UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 27, 2017

Mr. Edward D. Halpin
Senior Vice President, Generation
and Chief Nuclear Officer
Pacific Gas and Electric Company
Diablo Canyon Power Plant
P.O. Box 56, Mail Code 104/6
Avila Beach, CA 93424

SUBJECT: DIABLO CANYON POWER PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: REVISE LICENSING BASES TO ADOPT ALTERNATIVE SOURCE TERM (CAC NOS. MF6399 AND MF6400)

Dear Mr. Halpin:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 230 to Facility Operating License No. DPR-80 and Amendment No. 232 to Facility Operating License No. DPR-82 for the Diablo Canyon Power Plant, Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated June 17, 2015, as supplemented by letters dated August 31, October 22, November 2, November 6, and December 17, 2015; and February 1, February 10, April 21, June 9, 2016, September 15, October 6, and December 27, 2016.

The amendments revise the licensing bases to adopt alternative source term as allowed by Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67, "Accident source term," and approve the methodology for evaluating radiological consequences of design-basis accidents as described in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." The amendments revise TS 1.1, "Definitions"; TS 3.4.16, "RCS [Reactor Coolant System] Specific Activity"; TS 3.6.3, "Containment Isolation Valves"; TS 5.5.11, "Ventilation Filter Testing Program (VFTP)"; and TS 5.5.19, "Control Room Envelope Habitability Program," in support of the revised licensing bases. The amendments also add license conditions to Appendix D, "Additional Conditions," to Facility Operating License Nos. DPR-80 and DPR-82.

A copy of the related Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's next regular biweekly *Federal Register* notice.

Sincerely,

Balwant K. Singal, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-275 and 50-323

Enclosures:
1. Amendment No. 230 to DPR-80
2. Amendment No. 232 to DPR-82
3. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PACIFIC GAS AND ELECTRIC COMPANY

DOCKET NO. 50-275

DIABLO CANYON NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 230
License No. DPR-80

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Pacific Gas and Electric Company (the licensee), dated June 17, 2015, as supplemented by letters dated August 31, October 22, November 2, November 6, and December 17, 2015; and February 1, February 10, April 21, June 9, September 15, October 6, and December 27, 2016, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-80 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 230 are hereby incorporated in the license. Pacific Gas & Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

3. This license amendment is effective as of its date of issuance and shall be implemented within 365 days.

FOR THE NUCLEAR REGULATORY COMMISSION

R Pascarelli

Robert J. Pascarelli, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility
Operating License No. DPR-80
and Technical Specifications

Date of Issuance: April 27, 2017

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PACIFIC GAS AND ELECTRIC COMPANY

DOCKET NO. 50-323

DIABLO CANYON NUCLEAR POWER PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 232
License No. DPR-82

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Pacific Gas and Electric Company (the licensee), dated June 17, 2015, as supplemented by letters dated August 31, October 22, November 2, November 6, and December 17, 2015; and February 1, February 10, April 21, June 9, September 15, October 6, and December 27, 2016, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-82 is hereby amended to read as follows:

    (2) Technical Specifications (SSER 32, Section 8)* and Environmental Protection Plan

    The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 232, are hereby incorporated in the license. Pacific Gas & Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

3. This license amendment is effective as of its date of issuance and shall be implemented within 365 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert J. Pascarelli, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility
Operating License No. DPR-82
and Technical Specifications

Date of Issuance: April 27, 2017

ATTACHMENT TO LICENSE AMENDMENT NO. 230

TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 232 TO FACILITY OPERATING LICENSE NO. DPR-82

DIABLO CANYON POWER PLANT, UNITS 1 AND 2

DOCKET NOS. 50-275 AND 50-323

Replace the following pages of the Facility Operating License Nos. DPR-80 and DPR-82, Appendix A Technical Specifications, and Appendix D, Additional Conditions, with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Facility Operating License No. DPR-80

REMOVE	INSERT
3	3
10	10
--	Appendix D, Page 4

Facility Operating License No. DPR-82

REMOVE	INSERT
3	3
8	8
--	Appendix D, Page 4

Technical Specifications

REMOVE	INSERT
1.1-3	1.1-3
3.4-36	3.4-36
3.6-5	3.6-5
3.6-9	3.6-9
3.6-10	3.6-10
5.0-13	5.0-13
5.0-17a	5.0-17a

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This License shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The Pacific Gas and Electric Company is authorized to operate the facility at reactor core power levels not in excess of 3411 megawatts thermal (100% rated power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 230 are hereby incorporated in the license. Pacific Gas & Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

(3) Initial Test Program

The Pacific Gas and Electric Company shall conduct the post-fuel-loading initial test program (set forth in Section 14 of Pacific Gas and Electric Company's Final Safety Analysis Report, as amended), without making any major modifications of this program unless modifications have been identified and have received prior NRC approval. Major modifications are defined as:

a. Elimination of any test identified in Section 14 of PG&E's Final Safety Analysis Report as amended as being essential;

PG&E shall keep the staff informed on the progress of the reevaluation program as necessary, but as a minimum will submit quarterly progress reports and arrange for semi-annual meetings with the staff. PG&E will also keep the ACRS informed on the progress of the reevaluation program as necessary, but not less frequently than once a year.

(8) Control of Heavy Loads (SSER 27, Section IV.6)

Prior to startup following the first refueling outage, the licensee shall submit commitments necessary to implement changes and modifications as required to satisfy the guidelines of Section 5.1.2 through 5.1.6 of NUREG-0612 (Phase II: 9-month responses to the NRC Generic Letter dated December 22, 1980).

(9) Emergency Preparedness (SSER 27, Section IV.3)

In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's final rule, 44 CFR Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of preparedness, the provisions of 10 CFR Section 50.54(s)(2) will apply.

(10) Masonry Walls (SSER-27, Section IV.4: Safety Evaluation of November 2, 1984)

Prior to start-up following the first refueling outage, the licensee shall (1) evaluate the differences in margins between the staff criteria as set forth in the Standard Review Plan and the criteria used by the licensee, and (2) provide justification acceptable to the staff for those cases where differences exist between the staffs and the licensee's criteria.

(11) Spent Fuel Pool Modification

The licensee is authorized to modify the spent fuel pool as described in the application dated October 30, 1985 (LAR 85-13) as supplemented. Amendment No. 8 issued on May 30, 1986 and stayed by the U.S. Court of Appeals for the Ninth Circuit pending completion of NRC hearings is hereby reinstated.

Prior to final conversion to the modified rack design, fuel may be stored, as needed, in either the modified storage racks described in Technical Specification 5.6.1.1 or in the unmodified storage racks (or both) which are designed and shall be maintained with a nominal 21-inch center-to-center distance between fuel assemblies placed in the storage racks.

(12) Additional Conditions

The Additional Conditions contained in Appendix D, as revised through Amendment No. 230 are hereby incorporated into this license. Pacific Gas and Electric Company shall operate the facility in accordance with the Additional Conditions.

Amendment Number	Additional Conditions	Implementation Date
230	Implementation of the amendment adopting the alternative source term shall include the following plant modifications: Install shielding material, equivalent to that provided by the Control Room outer walls, at the external concrete west wall of the Control Room briefing room. Install a high efficiency particulate air filter in the Technical Support Center normal ventilation system. Re-classify a portion of the 40-inch Containment Penetration Area (GE/GW) Ventilation line from PG&E Design Class II to PG&E Design Class I and upgrade the damper actuators, pressure switches, and the damper solenoid valves to PG&E Design Class I. Update setpoints for the redundant safety related gamma sensitive area radiation monitors (1-RE 25/26, 2-RE 25/26).	The amendment is effective as of the date of its issuance and the condition shall be implemented within 365 days of its issuance

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This License shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The Pacific Gas and Electric Company is authorized to operate the facility at reactor core power levels not in excess of 3411 megawatts thermal (100% rated power) in accordance with the conditions specified herein.

(2) Technical Specifications (SSER 32, Section 8)* and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 232, are hereby incorporated in the license. Pacific Gas & Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

(3) Initial Test Program (SSER 31, Section 4.4.1)

Any changes to the Initial Test Program described in Section 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

*The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

(10) Pipeway Structure DE and DDE Analysis (SSER 32, Section 4)

Prior to start-up following the first refueling outage PG&E shall complete a confirmatory analysis for the pipeway structure to further demonstrate the adequacy of the pipeway structure for load combinations that include the design earthquake (DE) and double design earthquake (DDE).

(11) Spent Fuel Pool Modification

The licensee is authorized to modify the spent fuel pool as described in the application dated October 30, 1985 (LAR 85-13) as supplemented. Amendment No. 6 issued on May 30, 1986 and stayed by the U.S. Court of Appeals for the Ninth Circuit pending completion of NRC hearings is reinstated.

Prior to final conversion to the modified rack design, fuel may be stored, as needed, in either the modified storage racks described in Technical Specification 5.6.1.1 or in the unmodified storage racks (or both) which are designed and shall be maintained with a nominal 21-inch center-to-center distance between fuel assemblies placed in the storage racks.

(12) Additional Conditions

The Additional Conditions contained in Appendix D, as revised through Amendment No. 232, are hereby incorporated into this license. Pacific Gas and Electric Company shall operate the facility in accordance with the Additional Conditions.

D. Exemption (SSER 31, Section 6.2.6)

An exemption from certain requirements of Appendix J to 10 CFR Part 50 is described in the Office of Nuclear Reactor Regulation's Safety Evaluation Report, Supplement No. 9. This exemption is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest. Therefore, this exemption previously granted in Facility Operating License No. DPR-81 pursuant to 10 CFR 50.12 is hereby reaffirmed. The facility will operate, with the exemption authorized, in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission.

E. Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provision of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Diablo Canyon Power Plant, Units 1 and 2 Physical Security Plan, Training and Qualification Plan and Safeguards Contingency Plan," submitted by letter dated May 16, 2006.

PG&E shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The PG&E CSP was approved by License Amendment No. 212, as supplemented by a change approved by License Amendment No. 222.

Amendment Number	Additional Conditions	Implementation Date
232	Implementation of the amendment adopting the alternative source term shall include the following plant modifications: Install shielding material, equivalent to that provided by the Control Room outer walls, at the external concrete west wall of the Control Room briefing room. Install a high efficiency particulate air filter in the Technical Support Center normal ventilation system. Re-classify a portion of the 40-inch Containment Penetration Area (GE/GW) Ventilation line from PG&E Design Class II to PG&E Design Class I and upgrade the damper actuators, pressure switches, and the damper solenoid valves to PG&E Design Class I. Update setpoints for the redundant safety related gamma sensitive area radiation monitors (1-RE 25/26, 2-RE 25/26).	The amendment is effective as of the date of its issuance and the condition shall be implemented within 365 days of its issuance

1.1 Definitions (continued)

DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using the committed thyroid dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."
DOSE EQUIVALENT XE-133	DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-87, Kr-88, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."

(continued)

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.16.1	-----NOTE----- Only required to be performed in MODE 1. ----- Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 270.0 μCi/gm.	In accordance with the Surveillance Frequency Control Program
SR 3.4.16.2	-----NOTE----- Only required to be performed in MODE 1. ----- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm.	In accordance with the Surveillance Frequency Control Program AND Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period.

3.6 CONTAINMENT SYSTEMS

3.6.3 Containment Isolation Valves

LCO 3.6.3 Each containment isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

--NOTES--

1. Penetration flow path(s) except for 48-inch purge valve flow paths, may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each penetration flow path.
3. Enter applicable Conditions and Required Actions for systems made inoperable by containment isolation valves.
4. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. --------------NOTE------------- Only applicable to penetration flow paths with two containment isolation valves. ----------------------------------- One or more penetration flow paths with one containment isolation valve inoperable except for a containment purge supply and exhaust valve or pressure/vacuum relief valve leakage not within limit.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. <u>AND</u>	4 hours (continued)

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.3.1	Verify each 48 inch purge valve is sealed closed, except for one purge valve in a penetration flow path while in Condition D of this LCO.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.2	Verify each 12 inch vacuum/pressure relief valve is closed, except when these valves are open for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.3	-----------------------------NOTE--------------------------- Valves and blind flanges in high radiation areas may be verified by use of administrative controls. --- Verify each containment isolation manual valve and blind flange that is located outside containment and not locked, sealed or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.4	-----------------------------NOTE--------------------------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. --- Verify each containment isolation manual valve and blind flange that is located inside containment and not locked, sealed or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days
SR 3.6.3.5	Verify the isolation time of each automatic power operated containment isolation valve is within limits.	In accordance with the Inservice Testing Program
SR 3.6.3.6	Not used	

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.6.3.7	-----------------------------NOTE---------------------------- This surveillance is not required when the penetration flow path is isolated by a leak tested blank flange. --- Perform leakage rate testing for containment purge supply and exhaust and vacuum/pressure relief valves with resilient seals.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.8	Verify each automatic containment isolation valve that is not locked, sealed or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.9	Not used	
SR 3.6.3.10	Verify each 12 inch containment vacuum/pressure relief valve is blocked to restrict the valve from opening > 50°.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.11	Not used	

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal absorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and at the relative humidity specified below. Laboratory testing shall be completed at least once per 24 months and after every 720 hours of charcoal operation.

ESF Ventilation System	Penetration	RH
Control Room	2.5%	95%
Auxiliary Building	5.0%	95%
Fuel Handling Building	15.0%	95%

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is less than the value specified below when tested in accordance with ANSI N510-1980 at the system flowrate specified below ± 10% at least once per 24 months.

ESF Ventilation System	Delta P	Flowrate
Control Room	3.5 in. WG	2100 cfm
Auxiliary Building	3.7 in. WG	73,500 cfm
Fuel Handling Building	4.1 in. WG	35,750 cfm

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Waste Gas Holdup System, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in temporary unprotected outdoor liquid storage tanks.

The gaseous radioactivity quantities shall be determined following the methodology in Regulatory Guide 1.24 "Assumptions Used For Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure." The liquid radwaste quantities shall be maintained such that 10 CFR Part 20 limits are met.

(continued)

5.5 Programs and Manuals (continued)

5.5.19 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Ventilation System (CRVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident. The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition, including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CRVS, operating at the flow rate required by the VFTP, at a Frequency of 24 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 24 month assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the Frequencies required by paragraphs c and d for determining CRE unfiltered inleakage and assessing CRE habitability, and measuring CRE pressure and assessing the CRE boundary.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 230 TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 232 TO FACILITY OPERATING LICENSE NO. DPR-82

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON POWER PLANT, UNITS 1 AND 2

DOCKET NOS. 50-275 AND 50-323

1.0 INTRODUCTION

By application dated June 17, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15176A539), as supplemented by letters dated August 31, October 22, November 2, November 6, and December 17, 2015; and February 1, February 10, April 21, June 9, September 15, October 6, and December 27, 2016 (ADAMS Accession Nos. ML15243A363, ML15295A470, ML15321A235, ML15310A522, ML16004A363, ML16032A603, ML16041A533, ML16120A026, ML16169A267, ML16259A117, ML16287A776, and ML17006A051, respectively), Pacific Gas and Electric Company (PG&E, the licensee) requested changes to the Technical Specifications (TSs) for the Diablo Canyon Power Plant, (DCPP), Units 1 and 2.

The amendments would revise the licensing bases to adopt alternative source term (AST) as allowed by Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67, "Accident source term," and approve the methodology for evaluating radiological consequences of design-basis accidents (DBAs) as described in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000 (ADAMS Accession No. ML003716792). The amendments revise TS 1.1, "Definitions"; TS 3.4.16, "RCS [Reactor Coolant System] Specific Activity"; TS 3.6.3, "Containment Isolation Valves"; TS 5.5.11, "Ventilation Filter Testing Program (VFTP)"; and TS 5.5.19, "Control Room Envelope Habitability Program," in support of the revised licensing bases. The amendments also add license conditions to Appendix D, "Additional Conditions," to Facility Operating License Nos. DPR-80 and DPR-82.

The original proposed no significant hazards consideration determination was published in the *Federal Register* on October 13, 2015 (80 FR 61486). The letter dated September 15, 2016, changed the scope of the application, and the proposed no significant hazards consideration determination was republished in the *Federal Register* on November 8, 2016 (81 FR 78664). The supplemental letters dated October 6 and December 27, 2016, provided additional information that clarified the application, did not expand the scope of the application as noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's revised proposed no significant hazards consideration determination as published in the *Federal Register* on November 8, 2016.

2.0 REGULATORY EVALUATION

In 10 CFR 50.36, "Technical specifications," the Commission established its regulatory requirements related to the content of TS. Pursuant to 10 CFR 50.36, TS are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. The proposed changes to the TSs represent changes to the existing TS LCOs, SRs, and administrative controls.

The licensee's request was pursuant to 10 CFR 50.67, which provides a mechanism for licensed power reactors to replace the traditional source term used in the radiological consequence analyses of DBAs. With the exception of the fuel handling accident (FHA) in the fuel handling building (FHB), DCPP's current DBA radiological consequence analyses are based on the source term from U.S. Atomic Energy Commission (AEC) Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," dated March 23, 1962 (ADAMS Accession No. ML021720780). The NRC staff previously approved in Amendment Nos. 163 (Unit 1) and 165 (Unit 2) for DCPP, dated February 27, 2004 (ADAMS Accession No. ML040630575), a selective application of the AST for the FHA in the FHB. However, PG&E has reanalyzed the FHA in the FHB and has included it in this application for NRC staff review.

The NRC staff evaluated the licensee's analysis of the radiological consequences of the affected DBAs for implementation of the AST methodology and the associated changes to the TS proposed by the licensee against the radiological dose requirements specified in 10 CFR 50.67(b)(2) and the dose limits specified in 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 19[1], "Control room." Section 50.67(b)(2) of 10 CFR requires that the licensee's analysis demonstrates with reasonable assurance that:

i. An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv [Sieverts] (25 rem [roentgen equivalent man]) total effective dose equivalent (TEDE).

ii. An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).

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[1] As discussed in DCPP Updated Final Safety Analysis Report (UFSAR or FSARU), Revision 22 (ADAMS Accession No. ML15138A105), Chapter 3, Design of Structures, Components, Equipment, and Systems," although regulatory correspondence often refers to the 1971 criteria, the DCPP licensing basis remains the 1967 GDCs, except for GDCs 3, 4, 17, 18, 19, 54, 55, 56, and 57.

iii. Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

This safety evaluation (SE) addresses the impact of the proposed changes on previously analyzed DBA's radiological consequences and the acceptability of the revised analysis results. The regulatory requirements from which the NRC staff based its acceptance are the accident radiation dose values in 10 CFR 50.67; the accident specific guideline values in Regulatory Position 4.4, "Acceptance Criteria," of RG 1.183; Table 1 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition (SRP)," Section 15.0.1, Revision 0, "Radiological Consequence Analysis Using Alternative Source Terms," July 2000 (ADAMS Accession No. ML003734190). The licensee has not proposed any significant deviation or departure from the guidance provided in RG 1.183.

The NRC staff's evaluation is based upon the following regulations, regulatory guides, and standards:

- Section 50.34 of 10 CFR, "Contents of applications; technical information," defines the content requirements for the FSARU, including evaluations required to show that accident dose criteria are met.

