts Bar Nuclear Plant Units 1 and 2 - NUREG-0737 Item II.D.1 - Safety and Relief Valve Testing (TAC No. 79992)," July 11, 1991.

14. Letter P. Tam, NRC, to D. A. Nauman, TVA, "Watts Bar Nuclear Plant - NUREG-0737, Item II.D.1, Relief and Safety Valve Testing, (TAC No. M79992)," October 10, 1991.

15. Letter P. Tam, NRC, to M. O. Medford, TVA, "Watts Bar Nuclear Plant -

NUREG-0737, Item II.D.1, Relief and Safety Valve Testing, (TAC No. M79992)," April 15, 1993.

16. Letter W. J. Museler, TVA, to USNRC Document Control Desk, "Watts Bar Nuclear Plant (WBN) - NUREG-0737 Item II.D.1, Relief and Safety Valve Testing (TAC 79992)," December 26, 1992.

17. Letter D. E. Nunn, TVA, to USNRC Document Control Desk, "Watts Bar Nuclear Plant (WBN) - NUREG-0737 Item II.D.1, Relief and Safety Valve Testing (TAC M79992)," July 19, 1994.

18. Letter D. E. Nunn, TVA, to USNRC Document Control Desk, "Watts Bar Nuclear Plant (WBN) - NUREG-0737 Item II.D.1, Relief and Safety Valve Testing (TAC M79992)," December 1, 1994.

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APPENDIX DD

TECHNICAL EVALUATION REPORT: AUDIT OF THE ENVIRONMENTAL QUALIFICATION OF SELECTED SAFETY-RELATED ELECTRICAL EQUIPMENT AT THE WATTS BAR NUCLEAR PLANT, UNIT NO.1

INEL-95/0202

TECHNICAL EVALUATION REPORT

Audit of the Environmental Qualification of Selected Safety-Related Electrical Equipment at the Watts Bar Nuclear Plant, Unit No. 1

Docket No. 50-390

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ABSTRACT

The INEL audited Watts Bar Nuclear Plant Unit 1 to determine the environmental qualification of selected safety-related electrical equipment. This report summarizes the results of the audit.

JCN L1354, "Technical Assistance for TVA Reviews"

SUMMARY

The Idaho National Engineering Laboratory participated in an audit of the environmental qualification of safety-related electrical equipment at Watts Bar Nuclear Plant Unit 1 with a team headed by and including members of the Nuclear Regulatory Commission (NRC) staff. The audit concluded that the applicant has demonstrated conformance with the requirements for environmental qualification as detailed in 10 CFR 50.49; in the relevant parts of General Design Criteria 1, 2, 4, and 23 of Appendix A to 10 CFR 50; in the relevant parts of Sections II and XI of Appendix B to 10 CFR 50; and in NUREG-0588.

PREFACE

The INEL supplies this report as part of the "Technical Assistance for TVA Reviews." Lockheed Idaho Technologies Company, National Nuclear Operations Analysis Department, performed this review and audit inspection for the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Division of Systems Safety and Analysis, Plant Systems Branch.

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CONTENTS

AUDIT OF THE ENVIRONMENTAL QUALIFICATION OF SELECTED SAFETY-RELATED ELECTRICAL EQUIPMENT AT THE WATTS BAR NUCLEAR PLANT, UNIT NO. 1

1. INTRODUCTION

The equipment used to perform a necessary safety function must be demonstrated capable of maintaining functional operability under all service conditions postulated to occur during its installed life and for the time it is required to operate in response to any postulated accident conditions. General Design Criteria 1, 2, 4, and 23 of Appendix A (Reference 1) to 10 CFR 50 and Quality Assurance Sections III and XI of Appendix B (Reference 2) to 10 CFR 50 require this. This requirement applies to safety-related equipment located both inside and outside containment. 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important-to-Safety for Nuclear Power Plants," (Reference 3) and NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," (Reference 4) include more detailed requirements and guidance relating to the methods and procedures for demonstrating this capability. Regulatory Guide 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," (Reference 5) and NUREG-0588 endorse IEEE Standard 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," (Reference 6) which provides supplemen information about the qualification of electrical equipment.

The INEL reviewed and evaluated the Watts Bar environmental qualification (EQ) program. The INEL work effort consisted of pre-audit reviews of the Tennessee Valley Authority's (the applicant) environmental qualification submittals, audits of the applicant's central environmental qualification files for selected equipment items, and visual inspections of the equipment. The NRC staff performed an onsite audit of Unit No. 1 of the Watts Bar Nuclear Plant (WBN) from February 27, 1995, to March 10, 1995, to determine the status of the environmental qualification of selected safety-related electrical equipment. The audit team included NRC staff and contractors from the Idaho National Engineering Laboratory (INEL). Section 3 of this report provides a summary of the results of the INEL portion of the audit.

2. BACKGROUND

The NRC issued NUREG-0588 in December 1979 to promote a more orderly and systematic implementation of equipment qualification programs by industry and to provide guidance to the NRC staff for its use in ongoing licensing reviews. The positions contained in the report provide guidance on (a) how to establish environmental service conditions; (b) how to select methods considered appropriate for qualifying equipment in different areas of the plant; and (c) other specific topics, such as margin, aging, and documentation.

In February 1980, the NRC requested near term Operating License (OL) applicants to review and evaluate the environmental qualification documentation for each item of safety-related electric equipment and to identify the degree to which their qualification programs comply with the staff positions discussed in NUREG-0588. IE Bulletin 79-01B, "Environmental Qualification of Class 1E Equipment," (Reference 7) and its Supplements establish environmental qualification requirements and guidance for operating reactors and OL applicants.

The final rule on environmental qualification of electric equipment important-to-safety for nuclear power plants became effective on February 22, 1983. This rule, Section 49 of 10 CFR Part 50, specifies the requirements for demonstrating the environmental qualification of electrical equipment important-to-safety located in a harsh environment. In accordance with the requirements of 10 CFR 50.49, the electrical equipment in WBN could be qualified in accordance with the acceptance criteria delineated for Category II equipment in NUREG-0588, due to the date of the construction permit. Except as noted in Appendix A, the electrical equipment examined was qualified to NUREG-0588, Category I. The NUREG-0588 Category I qualification requirements are more conservative than the Category II qualification requirements.

In response to these requirements, the applicant submitted, "Watts Bar Nuclear Plant Unit 1, Summary Status Update Report of TVA's Compliance to 10 CFR 50.49 - Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," dated September 30, 1986. The applicant supplemented this submittal by a letter to the NRC, dated April 30, 1991 (Reference 8).

The INEL reviewed these submittals. The reviews identified deficiencies and open items in the Watts Bar Environmental Qualification program. As a result, the INEL prepared a request for additional information and submitted the request to the NRC for transmittal to the applicant. These concerns culminated in a working meeting of the INEL reviewers and the NRC with the applicant on March 5, 1992. That meeting resolved some of the deficiencies and open items. The NRC issued a Request for Additional Information on May 1, 1992. The applicant provided additional information and documented statements made in the working meeting in a letter to the NRC, dated February 17, 1993 (Reference 9). The INEL reviewed that response and found it acceptable.

Concurrently, the applicant continued updating their environmental qualification binders. That work was not complete sufficiently to support the on-site audit scheduled to begin October 31, 1994, or rescheduled to begin December 5, 1994. While the process of updating the environmental qualification binders was not complete at the time of the February 27 through March 10, 1995, audit, the binders reviewed as part of the audit had corrected the open items and deficiencies identified in the earlier review.

The scope of the audit of February 27 through March 10, 1995, included evaluating the qualification criteria that the equipment must meet and the environments in which the equipment must function, assessing the qualification documentation for the equipment in addressing the qualification criteria and environments, and assessing the physical installation of the equipment. The principal area of review was the qualification of safety-related electrical equipment that must function to prevent or mitigate the consequences of a loss-of-coolant accident (LOCA) or a high energy line break (HELB) inside or outside of containment, while subjected to the harsh environments associated with these accidents. The audit was limited to electric equipment that is part of systems tested and turned over to the applicant's operating department as operating systems.

3. EVALUATION

The evaluation of the applicant's environmental qualification program included a review of the applicants submittals regarding environmental qualification; an onsite examination of electrical equipment; audits of qualification documentation and environmental qualification documentation binders; and a review of the acceptability of the components, qualification methods, and accident environments. The criteria described in NUREG-0588 forms the basis for the evaluation of the adequacy of the applicant's qualification program. The auditors used Revision 1 of NUREG-0588 to clarify the NRC staff positions as required.

The applicant was directed to (a) establish a list of systems and components that are required to prevent or mitigate a LOCA or a HELB and (b) identify components needed to perform the function of safety-related display instrumentation, post-accident sampling and monitoring, and radiation monitoring. Based upon information in the applicant's submittal and additional information supplied, the INEL has verified and determined that the systems included in the applicant's submittal are those required to achieve or support: (a) emergency reactor shutdown, (b) containment isolation, (c) reactor core cooling, (d) containment heat removal, (e) core residual heat removal, and (f) prevention of significant release of radioactive material to the environment.

The NRC audit of the applicant's environmental qualification program occurred between the dates of February 27, 1995, and March 10, 1995. The INEL portion of the audit included a review of 27 binders of equipment qualification documentation (16 electrical equipment types and 11 cable constructions). The audit also included physical inspections of 32 pieces of installed electrical equipment and 24 associated cables that were connected to the equipment inspected. After identification of the cables, the supporting environmental qualification binder (of the 11 mentioned above) for those cables was examined. The results of the audit are discussed in Section 3.3 of this report.

The components inspected consisted of 2 level transmitters, 2 temperature elements, one hand switch, 2 solenoid valves, 3 position switches, 2 conduit seal assemblies, 2 junction boxes, one terminal block, one containment penetration, one hydrogen recombiner, 4 ventilation system drive motors, 3 pump motors, 4 motor-operated valve operators, 4 temperature switches, and 24 cables.