- Section 50.36 of 10 CFR, "Technical specifications," requires the technical specifications to include LCOs. LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a LCO of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.

- Section 50.49 of 10 CFR, "Environmental qualification of electric equipment important to safety for nuclear power plants," in part, requires that the electrical equipment important to safety, which are relied upon to remain functional during and following design-basis events be qualified for accident (harsh) environment.

- Section 50.67 of 10 CFR, "Accident source term," provides a mechanism for licensed power reactors to replace the traditional source term used in the radiological consequence analyses of DBAs.

- Section 50.90 of 10 CFR, "Application for amendment of license, construction permit, or early site permit," requires a holder of an operating license to fully describe the changes desired to amend the license.

- Appendix A to 10 CFR 50, "General Design Criteria for Nuclear Power Plants," establish minimum requirements for the principle design criteria for water-cooled nuclear power plants.

- Appendix A to 10 CFR 50, GDC 19, "Control room," requires that a control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents.

- Appendix A to 10 CFR Part 50, GDC 17, "Electric power systems," defines the requirements to provide onsite and offsite electric power systems to permit functioning of structures, systems, and components important to safety.

  Criterion 17 requires, in part, that:

  > An onsite electrical power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system... shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

- Appendix J to 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," provides containment leakage test requirements to ensure that (a) leakage through containments or systems and components, penetrating containments, does not exceed allowable leakage rates specified in the TS; and (b) integrity of the containment structure is maintained during the service life of the containment.

- Safety Guide 23, "Onsite Meteorological Programs," February 1972 (ADAMS Accession No. ML020360030).

- RG 1.23, Revision 1, "Meteorological Monitoring Programs for Nuclear Power Plants," Revision 1, March 2007 (ADAMS Accession No. ML070350028).

- RG 1.52, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water Cooled Nuclear Power Plants," Revision 3, June 2001 (ADAMS Accession No. ML011710176).

- RG 1.111, Revision 1 (for comments), "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977 (ADAMS Accession No. ML003740354).

- RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1, November 1982 (Reissued February 1983) (ADAMS Accession No. ML003740205).

- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000 (ADAMS Accession No. ML003716792).

- RG 1.194, "Revision 0, Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," June 2003 (ADAMS Accession No. ML031530505).

- RG 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors," Revision 1, January 2007 (ADAMS Accession No. ML063560144).

- RG 1.203, "Transient and Accident Analysis Methods," December 2005 (ADAMS Accession No. ML053500170).

- Information Notice (IN) 97-78, "Crediting of Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times" dated October 23, 1997 (ADAMS Accession No. ML031050065).

- SRP, Section 2.3.3, "Onsite Meteorological Measurements Program," Revision 3, March 2007 (ADAMS Accession No. ML063600394); Section 2.3.4, "Short-Term Atmospheric Dispersion Estimates for Accident Releases," Revision 3, March 2007 (ADAMS Accession No. ML070730398); Section 6.4, "Control Room Habitability System," Revision 3, March 2007 (ADAMS Accession No. ML070550069); Section 6.5.2, "Containment Spray as a Fission Product Cleanup System," Revision 4, March 2007 (ADAMS Accession No. ML070190178); Section 15.0, "Introduction –Transient and Accident Analysis," Revision 3, March 2007 (ADAMS Accession No. ML070710376); Section 15.0.1, "Radiological Consequence Analysis Using Alternative Source Terms," Revision 0,.July 2000 (ADAMS Accession No. ML003734190); Section 15.0.2, Review of Transient and Accident Analysis Methods," March 2007 (ADAMS Accession No. ML070820123); and Chapter 18, Section 18.0, "Human Factors Engineering," Revision 2, March 2007 (ADAMS Accession Nos. ML070670253) and Appendix 18-A, "Crediting Manual Operator Actions in Diversity and Defense-in-Depth Analyses, Revision 0, April 2014 (ADAMS Accession No. ML13115A156).

- NUREG-0696, "Functional Criteria for Emergency Response Facilities," Final Report, February 1981 (ADAMS Accession No. ML051390358).

- NUREG-0737, "Clarification of TMI [Three Mile Island] Action Plan Requirements," November 1980 (ADAMS Accession No. ML102560051) and "Clarification of TMI Action Plan Requirements - Requirements for Emergency Response Capability," Supplement No. 1, Reprinted February 1989 (ADAMS Accession No. ML102560009).

- NUREG/CR-2260 (NUS-3854), "Technical Basis for Regulatory Guide 1.145, 'Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants,'" October 1981 (ADAMS Accession No. ML12045A197).

- NUREG/CR-2858, "PAVAN: An Atmospheric-Dispersion Program for Evaluating Design-Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations," prepared by Pacific Northwest Laboratory (PNL-4413), November 1982, (ADAMS Accession No. ML12045A149).

- NUREG/CR-6331, "Atmospheric Relative Concentrations in Building Wakes," prepared by Pacific Northwest National Laboratory (PNNL-10521), Revision 1, May 1997.

- American National Standards Institute (ANSI)/American Nuclear Society (ANS) Standard 58.8-1994, "Time Response Design Criteria for Safety-Related Operator Actions."

3.0 TECHNICAL EVALUATION

3.1 Proposed TS Changes

The licensee has proposed the following TS changes:

TS 1.1 Definitions; DOSE EQUIVALENT I-131

Definition for Dose Equivalent Iodine (I)-131 currently reads:

> DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using thyroid dose conversion factors from Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or Table E-7 of Regulatory Guide 1.109, Revision 1, NRC, 1977, or International Commission on Radiological Protection (ICRP) Publication 30, 1979, Supplement to Part 1, pages 192-212, Table titled "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity," or Table 2.1 of EPA [Environmental Protection Agency] Federal Guidance Report No. 11, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

The definition for Dose equivalent I-131 would be revised to read:

> DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using the committed thyroid dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

SR 3.4.16.1

SR 3.4.16.1, currently states:

> Verify reactor coolant DOSE EQUIVALENT XE [xenon]-133 specific activity ≤ 600 μCi/gm [microcuries per gram].

SR 3.4.16.1 would be revised to state:

> Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 270.0 μCi/gm.

LCO 3.6.3

Note 1 for ACTIONS associated with LCO 3.6.3 currently states:

> Penetration flow path(s) except no more than two of three flow paths for containment purge supply and exhaust and containment vacuum/pressure relief paths at one time may be unisolated intermittently under administrative controls.

Revised Note 1 for ACTIONS associated with LCO 3.6.3 would state:

> Penetration flow path(s) except for 48-inch purge valve flow paths may be unisolated intermittently under administrative controls.

SR 3.6.3.1

SR 3.6.3.1 currently states:

Not used.

SR 3.6.3.1 would be revised to state:

> Verify each 48 inch purge valve is sealed closed, except for one purge valve in a penetration flow path while in Condition D of this LCO.

Also, the following will be added to the FREQUENCY for SR 3.6.3.1:

> In accordance with the Surveillance Frequency Control Program.

SR 3.6.3.2

SR 3.6.3.2 currently states:

> Verify each 48 inch containment purge supply and exhaust and 12 inch vacuum/pressure relief valve is closed, except when these valves are open for pressure control, ALARA [as low as reasonably achievable] or air quality considerations for personnel entry, or for Surveillances that require the valves to be open.

SR 3.6.3.2 would be revised to state:

> Verify each 12 inch vacuum/pressure relief valve is closed, except when these valves are open for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open.

SR 3.6.3.7

SR 3.6.3.7 FREQUENCY currently states:

In accordance with the Surveillance Frequency Control Program.

AND

For containment purge supply and exhaust valves only, within 92 days after opening the valve.

SR 3.6.3.7 FREQUENCY would be revised to state:

In accordance with the Surveillance Frequency Control Program.

TS 5.5.11.c

Paragraph 5.5.11.c currently states:

Demonstrate for each of the ESF [engineered safety feature] systems that a laboratory test of a sample of the charcoal absorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30 °C [degrees Centigrade] and at the relative humidity specified below. Laboratory testing shall be completed at least once per 24 months and after every 720 hours of charcoal operation.

ESF Ventilation System	Penetration	RH
Control Room	2.5%	95%
Auxiliary Building	15%	95%
Fuel Handling Building	15%	95%

Paragraph 5.5.11.c would be revised to state:

Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal absorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30 °C and at the relative humidity specified below. Laboratory testing shall be completed at least once per 24 months and after every 720 hours of charcoal operation.

ESF Ventilation System	Penetration	RH
Control Room	2.5%	95%
Auxiliary Building	5.0%	95%
Fuel Handling Building	15.0%	95%

TS 5.5.19, "Control Room Envelope Habitability Program"

TS 5.5.19 (first paragraph) currently states:

> A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Ventilation System (CRVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem whole body or its equivalent to any part of the body for the duration of the accident. The program shall include the following elements:

TS 5.5.19 (first paragraph) would be revised to state:

> A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Ventilation System (CRVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident. The program shall include the following elements:

3.2 Atmospheric Dispersion Estimates

The NRC staff's evaluation of the licensee's atmospheric dispersion modeling analyses and proposed X/Q values, which are input to the NRC staff's review of the licensee's radiological dose assessments at onsite and offsite receptor locations, is based upon the regulations and regulatory guidance documents 10 CFR Part 50, Appendix A, Criterion 19; NUREG-0696; NUREG-0737; SRP Section 2.3.3; SRP Section 2.3.4; SRP Section 6.4; Safety Guide 23; RG 1.23; RG 1.111, RG 1.145; RG 1.183; RG 1.194; NUREG/CR-2260; NUREG/CR-2858; and NUREG/- -6331.

The licensee performed atmospheric dispersion modeling analyses of hypothetical design-basis accident-related releases to air in order to evaluate the potential effects of radiological doses at onsite receptor locations associated with occupants of the control room (CR) and Technical Support Center (TSC), and at offsite receptors located on the exclusion area boundary (EAB) and the outer boundary of the low-population zone (LPZ) surrounding the DCPP site.

Meteorological data obtained from an onsite measurement program was key input to both modeling analyses. Onsite dispersion estimates were based on the results of the NRC-accepted ARCON96 dispersion model. Dispersion estimates at the offsite EAB and LPZ receptor locations were developed using a proprietary dispersion model, designated EN-113, which the licensee stated, implements applicable regulatory guidance for such analyses. The sections that follow, summarize the NRC staff's evaluation of these analyses, including the input information and assumptions necessary to run the ARCON96 and EN-113 atmospheric dispersion codes.

A series of requests for additional information (RAIs) were developed over the course of the NRC staff's review. In addition, a regulatory audit of specific model-related issues was also conducted on August 3 and 4, 2016, to better understand and verify various inputs to the atmospheric dispersion models (especially the EN-113 code) and assumptions made in developing those inputs. Licensee responses to those RAIs, including those issued following the regulatory audit, are summarized and referenced, where appropriate, throughout this section.

The initial set of RAIs was developed based on the original submittal dated June 17, 2015. Those RAIs primarily focused on acquiring meteorological data, input and output files for the ARCON96 dispersion modeling runs (including meteorological data formatted for input to that model), input files for the proprietary EN-113 dispersion model, and a description of the file structure and plant drawings used to confirm certain model input parameters. The licensee provided responses to these RAIs on November 2, 2015 (ADAMS Accession No. ML15321A235).

3.2.1 Meteorological Data

The meteorological variables relevant to the dispersion modeling analyses performed for this license amendment request (LAR) are wind speed, wind direction, and an indicator of atmospheric stability. The data was obtained from a monitoring program conducted at two onsite locations – a 76-meter (m) primary tower and a 60-m backup tower. The current licensing basis conforms to Safety Guide 23, the predecessor to RG 1.23.

Data from the primary tower was the primary input to the atmospheric dispersion models. Relevant variables from the 76-m primary tower consist of wind speed and wind direction at two levels (i.e., 10- and 76-m) and a determination of the vertical temperature difference (or delta-T) between the 76 and 10 m levels. The delta-T measurement is used to characterize atmospheric stability (or the diffusion potential of airborne releases). Measurements from the 60-m backup meteorological tower, which were used as a substitute for missing primary tower data, include wind speed and wind direction at the 10- and 60-m levels and delta-T between 60 and 10 m levels. Section 2.3.3 of the Final Safety Analysis Report Update (UFSAR or FSARU), Revision 22 (ADAMS Accession No. ML15138A105), provides additional information about the onsite monitoring program and its development at the DCPP site.

Two of the several changes to the current licensing basis (CLB) proposed in Section 2.1 of Revision 5 of the Technical Report submitted by letter dated December 27, 2016 (referenced as the TR hereafter), as well as in previous revisions of the TR that supports this LAR (i.e., Items 2 and 3) and that directly affect the dispersion modeling analyses, is the licensee's use of a more recent period of record (POR) of meteorological data than that discussed in the current Revision 22 (and previous revisions) of the FSARU. Meteorological data for a 5-year POR, from 2007 through 2011, was input to the ARCON96 onsite and EN-113 offsite modeling runs. The NRC staff considers a 5-year POR to be a reasonable duration for capturing year-to-year variations in conditions that effect atmospheric transport (i.e., wind speed and direction) and diffusion (i.e., atmospheric stability), which meets the SRP Section 2.3.4 criterion that three or more whole years of onsite meteorological data should be provided.

Because of this change to the CLB, the remainder of this section discusses the NRC staff's evaluation of the quality of this 5-year data set relative to the cited regulatory guidance and its development by the licensee for input to the dispersion modeling runs, and with respect to its long-term representativeness of conditions at the DCPP site.

3.2.1.1 Meteorological Data Quality Input to Dispersion Models

The NRC staff reviewed the meteorological data provided by the licensee in its letter dated November 2, 2015. This information included five text files of sequential hourly data formatted in accordance with Appendix A to RG 1.23. Five spreadsheet files of sequential hourly meteorological data were also provided in a format comparable to the input required by the ARCON96 dispersion model, based on Revision 1 of the user's guidance for that model (NUREG/CR-6331).

Quantitative and qualitative data validation checks performed by the NRC staff included, among others: data recovery and completeness; date/time stamps; joint recovery and characteristics of wind speed, wind direction, and atmospheric stability; variation of atmospheric stability conditions by time of day and other weather elements; and identification and frequency of calm wind conditions.

As one indicator of data quality, the NRC staff requested information about data recovery rates for the parameters relevant to the dispersion analyses, individually and for the composite recovery of concurrent wind speed, wind direction, and atmospheric stability for each year and for the 5-year POR combined. The licensee's responses to the referenced RAIs indicated that data recovery on an annual basis, for 10- and 76-m wind speed and wind direction, and for atmospheric stability, exceeded the 90 percent criterion specified in both Safety Guide 23 and RG 1.23. Although the letter dated November 2, 2015, did not provide composite data recovery rates for wind speed, wind direction, and atmospheric stability, the recovery rates noted for the individual variables suggest that the 90 percent criterion was likely met on a composite basis for those three parameters. The NRC staff confirmed this to be the case as a result of its own confirmatory dispersion modeling, which used the same meteorological data to develop that part of its model inputs.

As another check of data quality (e.g., an indication of how well the meteorological monitoring system is maintained, including the performance of individual instrumentation), the NRC staff requested an accounting of any data substitution that might have occurred over the 5-year POR. The licensee's letter dated November 2, 2015, indicated that between 2007 and 2011, about 12.5 percent of the data from the primary monitoring system was supplemented by measurements from the backup 60-m meteorological tower; on an annual basis the highest during 2008 (i.e., 28.78 percent) due to damage to the primary tower; and during 2011 (i.e., 14.52 percent) due to instrument replacement and calibrations.

The NRC staff noted, during its review of the referenced RAI responses, by letter dated November 2, 2015, that unlike the NRC-accepted PAVAN dispersion model, which uses joint frequency distributions (JFDs) of wind speed, wind direction, and atmospheric stability in calculating atmospheric dispersion factors (or X/Q values), the proprietary EN-113 code utilizes sequential hourly meteorological data. The NRC staff also noted that the units of measure for wind speed differed between the RG 1.23-formatted data files (i.e., meters per second) and the ARCON96 spreadsheet files (i.e., miles per hour).

The NRC staff made several observations during its review of the licensee's hourly RG 1.23-formatted and corresponding ARCON96-fomatted data files but was unable to determine whether the appropriate meteorological data was input to the licensee's dispersion modeling runs. These observations included:

- Numerous discrepancies between concurrent hourly stability class values in the RG 1.23 and ARCON96 data files; and

- Numerous periods of extended persistence of moderately to extremely stable atmospheric conditions (i.e., F and G Pasquill-Gifford stability classes) with very little or no variation, including two separate occasions lasting the better part of 5 consecutive days, with a only a few hours scattered over those time intervals not being designated F or G stability.

As a result, the NRC staff issued RAIs to the licensee on February 17, 2016 (ADAMS Accession No. ML16048A232); the licensee's responses were provided on April 21, 2016 (ADAMS Accession No. ML16120A026). Resolution of the first of these two issues is addressed below. Subsection 3.2.1.2 of this SE, includes a discussion of the licensee's response to the second issue, which contributed to the NRC staff's evaluation of the long-term representativeness of the 2007 through 2011 POR from an atmospheric dispersion standpoint.

In the RAI responses by letter dated April 21, 2016, the licensee confirmed that the atmospheric stability class data used in the ARCON96 model run files had been developed using an alternate methodology to the delta-T approach called for in RG 1.23, noting that approximately 30 percent of the concurrent pairs were different (see the background information in the original NRC staff RAIs of February 17, 2016, for additional details on the NRC staff's assessment). Further, the licensee confirmed that the delta-T data included in the previously submitted sequential hourly data files in RG 1.23 format are correct.

Consequently, the licensee revised the sequential hourly meteorological data in ARCON96 format and the text file of hourly meteorological data input to the proprietary EN-113 dispersion model incorporating stability classes based on delta-T in accordance with RG 1.23. The licensee also reran the ARCON96 (onsite) and EN-113 (offsite) modeling runs and performed a comparison of the X/Q results to determine the potential effects of the alternate stability classification approach on the X/Q values and related dose calculations.

The licensee's re-analyses showed that most of the X/Q values, based on the RG 1.23 stability classification methodology, were lower than those initially estimated using the alternate approach. Of the X/Q values that did increase, those dispersion factors did not represent controlling values input to the dose calculations for a given accident release scenario. Based on this evaluation, the licensee proposed the following in its letter dated April 21, 2016:

- Upon AST implementation, the updated X/Q values reported herein [i.e., the TR and proposed revisions to the FSARU] will become the DCPP licensing basis.

- The dose consequences reported in the LAR 15-03 [which are based on the original submitted set of X/Q values] are considered conservative and bounding and will remain unchanged.

The licensee's statements in the RAI responses submittal dated April 21, 2016, applied to the X/Q values and dose consequences estimated for both the onsite accident analysis associated with the CR and TSC, and at offsite receptors located on the EAB and the outer boundary of the LPZ.

Finally, the NRC staff generated a 5-year composite JFD for input to its PAVAN dispersion modeling runs based on the sequential hourly data sets initially provided by the licensee. Stability classes were determined from the reported delta-T values in accordance with RG 1.23. The JFD incorporated 13 wind speed classes (plus calm winds), taking into consideration the guidance indicated by Table 3 of Revision 1 to RG 1.23 (i.e., finer resolution of wind speed classes in the lower range) and the upper range of wind speeds observed in the 2007 to 2011 POR. In constructing this JFD, the NRC staff assumed a conservatively low value of 0.5 miles per hour (0.22 meter per second) as the threshold for calm winds, consistent with the value assumed by the licensee in its accident modeling analysis with the EN-113 code.

Based on the information provided by the licensee and its responses to the related RAIs, the NRC staff considers the meteorological data sets submitted by letter dated April 21, 2016, and used for the licensee's ARCON96 and EN-113 dispersion modeling analyses to be acceptable.

3.2.1.2 Long-Term Representativeness of Meteorological Data

The licensee's RAI responses by letter dated April 21, 2016, provided the requested technical and climatological justification for the numerous periods of extended persistence of moderately to extremely stable atmospheric conditions observed by the NRC staff, as a potential concern in establishing data validity.

The licensee attributes the apparent effects of the marine environment on the meteorological conditions measured at the DCPP plant site, which is located directly adjacent to the Pacific Ocean. These include upwelling of cold water temperatures (which affect low-level air temperatures and, therefore, the vertical temperature difference used to determine the stability class), in conjunction with persistent diurnal wind patterns and the presence and persistence of summertime coastal low-lying stratus clouds. The licensee correlated the frequency of occurrence of each of the seven stability classes (A through G) for the 2007 through 2011 POR, with summaries from the late 1980s, the 1990s, and the 2000s using the same classification approach. While some year-to-year variation is seen (and can be expected), these summaries illustrate a generally consistent distribution of stability class frequencies.

Therefore, given the information provided in the referenced letter dated April 21, 2016, the NRC staff considers the atmospheric stability conditions based on the delta-T approach in RG 1.23, as incorporated in the meteorological data sets used for the ARCON96 and EN-113 dispersion modeling analyses, to be representative of long-term conditions at the site.

3.2.2 Onsite (Control Room and Technical Support Center) Atmospheric Dispersion Factors

The licensee used the NRC-accepted ARCON96 dispersion model, which implements (with qualification) the guidance in RG 1.194, to estimate short-term, accident-related X/Q values at onsite receptor locations. The purpose of the NRC staff's evaluation is to confirm the reasonability of these estimated X/Q values, which are a direct input to the calculation of doses to occupants of the CR and TSC. To support the NRC staff's review, the licensee's letter dated November 2, 2015, provided an initial set of input and output files for the ARCON96 dispersion modeling analysis along with plant drawings used to confirm certain model input parameters.

The NRC staff's evaluation of the meteorological data input to the ARCON96 model runs is discussed in Section 3.2.1 of this SE.

The documentation accompanying the LAR submittal, indicates that seven onsite receptor points were modeled, including, for each of the two units, CR air intakes for normal operation and for emergency (pressurization) conditions, as well as a roof-level location at the center of the CR boundary to represent unfiltered in-leakage including that from ingress/egress to the CR. Similarly, a receptor was located at the normal operational intake for the TSC and another at roof-level at the center of the TSC, which was used to represent an average value for unfiltered in-leakage including that from ingress and egress to the TSC. The licensee indicated that the emergency (pressurization) intakes are common to the CR and TSC.

Accident scenarios considered by the licensee included potential releases due to a:

- Loss-of-Coolant Accident (LOCA);
- FHA in the FHB;
- FHA in the Containment;
- Locked Rotor Accident (LRA);
- Control Rod Ejection Accident (CREA);
- Main Steam Line Break (MSLB);
- Steam Generator Tube Rupture (SGTR); and
- Loss-of-Load (LOL) event.

Among these eight accident scenarios, 20 release points (10 associated with Unit 1 and 10 associated with Unit 2) were modeled: 1) containment building (CB) edge; 2) the plant vent; 3) the refueling water storage tank (RWST) vent; 4) the containment penetration location for Area GE[2]; 5) the containment penetration location for Area GW/FW; 6) the FHB; 7) the equipment hatch location; 8) main steam safety valves (MSSVS); 9) 10 percent atmospheric dump valves (ADVS); and 10) the assumed location for MSLB releases.

Appendix A to the TR provides additional information about these release points and receptor locations, including a site layout and arrangement relative to Plant North. The TR states that there is a 23-degree azimuth clockwise offset between True North and Plant North, which must be accounted for in the modeling runs because the wind direction data input to the ARCON96 model are referenced to True North.

The TR also provides a detailed summary of key parameters input to the more than 120 ARCON96 dispersion model runs covering meteorological information, source-related data (e.g., type and height of release, exhaust characteristics, building dimensions to account for structural wake effects on dispersion), receptor-related data (e.g., assumed distance between the source and receptor, receptor height, direction from the receptor to the source), as well as various default settings opted for by the licensee (consistent with the guidance in Table A-2 of RG 1.194).

Because of the release point locations and their heights in relation to the surrounding building complex, the licensee considered all releases to be ground-level sources. This approach is consistent with the guidance in Regulatory Positions C.3.2.1 and C.3.2.2 of RG 1.194. In

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[2] Location designations as shown in Figure A-1 of the TR, Revision 5 (ADAMS Accession No. ML17006A051).

addition, nearly all releases were assumed to be point sources with the exception of releases from the CB edge. The CB edge model runs were treated as diffuse area sources, which is intended to represent potential leakage from the surface area of the containment wall. As such, these diffuse area sources were assigned initial diffusion coefficients based on the height and width of the CB. In these cases, the closest edge of the Unit 1 or Unit 2 CB to the given receptor being modeled was determined to conservatively minimize the plume transport distance. The NRC staff reviewed the licensee's determination of these coefficients and agrees with this approach.

Using the plant drawings provided with the referenced RAI responses by letter dated November 2, 2015, the NRC staff's review also reviewed: distances between the modeled sources and receptors; the orientation between receptor locations and release points; and the height and cross-sectional areas of those buildings assumed (based on source-receptor orientations) to enhance the dispersion of a given release due the structure's turbulent wake effects.

The licensee also identified several changes to the CLB that directly affect the use of the results of the ARCON96 dispersion modeling analysis. These include Items 7, 9, and 10 as proposed in Section 2.1, "Proposed Changes to Current Licensing Basis," of the TR that supports the LAR, as summarized below:

- Item 7 – A proposed credit for the dual ventilation intake design (i.e., the emergency (pressurization) CR air intakes at each unit), which includes redundant radiation monitors at each intake and capability of automatic initial selection of the cleaner intake, and an assumption of proper intake selection manually throughout an event, which would allow for a reduction factor of 4 to be applied to the X/Q values for the more favorable CR air intake (i.e., the lower of the emergency intakes from either unit). This approach and criteria for acceptance are addressed in Regulatory Position C.3.3.2.3 of RG 1.194.

- Item 9 – Credit for a reduction factor of 5 is proposed to be applied to the calculated X/Q values for releases from the MSSVs and 10 percent ADVs. The licensee stated that these relief valves are uncapped and vertically oriented. As a result, these releases are considered to be "energetic" due to the high vertical velocity of the steam discharges. This approach and criteria for acceptance are addressed in Regulatory Position C.6 of RG 1.194. Such releases would be associated with MSLB, SGTR, LRA, CREA, and LOL accident scenarios. The reduced X/Q values would be input to the dose calculations for the applicable scenario.

- Item 10 – Credit is proposed to be taken for the fact that because the MSSV and 10 percent ADV release points are in close proximity to and slightly above the normal operation CR air intake of the affected unit (i.e., the unit at which the accident is occurring), and because such releases would be "energetic," the resultant post-accident plume would not contaminate the normal CR intake of the affected unit. The licensee implemented this proposed credit by not including ARCON96 modeling runs for either MSSV or 10 percent ADV releases to the normal operation CR intake of the same unit. As in Item 9, such releases would be associated with MSLB, SGTR, LRA, CREA, and LOL accident scenarios.

The NRC staff determined the X/Q reduction factor credit proposed in Item 7 above, on the basis of its review of the intake locations for each unit as illustrated by the site layout and source-receptor arrangement in Appendix A to the TR to be acceptable. The NRC staff also considers the licensee's approach to implementing the credit proposed in Item 10 above, to be acceptable on the basis of the related discussion in the TR, including the fact that the stated line-of-sight distance of separation between either the MSSV or 10 percent ADV release points and the normal operation CR air intake of the same unit is only 1.5 m. Regulatory Position C.3.4 of RG 1.194 indicates that 10-m represents the minimum distance of separation for ARCON96 model calculations.

However, the NRC staff was unable to initially conclude that one of the criteria for implementing the credit proposed in Item 9 above was met. Regulatory Position C.6 of RG 1.194 calls for the ratio between the exit velocities of vertically-oriented and uncapped releases (in the case of this LAR from the MSSVs or 10 percent ADVs) and the applicable 95th-percentile wind speed (defined in Regulatory Position C.6 of RG 1.194 as that wind speed that is not exceeded more than 5 percent of the time) at the release point height to be greater than a factor of 5 considering the time-dependent vertical velocity.

During the August 3 and 4, 2016, regulatory audit, the NRC staff discussed with the licensee the scope of the onsite dispersion modeling analysis, including topics related to specific ARCON96 model inputs and assumptions. One topic concerned demonstrating that the design-basis LOCA (i.e., the CLB event) still represents the highest potential dose to TSC occupants given that MSSV, 10 percent ADV, and MSLB accident release scenarios were unanalyzed and may have higher potential impacts at the TSC receptors than the LOCA event. Topics related to various model inputs and assumptions included: building dimensions and cross-sectional areas and their development; receptor-to-source orientations; and the need to correct estimated ratios between the exit velocities for MSSV and 10 percent ADV releases to ambient wind conditions at those release heights in order to justify a proposed credit for impact reduction from those sources. See the corresponding Audit Report dated October 27, 2016 (ADAMS Accession No. ML16279A343).

Subsequent to the regulatory audit, the NRC staff issued an RAI on September 7, 2016 (ADAMS Accession No. ML16251A091), which included requests for the licensee to:

- demonstrate, consistent with Regulatory Position C.1.3.2 of RG 1.183, "Re-Analysis Guidance," that the design-basis LOCA (i.e., the CLB event), still represents the worst-case accident scenario for onsite dose consequences given that X/Qs were unanalyzed at the TSC receptors in the initial LAR submittal for MSSV, 10 percent ADV, and MSLB release scenarios;

- correct the stated ratio of the vertical velocity of MSSV and 10 percent ADV releases to the improperly interpreted 95th-percentile 10-m wind speed that is used to justify the proposed change to the CLB (see Item 9 above) that would allow the application of a reduction factor of 5 to the X/Q values for "energetic" releases from these relief valves;

- reconcile the difference between the cross-sectional area of the containment (reactor) building as used in the licensing basis for this LAR and that appearing in Subsection 2.3.4.7 of the FSARU; and

- identify the cross-sectional areas of the CBs, RWSTs, and FHBs for Units 1 and 2 input to the ARCON96 model runs and explain how those cross-sectional areas were determined (indicating the applicable structures or portions of structures considered, building dimensions (width and height), and, if applicable, the method of handling portions of irregularly-shaped structures).

Subsection 3.2.1.1 of this SE, indicates that as a result of resolving the atmospheric stability classification issue, the meteorological data sets input to the ARCON96 and EN-113 dispersion models were revised. New sets of model runs were generated and provided to the NRC staff by the licensee's letter dated April 21, 2016. The ARCON96 dispersion modeling analysis was further supplemented based on the licensee's response to the RAI, issued post audit, regarding the need to demonstrate that the design-basis LOCA still represents the worst-case accident scenario for the previously unanalyzed modeling at the TSC receptors for MSSV, 10 percent ADV, and MSLB releases. These additional ARCON96 input and output files were included with the licensee's RAI responses by letter dated October 6, 2016 (ADAMS Accession No. ML16287A776).

The NRC staff verified the licensee's onsite atmospheric dispersion estimates by running the ARCON96 computer code and obtaining similar results for all short-term time intervals, as listed in Tables 5.2-2, 5.2-3, and 5.2-4 of Revision 4 to the TR, provided with the referenced October 6, 2016, RAI responses, and resubmitted as Revision 5 by letter dated December 27, 2016.

Further, based on the corrections to the 95th-percentile 10-m wind speed (and with the addition of the 95th-percentile wind speed at the 76-m measurement level), the NRC staff was able to conclude that the ratio of the vertical velocity of MSSV and 10 percent ADV releases to the 95th-percentile wind speed at the release point height would be greater than the required threshold value specified in Regulatory Position C.6 of RG 1.194. Consequently, the proposed crediting of the X/Q reduction factor in Item 9 above, is considered to be acceptable by the NRC staff. Finally, the NRC staff reviewed and found to be acceptable, the X/Q values listed as input to the dose consequence evaluation for the various accident release scenarios in Section 7 of the referenced TR.

3.2.3 Offsite (EAB and LPZ) Atmospheric Dispersion Factors

The licensee used a proprietary dispersion model designated EN-113 to estimate short-term, accident-related X/Q values at offsite receptors located on the EAB and outer boundary of the LPZ surrounding the DCPP site. This model represents an alternative approach to the NRC-accepted PAVAN dispersion model (NUREG/CR-2858, Revision 1, dated November 1982), which implements the guidance in RG 1.145. The acceptance criteria and review procedures in SRP Section 2.3.4 allow for the use of alternative approaches but calls on the NRC staff to review the applicant's evaluation of how the proposed alternative(s) provide an acceptable method of complying with the relevant acceptance criteria.

Section 3.0, "Computer Codes," of the TR identifies the EN-113 dispersion model, titled "Atmospheric Dispersion Factors," as one of the computer codes used in support of the LAR and is stated to "have been verified and validated under the CB&I [Chicago Bridge and Iron] S&W [Stone and Webster], Inc. NRC approved Quality Assurance Program, and...shown to be accurate and acceptable...." The description provided in that section of the TR summarizes some of the model's capabilities and indicates that EN-113 calculations at the EAB and LPZ follow "the methodology and logic outlined in NRC Regulatory Guide 1.145."

Consistent with the intent of the guidance to NRC staff in SRP Section 2.3.4, and given the proprietary nature of the EN-113 code, the NRC staff developed input data files for the PAVAN dispersion model that were equivalent, to the extent possible, to the licensee's EN-113 model runs. The PAVAN input files were based, in part, on the information provided by the licensee in its letter dated November 2, 2015, which included a description of the EN-113 input file structure and plant drawings to verify certain model input parameters. The NRC staff's evaluation of the meteorological data input to the EN-113 model runs is discussed in Section 3.2.1 of this SE.

The purpose of these PAVAN modeling runs was two-fold: first, to support the NRC staff's determination of the acceptance of the licensee's use of the EN-113 dispersion model as an alternate approach for this LAR submittal; and second, to confirm the reasonability of the estimated X/Q values at the EAB and outer boundary of the LPZ for input to related offsite dose calculations. The intent of this evaluation by the NRC staff was not to endorse the proprietary EN-113 dispersion model itself.

Many of the EN-113 model options and input information have direct parallels to the data and information used to set up PAVAN model runs. However, the proprietary EN-113 code is a legacy model of sorts and its scope and capabilities extend beyond that of the NRC-accepted PAVAN model. Consequently, it was necessary to understand the meaning of certain model options, to clarify the purpose and function of some of the selections made by the licensee, and to identify modeling analysis-related information gaps in the TR documentation and earlier proposed revisions to the FSARU. The modeling-related information was discussed during the August 3 and 4, 2016, regulatory audit of the licensee's EN-113 and ARCON96 dispersion modeling analyses. See the corresponding Audit Report dated October 27, 2016 (ADAMS Accession No. ML16279A343).

As indicated in Section 3.2.2 of this SE, subsequent to the regulatory audit, the NRC staff issued an RAI on September 7, 2016 (ADAMS Accession No. ML16251A091), which included requests for the licensee to:

- Specify the terrain adjustment factors applied to outer boundary of the LPZ model run;
- Explain how the height and cross-sectional area of the containment (reactor) building were determined (used in accounting for structural wake effects on dispersion);
- Reconcile a difference between the cross-sectional area of the containment (reactor) building as used in the licensing basis for this LAR and that appearing in Subsection 2.3.4.7 of the FSARU;
- Describe the basis for, location of, and distance to offshore EAB receptors; and
- Provide summary tables, with selected input and output, equivalent to the PAVAN model that includes:
    - ➢ distances to onshore and offshore EAB receptor locations, relative to Units 1 and 2 for each of the sixteen standard 22.5-degree direction sectors along with corresponding 0.5-percent sector-dependent 0- to 2-hour X/Q values;
    - ➢ distances to the outer boundary of the LPZ for each of the sixteen standard 22.5-degree direction sectors along with corresponding 0.5-percent sector-dependent 0- to 2-hour and annual average X/Q values; and

> 5-percent overall site 0- to 2-hour X/Q values for the Unit 1 and Unit 2 EAB and at the outer boundary of the LPZ, as well as X/Qs at the outer boundary of the LPZ for intermediate time periods (i.e., 2 to 8 hours, 8 to 24 hours, 1 to 4 days (24 to 96 hours), and 4 to 30 days (96 to 720 hours) for the assumed accident duration.

The licensee provided its responses to the RAI by letter dated October 6, 2016 (ADAMS Accession No. ML16287A776).

The NRC staff completed development of its confirmatory PAVAN dispersion model runs (i.e., one each at the EAB distances relative to Unit 1 and Unit 2 and one for the outer boundary of the LPZ), based on the above referenced information and the following assumptions:

- Meteorological data represented as a composite JFD of wind speed, wind direction, and atmospheric stability class for the 5-year POR from 2007 through 2011 (as opposed to the sequential hourly data input to EN-113) (see Subsection 3.2.1.1);

- A uniform distance to the outer boundary of the LPZ (i.e., 9650 m) with a release point centered between Units 1 and 2 corresponds to a radial distance of 6 miles previously designated in the FSARU as the LPZ and in the emergency plan as the nearest residential community – a conservative value compared to the current LPZ distance specified in Subsection 1.2.1.1 of the FSARU (Revision 22);

- All hypothetical accident releases from either Unit 1 or Unit 2 modeled as ground-level sources (default elevation of 10 m) with plume meander accounted for under low wind speed conditions in accordance with Regulatory Position C.1.3.1 of RG 1.145;

- Default values for terrain adjustment (recirculation) factors based on users guidance for the PAVAN model (Figure 4.2 of NUREG/CR-2858) (i.e., 4 for EAB distances and 1.25 for the outer boundary of the LPZ) in lieu of site-specific terrain adjustment factors;

- Building wake effects accounted for in estimating annual average X/Q values in accordance with Regulatory Position C.1.4 of RG 1.145 and Regulatory Position C.1.c of RG 1.111; and

- X/Q values at the outer boundary of the LPZ for the intermediate time periods indicated above estimated by logarithmic interpolation between the 5-percent overall site 0- 2-hour X/Q value and the maximum sector annual average X/Q value in accordance with Regulatory Position C.2.2.1 of RG 1.145.