3.1 Qualification Methods

NUREG-0588 provides detailed procedures for qualifying safety-related electrical equipment in a harsh environment. The NUREG-0588 criteria apply to equipment that is important-to-safety as defined in 10 CFR 50.49. Type tests of identical equipment in a sequence consisting of pre-aging (thermal, radiation, and mechanical), seismic and dynamic loading, and exposure to postulated LOCA/HELB conditions (if exposed to those environments) is the preferred method of qualification. The applicant extrapolated test data, using the Arrhenius methodology, to establish the qualified life prior to a LOCA/HELB harsh environment. The INEL reviewed this approach and found the method acceptable per NUREG-0588. Where a forty-year lifetime was not established, the WBN environmental qualification program requires replacement before the end of the qualified life of the equipment.

3.2 Service Conditions

NUREG-0588 defines methods for determining the environmental conditions associated with loss-of-coolant accidents or high energy line breaks, inside or outside of containment. The review and evaluation of the adequacy of these environmental conditions are described below. The qualification documentation was reviewed to assure the qualification conditions envelop the conditions established by the applicant.

3.2.1 Temperature, Pressure, and Humidity Conditions

The applicant determined the normal operating environments, the short-term abnormal environments, and the LOCA/HELB profiles used for equipment qualification. The application and methodology employed to determine these values were presented to the applicant in NUREG-0588. Reviewing the derivation of these postulated environments is outside the scope of the INEL review. The 47E235- series of drawings contains these conditions and profiles. For example, the normal temperature ranges between 60°F and 104°F (90°F average) in fan room 1, elevation 719 feet, 9 inches, (the location of 1-LT-063-0181-E, a containment level transmitter). The worst case abnormal temperature is 110°F for up to 8 hours. The applicant used the normal and abnormal conditions and applicable margins to set the criteria for aging the test samples. The peak calculated temperature from a LOCA is 327°F at the location of this transmitter. This temperature was exceeded in the LOCA simulation after accelerated aging of the equipment. The applicant's program requires aging and accident tests to envelope the required values or an analysis or evaluation to support the qualification in lieu of the enveloping test. The INEL determined this approach is acceptable.

3.2.2 Submergence

The applicant evaluated the effects of flooding on equipment to ensure that safe shutdown can be achieved. Reviewing the derivation of these postulated flood levels is outside the scope of the INEL review. The applicant has located all electrical equipment examined above the potential flood levels identified in the 47E235- series of drawings. For example, the potential flood level in fan room 1 (the location of 1-LT-063-0181-E) is 717.9 feet. This potential flood level is 1.85 feet below the bottom of the transmitter. Thus, this transmitter will not become submerged. The INEL finds the applicant's approach to submergence meets the NUREG-0588 criteria and is, therefore, acceptable.

3.2.3 Chemical Spray

Chemical spray is used for containment heat removal following a design basis accident. The applicant evaluated the effects of chemical spray impingement on electrical important-to-safety equipment. Chemical spray effects were included in the environmental qualification testing of in-containment equipment. For example, in fan room 1 (the location of 1-LT-063-0181-E) the effect of chemical spray on the transmitter was tested for 24-hours. The results of that testing was extended to 30-days by analysis. All environmental qualification binders examined either included testing to simulate the postulated chemical spray or provided details that established that the equipment would not be exposed to chemical spray. The INEL finds the applicant's approach to simulating chemical spray meets the NUREG-0588 criteria and is, therefore, acceptable.

3.2.4 Aging

The aging program requirements for WBN electrical equipment are defined in NUREG-0588. The applicant included the degrading influences of temperature, radiation, vibration, and mechanical stresses in the aging program. This requires the establishment of a qualified life with maintenance and replacement schedules based on the findings. For example, the test sequence for limit switch 1-ZS-081-0012B-A, PW RCS Pressurizer Relief Tank and RCP SP-VLV Position Switch conforms to the IEEE Standard 323-1974 test sequence. This test sequence is permissible per NUREG-0588. As such, this limit switch is qualified for a 40-year life at its location that has a maximum service temperature of 104°F. All environmental qualification binders examined conformed to such a test sequence. The INEL finds the applicant's approach to simulated equipment aging meets the NUREG-0588 criteria and is, therefore, acceptable.

3.2.5 Radiation (Inside and Outside Containment)

The applicant has determined the radiation levels postulated to exist following a LOCA. The application and methodology employed to determine these values were presented to the applicant in NUREG-0588. Reviewing the derivation of these postulated radiation levels is outside the scope of the INEL review. These radiation levels, which vary depending on location, are included in the 47E235- series of drawings for both inside primary containment and outside of primary containment in areas exposed to recirculating fluid lines.

The maximum total integrated radiation doses specified by the applicant are also location dependent. They are also included in the 47E235- series of drawings for both inside primary containment and outside of primary containment in areas exposed to recirculating fluid lines. The values contained in the drawing are acceptable for use in the qualification of electrical equipment important-to-safety. In all cases examined, the total integrated radiation dose applied to the test sample during testing was greater than the total integrated radiation dose analyzed for that location plus additional margin. The INEL finds the applicant's approach to radiation exposure meets the NUREG-0588 criteria and is, therefore, acceptable.

3.3 Environmental Qualification Audit

An audit was conducted of the applicant's qualification documentation and installed equipment between February 27, 1995 and March 10, 1995. The INEL staff reviewed 32 equipment items and 24 associated cables to determine that the test data and analyses in the files supported the qualification status determined by the applicant.

The equipment items selected for audit were:

TABLE 1 -- Audited Equipment and Supporting Binders

Plant ID Number & Use	Equipment	EQ Binder
1-LT-063-0181-E, Containment Sump Level	Barton dP Transmitter 764 with 351 Remote Sensor	WBNEQ-XMTR-001
1-LT-063-0182-F, Containment Sump Level	Barton dP Transmitter 764 with 351 Remote Sensor	WBNEQ-XMTR-001
1-TE-072-0006, CS Heat Exchanger B Outlet Temperature	Weed Instruments Model 612-1B-A-4C-13.25-0-0 Resistance Temperature Detector	WBNEQ-ITE-006
1-HS-074-0003B-A, Hand Switch, RHR Pump 1A-A Inlet Flow Control Valve	Cutler Hammer 10250T Hand Switch	WBNEQ-HS-002
1-TE-074-0039, RHR Heat Exchanger B Outlet Temperature	Weed Instruments Model 612-1B-A-4C-13.25-0-0 Resistance Temperature Detector	WBNEQ-ITE-006
1-FSV-081-0012-A, PW RCS Pressurizer Relief Tank	ASCO Solenoid Valve NP831654E	WBNEQ-SOL-006
1-ZS-081-0012A-A, Valve Position Switch	NAMCO Controls EA-180-14302	WBNEQ-IZS-005
1-ZS-081-0012B-A, Valve Position Switch	NAMCO Controls EA-180-15302	WBNEQ-IZS-005
1-CSC-072-0006, Conduit Seal for 1-TE-72-6	NAMCO Controls Quick Disconnect Conduit Seal, Series EC210-34000	WBNEQ-CSC-002
1-CSC-081-0012-A, Conduit Seal for 1-FSV-81-12-A	NAMCO Controls Quick Disconnect Conduit Seal, Series EC210-34000	WBNEQ-CSC-002
1-FSV-030-146A-A, Aux. Bldg. Gas Treatment Sys. (ABGTS) Fan A-A Exhaust Damper	ASCO Solenoid Valve 206-380-2RVU	WBNEQ-SOL-005
1-ZS-030-0146A-B, ABGTS Fan A-A Exhaust Damper Position Switch	NAMCO EA740-20100 Limit Switch	WBNEQ-IZS-003
1-JB-293-4340-G, Junction Box (two cable splices)	Raychem Corporation WCSF-N, NPKC, NPKS, NPK, NPKV, NMCK, NCBK, NESK, and NEIS cable splices (Junction Box is not required to be qualified)	WBNEQ-SPLC-001
1-JB-293-3201-B, Junction Box	JVB (10"x10"x6", NEMA 4)	WBNEQ-JBOX-001

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Plant ID Number & Use	Equipment	EQ Binder
1-TBLK-293-3201-B, Terminal Block	General Electric CR151B	WBNEQ-JBOX-001
1-PENT-293-27A, Primary Containment Electrical Penetration	Conax 7429-10002-05	WBNEQ-PENT-002
2-3PL-30-3753A, connected to 1-FSV-30-146A-A	American Insulated Wire Corporation PXMJ Cable, EPDM	WBNEQ-CABL-002
1V-9245-A, connected to 1-FSV-81-12-A	American Insulated Wire Corporation PXMJ Cable, FR-XLPE	WBNEQ-CABL-003
1V-9246-A, connected to 1-ZS-81-12B-A	American Insulated Wire Corporation PXMJ Cable, FR-XLPE	WBNEQ-CABL-003
1PM1930, connected to 1-TE-72-06	Rockbestos FR-XLPE, Type MS, Firewall III, KXL-760-D insulation	WBNEQ-CABL-037
1V-6211A, connected to 1-PENT-293-27A	Okonite Power and Control Cable, PXMJ, FR-CSPE	WBNEQ-CABL-051
1-3V-043-9963B, connected to 1-TBLK-293-3201-B	Rockbestos Cable, PXMJ, XLPE (irradiation), KXL-760G insulation	WBNEQ-CABL-052
1-3V-043-9973B, connected to 1-TBLK-293-3201-B	Rockbestos Cable, PXMJ, XLPE (irradiation), KXL-760G insulation	WBNEQ-CABL-052
1-3A-30-6206, connected to 1-ZS-030-0146AB	Rockbestos Cable, PXMJ, XLPE (irradiation), KXL-760G insulation	WBNEQ-CABL-052
1V-4919A, connected to 1-PENT-293-27A	Rockbestos Cable, PXMJ, XLPE (irradiation), KXL-760G insulation	WBNEQ-CABL-052
1V-8768A, connected to 1-PENT-293-27A	Rockbestos Cable, PXMJ, XLPE (irradiation), KXL-760G insulation	WBNEQ-CABL-052
1-2PM-68-0822G, connected to 1-TE-68-79A-G	Rockbestos YPS thermocouple extension lead wire cable, FR-XLPE, KXL-760G insulation	WBNEQ-CABL-055
1-2PM-68-0832G, connected to 1-TE-68-79B-G	Rockbestos YPS thermocouple extension lead wire cable, FR-XLPE, KXL-760G insulation	WBNEQ-CABL-055

Plant ID Number & Use	Equipment	EQ Binder
1-HTR-083-0001-A, Containment Hydrogen Recombiner A	Westinghouse Hydrogen Recombiner, Serial 61	WBNEQ-HTR-001
1-MTR-030-0146-A, ABGTS Exhaust Fan 1A Motor	Reliance Wound Motors - Outside Containment, Serial 1YF882365A1-YC	WBNEQ-MOT-003
1-MTR-030-0175-A, RHR Pump 1A-A Room Cooler Motor	Reliance Wound Motors - Outside Containment, Serial 1YF883397A4-VK	WBNEQ-MOT-003
1-MTR-030-0177-A, CS Pump 1A-A Room Cooler Motor	Reliance Wound Motors - Outside Containment, Serial 2YF883397A1-VK	WBNEQ-MOT-003
1-MTR-030-0180-A, SI Pump 1A-A Room Cooler Motor	Reliance Wound Motors - Outside Containment, Serial 1YF883397A1-VK	WBNEQ-MOT-003
1-MTR-063-0010-A, SI Pump 1A-A Motor	Westinghouse Motors on SIS, CS, and RHR, HSDP, Serial 1S-76	WBNEQ-MOT-001
1-MTR-072-0010-B, CS Pump 1B-B Motor	Westinghouse Motors on SIS, CS, and RHR, HSW2, Serial 1S-77	WBNEQ-MOT-001
1-MTR-074-0010-A, RHR Pump 1A-A Motor	Westinghouse Motors on SIS, CS, and RHR, VSW1, Serial 1S-75	WBNEQ-MOT-001
1-MVOP-063-0172-B, RHR to RCS Hot Leg 1 & 3 Flow Isolation Valve Operator	Limitorque Motorized Valve Operator with Type RH Insulation, SB-2, Serial 192138	WBNEQ-MOV-001
1-MVOP-074-0003-A, RHR Pump 1A-A Suction Shutoff Valve Operator	Limitorque Motorized Valve Operator with Type RH Insulation, SB-2, Serial 197000	WBNEQ-MOV-001
1-MVOP-074-0008-A, RHR System Isolation Bypass Valve Operator	Limitorque Motorized Valve Operator with Type RH Insulation, SB-1, Serial 241211	WBNEQ-MOV-001
1-MVOP-074-0012-A, RHR Pump 1A-A Minimum Flow Valve Operator	Limitorque Motorized Valve Operator, SMB-000, Serial 358806	WBNEQ-MOV-003
1-TS-030-5236A-A, RHR Pump 1A-A Room Ambient Air Temperature Switch A	Static-O-Ring Incorporated Temperature Switch (Vented), 201TA-BB125-JJTTX6, Serial 93-11-7022	WBNEQ-ITS-002

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Plant ID Number & Use	Equipment	EQ Binder
1-TS-030-5236B-B, RHR Pump 1A-A Room Ambient Air Temperature Switch B	Static-O-Ring Incorporated Temperature Switch (Vented), 201TA-BB125-JJTTX6, Serial 93-11-7018	WBNEQ-ITS-002
2-TS-030-0200A, EGTS 2A-A Room Ambient Air Temperature Switch A	Static-O-Ring Incorporated Temperature Switch (Unvented), 201TA-BB125-JJTTX6, Serial 85-3-3451	WBNEQ-ITS-002
2-TS-030-0207B, EGTS 2A-A Room Ambient Air Temperature Switch B	Static-O-Ring Incorporated Temperature Switch (Unvented), 201TA-BB125-JJTTX6, Serial 85-3-3458	WBNEQ-ITS-002
1PL 1062A, Power Cable for 1-HTR-083-0001-A	Brand-Rex Co., PXJ, PXMJ	WBNEQ-CABL-050
1PL 2981A, Power Cable for 1-MTR-030-0180-A	Triangle-Plastic Wire and Cable (PWC), CPJ, CPJJ	WBNEQ-CABL-032[1]
1PP 575A, Power Cable for 1-MTR-074-0010-A	Anaconda Wire and Cable Co., EPSJ	WBNEQ-CABL-005
1V 2346B[2], Control Cable for 1-MVOP-063-0172-B	Cyprus Wire and Cable Co., PJJ (1)	WBNEQ-CABL-015
1V 2343B[2], Power Cable for 1-MVOP-063-0172-B	Rockbestos Co., PXJ, PXMJ	WBNEQ-CABL-053
1-V-1922A[2], Control Cable for 1-MVOP-074-0003-A	Cyprus Wire and Cable Co., PJJ (1)	WBNEQ-CABL-015
1-V-1920A[2], Power Cable for 1-MVOP-074-0003-A	Rockbestos Co., PXJ, PXMJ	WBNEQ-CABL-053
1-3V-074-1218A[2], Control Cable: 1-MVOP-074-0008-A	Rockbestos Cable, PXMJ, XLPE (irradiation), KXL-760G insulation	WBNEQ-CABL-052
1V-2144A[2], Power Cable for 1-MVOP-074-0008-A	Rockbestos Cable, PXMJ, XLPE (irradiation), KXL-760G insulation	WBNEQ-CABL-052
1-3V-74-1937A[2], Control Cable: 1-MVOP-074-0012-A	Triangle-Plastic Wire and Cable (PWC), CPJ, CPJJ	WBNEQ-CABL-032[1]
1V 1935A[2], Power Cable for 1-MVOP-074-0012-A	Okonite Power and Control Cable, PXMJ, FR-CSPE	WBNEQ-CABL-051
2PL 3778A, Signal Cable for 2-TS-030-0200A	Rockbestos Cable, PXMJ, XLPE (irradiation), KXL-760G insulation	WBNEQ-CABL-052

NOTES FOR TABLE 1:

1. Revision 6, dated 1/26/95, to environmental qualification Binder WBNEQ-CABL-032 put a limited qualified life of 24.14 yr after initial criticality on the PWC cables, but failed to present a replacement maintenance requirement in the Qualification Maintenance Data Sheet (QMDS) in Tab G. Revision 7, issued during the inspection, corrected this.

2. The inspection verified the environmental qualification of only the power and control cables of each inspected Limitorque motor operator. WBN environmental qualification personnel said these two cables are the only internal wiring requiring qualification for functional operability of a motor-operated valve during a LOCA. In addition, the inspection verified correct hookup of these cables for 1-MVOP-063-0172-B, according to drawings made available for the inspection.

These files were reviewed to determine if qualification had been demonstrated based on the documents contained in the files. Appendix A contains details on these reviews. Based on these documents and additional information supplied by the applicant, the INEL staff determined there is adequate documentation establishing the qualification of the inspected equipment as claimed in all audited cases.

As part of the audit, the equipment as actually installed was inspected during a plant walkdown. The purpose of the walkdown was to verify manufacturer, model number, location, orientation (where such a requirement was observed in the environmental qualification testing), and proper installation consistent with the qualification documentation. No violations were discovered.

3.3.1 Temperature, Pressure, and Humidity Conditions

For all equipment items examined, the auditors found the documented testing conditions included the postulated environments plus additional margins. For example, after artificial aging samples of Rockbestos Flame Retardant XLPE, Type MS, Firewall III, Chemically Cross-linked Polyethylene, Insulation Type KXL-760-D, cable, saturated steam was used to bring the test chamber to 341°F at 121.9 psig for 105 days. This testing encompassed, with margin, the normal maximum temperature of 110°F and the maximum postulated accident temperature of 215°F. It is noted that this qualification is not for inside containment and other specified areas of the plant. The INEL finds the implementation of this step in the applicant's program for establishing the environmental qualification of electrical equipment meets the NUREG-0588 criteria and is, therefore, acceptable.

3.3.2 Submergence

All examined equipment was located above the calculated flood levels at the component location. None of the inspected equipment or cables are subject to submergence following a postulated event. For example, the potential flood level at the location of ASCO Model 260-380-2RVU, 1-FSV-030-0146A-A, Auxiliary Building Gas Treatment Fan A-A Exhaust Damper Solenoid Valve is 738 feet, 11 inches. The solenoid valve is located at 751 feet elevation. The potential flood level is 12 feet, 1 inch, below the bottom of the solenoid valve. Thus, this solenoid valve will not become submerged. The INEL finds the implementation of this step in the applicant's program for establishing the environmental qualification of electrical equipment meets the NUREG-0588 criteria and is, therefore, acceptable. Locating electrical equipment above the postulated flood level is acceptable.

3.3.3 Chemical Spray

For the equipment items examined that could be exposed to chemical spray, the auditors found that the documented testing included the simulation of the chemical spray with a solution that encompassed the pH and buffering of the chemical spray solution. This satisfies the chemical spray step in establishing the environmental qualification of the equipment. For equipment items that would not be exposed to chemical spray, no testing to simulate a chemical spray is required. The INEL finds the implementation of this step in the applicant's program for establishing the environmental qualification of electrical equipment meets the NUREG-0588 criteria and is, therefore, acceptable.

3.3.4 Aging

The applicant used the Arrhenius equation in establishing the simulated age of the component tested. This methodology (using test conditions that are more severe than worst case operating temperature plus additional margin) establishes the life of equipment qualification prior to a LOCA/HELB harsh environment. The difference between the test temperature and the worst case operating temperature plus additional margin is used by the Arrhenius equation to extend the test interval to a qualified life based on the activation energy of the most limiting component of the equipment. The INEL finds the applicant's implementation of simulated aging in their program for environmental qualification of electrical equipment meets the NUREG-0588 criteria and is, therefore, acceptable.