With the exception of the meteorological data in the form of a composite JFD required by the PAVAN code, the inputs, above, were common to the licensee's EN-113 and the NRC staff's confirmatory PAVAN modeling analyses, and were acceptable to the NRC staff based on licensee's submittals supporting the LAR, including responses to the RAIs.

Tables 3.2-1 through 3.2-3, of this SE, summarize the results of the licensee's X/Q estimates using the proprietary EN-113 dispersion model, and from the NRC staff's confirmatory modeling runs based on the NRC-accepted PAVAN model. The X/Q values for the Unit 1 and Unit 2 EAB model runs (see Tables 3.2-1 and 3.2-2) taken from the licensee's RAI responses by letter dated October 6, 2016, compare well with the results from the corresponding PAVAN model runs. The highest 0.5-percent sector-dependent 0- to 2-hour X/Q values occur in the same location (i.e., the northwest sector) with Unit 1 being the higher of the two release scenarios (i.e., 2.50 E-04 sec/m^3). The 0.5-percent sector-dependent 0- to 2-hour X/Q value represents the controlling dispersion factor when compared to 5-percent overall site X/Q (i.e., 1.89 E-04 seconds per cubic meter (sec/m^3)).

Both sets of results show that the relatively higher X/Qs typically occur in the northwest and southeast directional quadrants. Sector-to-sector variation of X/Qs is similar in the results for a given unit's release scenario. The EN-113 and PAVAN 0- to 2-hour X/Qs for the EAB model runs generally compare within about 10 percent of one another; the EN-113 values tend to be conservatively higher in the peak areas.

The X/Q values used to assess potential impacts at the outer boundary of the LPZ due to accident releases from either Unit 1 or Unit 2 (see Table 3.2-3 of this SE) show sector-to-sector variation similar to the EAB model runs with the highest 0.5-percent sector-dependent 0- to 2-hour X/Q value again occurring in the northwest sector (i.e., 2.00 E-05 sec/m^3). The EN-113 and PAVAN 0- to 2-hour X/Qs from the LPZ model runs generally compare within about 5 percent of one another with the EN-113 values again tending to be conservatively higher in the peak areas. As with the EAB model runs, the 5-percent overall site X/Q value (i.e., 1.46 E-05 sec/m^3) is less than the 0.5-percent sector-dependent 0- to 2-hour X/Q value. Consequently, the 0.5-percent X/Q value represents the controlling dispersion factor at the outer boundary of the LPZ for that initial accident time period.

As indicated above, the X/Q values for the intermediate 2- to 8-hour, 8- to 24-hour, 1- to 4-day, and 4- to 30-day time periods (see Table 3.2-3 of this SE) were logarithmically interpolated between the 5-percent overall site, 0- to 2-hour X/Q and the maximum sector annual average X/Q value at the outer boundary of the LPZ. The highest annual average X/Q value occurs in the southeast sector (i.e., 2.03 E-07 sec/m^3), the same directional quadrant with secondary peaks for the 0- to 2-hour time period shown in the results for the EAB and outer boundary of the LPZ model runs. The occurrence of the highest annual average X/Q value in the southeast sector is consistent with the frequency of winds from the northwest sector (about 35 percent of the time) as indicated in the licensee's RAI responses by letter dated April 21, 2016. The intermediate X/Q values estimated by the EN-113 model are conservatively higher by about 10 percent compared to the corresponding values from the NRC staff's PAVAN model runs.

Based on these evaluations, consistent with the intent of SRP Section 2.3.4, the NRC staff concluded that the licensee's use of the proprietary EN-113 dispersion model as an alternate approach for estimating offsite dispersion factors for the LAR is acceptable. Further, the NRC staff concludes that the X/Q values estimated with the EN-113 model are acceptable for input to the related DBA dose assessments at the EAB and outer boundary of the LPZ.

3.2.4 Conclusion

As stated in Sections 3.2.2 and 3.2.3 of this SE, the NRC staff concludes that the onsite X/Q values determined by use of the ARCON96 dispersion model, and offsite X/Qs values determined based on the use of the proprietary EN-113 dispersion model, are acceptable.

Table 3.2-1. Comparison of 0- to 2-Hour X/Q Values at the Exclusion Area Boundary (EAB) Based on Accident-Related Releases from DCPP, Unit 1

			Licensee EN-113 Model Run for Unit 1 Releases [1]		NRC Staff PAVAN Model Run for Unit 1 Releases		
Receptor Type	Downwind Sector	Distance (m)	0.5% 0-2 Hr X/Q (sec/m3)	Rank [2]	0.5% 0-2 Hr X/Q (sec/m3)	Rank [2]	% Difference ((EN-113 - PAVAN) / PAVAN) X 100
EAB	S	830	7.77E-05	16	8.76E-05	15	-11.3
EAB	SSW	830	8.39E-05	15	9.27E-05	14	-9.5
EAB	SW	780	1.12E-04	9	1.07E-04	9	4.7
EAB	WSW	780	1.04E-04	11	9.97E-05	11	4.3
EAB	W	750	1.47E-04	6	1.42E-04	6	3.5
EAB	WNW	750	2.02E-04	3	1.87E-04	3	8.0
EAB	NW	750	2.50E-04	1	2.45E-04	1	2.0
EAB	NNW	750	2.17E-04	2	1.90E-04	2	14.2
EAB	N	730	1.46E-04	7	1.47E-04	5	-0.7
EAB	NNE	730	1.16E-04	8	1.18E-04	8	-1.7
EAB	NE	740	9.99E-05	12	1.00E-04	10	-0.1
EAB	ENE	740	9.25E-05	13	9.52E-05	13	-2.8
EAB	E	890	8.75E-05	14	8.45E-05	16	3.6
EAB	ESE	890	1.52E-04	5	1.37E-04	7	10.9
EAB	SE	920	1.92E-04	4	1.82E-04	4	5.5
EAB	SSE	830	1.12E-04	9	9.97E-05	11	12.3
Maximum Sector 0.5% 0-2 Hr X/Q =			2.50E-04		2.45E-04		2.0
5% Overall Site 0-2 Hr X/Q =			1.89E-04		1.58E-04		19.6

(1) Based on proprietary EN-113 dispersion model runs and proposed markup of FSARU Table 2.3-145A in licensee's letter dated October 6, 2016 (ADAMS Accession No. ML16287A776).
(2) Rank is the same for identical X/Q values.

Table 3.2-2. Comparison of 0- to 2-Hour X/Q Values at the Exclusion Area Boundary (EAB) Based on Accident-Related Releases from DCPP, Unit 2

			Licensee EN-113 Model Run for Unit 2 Releases (1)		NRC Staff PAVAN Model Run for Unit 2 Releases		
Receptor Type	Downwind Sector	Distance (m)	0.5% 0-2 Hr X/Q (sec/m3)	Rank	0.5% 0-2 Hr X/Q (sec/m3)	Rank	% Difference ((EN-113 - PAVAN) / PAVAN) X 100
EAB	S	730	9.46E-05	13	1.08E-04	11	-12.4
EAB	SSW	730	1.02E-04	11	1.12E-04	10	-8.9
EAB	SW	740	1.22E-04	8	1.16E-04	9	5.2
EAB	WSW	780	1.04E-04	10	9.97E-05	12	4.3
EAB	W	780	1.38E-04	6	1.33E-04	6	3.8
EAB	WNW	780	1.89E-04	3	1.75E-04	3	8.0
EAB	NW	830	2.17E-04	1	2.11E-04	1	2.8
EAB	NNW	830	1.88E-04	4	1.64E-04	4	14.6
EAB	N	830	1.19E-04	9	1.20E-04	8	-0.8
EAB	NNE	830	9.53E-05	12	9.70E-05	13	-1.8
EAB	NE	820	8.51E-05	15	8.60E-05	15	-1.0
EAB	ENE	820	7.88E-05	16	8.17E-05	16	-3.5
EAB	E	870	9.00E-05	14	8.66E-05	14	3.9
EAB	ESE	870	1.56E-04	5	1.41E-04	5	10.6
EAB	SE	850	2.09E-04	2	2.08E-04	2	0.5
EAB	SSE	730	1.29E-04	7	1.26E-04	7	2.4
Maximum Sector 0.5% 0-2 Hr X/Q =			2.17E-04		2.11E-04		2.8
5% Overall Site 0-2 Hr X/Q =			1.88E-04		1.56E-04		20.5

(1) Based on proprietary EN-113 dispersion model runs and proposed markup of FSARU Table 2.3-145A in licensee's letter dated October 6, 2016.

Table 3.2-3. Comparison of 0- to 2-Hour and Intermediate X/Q Values at the Outer Low Population Zone (LPZ) Boundary Based on Accident-Related Releases from DCPP Unit 1 or Unit 2

			Licensee EN-113 Model Run for Unit 1/Unit 2 Releases (1)		NRC Staff PAVAN Model Run for Unit 1/Unit 2 Releases)		
Receptor Type	Downwind Sector	Distance (m)	0.5% 0-2 Hr X/Q (sec/m3)	Rank (2)	0.5% 0-2 Hr X/Q (sec/m3)	Rank	% Difference ((EN-113 - PAVAN) / PAVAN) X 100
LPZ	S	9650	4.73E-06	14	4.83E-06	14	-2.1
LPZ	SSW	9650	4.99E-06	11	5.18E-06	11	-3.7
LPZ	SW	9650	6.20E-06	9	6.26E-06	9	-1.0
LPZ	WSW	9650	5.80E-06	10	5.79E-06	10	0.2
LPZ	W	9650	8.17E-06	6	8.17E-06	6	0.0
LPZ	WNW	9650	1.38E-05	4	1.27E-05	4	8.7
LPZ	NW	9650	2.00E-05	1	1.85E-05	1	8.1
LPZ	NNW	9650	1.49E-05	3	1.31E-05	3	13.7
LPZ	N	9650	7.19E-06	7	7.29E-06	7	-1.4
LPZ	NNE	9650	4.76E-06	13	4.84E-06	13	-1.7
LPZ	NE	9650	4.28E-06	15	4.29E-06	15	-0.2
LPZ	ENE	9650	3.89E-06	16	3.87E-06	16	0.5
LPZ	E	9650	4.99E-06	11	5.16E-06	12	-3.3
LPZ	ESE	9650	1.33E-05	5	1.22E-05	5	9.0
LPZ	SE	9650	1.89E-05	2	1.84E-05	2	2.7
LPZ	SSE	9650	6.75E-06	8	6.56E-06	8	2.9
Maximum Sector 0.5% 0-2 Hr X/Q =			2.00E-05	1	1.85E-05	1	8.1
5% Overall Site 0-2 Hr X/Q =			1.46E-05	----	1.33E-05	----	9.8
2-8 Hr X/Q =			7.20E-06	----	6.51E-06	----	10.6
8-24 Hr X/Q =			5.06E-06	----	4.56E-06	----	11.0
24-96 Hr (1-3 Day) X/Q =			2.35E-06	----	2.10E-06	----	11.9
96-720 Hr (4-30 Day) X/Q =			7.81E-07	----	6.92E-07	----	12.9
Maximum Annual Average X/Q =			2.03E-07	----	1.78E-07	----	14.0

(1) Based on proprietary EN-113 dispersion model runs and proposed markup of FSARU Table 2.3-145A in the licensee's letter dated October 6, 2016
(2) Rank is the same for identical X/Q values.
(3) Intermediate time period X/Qs estimated by logarithmic interpolation between 0-2 hour and annual average X/Qs per RG 1.145.

3.3 Radiological Consequences Analyses

Radiological consequences of LOCA, FHA in containment, FHA in FHB, LRA, CREA, MSLB, SGTR, and LOL, are discussed in the following sections.

3.3.1 Radiological Consequences of DBAs

The licensee has proposed a licensing basis change for its offsite and CR DBA dose consequence analysis for DCPP. The proposed change will implement an AST methodology for determining DBAs offsite and CR doses. For full implementation of the AST DBAs analysis methodology, the dose acceptance criteria specified in 10 CFR 50.67 provides an alternative to the previous whole body and thyroid dose guidelines stated in 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance"; and 10 CFR Part 50, Appendix A, GDC 19.

As discussed in RG 1.183, Regulatory Position C.1.2.1, "Full Implementation," full implementation is a modification of the facility design basis that addresses all characteristics of the AST, that is, composition and magnitude of the radioactive material, its chemical and physical form, and the timing of its release. Full implementation revises the plant licensing basis to specify the AST in place of the previous accident source term and establishes the TEDE dose as the new acceptance criteria. This applies not only to the analyses performed in this application, but also to all future design-basis dose consequence analyses at DCPP. At a minimum for full implementation of the AST, the DBA LOCA must be reanalyzed. Since, upon issuance of this LAR, the AST and TEDE criteria will become part of the design basis for DCPP, new applications of the AST would not require prior NRC approval unless stipulated by 10 CFR 50.59, "Changes, tests, and experiments," or unless the new application involved a change to a TS. However, a change from an approved AST to a different AST that is not approved for use at DCPP would require a license amendment under 10 CFR 50.67.

As stated in RG 1.183, Regulatory Position C.5.2, "Accident-Specific Assumptions," the DBAs addressed in the appendices of RG 1.183 were selected from accidents that may involve damage to irradiated fuel. RG 1.183 does not address DBAs with radiological consequences based on technical specification reactor or secondary coolant specific activities only. The inclusion or exclusion of a particular DBA in RG 1.183 should not be interpreted as indicating that an analysis of that DBA is required or not required. Licensees should analyze the DBAs that are affected by the plant-specific proposed applications of an AST.

The licensee performed analyses for the full implementation of the AST, in accordance with the guidance in RG 1.183, and Section 15.0.1 of the SRP. Also, the licensee's AST analyses were based on the pressurized-water reactor (PWR) DBAs identified in RG 1.183 that could potentially result in significant CR and offsite doses.

The licensee has performed its evaluation based on full implementation of the AST as defined in RG 1.183, with the exception of the equipment qualification (EQ). The licensee has determined that the current TID-14844 for accident source term will remain the licensing basis for EQ.

Regulatory Position C.6, "Assumptions for Evaluating the Radiation Doses for Equipment Qualification," of RG 1.183 states, in part, that "[t]he NRC staff is assessing the effect of increased cesium releases on EQ doses to determine whether licensee action is warranted. Until such time as this generic issue is resolved, licensees may use either the AST or the

TID-14844 assumptions for performing the required EQ analyses." This issue has been resolved as documented in a memorandum dated April 30, 2001, "Initial Screening of Candidate Generic Issue 187, "The Potential Impact of Postulated Cesium Concentration on Equipment Qualification in the Containment Sump" (ADAMS Accession No. ML011210348), and in NUREG-0933, "A Prioritization of Generic Safety Issues," Supplement 25, June 2001, (ADAMS Accession No. ML012190402).

In the memorandum dated April 30, 2001, the conclusion of Generic Issue 187 states, in part, the following:

> The panel concludes that there is no clear basis for back-fitting the requirement to modify the design basis for equipment qualification to adopt the AST. There would be no discernible risk reduction associated with such a requirement. Licensees should be aware, however, that a more realistic source term would potentially involve a larger dose for equipment exposed to sump water for long periods of time. Longer term equipment operability issues associated with severe fuel damage accidents, (with which the AST is associated) could also be addressed under accident management or plant recovery actions as necessary.

Therefore, in consideration of the above-cited references, the NRC staff finds that it is acceptable for the TID-14844 accident source term to remain the licensing basis for EQ at DCPP.

RG 1.183, Regulatory Position C.4.3, "Other Dose Consequences," states, in part, that:

> The guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737 [Reference 2 of RG 1.183]. Design envelope source terms provided in NUREG-0737 should be updated for consistency with the AST. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be expressed in terms of TEDE.

The licensee performed a review of its shielding study developed in response to NUREG-0737, Item II.B.2 and submitted to the NRC in July 12, 1984. This report documented the estimated radiation exposure to plant personnel performing vital missions in support of accident mitigation and safe shutdown following a LOCA. The source terms were based on traditional TID-14844 assumptions. The nuclide release fractions specified in the AST methodology outlined in RG 1.183, differ from those outlined in TID-14844/NUREG-0737. The difference in the release fractions has the potential to affect the dose rates in vital areas where piping containing post-LOCA sump fluid are located.

NUREG-0933, Generic Issue 187 showed that exposure to containment atmosphere sources developed, based on traditional source term methodology and AST methodology, produced similar integrated doses and that the integrated AST doses from exposure to post-LOCA sump fluid did not exceed those based on TID-14844 assumptions until 42 days after an event at a PWR.

Based on NUREG-0933, the licensee concluded, and the NRC staff agrees, that the differences in the release fractions associated with AST methodology would have little impact on the local dose rates during the 30-day post-LOCA mission time. Since the local dose rates are not expected to be significantly impacted by AST during the first 30 days following a LOCA, the conclusions of the shielding study, with respect to operator exposure, would not significantly change by expressing the mission doses in terms of TEDE.

A full implementation of the AST is proposed for DCPP. Therefore, to support the licensing and plant operation changes discussed in the LAR, the licensee analyzed the following accidents employing the AST as described in RG 1.183.

- LOCA
- FHA in the Containment
- FHA in the FHB
- LRA
- CREA
- MSLB
- SGTR
- LOL

The DBA dose consequence analyses evaluated the integrated TEDE dose at the EAB for the worst 2-hour period following the onset of the accident. The integrated TEDE doses at the outer boundary of the LPZ during the entire period of the passage of the radioactive cloud resulting from postulated release of fission products, and the integrated dose to a DCPP CR operator, were evaluated for the duration of the accident. The dose consequence analyses were performed by the licensee using the "RADTRAD: Simplified Model for RADionuclide Transport and Removal and Dose Estimation," Version 3.03, computer code. The development of the RADTRAD radiological consequence computer code was sponsored by the NRC, as described in NUREG/CR-6604, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal and Dose Estimation," April 1998 (ADAMS Accession No. ML15092A284) and was developed by Sandia National Laboratories for the NRC. The code estimates transport and removal of radionuclides and radiological consequence doses at selected receptors. The NRC staff performs independent confirmatory dose evaluations using the RADTRAD computer code. The results of the evaluations performed by the licensee, as well as the applicable dose acceptance guidelines from RG 1.183, are shown in Table 3.3-1 of this SE.

Each DBA radiological source term used in the AST analyses was developed based on a core power level of 3580 megawatts thermal (MWt). The core power level represents the licensed power of 3411 MWt with a 5 percent increase to account for measurement uncertainties. The use of 3580 MWt for the AST DBA radiological source term analyses bounds the current licensed core thermal power level of 3411 MWt and is, therefore, acceptable to the NRC staff for use in the full implementation of the AST at DCPP.

RG 1.183, Regulatory Position C.3.1, "Fission Product Inventory," states, in part:

> The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the ECCS [Emergency Core Cooling System] evaluation uncertainty [the uncertainty factor used in determining the core inventory should be that value provided in Appendix K to 10 CFR 50, typically 1.02]. The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. The core

> inventory should be determined using an appropriate isotope generation and depletion computer code such as [ORIGEN 2 or ORIGEN-ARP].

In accordance with RG 1.183, the licensee developed the equilibrium core activity inventory and the decayed fuel inventories after shutdown (for FHA) using the SAS2 and ORIGEN-S modules of the NRC sponsored SCALE code package. The determination of core inventory is dependent on the level of fuel enrichment. Since the DCPP core contains fuel assemblies with different levels of enrichment, core inventory calculations were performed for enrichments of 4.2 and 5.0 percent. The highest activity for each isotope from the two enrichments was chosen to represent the inventory of that isotope in the equilibrium core. A 19-month average fuel cycle, which has 3 fuel cycles every 5 years and refueling outages in the spring or fall, is used. The core inventory is based on a maximum core average burnup of 50 gigawatt day per metric ton of uranium (GWD/MTU) and the peak rod burnup limit at the end of cycle is not allowed to exceed 62,000 megawatt day per metric ton of uranium (MWD/MTU). In addition, a 4 percent margin has been included in the final isotopic inventory. The inventories, consisting of the curie levels for 272 isotopes at end of fuel cycle, formed the source term input for the RADTRAD dose evaluations. The NRC staff finds this approach to be consistent with current regulatory guidance and, therefore, acceptable.

As stated in Draft Regulatory Guide (DG)-1199 (proposed Revision 1 of RG 1.183) "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," October 2009 (ADAMS Accession No. ML090960464), the release fractions associated with the LWR core inventory released into the containment for the DBA LOCA and non-LOCA events have been determined to be acceptable for use with currently approved LWR fuel with a peak rod average burnup up to 62,000 MWD/MTU provided that the licensee operates within the maximum allowable power operating envelope for non-LOCA gap fractions shown in Figure 1 of DG-1199. DCPP stated that it falls within, and intends to operate within, the maximum allowable power operating envelope for non-LOCA gap fractions, as shown in Figure 1 of DG-1199. Based on its review, the NRC staff finds this approach to be acceptable.

The licensee used committed effective dose equivalent and effective dose equivalent dose conversion factors (DCFs) from FGRs 11 and 12 to determine the TEDE dose in accordance with AST evaluations. The use of ORIGEN and DCFs from FGR-11 and FGR-12 is in accordance with RG 1.183 guidance and is, therefore, acceptable to the NRC staff.

3.3.2 Loss-of-Coolant Accident (LOCA)

A DBA LOCA is a failure of the reactor coolant system (RCS) that results in the loss of reactor coolant, which, if not mitigated, could result in fuel damage including core melt. Analyses are performed using a spectrum of RCS break sizes to evaluate fuel and ECCS performance. A large-break LOCA is postulated as the failure of the largest pipe in the RCS. RG 1.183 establishes the large-break LOCA as the licensing basis LOCA with regards to radiological consequences since this represents the larger challenge to plant safety features designed to mitigate the release of radionuclides to the environment in the unlikely event that ECCS is not effective. Evaluation of the effectiveness of plant safety features, such as ECCS, has shown that core melt is unlikely. The objective of this DBA is to evaluate the ability of the plant design to mitigate the release of radionuclides to the environment in the unlikely event that ECCS is not effective in preventing core damage.

The fission product release is assumed to occur in phases over a 2-hour period. When using the AST for the evaluation of a design-basis LOCA for a PWR, it is assumed that the initial fission product release to the containment will last for 30 seconds and will consist of the radioactive materials dissolved or suspended in the RCS liquid. After 30 seconds, fuel damage is assumed to begin and is characterized by clad damage that releases the fission product inventory assumed to reside in the fuel gap. The fuel gap release phase is assumed to continue until 30 minutes after the initial breach of the RCS. As core damage continues, the gap release phase ends and the early in-vessel release phase begins. The early in-vessel release phase continues for the next 1.3 hours. The licensee used the LOCA source term release fractions, timing characteristics, and radionuclide grouping, as specified in RG 1.183, for evaluation of the AST.

In the evaluation of the LOCA design-basis radiological analysis, the licensee considered dose contributions from the following potential activity release pathways:

- Containment leakage directly to the atmosphere,
- Release from the containment pressure/vacuum relief to the atmosphere,
- Engineered safety feature (ESF) systems leakage,
- Residual heat removal (RHR) pump seal failure,
- Miscellaneous equipment drain tank (MEDT) leakage to the atmosphere, and
- RWST leakage to the atmosphere.

3.3.2.1 LOCA Source Term

The licensee followed all aspects of the guidance outlined in RG 1.183, Regulatory Position C.3, "Accident Source Term," regarding the fission product inventory, release fractions, timing of the release phases, radionuclide composition, and chemical form for the evaluation of the LOCA. For the DBA LOCA, the licensee uses the core average inventory, as discussed above, and assumes that all the fuel assemblies in the core are affected. The LOCA analysis assumes that iodine will be removed from the containment atmosphere by both containment sprays and natural diffusion to the containment walls. As a result of these removal mechanisms a large fraction of the released activity will be deposited in the containment sump. The sump water will retain soluble gases and soluble fission products, such as iodine and cesium, but not noble gases. The guidance from RG 1.183 specifies that the iodine deposited in the sump water can be assumed to remain in solution as long as the containment sump pH is maintained at or above 7.

The licensee conducted an evaluation of containment sump pH in order to ensure that particulate iodine deposited into the containment sump water does not re-evolve beyond the amount recognized in the DBA LOCA analysis. The licensee's determination of pH was performed using the methodology outlined in NUREG/CR-5732, "Iodine Chemical Forms in LWR Severe Accidents," April 1992 (ADAMS Accession No. ML003726825), and NUREG/CR-5950, "Iodine Evolution and pH Control," December 1992 (ADAMS Accession No. ML063460464). The licensee's evaluation concluded that the sump pH will remain greater than 7.0 for the 30-day duration of the accident and that the post-LOCA dose consequence analyses need not consider iodine re-evolution from the sump fluid in accordance with RG 1.183. The licensee's evaluation is based on the NRC guidance and based on the review, the NRC staff agreed with licensee's conclusion.

3.3.2.2 Assumptions on Transport in the Primary Containment

3.3.2.2.1 Containment Mixing, Natural Deposition and Leak Rate

In accordance with RG 1.183, the licensee assumed that the activity released from the fuel is mixed instantaneously and homogeneously throughout the free air volume of the containment. The licensee used the core release fractions and timing as specified in RG 1.183 with the termination of the release into containment set at the end of the early in-vessel phase.

The licensee credited the reduction of airborne radioactivity in the containment by natural deposition. Specifically, the licensee credited an elemental iodine natural deposition removal coefficient of 0.57 per hour until 6.25 hours post-LOCA to the sprayed volume of the containment. The licensee did not credit the removal of organic iodine by natural deposition.

RG 1.183, Appendix A, Regulatory Position 3.7 states that the primary containment should be assumed to leak at the peak pressure TS leak rate for the first 24 hours and that for pressurized water reactors, the leak rate may be reduced after the first 24 hours to 50 percent of the TS leak rate. Accordingly, the licensee assumed a containment leak rate of 0.1 percent per day for the first 24 hours, after which the containment leak rate is reduced to 0.05 percent per day for the duration of the accident consistent with DCPP's TS. The licensee assumes the leakage is from both the sprayed and unsprayed regions of the containment to the environment.

The licensee's analysis for containment mixing and natural deposition and leak rate follows the applicable regulatory guidance, and is therefore acceptable.

3.3.2.2.2 Containment Spray Assumptions

RG 1.183, Appendix A, Regulatory Position 3. states, in part, that:

> The mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment building, provided that adequate flow exists between these regions, is assumed to be two turnovers of the unsprayed regions per hour, unless other rates are justified. The containment building atmosphere may be considered a single, well-mixed volume if the spray covers at least 90% of the volume and if adequate mixing of unsprayed compartments can be shown.

For DCPP, the volume of the sprayed region is 2.103E+06 cubic feet (ft^3) and the volume of the unsprayed region is 4.470E+05 ft^3. The internal design of DCPP containment structures allows air to circulate freely and the volume above the operating floor, which comprises the majority of the containment net free volume, does not have significant barriers to obstruct mixing. In addition, the cubicles and compartments within the containment below the operating floor are provided with openings near the top as well as bottom to allow air circulation. In accordance with RG 1.183 Appendix A, Regulatory Position 3.3, prior to containment fan cooler unit's (CFCU's) operation, the licensee used the mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment building of two turnovers of the unsprayed regions per hour. Since the sprayed region represents approximately 82.5 percent of the total containment volume the licensee used a two volume model to represent the sprayed and unsprayed regions of the containment.

The licensee credited two out of the five CFCUs for mixing between the sprayed and unsprayed regions of the containment in the AST LOCA analysis. In addition, credit was given for the effects of natural convection in the determination of the mixing rate between the sprayed and unsprayed volumes for the first 86 seconds after LOCA initiation. At 86 seconds after LOCA initiation, the CFCUs are initiated and they operate for the duration of the accident. The licensee considered the forced convection induced by the CFCUs to model the transfer rate between the sprayed and unsprayed containment regions. The containment mixing rate between the sprayed and unsprayed regions, following a LOCA, is determined to be 9.13 turnovers of the unsprayed regions per hour while the CFCUs are in operation.

For DCPP, the containment spray (CS) in the injection mode is initiated at 111 seconds after the LOCA initiation and terminates at 63.3 minutes. Manual operation is credited to reinitiate CS in the recirculation mode 12 minutes after injection spray is terminated and the CS continues until termination at 6.25 hours.

Using the guidance from SRP Section 6.5.2, "Containment Spray as a Fission Product Cleanup System," and NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays," June 1993 (not publicly available), the licensee determined the aerosol removal rate from the effects of the CS system particle growth due to agglomeration, gravitational settling of particles, and diffusiophoresis. The aerosol removal rate is based on the calculated time-dependent airborne aerosol mass. The aerosol removal rates are listed in Table 3.3-2 of this SE. Regulatory Position 3.3 of Appendix A to RG 1.183 states, in part:

> The reduction in the removal rate is not required if the removal rate is based on the calculated time-dependent airborne aerosol mass. There is no specified maximum DF [decontamination factor] for aerosol removal by sprays.

Therefore, in the AST analysis, the licensee did not limit the DF for aerosol removal. In the licensee's analysis, sprays are assumed to run for 6.25 hours post-LOCA with a 12-minute gap after injection spray is terminated.

Using the guidance from SRP Section 6.5.2, the licensee determined the elemental iodine removal rate from the effects of the CS system in injection mode is in excess of 20 per hour. However, in accordance with the guidance in SRP Section 6.5.2, the licensee limited the removal rate constant for elemental iodine to 20 per hour. During recirculation spray operation, the licensee determined the elemental iodine removal rate is 19.34 per hour. The elemental iodine removal rates are listed in Table 3.3-2 of this SE. In the sprayed and unsprayed regions, prior to spray actuation, the licensee determined the wall deposition removal coefficient is 2.74 per hour, and in the sprayed region while sprays are in operation the wall deposition removal coefficient is 0.57 per hour. During the period of spray operation, the elemental iodine removal rate constant from sprays was added to the elemental iodine removal rate constant from wall deposition to yield an effective elemental iodine removal rate constant. The licensee applied this effective removal rate in the radiological dose analysis from the time of spray actuation until 6.25 hours post-LOCA, with the exception of the 12-minute time between injection and recirculation mode. No credit is taken for elemental iodine removal in the unsprayed region.

The NRC staff has reviewed the licensee's application of credit for iodine removal from the operation of the CS system and found that the analysis follows the applicable regulatory guidance, is conservative, and is therefore, acceptable.

3.3.2.3 Assumptions on ESF System Leakage

To evaluate the radiological consequences of ESF leakage, the licensee used the deterministic approach, as described in RG 1.183. This approach assumes, with the exception of noble gases, all the fission products released from the fuel to the containment instantaneously and homogeneously, mix in the containment sump water at the time of release from the core. Except for iodine, all of the radioactive materials in the containment sump are assumed to be in aerosol form and retained in the liquid phase. As a result, the licensee assumed that the fission product inventory available for release from ESF leakage consists of 40 percent of the core inventory of iodine. This amount is the combination of 5 percent released to the containment sump water during the gap release phase and 35 percent released to the containment sump water during the early in-vessel release phase. This source term assumption is conservative in that 100 percent of the radioiodine released from the fuel is assumed to reside in both the containment atmosphere and in the containment sump concurrently. ECCS leakage develops when ESF systems circulate containment sump water outside containment and leaks develop through packing glands, pump shaft seals and flanged connections.

RG 1.183, Appendix A, Regulatory Position 5.5, states, in part, that, "[i]f the temperature of the leakage is less than 212 °F [degrees Fahrenheit] or the calculated flash fraction is less than 10%, the amount of iodine that becomes airborne should be assumed to be 10% of the total iodine activity in the leaked fluid...." The licensee has determined that the maximum temperature of the recirculation fluid is 259.9 °F and that the flash fraction is less than 10 percent, therefore per RG 1.183 the licensee assumes that 10 percent of the iodine activity associated with this leakage is airborne. In addition, in accordance with RG 1.183, for ESF leakage, the licensee assumes that the chemical form of the released iodine is 97 percent elemental and 3 percent organic.

For the LOCA analysis of ESF leakage, the licensee used two different pathways. The first pathway is via the closest structural opening in the Containment Penetration area GE and areas GW and FW and consists of the sum of the maximum allowable simultaneous leakage from all components in the ESF recirculation systems located in the containment penetration areas, which is 12 cubic centimeters per minute (cc/min), representing two times the maximum permitted recirculation loop leakage of 6 cc/min, as specified in RG 1.183, Appendix A, Regulatory Position 5.2. The second pathway is via the plant vent and consists of the sum of the maximum allowable simultaneous leakage from all components in the ESF recirculation systems located in the auxiliary building, which is 240 cc/min, representing two times the maximum permitted recirculation loop leakage of 120 cc/min, as specified in RG 1.183, Appendix A, Regulatory Position 5.2. As stated above, actual ECCS leakage starts when the recirculation phase of the accident begins.

For conservatism, with the exception of the RHR pump rooms, the licensee assumes the plant vent release bypasses the ventilation filters and the containment penetration area release is unfiltered. The licensee's analysis for ESF system leakage follows the applicable regulatory guidance, RG 1.183, and is therefore acceptable.

3.3.2.3.1 Assumptions on ESF System Back-Leakage to the RWST

Although the RWST is isolated during recirculation, design leakage through ECCS valves provides a pathway for back leakage of the containment sump water to the RWST. The RWST is vented to the atmosphere. Since this release path represents a bypass of the containment, the radiological dose consequences are considered. The concentration of radionuclides in the containment sump water is as modeled above for ESF leakage. DCPP assumes that containment sump water leaks into the RWST at a rate of 2 gallons per minute (gpm) representing two times the maximum permitted back leakage of 1 gpm, as specified in RG 1.183, Appendix A, Section 5.2. The back leakage to the RWST starts at 829 seconds post-LOCA. At 829 seconds, the iodine is projected to be released via the RWST vent and continue for 30 days. The licensee assumes that a portion of the iodine dissolved in the back leakage will be retained within the RWST. The time-dependent iodine release fractions used by the licensee are illustrated in Table 3.3-3 of this SE. Values range from about 9.451E-5 at 829 seconds post-accident, to a minimum value of about 1.483E-6 at 30 days. The time dependent iodine partition coefficient reflects the temperature and pH of the RWST liquid and containment sump fluid, the RWST liquid and gas volumes, and the temperature, pH and volume of the incoming leakage. The equilibrium iodine concentration in the RWST gas space is based on the iodine mass in the sump fluid entering the RWST vapor space as back leakage or the total iodine mass contained in the RWST liquid, whichever results in higher RWST vapor phase concentration. The licensee assumed that this activity is exhausted without filtration or tank holdup, and that the chemical form of the iodine released is 97 percent elemental and 3 percent organic.

The licensee used conservative assumptions to evaluate the RWST back leakage contribution to the LOCA dose, and, therefore, the NRC staff finds this evaluation acceptable for the AST LOCA analysis.

3.3.2.3.2 Assumption on ESF System leakage to the MEDT

The MEDT is located in the auxiliary building and is vented to the auxiliary building ventilation ductwork, and therefore, to the atmosphere via the plant vent. Following a LOCA, the MEDT will receive both ESF system leakage from the accident unit, as well as nonradioactive fluids from equipment drains and RWST leakage from the non-accident unit. Since this release path represents a bypass of the containment, the radiological dose consequences are considered. The concentration of radionuclides in the containment sump water is as modeled above for ESF leakage. DCPP assumes that ESF system leakage into the MEDT is at a rate of 1900 cc/min and nonradioactive fluids leak into the MEDT at 968 cc/min representing two times the maximum permitted back leakage of 950 cc/min and 484 cc/min, respectively, as specified in RG 1.183, Appendix A, Regulatory Position 5.2. The back leakage to the MEDT starts at 829 seconds post-LOCA. At 829 seconds, the iodine is projected to be released via the MEDT vent and continue for 30 days.

For conservatism, the licensee assumes that: (1) the boron concentration of the preexisting fluid in the MEDT, as well as the incoming leakage, is at its upper bound levels, (2) the LOCA occurs when the MEDT water level is at the normal maximum setpoint to initiate auto transfer, (3) the auto transfer capability is not initiated because it is not a safety-related function, and (4) the MEDT contents will spill over into the equipment drain receiver tank (EDRT) room after the tank is full. The EDRT room drains into the auxiliary building sump, which overflows into the Unit 1/Unit 2 pipe tunnels. The licensee analyzed this leakage in two parts: prior to MEDT overflow and post-MEDT overflow.

Prior to MEDT overflow, the iodine in the MEDT gas space is released to the atmosphere via the plant vent, at a rate based on the temperature transient in the MEDT, the increase in the liquid inventory in the MEDT, and the gases evolving out of the incoming leakage.

After MEDT overflow, the iodine distribution is conservatively assumed to be between the iodine concentrations in the MEDT overflow liquid and the EDRT room or Unit 1/Unit 2 pipe tunnels ventilation flow, which maximizes the iodine release rate. The iodine released is a sum of the following:

- The iodine in the EDRT room air space is released to the atmosphere via the plant vent at the vent rate established by the EDRT room ventilation system.
- The iodine in the Unit 1/Unit 2 pipe tunnel air space is released to the atmosphere via the plant vent at the vent rate established by the Unit1/Unit 2 pipe tunnel ventilation system.

The average time-dependent iodine release fractions used by the licensee are illustrated in Table 3.3-4 of this SE. Values range from about 4.521E-7 at 829 seconds post-accident, to a maximum value of about 2.166E-2 at 30 days. The time dependent iodine partition coefficient reflects the temperature of the MEDT. The equilibrium iodine concentration in the MEDT is based on the iodine mass in the fluids entering the MEDT. The licensee assumes that this activity is exhausted without filtration or tank/room holdup, and that the chemical form of the iodine released is 97 percent elemental and 3 percent organic.

The licensee used conservative assumptions to evaluate the MEDT back leakage contribution to the LOCA dose, and, therefore, the NRC staff finds this evaluation acceptable for the AST LOCA analysis.

3.3.2.3.3 RHR Pump Seal Failure

DCPP's CLB defines a passive failure as, the structural failure of a static component that limits the component's effectiveness in carrying out its design function, and when applied to a fluid system, this means a break in the pressure boundary resulting in abnormal leakage not exceeding 50 gpm for 30 minutes. For the LOCA analysis, the passive failure is assumed to be an RHR pump seal failure occurring at 24 hours following a LOCA for a duration of 30 minutes.