3.3.5 Radiation (Inside and Outside Containment)

The maximum total radiation dose specified by the applicant, included in the 47E235- series of drawings, for both inside primary containment and outside primary containment in areas exposed to recirculating fluid lines, plus additional margins, were enveloped by the total integrated dose provided in the radiation testing of the equipment. For example, ASCO Solenoid Valve Model NP831654E, 1-FSV-081-0012-A, had radiation aging simulation of 50 megarads gamma (not exceeding 1 megarad/hour) to simulate non-accident radiation exposure. Later in the test sequence, 150 megarads gamma (not exceeding 1 megarad/hour) simulated the accident radiation exposure. The maximum calculated accident radiation exposure at the equipment location is 136 megarads gamma. Thus, the test included the expected radiation level plus margin. All examined equipment radiation testing encompassed the expected radiation levels at their location. Thus, the INEL finds the radiation testing provided is acceptable in establishing the environmental qualification of the equipment.

3.3.6 Inspection Observations

The auditors noted that procedures require a degradation inspection for Class 1E components when opening and before closing the component. That inspection includes steps to monitor and correct the following:

- Cleanliness, freedom from debris, trash, foreign materials, dirt, dust, and shavings.
- Assessment or replacement of gaskets.
- Terminal blocks - verification of manufacturer and type, proper mounting, and coating of the terminations with Dow-Corning RTV-3140 in containment and high moisture areas is complete.
- Junction boxes - verification of two open 1/4-inch weep holes in the bottom of the box and qualified wire is used (Rockbestos Firewall SIS).

The inspection includes observations for rust, corrosion, moisture, excessive dust or dirt, pliability, brittleness, cracks, or deformation; damaged, broken, loose, missing, or improperly installed parts; evidence of arcing or overheating; damage to cable jackets, insulation, conductors, shielding, or splices; and leakage or seepage of lubricants.

A degradation inspection noted the Dow Corning RTV-3140 coating of terminal block 1-TB-293-3201-B was not uniform nor applied to coat all terminations. The applicant's degradation inspector issued a work request to correct the condition. In another instance, flakes of paint in the bottom of junction box 1-JB-293-4360-E were cleaned out before the box was secured. Another degradation inspection noticed a spare wire with a Raychem end cap that was not appropriate for the application in Limitorque valve motor operator 1-MVOP-063-0172-B. The applicant's degradation inspector issued a work request to change the end cap from one for rubber wire insulation to one for plastic wire insulation at that location and to inspect for and correct similar conditions in other Limitorque operators. The applicant is performing degradation inspections as directed by procedures. Degraded conditions found during these inspections are corrected. From these observations, it is apparent that the program for degradation inspection is an acceptable compliment to the environmental qualification program.

As part of the audit, the installed equipment was inspected during plant walkdowns. The purpose of the walkdowns was to verify manufacturer, model number, serial number, location, and proper installation consistent with the qualification documents, and that no damage to the equipment was evident. For the 32 pieces of electrical equipment and 24 associated cables, each of these attributes was verified and found consistent with the qualification documentation. No violations were discovered.

4. CONCLUSION

The Watts Bar Nuclear Plant, Unit No. 1, program for the environmental qualification of safety-related electrical equipment has been examined. This review included the environmental conditions resulting from design basis accidents, the methods used for qualification, and the documentation for specific items of equipment. Based on the results of this limited audit, it is concluded that the applicant's environmental qualification program conforms with the requirements of 10 CFR 50.49 and relevant parts of General Design Criteria 1, 2, 4, and 23 of Appendix A, Sections III and XI of Appendix B, 10 CFR 50, and the criteria specified in NUREG-0588.

5. REFERENCES

1. Code of Federal Regulations, 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

2. Code of Federal Regulations, 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

3. Code of Federal Regulations, 10 CFR 50.49, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants."

4. "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," NUREG-0588, Revision 1, July 1981.

5. "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," U. S. Nuclear Regulatory Commission, Regulatory Guide 1.89, Revision 1, June 1984.

6. "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," IEEE Std 323-1974.

7. U. S. Nuclear Regulatory Commission IE Bulletin 79-01B, "Environmental Qualification of Class 1E Equipment," January 14, 1980, and Supplements dated February 29, 1980, September 30, 1980, and October 24, 1980.

8. "Watts Bar Nuclear Plant, (Unit 1), Summary Status Update Report of TVA's Compliance to 10 CFR 50.49 - Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants" September 1986, supplemented by a letter to the NRC dated April 30, 1991.

9. Letter, Tennessee Valley Authority (W. J. Museler) to NRC, "Watts Bar Nuclear Plant (WBN) - Response to NRC Request for Additional Information (RAI) - Environmental Qualification (EQ) Special Program," February 17, 1993.

10. "IEEE Standard for Type Test of Class 1E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations," IEEE Std 383-1974.

11. "Qualification Tests of Electric Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants," U. S. Nuclear Regulatory Commission, Regulatory Guide 1.131, August 1977.

APPENDIX A
REVIEW OF WATTS BAR EQ DOCUMENTATION FILES

APPENDIX A

REVIEW OF WATTS BAR EQ DOCUMENTATION FILES

EQ Binder	WBNEQ-CABL-002
Equipment Type	ethylene propylene diene monomer (EPDM) insulated cable
Manufacturer	American Insulated Wire Corporation
Model Number	PXJ and PMXJ
Plant ID Number	2-3PL-30-3753A

The environmental qualification of this cable is documented in EQ Binder WBNEQ-CABL-002. This cable is spliced to 1-FSV-30-146A-A, located in the auxiliary building, room A05, at an elevation of 751 feet. The flood level in room A05 is 716 feet, 2 inches. With a flood level 23 inches above the floor, flooding of this cable is not postulated. The normal ambient temperature ranges up to 104°F. The peak accident temperature for this location is 119°F. The applicant's evaluation utilizes this data in qualifying these components.

The samples were artificially aged with the test sequence outlined in IEEE Standard 323-1974. The radiation dose exposure of 9.2×10^7 rads covered the accident and normal service conditions, with margin. A DBE exposure simulation encompassed the limiting DBE temperature, pressure, and 100% relative humidity. While a chemical spray is not required outside containment, it was simulated in the testing of this cable. The temperature test profile lasted 34 days with a peak temperature of 381°F at 89.7 psig, demonstrating margin. The binder included an analysis that extends the testing results to the required 100 days.

It is concluded that the American Insulated Wire PXJ and PXMJ EPDM-insulated cables, as documented in EQ Binder WBNEQ-CABL-002, are qualified for the parameters specified and the applicant has adequate documentation in this binder to support that conclusion.

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REVIEW OF WATTS BAR EQ DOCUMENTATION FILES

EQ Binder	WBNEQ-CABL-003
Equipment Type	ethylene propylene diene monomer (EPDM) insulated cable
Manufacturer	American Insulated Wire Corporation
Model Number	PXJ and PMXJ (TVA Type WHB)
Plant ID Number	1V-9245-A and 1V-9246-A

The environmental qualification of this cable is documented in EQ Binder WBNEQ-CABL-003. Cable 1V-9245-A is spliced to 1-FSV-081-0012-A, located in the auxiliary building, A28 pipe chase room, at an elevation of 716 feet, 6 inches. Cable 1V-9246-A is spliced to 1-ZS-081-0012B-A, located in the auxiliary building, A28 pipe chase room, at an elevation of 715 feet. The flood level in the A28 pipe chase room is 713 feet, 3 inches. Therefore, flooding of this cable is not postulated. The normal ambient temperature ranges up to 104°F. The peak accident temperature for this location is 110°F. The temperature will remain above 104°F for up to 30 days. The applicant's evaluation utilizes this data in qualifying these components.

The samples were artificially aged with the test sequence outlined in IEEE Standard 323-1974. The radiation dose exposure of 6.5×10^7 rads covered the accident and normal service conditions, with margin. A DBE exposure simulation encompassed the limiting DBE temperature, pressure, and 100% relative humidity, with margin. A chemical spray is not required outside containment.

It is concluded that the American Insulated Wire TVA Type WHB cable, as documented in EQ Binder WBNEQ-CABL-003, is qualified for the parameters specified and the applicant has adequate documentation in this binder to support that conclusion.

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REVIEW OF WATTS BAR EQ DOCUMENTATION FILES

EQ Binder	WBNEQ-CABL-005
Equipment Type	Power Cable with EPR insulation for 1-MTR-074-0010-A
Manufacturer	Anaconda Wire and Cable Company
Model Number	TVA Type EPSJ
Plant ID Number	1PP 575A

EQ Binder WBNEQ-CABL-005 documents the environmental qualification of this power cable to the RHR Pump 1A-A Motor. The pump moves water through a closed cooling loop including the reactor vessel and a heat exchanger to remove residual nuclear decay heat from the reactor core following a shutdown. It is located in Auxiliary Building Room A11, elevation 676 feet. The maximum flood level inside the room is 7 inches above the floor. The power cable enters the motor housing through a conduit entering from above the motor. Since the conduit enters the room near the ceiling, the cable is not subject to submergence. The normal ambient temperature ranges between 60°F and 104°F. The peak accident temperature is 110°F. The applicant's evaluation uses this data in qualifying this cable.

The tested cable was artificially aged with the test sequence outlined in IEEE Standard 323-1974. The radiation dose exposure of $2x10^8$ rads covered the accident and normal service conditions, with margin. A DBE exposure simulation included steam exposure and encompassed the limiting DBE temperature, pressure, and humidity. The test profile included 30 days at 346°F, 75 psig, and 100%, demonstrating qualification with margin. Chemical spray is not postulated for this location.

The INEL concluded that the RHR Pump 1A-A power cable, documented in EQ Binder WBNEQ-HTR-001, is qualified for the parameters specified. The applicant has adequate documentation in this binder to support that conclusion.

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REVIEW OF WATTS BAR EQ DOCUMENTATION FILES

EQ Binder	WBNEQ-CABL-015
Equipment Type	Control Cables with PE insulation to 1-MVOP-063-0172-B and 1-MVOP-074-0003-A
Manufacturer	Cyprus Wire and Cable Company
Model Number	TVA Type PJJ (1)
Plant ID Number	1V 2346B and 1-V-1922A, respectively

EQ Binder WBNEQ-CABL-015 documents the environmental qualification of these motor operated valve (MOV) operator control cables. Valve operator 1-MVOP-063-0172-B sits atop the RHR to RCS Hot Leg 1 & 3 flow isolation valve, 16 feet, 4 inches, above the floor in Auxiliary Building Pipe Chase A28, elevation 713 feet. The maximum flood level inside the pipe chase is 1 inch above the floor. Valve operator 1-MVOP-074-0003-A is on top of the RHR Pump 1A-A Suction shutoff valve, and is located some 12 feet above the floor in Auxiliary Building Room A11, elevation 676 feet. The maximum flood level in the room is 7 inches above the floor. Thus, the cables are not subject to submergence. The normal ambient temperatures range between 60°F and 104°F. The peak accident temperatures are 110°F. The applicant's evaluation uses these data in qualifying these cables.

The tested cable was artificially aged with the test sequence outlined in NUREG-0588 Category II. The radiation dose exposure of $6.5x10^7$ rads covered the accident and normal service conditions, with margin. A DBE exposure simulation included steam exposure and encompassed the limiting DBE temperature, pressure, and humidity. The test profile included 319 hours at 340°F, 30.4 psig, and 100%, demonstrating margin. Chemical spray is not postulated for these locations.

The INEL concluded that these MOV control cables, documented in EQ Binder WBNEQ-CABL-015, are qualified for the parameters specified. The applicant has adequate documentation in this binder to support that conclusion.

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REVIEW OF WATTS BAR EQ DOCUMENTATION FILES

EQ Binder	WBNEQ-CABL-032
Equipment Type	Plastic Power Cable for 1-MTR-030-0180-A and Plastic Control Cable for 1-MVOP-074-0012-A
Manufacturer	Triangle-Plastic Wire and Cable (PWC)
Model Number	TVA Type CPJ, CPJJ
Plant ID Number	1PL 2981A and 1-3V-74-1937A, respectively

EQ Binder WBNEQ-CABL-032 documents the environmental qualification of these PWC cables. Cable 1PL 2981A provides power to the motor of the room cooler in Room A13, elevation 692 feet, of the Auxiliary Building where SI Pump 1A-A is located. The room cooler maintains proper room ambient temperature for the operating pump and the pump motor. The cable conduit enters from above and through the top of the electrical connection housing on the side of the room cooler motor. The motor is mounted on the top of the room cooler housing, 89 inches above the floor of the room. The maximum flood level in the room is 2 inches above the floor. Thus, the cable is not subject to submergence. The normal ambient temperature ranges between 60°F and 104°F. The peak accident temperature is 110°F. Cable 1-3V-74-1937A is the control cable for 1-MVOP-074-0012 A, the motor operator for the RHR Pump 1A-A Minimum Flow Valve. Motor operator 1-MVOP-074-0012-A is mounted on the minimum flow isolation valve for RHR Pump 1A-A, located over 10 feet above the floor in Pipe Chase A16, elevation 676 feet, in the Auxiliary Building. The control cable is hooked to the operator from above through flexible conduit that enters the room near the ceiling. Since the maximum flood level in the room is 7 inches above the floor, the cable is not subject to submergence. Again, the normal ambient temperature ranges between 60°F and 104°F, and the peak accident temperature is 110°F. The applicant's evaluation uses this data in qualifying these PWC cables.

The tested cable was artificially aged with the test sequence generally outlined in IEEE Standard 323-1974 and according to the criteria of NUREG-0588 Category II. The radiation dose exposure of 7.5×10^7 rads covered the accident and normal service conditions, with margin. A DBE exposure simulation included steam exposure and encompassed the limiting DBE temperature, pressure, and humidity, with margin. The respective test profile included 13 days at 240°F, 1.9 psig, and 95-100%. Chemical spray not is postulated for these locations.

The INEL concluded that the cables documented in EQ Binder WBNEQ-CABL-032 are qualified for the parameters specified. The applicant has adequate documentation in this binder to support that conclusion.

NOTE: Revision 6, dated 1/26/95, to EQ Binder WBNEQ-CABL-032 put a limited qualified life of 24.14 yr after initial criticality on the PWC cables, but failed to present a replacement maintenance requirement in the Qualification Maintenance Data Sheet (QMDS) in Tab G. Revision 7, issued during the inspection, corrected this.

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EQ Binder	WBNEQ-CABL-037
Equipment Type	Chemically cross-linked polyethylene (XLPE) insulated cable
Manufacturer	Rockbestos
Model Number	KXL-760-D Type MS
Plant ID Number	1PM1930

The environmental qualification of this cable is documented in EQ Binder WBNEQ-CABL-037. Cable 1PM1930 is spliced to 1-TE-072-0006, located in the auxiliary building, room A11, at an elevation of 717 feet, 6 inches, and above. The flood level in room A11 is 714 feet, 10 inches. Therefore, flooding is not postulated. The normal ambient temperature ranges up to 120°F. The peak accident temperature for this location is 327°F. The temperature would be above 120°F for up to 30 days. The applicant's evaluation utilizes this data in qualifying these components.

The samples were artificially aged with the test sequence outlined in IEEE Standard 323-1974. The radiation dose exposure of 1.8489×10^8 rads gamma covered the accident and normal service conditions, with margin. A DBE exposure simulation used saturated steam and encompassed the limiting DBE temperature, pressure, and 100% relative humidity, with margin. The peak test temperature was 341°F at 121.9 psig. The DBE testing lasted for 105 days with a final temperature of 229°F, demonstrating margin throughout the test period. A simulated chemical spray was included in the testing sequence.

It is concluded that the Rockbestos KXL-760-D, Type MS cable, as documented in EQ Binder WBNEQ-CABL-037, is qualified for the parameters specified and the applicant has adequate documentation in this binder to support that conclusion.

REVIEW OF WATTS BAR EQ DOCUMENTATION FILES

EQ Binder WBNEQ-CABL-050

Equipment Type High Voltage Power Cable for 1-HTR-083-0001-A

Manufacturer Brand-Rex Company

Model Number TVA type PXJ, PXMJ

Plant ID Number 1PL 1062A

EQ Binder WBNEQ-CABL-050 documents the environmental qualification of this power cable to the hydrogen recombiner. The equipment recombines hydrogen with oxygen to produce water to prevent hydrogen explosion inside primary containment following a LOCA. It is located on a platform inside the polar crane wall at an elevation of 779 feet, 11 inches. The maximum flood level inside the crane wall reaches an elevation of 722 feet. Thus, the cable is not subject to submergence. The normal ambient temperature ranges between 60°F and 120°F. The peak accident temperature is 327°F. The applicant's evaluation uses this data in qualifying this equipment.

The tested equipment was artificially aged with the test sequence outlined in IEEE Standard 323-1974. The radiation dose exposure of 2.2×10^{8} rads covered the accident and normal service conditions, with margin. A DBE exposure simulation included steam exposure and encompassed the limiting DBE temperature, pressure, and humidity, with margin. Chemical spray is postulated for this location.

The INEL concluded that the cable documented in EQ Binder WBNEQ-CABL-050 is qualified for the parameters specified. The applicant has adequate documentation in this binder to support that conclusion.

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REVIEW OF WATTS BAR EQ DOCUMENTATION FILES

EQ Binder	WBNEQ-CABL-051
Equipment Type	Flame resistant XLPE insulation with CSPE overall jacket cable
Manufacturer	Okonite
Model Number	PMXJ
Plant ID Number	1V6211A

The environmental qualification of this cable is documented in EQ Binder WBNEQ-CABL-051. Cable 1V6211A splices to penetration 1-PENT-293-0027-A in the reactor annulus at an elevation of 728 feet. Flooding of this cable is not postulated at this elevation. The normal ambient temperature ranges up to 140°F. The peak accident temperature for this location is 419°F. The applicant's analysis shows the temperature inside the conduit and junction boxes where this cable is routed peaks at 340°F. The applicant's evaluation utilizes this data in qualifying these components. The use of this cable is limited to non-power applications, limiting the self-generated conductor heat rise.

The samples were artificially aged with the test sequence outlined in IEEE Standard 323-1974. The radiation dose exposure of 2.014×10^8 rads gamma ±8% covered the accident and normal service conditions, with margin. A DBE exposure simulation used saturated steam, encompassed the limiting DBE temperature, pressure, and 100% relative humidity. The peak temperature was 341°F at 112 psig. The DBE testing lasted for 130 days with a final temperature of 212°F. A simulated chemical spray was included in the testing sequence even though not needed for this application. The applicant's binder for this cable notes the 1°F margin between the postulated accident temperature and the peak test temperature, and provides justification for that small margin. The required temperature profile is for 620 seconds, from 140°F to 453°F to 140°F, resulting in a maximum required duration at 340°F of less than 7 minutes. The peak test temperature was maintained for 175 minutes.

The binder also contained a separate engineering report demonstrating qualification to main steamline break temperatures of 455°F, well above the 340°F required of this application.

It is concluded that the Okonite Type PMXJ cable, as documented in EQ Binder WBNEQ-CABL-051, is qualified for the parameters specified and the applicant has adequate documentation in this binder to support that conclusion.

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REVIEW OF WATTS BAR EQ DOCUMENTATION FILES

EQ Binder	WBNEQ-CABL-052
Equipment Type	Power and control cable, XLPE (irradiation)
Manufacturer	Rockbestos
Model Number	PXMJ
Plant ID Number	1-3V-043-9963B, 1-3V-043-9973B, 1-3A-30-6206, 1V-4919A, and 1V-8768A

The environmental qualification of these cables is documented in EQ Binder WBNEQ-CABL-052. Cables 1-3V-043-9963B, 1-3V-043-9973B, 1V-4919A, and 1V-8768A are located in containment, are installed in conduit, and are not subject to flooding. Cable 1-3A-30-6206 is located in room A05 of the auxiliary building, is installed in conduit, and is not subject to flooding. The normal ambient temperature for these locations ranges up to 140°F. The peak accident temperature for this location is 327°F. The applicant's evaluation utilizes this data in qualifying these components.