To evaluate the radiological consequences of RHR pump seal failure, the licensee used the deterministic approach for ESF leakage as described in RG 1.183. This approach assumes with the exception of noble gases, all the fission products released from the fuel to the containment instantaneously and homogeneously mix in the containment sump water at the time of release from the core. Except for iodine, all of the radioactive materials in the containment sump are assumed to be in aerosol form and retained in the liquid phase. As a result, the licensee assumed that the fission product inventory available for release from ECCS leakage consists of 40 percent of the core inventory of iodine. This amount is the combination of 5 percent released to the containment sump water during the gap release phase and 35 percent released to the containment sump water during the early in-vessel release phase. This source term assumption is conservative in that 100 percent of the radioiodine released from the fuel is assumed to reside in both the containment atmosphere and in the containment sump concurrently.

RG 1.183, Appendix A, Regulatory Position 5.5, states, in part, that "[i]f the temperature of the leakage is less than 212 °F or the calculated flash fraction is less than 10%, the amount of iodine that becomes airborne should be assumed to be 10% of the total iodine activity in the leaked fluid...." The licensee has determined that the maximum temperature of the recirculation fluid is 259.9 °F and that the flash fraction is less than 10 percent, therefore per RG 1.183 the licensee assumes that 10 percent of the iodine activity associated with this leakage is airborne. In addition, in accordance with RG 1.183, for ESF leakage, the licensee assumes that the chemical form of the released iodine is 97 percent elemental and 3 percent organic.

The licensee credits the auxiliary building ventilation system for filtration of the release. The release is routed through the auxiliary building ventilation system charcoal filter prior to being released to the environment via the plant vent. The auxiliary building ventilation system charcoal filter efficiency for elemental and organic iodine is 88 percent.

The RHR pump failure occurs at 24 hours following a LOCA and continues for a duration of 30 minutes, during which time the RHR pump is in the recirculation mode with its suction source aligned to the containment sump. The licensee used the deterministic approach for ESF leakage as described in RG 1.183, which is conservative to evaluate the RHR pump seal failure. Therefore, the NRC staff finds this evaluation to be acceptable for the AST LOCA analysis.

3.3.2.4 Assumptions on Containment Purging

The licensee evaluated the radiological consequences of containment leakage via the 12 inch containment vacuum/over pressure relief valves, which are assumed to be open to the extent allowed by the DCPP TS at the initiation of the LOCA, and are terminated as part of the containment isolation. The assumed volumetric flow rate from the 12-inch containment vacuum/over pressure relief valves is 218 cubic feet per second (cfs) and is released directly to the environment until terminated by the containment isolation at 13 seconds post-LOCA.

During this time period of 13 seconds following accident onset, the licensee assumes that fuel failure has not occurred. This assumption follows the guidance in Table 4 of RG 1.183, which indicates that the initial release of the RCS into containment for a PWR would occur within the first 30 seconds of the accident prior to the onset of fuel damage. Consistent with RG 1.183, RCS radionuclide concentrations for the AST analysis is based on the TS RCS equilibrium activity. The licensee calculated a flashing fraction of 40 percent based on the thermodynamic conditions in the reactor coolant at full power. Therefore, this conservative approach for the evaluation of the radiological dose consequence is acceptable to the NRC staff.

The licensee used conservative assumptions to evaluate the containment purge contribution to the LOCA dose, and, therefore, the NRC staff finds this evaluation to be acceptable for the AST LOCA analysis.

3.3.2.5 CR Habitability

The DCPP CR is common to Units 1 and 2. The CR is located at elevation 140 feet of the auxiliary building as shown in the FSARU, Figure 1.2-2. The CR ventilation system (CRVS) is designed to maintain the CR envelope (CRE) at a positive pressure relative to the surrounding area, following postulated accidents with the exception of toxic gas/smoke releases called CRVS Mode 4 operation. CRVS Mode 4 operation is activated on a safety injection (SI) signal and/or high radiation in the normal outside air intakes. CRVS Mode 4 is designed to introduce pressurization flow, which is approximately equivalent to the expected exfiltration air during plant emergency conditions. The design pressurization flow ranges between 650 to 900 cubic feet per minute (cfm) and is drawn from one of the two intakes on the north or south sides of the turbine building. The pressurization flow includes 100 cfm that enters the CR unfiltered due to backdraft damper leakage. Unfiltered in-leakage into the CR during Mode 1 and Mode 4 is assumed to be 70 cfm, which includes 10 cfm for CR ingress and egress.

Additionally, during postulated accident conditions, on detection of high radiation in the normal outside air intakes, or SI signal, the normal outside air supply for the CR is automatically routed through the less contaminated pressurization air intake of Unit 1 or Unit 2. The CR pressurization flow is routed through charcoal and high efficiency particulate air filters. Furthermore, during postulated accident conditions, part of the CR pressurization flow is recirculated and a portion of the recirculated flow is filtered through the same filtration unit as the pressurization flow at a flow rate of 1250 cfm.

3.3.2.5.1 CR Ventilation Assumptions

The licensee's assumption of 70 cfm unfiltered in-leakage is validated by in-leakage testing conducted during December 2012. The testing was conducted in accordance with DCPP procedure STP M-57, "Control Room Ventilation System Tracer Gas Test," using the tracer gas method described in American Society for Testing and Materials (ASTM) E741-00, "Standard Test Method for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution." The flow values listed below, stated in licensee's letter dated February 1, 2016, encompass the test results.

Mode 4 Control Room Parameters	Minimum Flow (cfm)	Maximum Flow (cfm)
Pressurization Flow	650	900
Backdraft Damper Leakage	100	100
Filtered Intake	550	800
Charcoal Filter Flow	1800	2200
Filtered Recirculation Flow	1250	1400
Unfiltered in Leakage	70	70
Control Room Exhaust Flow	720	970

The amount of unfiltered in-leakage into the pressurized CRE was determined using the constant injection method for each of the fans supplying pressurization air. Based on the test results of each pressurization fan, two additional tests were performed. Testing was then performed with the Unit 2 fan and the TSC ventilation system in operation, and the final test had the Unit 2 fan in operation, Unit 1 in Mode 3 and Unit 2 in Mode 4. The selected alignments were based on previous tests performed at DCPP. A total of four tests were performed at DCPP to determine the total in-leakage into the CRE under various system operational modes. Summaries of these tests, as described in licensee's letter dated February 1, 2016, are:

- 32 ± 5 standard cubic feet per minute (scfm) with Unit 2 in Mode 4 and Unit 1 idle,
- 25 ± 10 scfm with Unit 1 in Mode 4 and Unit 2 idle,
- 23 ± 7 scfm with Unit 2 in Mode 4, Unit 1 idle and TSC in operation, and
- 7 ± 9 scfm with Unit 2 in Mode 4 and Unit 1 in Mode 3.

Based on the review, the NRC staff has determined that an unfiltered in-leakage assumption of 70 cfm, conservatively, bounds the test results.

The CRVS automatically transfers to the pressurization mode of operation after the initiation of the SI on a containment pressure or pressurizer pressure signal. The licensee determined that the time to generate the SI signal will be 6 seconds following LOCA initiation. The licensee assumes the CRVS normal intake dampers of the accident unit start to close after a 28.2 second delay due to diesel generator loading onto the emergency buses, and 10 seconds for damper closure giving an overall delay time of 44.2 seconds for CRVS pressurization.

The CRVS is designed to maintain the CRE at a positive pressure relative to the surrounding area, following a postulated LOCA. Upon CRVS Mode 4 initiation the normal outside air supply to the CR is automatically routed through the less contaminated pressurization air intake of Unit 1 or Unit 2. Pressurization flow, which ranges between 650 to 900 cfm, is drawn from one of the two intakes on the north or south sides of the turbine building. The CR pressurization flow is routed through charcoal and high efficiency particulate air (HEPA) filters. The CR charcoal filter efficiency for elemental and organic iodine is 93 percent and the HEPA filter efficiency for particulates is 98 percent. Additionally, during postulated accident conditions, the air in the CR is recirculated and a portion of the recirculated flow is filtered through the same filtration unit as the pressurization flow at a flow rate of 1250 cfm. The control room ventilation assumptions are conservative, reflective of the CRVS design, and are acceptable.

3.3.2.5.2 Direct Shine Dose Evaluations

The total CR LOCA radiological dose includes direct shine contributions from the following DBA LOCA radiation sources:

- Direct shine from radioactive material in the containment including shine through one of the main steam line penetrations and the personnel hatch facing the CR.
- Direct shine from the external radioactive plume outside the CR pressure boundary resulting from containment leakage, ESF system leakage, RHR pump seal leakage, RWST back leakage, and MEDT leakage.
- Radiation shine from scattered gamma radiation through wall penetrations from the CRVS filters.
- Direct shine from the containment sump fluid that is postulated to collect in the RWST.

RG 1.196 defines the CRE as follows:

> The plant area, defined in the facility's licensing basis that in the event of an emergency, can be isolated from the plant areas and the environment external to the CRE. This area is served by an emergency ventilation system, with the intent of maintaining the habitability of the control room. This area encompasses the control room, and may encompass other non-critical areas to which frequent personnel access or continuous occupancy is not necessary in the event of an accident.

The licensee evaluated the DBA LOCA radiation dose to personnel in the CR from the following sources: gamma shine from the primary containment airborne activity, gamma shine from the RWST, activity in the radioactive cloud surrounding the plant structures, and from trapped activity on CRVS filters. The result of the licensee's evaluation of the CR gamma shine dose are included in Table 3.3-1 of this SE.

The licensee used conservative assumptions to evaluate the direct shine dose to the CR, and therefore, the NRC staff finds this evaluation to be acceptable for the AST LOCA analysis.

3.3.2.5.3 Control Room Operator Dose during Ingress and Egress

Section 50.67(b)(2) of 10 CFR requires that the licensee's analysis demonstrates with reasonable assurance that adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident.

The licensee evaluated the dose received by the CR operators during routine access to the CR for the 30-day period following the LOCA. The total CR operator dose from ingress and egress includes direct shine contributions from the following DBA LOCA radiation sources:

- Direct shine from radioactive material in the containment.
- Direct shine from the external radioactive plume from containment leakage.

The post-LOCA egress/ingress calculation is based on 27 outbound transits from the CR to the site boundary, and 26 inbound transits from the site boundary to the CR. The licensee estimated that each transition would take 5 minutes and no credit was taken for breathing apparatus or special whole body shielding. The licensee assumes that transit to and from the CR is only expected after the first 24 hours following the accident by which time the airborne levels inside containment are reduced due to the use of containment sprays and radioactive decay. In addition, the licensee's evaluation accounts for shielding from the buildings that house the ESF leakage.

The results of the licensee's evaluations of the CR operator access dose are included in Table 3.3-1 of this SE. The licensee used conservative assumptions to evaluate the CR operator dose during ingress and egress, and therefore, the NRC staff finds this evaluation to be acceptable for the AST LOCA analysis.

3.3.2.6 TSC Habitability

The DCPP onsite TSC is common to Units 1 and 2. The TSC is located at elevation 104 feet on the south-west side of the Unit 2 turbine building. The TSC is large enough to house 25 people along with the necessary data and information displays. During normal plant operation, the TSC ventilation system intake flow rate of 500 cfm is processed through a HEPA filter. Unfiltered in-leakage during normal operation and Mode 4 operation is 60 cfm.

Following a LOCA, the TSC is manually isolated and the ventilation system is switched to pressurization mode (Mode 4) within 2 hours of the LOCA. In pressurization mode, the TSC air is recirculated through the same filtration unit as the pressurization flow. The pressurization flow is routed through the CRVS pressurization intakes and the TSC charcoal and HEPA filters. The TSC charcoal filter efficiency for elemental and organic iodine is 93 percent and the HEPA filter efficiency for particulates is 98 percent. The recirculation flow rate is 500 cfm and the pressurization flow rate is 500 cfm. Unfiltered in-leakage during pressurization mode is 60 cfm, which includes 10 cfm for CR ingress and egress. The TSC ventilation parameters used in the AST analyses, as described in licensee's letter dated February 1, 2016, are shown below. The TSC assumptions are conservative, reflective of the TSC design, and are therefore acceptable.

Mode 4 TSC Parameters	TSC Ventilation Flow Values (cfm)
Filtered Intake	500
Filtered Recirculation Flow	500
Unfiltered In-leakage	60
TSC Exhaust Flow	560

3.3.2.6.1 Direct Shine Dose Evaluation

The total TSC LOCA dose includes direct shine contributions from the following DBA LOCA radiation sources:

- Direct shine from radioactive material in the containment including shine through the personnel hatch facing the TSC.
- Direct shine from the external radioactive plume outside the TSC pressure boundary resulting from containment leakage, ESF system leakage via the plant vent and the containment penetration areas, RHR pump seal leakage, and MEDT leakage.
- Radiation shine from direct and scattered gamma radiation from the TSC ventilation filters.

The result of the licensee's evaluation of the TSC radiological dose from inhalation, submersion, and direct gamma shine is 4.1 rem TEDE. The licensee used conservative assumptions to evaluate the radiological dose to the TSC, and therefore, the NRC staff finds this evaluation to be acceptable for the AST LOCA analysis.

3.3.2.7 LOCA Radiological Consequences Conclusion

The licensee evaluated the radiological consequences resulting from the postulated LOCA and concluded that the radiological consequences at the TSC, EAB, LPZ, and CR are within the radiation dose reference values provided in 10 CFR 50.67 and the accident specific dose criteria specified in SRP Section 15.0.1. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the NRC staff are presented in Table 3.3-5 and the licensee's calculated dose results are given in Table 3.3-1 of this SE. The NRC staff performed independent confirmatory dose evaluations, as necessary, to ensure a thorough understanding of the licensee's methods. The NRC staff finds that the EAB, LPZ, CR, and TSC radiological doses, estimated by the licensee for the LOCA, meet the applicable accident dose criteria, and are therefore, acceptable.

3.3.3 Fuel Handling Accident

The FHA involves the drop of a fuel assembly on top of other fuel assemblies during refueling operations. The mechanical part of the licensee's analysis remains unchanged from the CLB and it assumes that the total number of failed fuel rods is 264 out of the 50,952 rods in an entire core. The depth of water over the damaged fuel is not less than 23 feet and is controlled by TS 3.7.15, "Spent Fuel Pool Water Level," and TS 3.9.7, "Refueling Cavity Water Level." Following reactor shutdown, decay of short-lived fission products greatly reduces the fission product inventory present in irradiated fuel. The proposed AST amendment takes credit for the normal decay of irradiated fuel.

The analysis was performed, assuming a decay period of 72 hours after shut down and a ground-level release. An FHA in the FHB would involve a release via the plant vent and directly from the closest edge of the FHB to the CR normal intake. An FHA in the containment would involve a release via the plant vent or directly from the containment equipment hatch. However, a release directly from the plant vent would experience more favorable atmospheric dispersion on the path to the CR normal air intake than a release directly from the containment equipment hatch because of the greater distance involved. Therefore, the licensee used the equipment hatch release pathway for the FHA in containment to ensure conservative results.

3.3.3.1 Source Term

The fission product inventory that constitutes the source term for this event is the gap activity in the fuel rods assumed to be damaged as a result of the postulated design basis FHA. Volatile constituents of the core fission product inventory migrate from the fuel pellets to the gap between the pellets and the fuel rod cladding during normal power operations. The fission product inventory in the fuel rod gap of the damaged fuel rods is assumed to be instantaneously released to the surrounding water as a result of the accident. The licensee performed a detailed analysis to ensure that the most restrictive case would be considered for the FHA dose consequence analysis.

Fission products released from the damaged fuel are decontaminated by passage through the overlaying water in the reactor cavity or spent fuel pool depending on their physical and chemical form. Following the guidance in RG 1.183, Appendix B, Regulatory Position 1.3, the licensee assumes: (1) that the chemical form of radioiodine released from the fuel to the spent fuel pool consists of 95 percent cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide, (2) the CsI released from the fuel completely dissociates in the pool water, and (3) because of the low pH of the pool water, the CsI re-evolves and releases elemental iodine. This results in a final iodine distribution of 99.85 percent elemental iodine and 0.15 percent organic iodine. The licensee assumes that the release to the pool water and the chemical redistribution of the iodine species occurs instantaneously.

As corrected by Item 8 of Regulatory Issue Summary 2006-04, "Experience with Implementation of Alternative Source Terms" (ADAMS Accession No. ML053460347), RG 1.183, Appendix B, Regulatory Position 2, "Water Depth," should be read, in part, as follows:

> If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are [285] and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.85%) and organic iodine (0.15%) species results in the iodine above the water being composed of 57% elemental and 43% organic species.

In accordance with RG 1.183, Appendix B, Regulatory Position 2, "Water Depth," the licensee credits an overall iodine DF of 200 for a water cover depth of 23 feet. Consistent with RG 1.183, the licensee credits an infinite DF for the remaining particulate forms of the radionuclides contained in the gap activity and did not credit decontamination from water scrubbing for the noble gas constituents of the gap activity.

The licensee used ORIGEN-S to calculate plant-specific fission product inventories for use in the FHA dose analyses. The fraction of the core that is damaged is assumed to be 1 fuel assembly, which consists of 264 fuel rods out of the 50,952 rods in the full core. A peaking factor of 1.65 is applied to the fission product inventory of the damaged rods.

As stated in DG-1199 (proposed Revision 1 of RG 1.183), for non-LOCA DBAs where only the cladding is postulated to be breached, Table 3.3-6, of this SE gives the fractions of the core inventory for the various radionuclides assumed to be in the gap for a fuel rod. The release fractions from Table 3.3-6 of this SE are used in conjunction with the calculated fission product inventory calculated with the maximum core radial peaking factor. The applicability of Table 3.3-6 non-LOCA fission product gap fractions is limited to fuel assemblies with peak rod power histories below the nodal power envelope depicted in Figure 1 of DG-1199, "Maximum Allowable Power Operating Envelope for Non-LOCA Gap Fractions." DCPP stated that they fall within, and intends to operate within, the maximum allowable power operating envelope for non-LOCA gap fractions shown in Figure 1 of DG-1199.

The licensee analyzed the FHA based on the fuel rod gap activity release fractions of 8 percent of the core I-131 inventory, 23 percent of the core I-132 inventory, 35 percent of the Krypton (Kr)-85 inventory, 4 percent of the remaining noble gas, 5 percent of the remaining halogen isotopes, and 46 percent of the core alkali metals. The licensee stated that per DCPP core-reload design documentation, the peak rod burnup limit at the end of cycle is not allowed to exceed 62,000 MWD/MTU. In addition, the equilibrium core inventory is based on a maximum core average burnup of 50 GWD/MTU. The licensee's approach is consistent with the guidance provided by DG-1199 and RG 1.183 as discussed above, and therefore the NRC staff finds the approach to be acceptable.