The samples were artificially aged with the test sequence outlined in IEEE Standard 323-1974. The radiation dose exposure of 1.84×10^8 rads gamma covered the accident and normal service conditions, with margin. A DBE exposure simulation used saturated steam, encompassed the limiting DBE temperature, pressure, and 100% relative humidity. The peak temperature was 341°F at 117.8 psig, demonstrating margin. The DBE testing lasted for 100 days with a final temperature of 200°F. No simulated chemical spray was included in the testing sequence.

It is concluded that the Rockbestos PXMJ cable, as documented in EQ Binder WBNEQ-CABL-052, is qualified for the parameters specified and the applicant has adequate documentation in this binder to support that conclusion.

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REVIEW OF WATTS BAR EQ DOCUMENTATION FILES

EQ Binder	WBNEQ-CABL-053
Equipment Type	Medium Voltage Power Cable for 1-MVOP-063-0172-B
Manufacturer	Rockbestos Company
Model Number	PXJ, PXMJ
Plant ID Number	1V 2343B

EQ Binder WBNEQ-CABL-053 documents the environmental qualification of this MOV power cable. Valve 1-MVOP-063-0172-B is the RHR to RCS Hot Leg 1 & 3 flow isolation valve, and is located 16 feet, 4 inches, above the floor in Auxiliary Building Pipe Chase A28, elevation 713 feet. The maximum flood level inside the pipe chase is 1 inch above the floor. Thus, the equipment is not subject to submergence. The normal ambient temperature ranges between 60°F and 104°F. The peak accident temperature is 110°F. The applicant's evaluation uses this data in qualifying this cable.

The tested equipment was artificially aged with the test sequence outlined in IEEE Standard 323-1974. The tested radiation dose exposure covered the accident and normal service conditions, with margin. A DBE exposure simulation included steam exposure and encompassed the limiting DBE temperature, pressure, and humidity, with margin. Chemical spray is not postulated for this location.

The INEL concluded that cable documented in EQ Binder WBNEQ-CABL-053 is qualified for the parameters specified. The applicant has adequate documentation in this binder to support that conclusion.

EQ Binder	WBNEQ-CABL-055
Equipment Type	Thermocouple extension cable, copper/constantan with FR-XLPE insulation and CSPE jacket
Manufacturer	Rockbestos
Model Number	YPS
Plant ID Number	1-2PM-68-0832G

The environmental qualification of this cable is documented in EQ Binder WBNEQ-CABL-055. Cable 1-2PM-68-0832G is spliced to 1-TE-068-79B-G, located in the containment at an elevation of 724 feet. The worst case flood level inside the crane wall is 722 feet. Therefore, flooding is not postulated. The normal ambient temperature ranges up to 140°F. The peak accident temperature for this location is 419°F. The applicant's analysis shows the temperature inside the conduit and junction boxes for this cable peaks at 340°F, based on an ambient temperature of 453°F, providing margin above the 419°F maximum accident temperature. Thermocouple signals do not subject the cable to any conductor self heating. The applicant's evaluation utilizes this data in qualifying these cables.

The samples were artificially aged with the test sequence outlined in IEEE Standard 323-1974. The radiation dose exposure covered the accident and normal service conditions, with margin. A DBE exposure simulation used saturated steam, encompassing the limiting DBE temperature, pressure, and 100% relative humidity. A simulated chemical spray was included in the testing sequence. The applicant's binder for this cable notes the 1.8°F margin between the postulated accident temperature and the peak test temperature, and provides justification for that small margin. The maximum conservatively required temperature of 340°F inside the metal enclosures is for five minutes based on a maximum accident ambient temperature of 453°F, whereas the postulated maximum ambient temperature is 419°F. This difference in the baseline temperature used demonstrates the required margin. The peak test temperature was maintained for 16 minutes.

It is concluded that the Rockbestos KXL-760-D, Type YPS cable, as documented in EQ Binder WBNEQ-CABL-055, is qualified for the parameters specified and the applicant has adequate documentation in this binder to support that conclusion.

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EQ Binder	WBNEQ-CSC-002
Equipment Type	Conduit Seal Assembly
Manufacturer	NAMCO Controls
Model Number	EC210-34000
Plant ID Number	CSC-072-0006 for 1-TE-072-0006, CSC-081-0012A for 1-FSV-081-0012-A

The environmental qualification of these conduit seal assemblies is documented in EQ Binder WBNEQ-CSC-002. CSC-072-0006 is located in the auxiliary building, room A-11, at an elevation of 717 feet, 6 inches. CSC-081-0012-A is located in the A-28 pipe chase area of the auxiliary building at an elevation of 716 feet, 6 inches. The flood level in room A-11 is 714 feet, 10 inches. The flood level in A-28 pipe chase is 713 feet, 3 inches. Thus, the conduit seal assemblies are not subject to submergence. The normal ambient temperature ranges up to 104°F (A-28) and 120°F (A-11). The peak accident temperature is 110°F and 327°F, respectively. The applicant's evaluation utilizes this data in qualifying these components.

The samples were artificially aged with the test sequence outlined in IEEE Standard 323-1974. The radiation dose exposure of 2.04×10^8 rd covered the accident and normal service conditions, with margin. A DBE exposure simulation encompassed the limiting DBE temperature, pressure, and 100% relative humidity, with margin. The peak test temperature was 350°F at 70 psig. The DBE testing lasted for 30 days with a final temperature of 205°F. The binder included an analysis that extends the testing results to the required 100 days. A chemical spray was included in testing this RTD.

It is concluded that the NAMCO Control, Series EC210-34000 conduit seal assembly, as documented in EQ Binder WBNEQ-CSC-002, is qualified for the parameters specified and the applicant has adequate documentation in this binder to support that conclusion.

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REVIEW OF WATTS BAR EQ DOCUMENTATION FILES

EQ Binder	WBNEQ-HS-002
Equipment Type	Hand Switch
Manufacturer	Cutler-Hammer
Model Number	10250T
Plant ID Number	1-HS-074-0003B-A

The environmental qualification of this hand switch is documented in EQ Binder WBNEQ-HS-002. The switch is a manual control for the RHR Pump A-A Inlet Flow Control Valve, located in the room A-11 of the Auxiliary Building at an elevation of 676 feet, some 6 feet above the flood level in that room. The normal ambient temperature ranges up to 104°F. The peak accident temperature (HELB) is 215°F. The applicant's evaluation utilizes this data in qualifying these valves.

The samples were artificially aged with the test sequence outlined in IEEE Standard 323-1974. Radiation dose exposure of 1.8×10^7 rads covers the accident and normal service conditions, with margin. A DBE exposure simulation included steam exposure and encompassed the limiting DBE temperature, pressure, and 100% relative humidity. The peak test temperature was 330°F, showing margin. The DBE testing lasted for 26.5 hours. The binder included an analysis that extends the testing results to the required 100 days. Chemical spray is not postulated for this switch location.

It is concluded that the Cutler-Hammer 10250T switch, as documented in EQ Binder WBNEQ-HS-002, is qualified for the parameters specified and the applicant has adequate documentation in this binder to support that conclusion.

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REVIEW OF WATTS BAR EQ DOCUMENTATION FILES

EQ Binder	WBNEQ-HTR-001
Equipment Type	Primary Containment Hydrogen Recombiner
Manufacturer	Westinghouse Electric Corporation
Model Number	A
Plant ID Number	1-HTR-083-0001-A

EQ Binder WBNEQ-HTR-001 documents the environmental qualification of this hydrogen recombiner. The equipment recombines hydrogen with oxygen to produce water to prevent hydrogen explosion inside primary containment following a LOCA. It is located on a platform inside the polar crane wall at an elevation of 779 feet 11 inches. The maximum flood level inside the crane wall reaches an elevation of 722 feet. Thus, the equipment is not subject to submergence. The normal ambient temperature ranges between 60°F and 120°F. The peak accident temperature is 327°F. The applicant's evaluation uses this data to qualify this equipment.

The tested equipment was artificially aged with the test sequence outlined in IEEE Standard 323-1974. The radiation dose exposure covered the accident and normal service conditions, with margin. A DBE exposure simulation included steam exposure and encompassed the limiting DBE temperature, pressure, and humidity, with margin. Chemical spray is postulated for this location.

The INEL concluded that the Westinghouse Model A Hydrogen Recombiner, documented in EQ Binder WBNEQ-HTR-001, is qualified for the parameters specified. The applicant has adequate documentation in this binder to support that conclusion.

APPENDIX A

REVIEW OF WATTS BAR EQ DOCUMENTATION FILES

EQ Binder	WBNEQ-ITE-006
Equipment Type	Resistance Temperature Detector (RTD)
Manufacturer	Weed Instruments
Model Number	612-1B-A-4C
Plant ID Number	1-TE-074-0039, 1-TE-072-0006

The environmental qualification of these RTDs is documented in EQ Binder WBNEQ-ITE-006. RTD 1-TE-074-0039, which measures the RHR Heat Exchanger B Outlet Temperature, is located in the auxiliary building, room A-11, elevation 713, at an elevation of 719 feet, 6 inches. RTD 1-TE-072-0006, which measures the Containment Spray Heat Exchanger B Outlet Temperature, is located in the auxiliary building, room A-11, elevation 713, at an elevation of 717 feet, 6 inches. The flood level in room A-11 is 716 feet, 2 inches. Thus, the RTDs are not subject to submergence. The normal ambient temperature ranges up to 110°F. The peak accident temperature is 215°F. The applicant's evaluation utilizes this data in qualifying these components.

The samples were artificially aged with the test sequence outlined in IEEE Standard 323-1974. The radiation dose exposure covered the accident and normal service conditions, with margin. A DBE exposure simulation encompassed the limiting DBE temperature, pressure, and 100% relative humidity. The peak test temperature was 503°F at 75 psig, demonstrating margin, and for applications for higher accident temperature locations. The DBE testing lasted for 26 days with a final temperature of 300°F. The binder included an analysis that extends the testing results to the required 100 days. A chemical spray was included in the testing of this RTD.

It is concluded that the Weed Instruments Model 612-1B-A-4C RTD, as documented in EQ Binder WBNEQ-ITE-006, is qualified for the parameters specified and the applicant has adequate documentation in the binder to support that conclusion.

APPENDIX A

REVIEW OF WATTS BAR EQ DOCUMENTATION FILES

EQ Binder	WBNEQ-ITS-002
Equipment Type	RHR Pump 1A-A Room Ambient Air Temperature Switch A (Vented) RHR Pump 1A-A Room Ambient Air Temperature Switch B (Vented) EGTS 2A-A Room Ambient Air Temperature Switch A (Unvented) EGTS 2A-A Room Ambient Air Temperature Switch B (Unvented)
Manufacturer	Static-O-Ring Incorporated
Model Number	201TA-BB125-JJTTX6
Serial Numbers	93-11-7022, 93-11-7018, 85-3-3451, and 85-3-3458, respectively
Plant ID Numbers	1-TS-030-5236A-A, 1-TS-030-5236B-B, 2-TS-030-0200A, and 2-TS-030-0207B, respectively

EQ Binder WBNEQ-ITS-002 documents the environmental qualification of these room ambient air temperature switches. The lowest points of the two switches for RHR Pump 1A-A are located 4.5 feet above the floor in Auxiliary Building Room A11, elevation 676 feet. The maximum flood level in the room is 7 inches above the floor. Thus, these switches are not subject to submergence. The normal ambient temperature ranges between 60°F and 104°F and the peak accident temperature is 110°F. The lowest points of the two switches for EGTS 2A-A are located 4.5 feet above the floor in Auxiliary Building Room A16 on Elevation 757 feet. The room is above the maximum possible flood level specified in plant design criteria. Thus, these switches are not subject to submergence. The normal ambient temperature ranges between 60°F and 104°F. The peak accident temperature is 110°F. The applicant's evaluation uses this data in qualifying this equipment.

The tested switch was artificially aged with the test sequence outlined in IEEE Standard 323-1974. The test radiation dose exposure covered the accident and normal service conditions, with margin. A DBE exposure simulation included steam exposure and encompassed the limiting DBE temperature, pressure, and humidity, with margin. Chemical spray is not postulated for these locations.

The INEL concluded that the temperature switches documented in EQ Binder WBNEQ-ITS-002, are qualified for the parameters specified. The applicant has adequate documentation in this binder to support that conclusion.

APPENDIX A

REVIEW OF WATTS BAR EQ DOCUMENTATION FILES

EQ Binder	WBNEQ-IZS-003
Equipment Type	Limit Switch
Manufacturer	NAMCO Controls
Model Number	EA740
Plant ID Number	ZS-030-146AB

The environmental qualification of this limit switch is documented in EQ Binder WBNEQ-IZS-003. The limit switch monitors the position of the Auxiliary Building Gas Treatment Fan A-A Exhaust Damper in Room A05, elevation 737, of the Auxiliary Building. The switch is mounted at an elevation of 751 feet, 2 inches. The flood level is 738 feet, 11 inches. Thus, the component is not subject to submergence. The normal ambient temperature ranges up to 104°F. The peak accident temperature at this location is 119°F. The peak accident temperature in any examined location for this component is 219°F. The applicant's evaluation utilizes this data in qualifying these components.

The samples were artificially aged with the test sequence outlined in IEEE Standard 323-1974. The radiation dose exposure covered the accident and normal service conditions, with margin. A DBE exposure simulation included steam exposure and encompassed the limiting DBE temperature, pressure, and 100% relative humidity. Chemical spray, postulated for this equipment, was simulated. The peak test temperature was 347°F at 72 psig, demonstrating margin. The DBE testing lasted for 32 days with a final temperature of 203°F. The binder included an analysis that extends the testing results to the required 100 days.

It is concluded that the NAMCO Controls, Series EA740 limit switch, as documented in EQ Binder WBNEQ-IZS-003, is qualified for the parameters specified and the applicant has adequate documentation in this binder to support that conclusion.

APPENDIX A

REVIEW OF WATTS BAR EQ DOCUMENTATION FILES

EQ Binder	WBNEQ-IZS-005
Equipment Type	Limit Switch
Manufacturer	NAMCO Controls
Model Number	EA-180
Plant ID Number	ZS-081-0012A-A and ZS-081-0012B-A

The environmental qualification of this limit switch is documented in EQ Binder WBNEQ-IZS-005. These limit switches monitor the position of the Pressurizer Relief Tank and Reactor Coolant Pump SP Valve in Pipe Chase A28 of the Auxiliary Building at an elevation of 715 feet. The flood level is 713 feet, 3 inches. Thus, these limit switches are not subject to submergence. The normal ambient temperature ranges up to 104°F. The peak accident temperature at this location is 110°F for up to 30 days. The applicant's evaluation utilizes this data in qualifying these components.

The samples were artificially aged with the test sequence outlined in IEEE Standard 323-1974. The radiation dose exposure covered the accident and normal service conditions, with margin. A DBE exposure simulation included steam exposure and encompassed the limiting DBE temperature, pressure, and 100% relative humidity. The peak test temperature was 339°F, demonstrating margin. The DBE testing lasted for 30 days. The binder included an analysis that extends the testing results to the required 100 days. Chemical spray, postulated for this equipment, was simulated.

It is concluded that the NAMCO Controls, Series EA-180 limit switch, as documented in EQ Binder WBNEQ-IZS-005, is qualified for the parameters specified and the applicant has adequate documentation in this binder to support that conclusion.

APPENDIX A

REVIEW OF WATTS BAR EQ DOCUMENTATION FILES

EQ Binder	WBNEQ-JBOX-001	
Equipment Type	Junction Box	Terminal Strip
Manufacturer	Various, TVA Type JVB	General Electric
Model Number	JVB, 10x10x6, NEMA 4	CR151B
Plant ID Number	1-JB-293-3201-B	1-TBLK-293-3201-B

The environmental qualification of these components is documented in EQ Binder WBNEQ-JBOX-001. The junction box and enclosed terminal strip are located in Accumulator Room 4 of the reactor building at an elevation of 720 feet, which is 2 feet, 3 inches above the maximum flood level. The normal ambient temperature ranges up to 140°F. The peak accident temperature (HELB) is 327°F (for the junction box) and 320.9°F for the terminal strip inside the junction box. The applicant's evaluation utilizes this data in qualifying these components.

The junction box, of metal construction, is not subject to age degradation. The door gasket material is not relied on to seal the junction box or provide structural integrity. As the terminal strip materials were not deemed age-sensitive to time-temperature effects, the terminal block samples were both coated and not coated with Dow-Corning RTV 3140. Only the coated blocks apply to inside containment installations. Those samples were artificially aged with the test sequence outlined in IEEE Standard 323-1974. The radiation dose covered the accident and normal service conditions, with margin. A DBE test simulation included steam exposure. The peak temperature was 350°F at 35.9 psig, demonstrating margin. The DBE testing lasted for 83 hours. The binder included an analysis that extends the testing results to the required 100 days. A simulated chemical spray was included in the test sequence.

It is concluded that the junction boxes and terminal strips documented in EQ Binder WBNEQ-JBOX-001 are qualified for the parameters specified and the applicant has adequate documentation to support that conclusion.

APPENDIX A

REVIEW OF WATTS BAR EQ DOCUMENTATION FILES

EQ Binder	WBNEQ-MOT-001
Equipment Type	SI Pump 1A-A Motor, CS Pump 1B-B Motor, and RHR Pump 1A-A Motor
Manufacturer	Westinghouse Electric Corporation
Model Number	HSDP, HSW2, and VSW1, respectively
Plant ID Number	1-MTR-063-0010-A, 1-MTR-072-0010-B, and 1-MTR-074-0010-A, respectively

EQ Binder WBNEQ-MOT-001 documents the environmental qualification of these pump motors. The SI Pump 1A-A motor is located on a 6-inch metal skid mounted on top of a 6-inch concrete platform in Auxiliary Building Room A13, elevation 692 feet. The maximum flood level inside the room is 2 inches above the floor. The CS Pump 1B-B motor is located on a 6-inch metal skid mounted on top of a 6-inch concrete platform in Auxiliary Building Room A08, elevation 676 feet. The maximum flood level inside the room is 7 inches above the floor. The RHR Pump 1A-A motor is located on a 6-inch metal skid mounted on top of a 6-inch concrete platform in Auxiliary Building Room A11, elevation 676 feet. The maximum flood level inside the room is 7 inches above the floor. Thus, none of the motors is subject to submergence. In all rooms, the normal ambient temperatures range between 60°F and 104°F, and the peak accident temperatures are 110°F. The applicant's evaluation uses these data in qualifying these motors.

The tested equipment was artificially aged with the test sequence outlined in IEEE Standard 323-1974. The radiation dose exposure covered the accident and normal service conditions where applicable, with margin. A DBE exposure simulation included steam exposure and encompassed the limiting DBE temperature, pressure, and humidity. The test profiles included adequate margins. Chemical spray is not postulated for these locations.

The INEL concluded that the Westinghouse Electric Motors for SI, CS, and RHR, documented in EQ Binder WBNEQ-MOT-001, are qualified for the specified parameters. The applicant has adequate documentation in this binder to support that conclusion.