3.3.3.2 Transport

Releases from the FHB are via the plant vent and FHB leakage. Releases from the containment are through the containment equipment hatch and the plant vent to the atmosphere. The atmospheric dispersion factors for a release from the containment equipment hatch to the CR normal intake are more limiting than for releases from the plant vent since the containment equipment hatch is much closer to the CR normal air intake than the plant vent. Therefore, to evaluate the FHA release in containment, the licensee used the atmospheric dispersion factors from the equipment hatch to the CR normal intake to conservatively encompass the FHA in containment scenario.

Although RG 1.183 allows for the environmental release of activity from an FHA to occur over a 2-hour period, the licensee conservatively assumed that the environmental release from the FHA in the FHB occurs at a faster release rate due to the FHB ventilation system. Consistent with RG 1.183, the FHA in containment is released over a 2-hour period. In addition, for both FHA scenarios, the licensee assumes a release of the fission products to the environment with no credit for holdup or dilution in the surrounding structures. The FHA transport assumptions are consistent with RG 1.183 and are, therefore, acceptable.

3.3.3.3 CR Habitability for the FHA

The licensee evaluated CR habitability for the FHA assuming that the activity is released directly to the CRVS normal intake from the plant vent using the plant vent to control room atmospheric dispersion factors for the release in the FHB, and from the containment equipment hatch using the equipment hatch to CR atmospheric dispersion factors for the release in containment. In the

FHA analysis, the licensee takes credit for the CR normal intake radiation monitors to initiate CRVS Mode 4, which pressurizes the CR. The licensee determined that upon an FHA, the radiation environment at the CR normal intakes will exceed the radiation monitors analytical limit almost instantaneously, therefore, initiating the CRVS in pressurization mode. The licensee conservatively assumes a 20-second delay time in the initiation of the CRVS by the control room normal intake radiation monitors, a 2-second signal processing time, 10 seconds for damper closure giving an overall delay time of 32 seconds for CRVS pressurization.

The CRVS is designed to maintain the CRE at a positive pressure relative to the surrounding area, following a postulated FHA. CRVS Mode 4 operation is activated on detection of high radiation in the normal outside air intakes. Additionally, on detection of high radiation in the normal outside air intakes, the normal outside air supply for the CR is automatically routed through the less contaminated pressurization air intake of Unit 1 or Unit 2. Pressurization flow, which ranges between 650 to 900 cfm, is drawn from one of the two intakes on the north or south sides of the turbine building. The CR pressurization flow is routed through charcoal and HEPA filters. The CR charcoal filter efficiency for elemental and organic iodine is 93 percent and the HEPA filter efficiency for particulates is 98 percent. Additionally, during postulated accident conditions, the air in the CR is recirculated and a portion of the recirculated flow is filtered through the same filtration unit as the pressurization flow at a flow rate of 1250 cfm. The CRVS parameters, used in the AST analyses, are shown in Section 3.3.2.5.1 of this SE.

The licensee evaluated the radiological consequences resulting from a postulated FHA and concluded that the radiological consequences at the EAB, LPZ, and CR are within the radiological dose guidelines provided in 10 CFR 50.67 and accident-specific dose criteria specified in SRP Section 15.0.1. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the NRC staff are presented in Table 3.3-6 and the licensee's calculated dose results are given in Table 3.3-1 of this SE. The NRC staff performed independent confirmatory dose evaluations, as necessary, to ensure a thorough understanding of the licensee's methods. The NRC staff finds that the EAB, LPZ, and CR doses, estimated by the licensee for the FHA, meet the applicable accident dose criteria and are, therefore, acceptable.

3.3.3.4 TSC Habitability

The DCPP onsite TSC is common to Units 1 and 2. During normal plant operation, the TSC ventilation system intake flow rate of 500 cfm is processed through a HEPA filter. Unfiltered in-leakage during normal operation and Mode 4 operation is 60 cfm, which includes 10 cfm for CR ingress and egress.

In the LAR, the licensee performed a simplified analysis of the TSC dose during an FHA in containment and in the FHB, and concluded that the TSC dose is bounded by the TSC LOCA analysis. The simplified approach conservatively estimates the 30-day integrated inhalation and submersion dose in the TSC for each non-LOCA event. The analysis utilized the RADTRAD LPZ dose model and adjusts the X/Q's and breathing rates for the TSC. Specifically, the X/Q values used for the FHA analysis are applicable to the TSC roof and the RG 1.183 CR breathing rates are used. The licensee did not take credit for either the TSC ventilation systems or the TSC structure. Essentially, the analysis is reflective of an operator located on the roof of the TSC during an FHA.

The NRC staff's review has found that the licensee used conservative analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The NRC staff performed independent confirmatory calculations of the dose consequences of the postulated FHA releases, using the licensee's assumptions for input to the RADTRAD computer code to ensure a thorough understanding of the licensee's methods. The NRC staff's calculations confirmed the licensee's dose results, and therefore, the NRC staff finds the TSC dose to be bounded by the AST LOCA analysis.

3.3.3.5 FHA Radiological Consequences Conclusion

The licensee evaluated the radiological consequences resulting from the postulated FHA and concluded that the radiological consequences at the EAB, LPZ, and CR are within dose guidelines provided by 10 CFR 50.67 and accident specific-criteria specified by SRP Section 15.0.1. Also, based on the results of the independent confirmatory calculations, the NRC staff concluded that TSC dose is bounded by the LOCA AST analysis. The NRC staff finds that the EAB, LPZ, CR, and TSC radiological doses, estimated by the licensee for the FHA, meet the applicable accident does criteria, and therefore, acceptable.

3.3.4 Locked Rotor Accident

The Locked Rotor Accident (LRA) considers the instantaneous seizure of a reactor coolant pump (RCP) rotor, which causes a rapid reduction in the flow through the affected RCS loop. The sudden decrease in core coolant flow causes a reactor trip on a low flow signal. The licensee's evaluation indicates that fuel cladding damage will occur as a result of this accident. Activity from the fuel cladding damage is transported to the secondary side due to primary-to-secondary side leakage. Radioactivity is released to the outside atmosphere from the secondary coolant system via the 10 percent atmospheric dump valves (ADVs) and main steam safety valves (MSSVs). Following reactor trip and based on a coincident assumption of loss of offsite power, the condenser is unavailable and reactor cooldown is achieved using steam releases from the steam generator (SG) MSSVs and 10 percent ADVs until initiation of shutdown cooling. For conservatism, the licensee assumes a total primary-to-secondary leak rate equal to 0.75 gpm from all four SGs, which is higher than the TS total allowable leak rate of 150 gallons per day (gpd) through any one SG to account for any accident induced leakage.

3.3.4.1 Source Term

The licensee assumed that the instantaneous seizure of the RCP rotor associated with the LRA, results in a small percentage of fuel clad damage. The radiological dose analysis for this event conservatively assumes 10 percent fuel clad damage with no fuel melt predicted. Therefore, the source term available for release is associated with this fraction of damaged fuel cladding and the fraction of core activity existing in the gap. A radial peaking factor of 1.65 was applied to the fission product inventory of the damaged rods. The activity released from the fuel is assumed to be released instantaneously and homogeneously through the RCS. Following the guidance in RG 1.183, Appendix G, Regulatory Position 4, the licensee assumes that the chemical form of radioiodine released from the fuel to the reactor coolant consists of 95 percent CsI, 4.85 percent elemental iodine, and 0.15 percent organic iodide, and that the iodine releases from the SG to the environment is 97 percent elemental and 3 percent organic.

As stated in DG-1199 (proposed revision 1 of RG 1.183) for non-LOCA DBAs, where only the cladding is postulated to be breached, Table 2, "PWR Core Inventory Fraction Released into Containment Atmosphere," gives the fractions of the core inventory for the various radionuclides assumed to be in the gap for a fuel rod. The release fractions from Table 3, "Non-LOCA Fraction of Fission Product Inventory in Gap," are used in conjunction with the calculated fission product inventory calculated with the maximum core radial peaking factor. The applicability of Table 3 non-LOCA fission product gap fractions is limited to fuel assemblies with peak rod power histories below the nodal power envelope depicted in Figure 1 of DG-1199. DCPP stated that they fall within, and intend to operate within, the maximum allowable power operating envelope for non-LOCA gap fractions shown in Figure 1 of DG-1199.

The licensee analyzed the LRA based on the fuel rod gap activity release fractions of 8 percent of the core I-131 inventory, 23 percent of the core I-132 inventory, 35 percent of the Kr-85 inventory, 4 percent of the remaining noble gas, 5 percent of the remaining halogen isotopes, and 46 percent of the core alkali metals. The licensee stated that per DCPP core-reload design documentation, the peak rod burnup limit at the end of cycle is not allowed to exceed 62,000 MWD/MTU. In addition, the equilibrium core inventory is based on a maximum core average burnup of 50 GWD/MTU. The NRC staff finds this approach to be consistent with the draft regulatory guidance and, therefore, acceptable.

3.3.4.2 Release Transport

The activity that originates in the RCS is released to the secondary coolant by means of the primary-to-secondary coolant leak rate. The licensee assumes a conservative value for the design-basis leak rate of 0.75 gpm from all four SGs. This equates to a total of 1080 gpd, which is greater than the maximum allowable operational leakage of 150 gpd for any one SG imposed in TS 3.4.13d. A loss of offsite power is assumed to occur concurrently with the reactor trip, which results in releases to the environment associated with the secondary coolant steaming from the SGs.

Because of the release dynamic of the activity from the SGs, RG 1.183 allows for a reduction in the amount of activity released to the environment based on partitioning of nuclides between the liquid and gas states of water for this release path. For iodine, because the SG tubes remain covered for the duration of the LRA, the partition coefficient of 100 was taken directly from the suggested guidance. Because of their volatility, 100 percent of the noble gases are assumed to be released. The licensee assumes that the steaming release from the 10 percent ADVs, MSSVs, and primary-to-secondary coolant leakage end after 10.73 hours, at which time the shutdown cooling is initiated via the RHR system.

The licensee used the RADTRAD 3.03 computer code to model the time dependent transport of radionuclides, from the primary-to-secondary side and consequently to the environment via the 10 percent ADVs and MSSVs. The licensee's analysis is consistent with Appendix G of RG 1.183, "Assumptions for Evaluating the Radiological Consequences of a PWR Locked Rotor Accident," which identifies acceptable radiological analysis assumptions for an LRA. The licensee determined that the LRA does not initiate any signal which could automatically start the CRVS Mode 4. Therefore, the CRVS remains in normal operation mode and the licensee does not credit CR pressurization or any other safety functions of the CRVS in the LRA analysis.

The licensee evaluated the radiological consequences resulting from the postulated LRA and concluded that the radiological consequences at the EAB, LPZ, and CR are within the radiological dose guidelines provided in 10 CFR 50.67 and accident dose criteria specified in SRP Section 15.0.1. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the NRC staff are presented in Table 3.3-7 and the licensee's calculated dose results are given in Table 3.3-1 of this SE. The NRC staff performed independent confirmatory dose evaluations, as necessary, to ensure a thorough understanding of the licensee's methods. The NRC staff review confirmed that the EAB, LPZ, and CR doses estimated by the licensee for the LRA meet the applicable accident dose criteria and are, therefore, acceptable.

3.3.4.3 TSC Habitability

The DCPP onsite TSC is common to Units 1 and 2. During normal plant operation, the TSC ventilation system intake flow rate of 500 cfm is processed through a HEPA filter. Unfiltered in-leakage during normal operation and Mode 4 operation is 60 cfm, which includes 10 cfm for CR ingress and egress.

In the LAR, the licensee performed a simplified analysis of the TSC dose during an LRA and concluded that the TSC dose is bounded by the TSC LOCA analysis. The simplified approach conservatively estimates the 30-day integrated inhalation and submersion dose in the TSC for each non-LOCA event. The analysis utilized the RADTRAD LPZ dose model and adjusted the X/Q's and breathing rates for the TSC. Specifically, the X/Q values used for the LRA analysis are applicable to the TSC roof and the RG 1.183 CR breathing rates are used. The licensee did not take credit for either the TSC ventilation systems or the TSC structure. Essentially, the analysis is reflective of an operator located on the roof of the TSC during an LRA.

The NRC staff's review has found that the licensee used conservative analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The NRC staff performed independent confirmatory calculations of the dose consequences of the postulated LRA releases, using the licensee's assumptions for input to the RADTRAD computer code, to ensure a thorough understanding of the licensee's methods. The NRC staff's calculations confirmed the licensee's dose results, and therefore, the NRC staff finds the TSC dose to be bounded by the AST LOCA analysis.

3.3.4.4 LRA Radiological Consequences Conclusion

The licensee evaluated the radiological consequences resulting from the postulated LRA and concluded that the radiological consequences at the EAB, LPZ, and CR are within dose guidelines provided by 10 CFR 50.67 and accident specific-criteria specified by SRP Section 15.0.1. Also, based on the results of the independent confirmatory calculations, the NRC staff concluded that TSC dose is bounded by the LOCA AST analysis. The NRC staff finds that the EAB, LPZ, CR, and TSC radiological doses, estimated by the licensee for the LRA, meet the applicable accident does criteria, and therefore, acceptable.

3.3.5 Control Rod Ejection Accident (CREA)

DCPP FSARU, Section 15.4.6, describes the control rod ejection accident (CREA) as the mechanical failure of a control rod mechanism pressure housing resulting in the ejection of a rod cluster control assembly (RCCA) and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion and system depressurization together with an adverse core power distribution, possibly leading to localized fuel rod damage. Following the applicable guidance, the licensee evaluated two separate release scenarios for the CREA. In the first case, the failed fuel resulting from the CREA is released in its entirety into the containment via the ruptured control rod drive mechanism housing, is mixed in the free volume of the containment, and then released to the environment at the containment technical specification leak rate for the first 24 hours and at half that value for the remaining 29 days.

For the second case, the radiological consequences from a CREA is evaluated assuming that the RCS boundary remains intact and that fission products are released to the environment from the secondary system. In this case, fission products from the damaged fuel are assumed to be released to the primary coolant and transported to the secondary system through primary-to-secondary leakage in the SGs. Both CREA cases are analyzed with the assumption of a concurrent loss of offsite power, which causes steam releases from the secondary system to occur through the MSSVs and 10 percent ADVs to the environment.

3.3.5.1 Source Term

The source term for the CREA is assumed to result in fuel damage consisting of localized damage to fuel cladding with no fuel melt occurring in the damaged rods. The source term for the CREA is described in RG 1.183, Appendix H, "Assumptions for Evaluating the Radiological consequences of a PWR Rod Ejection Accident," Regulatory Position 1, which states that:

> Assumptions acceptable to the NRC staff regarding core inventory are in Regulatory Position 3 of this guide. For the rod ejection accident, the release from the breached fuel is based on the estimate of the number of fuel rods breached and the assumption that 10% of the core inventory of the noble gases and iodines is in the fuel gap. The release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting and the assumption that 100% of the noble gases and 25% of the iodines contained in that fraction are available for release from containment. For the secondary system release pathway, 100% of the noble gases and 50% of the iodines in that fraction are released to the reactor coolant.

The licensee assumed that as a result of the CREA, 10 percent of the fuel experiences departure from nucleate boiling resulting in cladding damage. A peaking factor of 1.65 was applied to the fission product inventory of the damaged rods. Consistent with the guidance provided in RG 1.183, Appendix H, the licensee assumed that 10 percent of the core inventory of noble gases and iodine reside in the fuel gap and will be available for release in both the ruptured control rod drive mechanism housing scenario and the secondary-side release scenario. The licensee assumes no melted fuel for either scenario.

In accordance with RG 1.183, Appendix H, Regulatory Position 3, 100 percent of the released activity is assumed to be released instantaneously and mixed homogeneously throughout the containment atmosphere for the ruptured control rod drive mechanism housing; and, 100 percent of the released activity is assumed to be released instantaneously and completely dissolved in the primary coolant and available for release to the secondary containment in the secondary-side release scenario. The NRC staff finds the CREA source term assumptions to be consistent with the RG 1.183 and, therefore, acceptable.

3.3.5.2 Transport from Containment

The licensee used the minimum containment free air volume to conservatively maximize the radioactive concentration in containment. The licensee assumes that the activity released to the containment through the rupture in the reactor vessel head mixes instantaneously throughout the containment with no credit assumed for removal of iodine or noble gas in the containment due to containment sprays or natural deposition. The licensee assumes that all containment leakage is at the TS limit of 0.1 percent per day for the first 24 hours and 0.05 percent per day thereafter. The licensee assumes that the iodine released to the containment from the fuel consists of 95 percent particulate, 4.85 percent elemental, and 0.15 percent organic per RG 1.183, Appendix H, Regulatory Position 4. However, because no credit is taken for actuation of containment sprays or pH control, the iodine released via the containment leakage pathway is assumed to have the same composition as the iodine activity released to the environment from the secondary system release pathway. The licensee assumes that 97 percent of all halogens available for release to the environment are elemental, while the remaining 3 percent is organic. The NRC staff finds the CREA containment transport assumptions to be consistent with the RG 1.183 and, therefore, acceptable.

3.3.5.3 Transport from Secondary System

In accordance with RG 1.183, Appendix H, Regulatory Position 7, the licensee evaluated the transport of activity from the RCS to the SGs secondary side assuming a total primary-to-secondary leak rate equal to 0.75 gpm from all four SGs, which is higher than the TS 3.4.13d total allowable leak rate of 150 gpd through any one SG, to account for any accident induced leakage. The licensee assumes that this leak rate persists for a period of 10.73 hours until shutdown cooling is in operation and the RCS and the SG pressures have equalized. In accordance with RG 1.183, the licensee assumes that all noble gas radionuclides released from the primary system are released to the environment without reduction or mitigation. Following the guidance from RG 1.183, Appendix E, "Assumptions for Evaluation the Radiological Consequences of a PWR Main Steam Line Break Accident," Regulatory Position 5.5, the licensee assumes that all of the primary-to-secondary leakage in the SGs mix with the secondary water without flashing. For iodine, because the SG tubes remain covered for the duration of the CREA, the partition coefficient of 100 was taken directly from RG 1.183. The licensee assumes a loss of offsite power coincident with the reactor trip making the condenser unavailable and reactor cooldown is achieved using steam releases from the SG MSSVs and 10 percent ADVs until initiation of shutdown cooling. Following the guidance from RG 1.183, Appendix E, Regulatory Position 5, the licensee assumed that 97 percent of all halogens available for release to the environment are elemental, while the remaining 3 percent is organic. The NRC staff finds the CREA secondary system transport assumptions to be consistent with the RG 1.183 and, therefore, acceptable.

3.3.5.4 CR Habitability for the CREA

The licensee evaluated CR habitability for the CRE assuming that the CRVS automatically transfers to the pressurization mode of operation after the initiation of SI on a containment high pressure signal. As discussed earlier, releases to the containment following a CREA is through a ruptured control rod drive mechanism housing. The licensee compared the calculated ruptured control rod drive mechanism housing size to that of a 2-inch small break LOCA and determined that the time to generate the containment high-pressure signal will conservatively be 300 seconds. The licensee assumes the CRVS normal intake dampers of the accident unit start to close after a 28.2 second delay due to diesel generator loading onto the emergency buses and 10 seconds for damper closure, giving an overall delay time of 338.2 seconds for CRVS pressurization.