APPENDIX A

REVIEW OF WATTS BAR EQ DOCUMENTATION FILES

EQ Binder	WBNEQ-MOT-003
Equipment Type	ABGTS Exhaust Fan 1A Motor, RHR Pump 1A-A Room Cooler Motor, CS Pump 1A-A Room Cooler Motor, and SI Pump 1A-A Room Cooler Motor
Manufacturer	Reliance Motors Company
Serial Numbers	1YF882365A1-YC, 1YF883397A4-VK, 2YF883397A1-VK, and 1YF883397A1-VK, respectively
Plant ID Numbers	1-MTR-030-0146-A, 1-MTR-030-0175-A, 1-MTR-030-0177-A, and 1-MTR-030-0180-A, respectively

EQ Binder WBNEQ-MOT-003 documents the environmental qualification of these ventilation system motors. The ABGTS Exhaust Fan 1A motor is mounted to the side of the fan housing, and the bottom of the motor is about 8 inches above the floor in Room A05, elevation 737 feet, in the Auxiliary Building. The maximum flood level during a moderate energy line break inside Room A05, elevation 737 feet, is 23 inches above the floor. However, the ABGTS is only required to operate to mitigate a LOCA condition that has no flooding impact on this room. The RHR Pump 1A-A Room Cooler motor is mounted atop the cooler housing, and the bottom of the motor is about 89 inches above the floor in Room A11, elevation 676 feet, in the Auxiliary Building. The maximum flood level inside Room A11, Elevation 676 feet, is 7 inches above the floor. The CS Pump 1A-A Room Cooler motor is mounted to the bottom of the cooler housing. The bottom of the motor is about 100 inches above the floor in Room A09, elevation 676 feet, in the Auxiliary Building. The maximum flood level inside Room A09, elevation 676 feet, is 6 inches above the floor. The SI Pump 1A-A Room Cooler motor is mounted on the top of the cooler housing, and the bottom of the motor is about 89 inches above the floor in Room A13, elevation 692 feet, in the Auxiliary Building. The maximum flood level inside Room A13, elevation 692 feet, is 7 inches above the floor. Thus, none of the motors is subject to submergence. In all rooms, the normal ambient temperatures range between 60°F and 104°F, and the peak accident temperatures are 110°F. The applicant's evaluation uses this data in qualifying this equipment.

The tested equipment was artificially aged with the test sequence outlined in IEEE Standard 323-1974. The radiation dose exposure of 2.2×10^8 rads covered the accident and normal service conditions, with margin. A DBE exposure simulation included steam exposure and encompassed the limiting DBE temperature, pressure, and humidity, with margin. The test profile included 155°F, atmospheric pressure, and 100% relative humidity. Chemical spray is not postulated for these locations.

The INEL concluded that the motors documented in EQ Binder WBNEQ-MOT-003 are qualified for the parameters specified. The applicant has adequate documentation in this binder to support that conclusion.

EQ Binder	WBNEQ-MOV-001
Equipment Type	RHR to RCS Hot Leg 1 & 3 Flow Isolation Valve Operator, RHR Pump 1A-A Suction Shutoff Valve Operator, and RHR System Isolation Bypass Shutoff Valve Operator
Manufacturer	Limitorque Motorized Valve Operator with Type RH Insulation
Model Number	SB-2, Serial 192138; SB-2, Serial 197000; SB-1, Serial 241211, respectively
Plant ID Number	1-MVOP-063-0172-B, 1-MVOP-074-0003-A, and 1-MVOP-074-0008-A, respectively

EQ Binder WBNEQ-MOV-001 documents the environmental qualification of these MOV operators. Valve operator 1-MVOP-063-0172-B sits atop the RHR to RCS Hot Leg 1 & 3 flow isolation valve, and is located 16 feet, 4 inches, above the floor in Auxiliary Building Pipe Chase A28, elevation 713 feet. The maximum flood level inside this pipe chase is 3 inches above the floor. Operator 1-MVOP-074-0003-A is on top of the RHR Pump 1A-A Suction shutoff valve, and is located some 12 feet above the floor in Auxiliary Building Room A11 on Elevation 676 feet. The maximum flood level in the room is 7 inches. Valve operator 1-MVOP-074-0008-A is on the top of the RHR System Isolation Bypass Shutoff Valve, located in the Reactor Building Room AC4 on Elevation 722 feet. Plant design criteria do not postulate any flooding in this room. Thus, none of these valve operators is subject to submergence. In all cases, the normal ambient temperatures range between 60°F and 104°F, and the peak accident temperatures are 110°F. The applicant's evaluation uses these data in qualifying this equipment.

The tested equipment was artificially aged with the test sequence outlined in IEEE Standard 323-1974. The radiation dose exposure of 2.04×10^{8} rads covered the accident and normal service conditions, with margin. A DBE test simulation included steam exposure and encompassed the limiting DBE temperature, pressure, and humidity. The test profile included 30 days at 315°F, 78 psig, and 100% relative humidity, demonstrating margin. Chemical spray is not postulated for these locations.

The INEL concluded that the valve operators documented in EQ Binder WBNEQ-MOV-001 are qualified for the parameters specified. The applicant has adequate documentation in this binder to support that conclusion.

APPENDIX A

REVIEW OF WATTS BAR EQ DOCUMENTATION FILES

EQ Binder	WBNEQ-MOV-003
Equipment Type	RHR Pump 1A-A Minimum Flow Shutoff Valve Operator
Manufacturer	Limitorque Motorized Valve Operator
Model Number	SMB-000, Serial 358806
Plant ID Number	1-MVOP-074-0012-A

EQ Binder WBNEQ-MOV-003 documents the environmental qualification of this valve operator. Motor operator 1-MVOP-074-0012-A is mounted on the minimum flow isolation valve for RHR Pump 1A-A, located over 10 feet above the floor in Pipe Chase A16, elevation 676 feet, in the Auxiliary Building. The maximum flood level in the room is 7 inches above the floor. Thus, the operator is not subject to submergence. The normal ambient temperature ranges between 60°F and 104°F, and the peak accident temperature is 110°F. The applicant's evaluation uses this data in qualifying this operator.

The tested valve operator was artificially aged with the test sequence outlined in IEEE Standard 323-1974 as applied to valve operators. The radiation dose exposure of 2.04×10^8 rads covered the accident and normal service conditions, with margin. A DBE exposure simulation included steam exposure and encompassed the limiting DBE temperature, pressure, and humidity. The test profile included 16 days at 250°F, 25 psig, and 100% relative humidity, demonstrating margin. Chemical spray is not postulated for this location.

The INEL concluded that the operator documented in EQ Binder WBNEQ-MOV-003 is qualified for the parameters specified. The applicant has adequate documentation in this binder to support that conclusion.

APPENDIX A

REVIEW OF WATTS BAR EQ DOCUMENTATION FILES

EQ Binder	WBNEQ-PENT-002
Equipment Type	Primary Containment Electrical Penetration
Manufacturer	CONAX
Model Number	7429-10002-05
Plant ID Number	1-PENT-293-0027-A

The environmental qualification of this penetration is documented in EQ Binder WBNEQ-PENT-002. The penetration carries low voltage and control circuits between the auxiliary building and the reactor building, Fan Room 1, at an elevation of 728 feet. The flood level in Fan Room 1 is 717.9 feet. Thus, the penetration is not subject to submergence. The normal ambient temperature ranges between 60°F and 120°F. The peak accident temperature is 327°F. The applicant's evaluation utilizes this data in qualifying these components.

The samples were artificially aged with the test sequence outlined in IEEE Standard 323-1974. The radiation dose exposure covered the accident and normal service conditions, with margin. A DBE exposure simulation included steam exposure and encompassed the limiting DBE temperature, pressure, and 100% relative humidity. The peak test temperature was 370°F at 75 psig, demonstrating margin. The DBE testing lasted for 83 hours. The binder included an analysis that extends the testing results to the required 100 days. Chemical spray was not postulated for this penetration.

It is concluded that the CONAX 7429-10002-05 penetration, as documented in EQ Binder WBNEQ-PENT-002, is qualified for the parameters specified and the applicant has adequate documentation in this binder to support that conclusion.

It is concluded that the ASCO NP831654E solenoid valve, as documented in EQ Binder WBNEQ-SOL-006, is qualified for the parameters specified and the applicant has adequate documentation to support that conclusion.

APPENDIX A

REVIEW OF WATTS BAR EQ DOCUMENTATION FILES

EQ Binder	WBNEQ-SPLC-001
Equipment Type	Nuclear grade cable connection heat shrink splices and terminations
Manufacturer	Raychem
Model Number	Type-52 or WCSF
Plant ID Number	WBN-SPL-11256

The environmental qualification of this cable splice is documented in EQ Binder WBNEQ-SPLC-001. Cable 1-2PM-68-0832G is spliced, via splice WBN-SPL-11256 to 1-TE-068-79B-G, located in JB293-4340G inside the crane wall in containment at an elevation of 724 feet. Flooding is not postulated because the maximum flood level is at 722 feet. The normal ambient temperature ranges up to 140°F. The peak accident temperature for this location is 327°F inside the junction box where this splice is located. The applicant's evaluation utilizes this data in qualifying these components.

The samples were artificially aged with the test sequence outlined in IEEE Standard 323-1974. The radiation dose exposure of the samples, 2.2×10^8 rads, covered the accident and normal service conditions, with margin. A DBE exposure simulation used saturated steam, encompassed the limiting DBE temperature, pressure, and 100% relative humidity, with margin. Though the splice is located in a junction box and not subject to the direct impingement of chemical spray, a simulated chemical spray was included in the testing sequence.

It is concluded that the Raychem Type-52 or WCSF cable splices, as documented in EQ Binder WBNEQ-SPLC-001, are qualified for the parameters specified for in containment use. It is also concluded that the applicant has adequate documentation in this binder to support that conclusion.

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Docket Nos. 50-390 and 50-391

11. ABSTRACT *(200 words or less)*

Supplement No. 15 to the Safety Evaluation Report for the application filed by the Tennessee Valley Authority for license to operate Watts Bar Nuclear Plant, Units 1 and 2, Docket Nos. 50-390 and 50-391, located in Rhea County Tennessee, has been prepared by the Office of Nuclear Reactor Regulation of the Nuclear Regulatory Commission. The purpose of this supplement is to update the Safety Evaluation with (1) additional information submitted by the applicant since Supplement No. 14 was issued, and (2) matters that the staff had under review when Supplement No. 14 was issued.

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Watts Bar Nuclear Plant
Docket Nos. 50-390/50-391

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