The CRVS is designed to maintain the CRE at a positive pressure relative to the surrounding area, following a postulated CREA. Upon CRVS Mode 4 initiation, the normal outside air supply to the CR is automatically routed through the less contaminated pressurization air intake of

Unit 1 or Unit 2. Pressurization flow, which ranges between 650 to 900 cfm and is drawn from one of the two intakes on the north or south sides of the turbine building. The CR pressurization flow is routed through charcoal and HEPA filters. The CR charcoal filter efficiency for elemental and organic iodine is 93 percent and the HEPA filter efficiency for particulates is 98 percent. Additionally, during postulated accident conditions, part of the air in is recirculated and a portion of the recirculated flow is filtered through the same filtration unit as the pressurization flow at a flow rate of 1250 cfm. The CRVS parameters used in the AST analyses are shown in Section 3.3.2.5.1 of this SE.

The licensee evaluated the radiological consequences resulting from the postulated CREA and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 50.67 and accident specific dose criteria specified in SRP Section 15.0.1. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the NRC staff are presented in Table 3.3-8 and the licensee's calculated dose results are given in Table 3.3-1 of this SE. The NRC staff performed independent confirmatory dose evaluations, as necessary, to ensure a thorough understanding of the licensee's methods. The EAB, LPZ, and CR doses estimated by the licensee for the CREA were found to meet the applicable accident dose criteria and are, therefore, acceptable.

3.3.5.5 TSC Habitability for CREA into the Secondary System

The DCPP onsite TSC is common to Units 1 and 2. During normal plant operation, the TSC ventilation system intake flow rate of 500 cfm is processed through a HEPA filter. Unfiltered in-leakage during normal operation and Mode 4 operation is 60 cfm, which includes 10 cfm for CR ingress and egress.

In the LAR, the licensee performed a simplified analysis of the TSC dose during a CREA release into the secondary system and concluded that the TSC dose is bounded by the TSC LOCA analysis. The simplified approach conservatively estimates the 30-day integrated inhalation and submersion dose in the TSC for each non-LOCA event. The analysis utilized the RADTRAD LPZ dose model and adjusted the X/Q's and breathing rates for the TSC. Specifically, the X/Q values used for the CREA analysis are applicable to the TSC roof and the RG 1.183 CR breathing rates are used. The licensee did not take credit for either the TSC ventilation systems or the TSC structure. Essentially, the analysis is reflective of an operator located on the roof of the TSC during a CREA release into the secondary system.

The NRC staff's review has found that the licensee used conservative analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The NRC staff performed independent confirmatory calculations of the dose consequences of the postulated CREA into the secondary system, using the licensee's assumptions for input to the RADTRAD computer code, to ensure a thorough understanding of the licensee's methods. The NRC staff's calculations confirmed the licensee's dose results and therefore, the NRC staff finds the TSC dose to be bounded by the AST LOCA analysis.

3.3.5.6 TSC Habitability for CREA Release into Containment

The licensee's simplistic evaluation, discussed above in Section 3.3.5.5, did not demonstrate that the LOCA dose bounded the CREA release into containment, therefore the licensee evaluated the TSC dose using the full activity transport model. The DCPP onsite TSC is common to Units 1 and 2. During normal plant operation, the TSC ventilation system intake flow rate of 500 cfm is processed through a HEPA filter. Unfiltered in-leakage during normal operation and Mode 4 operation is 60 cfm, which includes 10 cfm for CR ingress and egress.

Following a CREA, the TSC is manually isolated and the ventilation system is switched to pressurization mode within 2 hours of the CREA. In pressurization mode, the TSC air is recirculated through the same filtration unit as the pressurization flow. The pressurization flow is routed through the CRVS pressurization intakes and the TSC charcoal and HEPA filters. The TSC charcoal filter efficiency for elemental and organic iodine is 93 percent and the HEPA filter efficiency for particulates is 98 percent. The recirculation flow rate is 500 cfm and the pressurization flow rate is 500 cfm. Unfiltered in-leakage during pressurization mode is 60 cfm. The TSC ventilation parameters used in the AST analyses are shown in Section 3.3.2.6 of this SE.

The result of the licensee's evaluation of the TSC radiological dose from inhalation and submersion is 4 rem TEDE. The licensee used conservative assumptions to evaluate the radiological dose to the TSC, and therefore, the NRC staff finds this evaluation to be acceptable for the AST CREA release into containment analysis.

3.3.5.7 CRA Radiological Consequence Conclusion

The licensee evaluated the radiological consequences resulting from the postulated CRA and concluded that the radiological consequences at the EAB, LPZ, and CR are within dose guidelines provided by 10 CFR 50.67 and accident specific-criteria specified by SRP Section 15.0.1. Also, based on the results of the independent confirmatory calculations, the NRC staff concluded that TSC dose is bounded by the LOCA AST analysis for the CREA release into the secondary system. For the CREA release into the containment, the licensee used conservative assumptions to evaluate the radiological dose to the TSC, and therefore, the NRC staff finds the evaluation to be acceptable. The NRC staff finds that the EAB, LPZ, CR, and TSC radiological doses, estimated by the licensee for the CRA, meet the applicable accident does criteria, and therefore, acceptable.

3.3.6 Main Steam Line Break Accident

The postulated main steam line break (MSLB) accident assumes a double-ended break of one main steam line outside the primary containment. This leads to an uncontrolled release of steam from the steam system. The resultant depressurization of the steam system causes the main steam isolation valves to close and, if the plant is operating at power when the event is initiated, causes a reactor scram. For the MSLB DBA radiological consequence analysis, a loss of offsite power occurs coincident with the reactor trip. Following a reactor trip and turbine trip, the radioactivity is released to the environment through the MSSVs and 10 percent ADVs on the intact SGs and from the break point on the faulted SG. Because the loss of offsite power renders the main condenser unavailable, the plant is cooled down by releasing steam to the environment via the MSSVs and 10 percent ADVs.

The radiological consequences of an MSLB outside containment will bound the consequences of a break inside containment. Therefore, only the MSLB outside of containment is considered with regard to the radiological consequences. The affected SG, hereafter referred to as the faulted SG, rapidly depressurizes and releases its initial contents to the environment. The MSLB accident is described in DCPP FSARU, Section 15.4.2. RG 1.183, Appendix E, identifies acceptable radiological analysis assumptions for a pressurized water reactor MSLB.

As stated above, the steam release from a rupture of a main steam line would result in an initial increase in steam flow, which decreases during the accident as the steam pressure decreases. The increased energy removal from the RCS causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity. If the most reactive RCCA is stuck in its fully withdrawn position after the reactor trip, there is an increased possibility that the core will become critical and return to power. The core is ultimately shut down by the boric acid delivered by the SI system.

3.3.6.1 Source Term

Appendix E of RG 1.183 identifies acceptable radiological analysis assumptions for a PWR MSLB accident. RG 1.183, Appendix E, Regulatory Position 2, states that if no or minimal fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by the TSs including the effects of pre-accident and concurrent iodine spiking. The licensee's evaluation indicates that no fuel damage would occur as a result of an MSLB accident.

Therefore, the licensee considered the two radioiodine spiking cases described in RG 1.183. The first case is referred to as a pre-accident iodine spike and assumes that a reactor transient has occurred prior to the postulated MSLB that has raised the primary coolant iodine concentration to the maximum value permitted by the TS for a spiking condition. For DCPP, the maximum iodine concentration allowed by TS 3.4.16 as the result of an iodine spike is 60 μCi/gm of Dose Equivalent I-131 (DEI).

The second case assumes that the primary system transient associated with the MSLB causes an iodine spike in the primary system. This case is referred to as a concurrent iodine spike. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value specified in DCPP TS. For DCPP, the RCS TS 3.4.16, "RCS Specific Activity," limit for equilibrium or normal operation is 1.0 μCi/gm DEI. The duration of the concurrent iodine spike is assumed to be 8 hours in accordance with RG 1.183.

For the MSLB accident, the licensee evaluated the radiological dose contribution from the release of secondary-side activity using the equilibrium secondary-side specific activity found in DCPP TS 3.7.18, "Secondary Specific Activity," as 0.1 μCi/gm DEI. The licensee assumes that the chemical form of iodine released from the SGs to the environment is 97 percent elemental and 3 percent organic consistent with RG 1.183, Appendix E, Regulatory Position 4. The NRC staff finds the MSLB source term assumptions to be consistent with RG 1.183 and, therefore, acceptable.

3.3.6.2 Release Transport

The licensee followed the guidance, as described in RG 1.183, Appendix E, Regulatory Position 5, in all aspects of the transport analysis for the MSLB. For additional conservatism the licensee assumes a total primary-to-secondary leak rate equal to 0.75 gpm (1080 gpd), which is higher than the TS 3.4.13d total allowable leak rate of 150 gpd per SG, which is a total of 600 gpd from all 4 SGs. The licensee modeled the assumed primary-to-secondary leakage of 0.75 gpm into the faulted SG.

RG 1.183, Appendix E, Regulatory Position 5.2, states:

> The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr [pounds of mass per hour]) should be consistent with the basis of the parameter being converted. The ARC [alternate repair criteria] leak rate correlations are generally based on the collection of cooled liquid. Surveillance tests and facility instrumentation used to show compliance with leak rate technical specifications are typically based on cooled liquid. In most cases, the density should be assumed to be 1.0 gm/cc [grams per cubic centimeter] (62.4 lbm/ft^3 [pounds of mass per cubic foot]).

The licensee assumes a leakage density of 62.4 lbm/ft^3. RG 1.183, Appendix E, Regulatory Position 5.3, states:

> The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100 °C (212 °F). The release of radioactivity from unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.

In accordance with RG 1.183, the licensee assumes that primary-to-secondary leakage continues until the RCS reaches 212 °F, which is 30 hours after the MSLB, while the intact SG releases terminate at 10.73 hours, at which time shutdown cooling is initiated using the RHR system. In accordance with RG 1.183, the licensee assumes that all noble gas radionuclides released from the primary system are released through the faulted SG to the environment without reduction or mitigation. Following the guidance from RG 1.183, Appendix E, Regulatory Positions 5.5.1, 5.5.2, and 5.5.3, the licensee assumes that all of the primary-to-secondary leakage into the faulted SG will flash to vapor, and be released to the environment with no mitigation. For the unaffected SGs that are used for plant cooldown, the licensee assumes that no primary-to secondary leakage exists.

RG 1.183, Appendix E, Regulatory Position 5.5.4, states:

> The radioactivity in the bulk water is assumed to become vapor at a rate that is the function of the steaming rate and the partition coefficient. A partition coefficient for iodine of 100 may be assumed. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators.

Accordingly, the licensee assumes that the radioactivity in the initial bulk water of the unaffected SGs becomes vapor at a rate that is a function of the steaming rate and the inverse of the partition coefficient. The licensee used a partition coefficient of 100 for iodine released from the intact SGs. The licensee assumes that noble gases are released freely to the environment without retention in the SGs. The iodine releases to the environment from the unaffected SGs are assumed to be 97 percent elemental and 3 percent organic, which is consistent with Regulatory Position 4 in RG 1.183, Appendix E.

The total release from the faulted SG is 182,544 pounds per mass (lbm) initially plus 0.75 gpm from the primary-to-secondary leakage for 30 hours. Thirty hours after the accident, no further steam containing radionuclides is released from the faulted SG to the environment. The NRC staff finds the MSLB transport assumptions to be consistent with RG 1.183 and, therefore, acceptable.

3.3.6.3 CR Habitability for the MSLB

The licensee evaluated CR habitability for the CRE assuming that the CRVS automatically transfers to the pressurization mode of operation after the initiation of SI. The licensee determined that the time to generate the SI signal will conservatively be 0.6 seconds. The licensee assumes the CRVS normal intake dampers of the accident unit start to close after a 28.2-second delay due to diesel generator loading onto the emergency buses, and 10 seconds for damper closure giving an overall delay time of 38.8 seconds for CRVS pressurization.

The CRVS is designed to maintain the CRE at a positive pressure relative to the surrounding area, following a postulated MSLB. Upon CRVS Mode 4 initiation, the normal outside air supply to the CR is automatically routed through the less contaminated pressurization air intake of Unit 1 or Unit 2. Pressurization flow, which ranges between 650 to 900 cfm, and is drawn from one of the two intakes on the north or south sides of the turbine building. The CR pressurization flow is routed through charcoal and HEPA filters. The CR charcoal filter efficiency for elemental and organic iodine is 93 percent and the HEPA filter efficiency for particulates is 98 percent. Additionally, during postulated accident conditions, the air in the CR is recirculated and a portion of the recirculated flow is filtered through the same filtration unit as the pressurization flow at a flow rate of 1250 cfm. The CRVS parameters used in the AST analyses are shown in Section 3.3.2.5.1 of this SE.

The licensee determined that an atmospheric dispersion factor cannot be accurately determined while the CR normal intake of the faulted unit is in normal operation because of its close proximity to the MSLB point. The licensee is not crediting atmospheric dispersion when determining the CR operator dose from the secondary coolant discharge or the primary-to-secondary SG tube leakage released from the faulted SG via the break point. In the analysis, the licensee considers the primary-to-secondary SG tube leakage into the faulted SG is conservatively assumed to be piped directly into the CR and the secondary coolant discharge activity concentration entering the CR is assumed to be the same as the concentration of the flash liquid at the break point until the control room normal intake is isolated and the CRVS realigned to pressurization mode.

The licensee evaluated the radiological consequences resulting from the postulated MSLB and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 50.67 and accident specific dose criteria specified in SRP Section 15.0.1. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the NRC staff are presented in Table 3.3-9 and the licensee's calculated dose results are given in Table 3.3-1 of this SE. The NRC staff performed independent confirmatory dose evaluations, as necessary, to ensure a thorough understanding of the licensee's methods. The EAB, LPZ, and CR doses, estimated by the licensee for the MSLB, were found to meet the applicable accident dose criteria, and are therefore, acceptable.

3.3.6.4 TSC Habitability for the MSLB

The DCPP onsite TSC is common to Units 1 and 2. During normal plant operation, the TSC ventilation system intake flow rate of 500 cfm is processed through a HEPA filter. Unfiltered in-leakage during normal operation and Mode 4 operation is 60 cfm, which includes 10 cfm for CR ingress and egress.

Following an MSLB, the TSC is manually isolated and the ventilation system is switched to pressurization mode within 2 hours of the MSLB. In pressurization mode, the TSC air is recirculated through the same filtration unit as the pressurization flow. The pressurization flow is routed through the CRVS pressurization intakes and the TSC charcoal and HEPA filters. The TSC charcoal filter efficiency for elemental and organic iodine is 93 percent and the HEPA filter efficiency for particulates is 98 percent. The recirculation flow rate is 500 cfm and the pressurization flow rate is 500 cfm. Unfiltered in-leakage during pressurization mode is 60 cfm. The TSC ventilation parameters used in the AST analyses are shown in Section 3.3.2.6 of this SE.

The result of the licensee's evaluation of the TSC radiological dose from inhalation and submersion is 0.7 rem TEDE. The licensee used conservative assumptions to evaluate the radiological dose to the TSC, and therefore, the NRC staff finds this evaluation to be acceptable for the AST MSLB analysis.

3.3.6.5 MSLB Radiological Consequence Conclusion

The licensee evaluated the radiological consequences resulting from the postulated MSLB and concluded that the radiological consequences at the EAB, LPZ, and CR are within dose guidelines provided by 10 CFR 50.67 and accident specific-criteria specified by SRP Section 15.0.1. Also, the NRC staff concluded that the licensee used conservative assumptions to evaluate the radiological dose to the TSC, and find it acceptable. The NRC staff finds that the EAB, LPZ, CR, and TSC radiological doses, estimated by the licensee for the MSLB, meet the applicable accident does criteria, and therefore, acceptable.

3.3.7 Steam Generator Tube Rupture Accident

The steam generator tube rupture (SGTR) accident assumes an instantaneous and complete severance of a single SG tube. The postulated break allows primary coolant to leak to the secondary side of the ruptured SG. The radioactivity from the leaking SG tube mixes with the shell-side water in the affected SG. For the SGTR DBA radiological consequence analysis, prior to the reactor trip, the radioactivity in the steam is released to the environment from the air ejector, which discharges to the plant vent. The licensee assumes that offsite power is lost coincident with the reactor trip at 179 seconds after the SGTR. Following the reactor trip and turbine trip, the radioactivity in the steam is released to the environment through the SG MSSVs and 10 percent ADVs. Subsequently, a 10 percent ADV of the ruptured SG is stuck open for 30 minutes. Because the loss of offsite power renders the main condenser unavailable, the plant is cooled down by releasing steam to the environment from the MSSVs and 10 percent ADVs of the intact SGs.

3.3.7.1 Source Term

Appendix F of RG 1.183, "Assumptions for Evaluation the Radiological Consequences of a PWR Steam Generator Tube Rupture Accident," identifies acceptable radiological analysis assumptions for an SGTR accident. RG 1.183, Appendix F, Regulatory Position 2, states that "if no or minimal fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by technical specification," and that "two cases of iodine spiking should be assumed." The licensee's evaluation indicates that no fuel damage would occur as a result of a SGTR accident. Therefore, consistent with RG 1.183, the licensee performed the SGTR accident analyses for two radioiodine spiking cases. The first case is referred to as a pre-accident iodine spike and assumes that a reactor transient has occurred prior to the postulated SGTR that has raised the primary coolant iodine concentration to the maximum value permitted by the TS for a spiking condition. For DCPP, the maximum iodine concentration allowed by TS 3.4.16, as a result of an iodine spike, is 60 μCi/gm DEI.

The second case assumes that the primary system transient associated with the SGTR causes an iodine spike in the primary system. This case is referred to as a concurrent iodine spike. Initially, the plant is assumed to be operating with the RCS iodine activity at the TS limit for normal operation. For DCPP, the RCS TS 3.4.16 limit for normal operation is 1 μCi/gm DEI. The increase in primary coolant iodine concentration for the concurrent iodine spike case is estimated using a spiking model that assumes that as a result of the accident, iodine is released from the fuel rods to the primary coolant at a rate that is 335 times greater than the iodine equilibrium release rate. The iodine release rate at equilibrium is equal to the rate at which iodine is lost due to radioactive decay, RCS purification, and RCS leakage. The iodine release rate is also referred to as the iodine appearance rate. The concurrent iodine spike is assumed to persist for a period of 8 hours. The licensee assumes that the activity released from the iodine spiking mixes instantaneously and homogeneously throughout the primary coolant system.

In accordance with RG 1.183, Appendix F, Regulatory Position 4, the licensee assumes the speciation for iodine release from the SGs is 97 percent elemental and 3 percent organic. In addition, the licensee included the radiological dose contribution from the release of secondary coolant iodine activity at the TS 3.7.18 limit of 0.1 μCi/gm DEI. The NRC staff finds the SGTR source term assumptions to be consistent with RG 1.183 and, therefore, acceptable.

3.3.7.2 Release Transport

The licensee followed the guidance as described in RG 1.183, Appendix F, Regulatory Position 5, in all aspects of the transport analysis for the SGTR. For additional conservatism the licensee assumes a total primary-to-secondary leak rate equal to 0.75 gpm (1080 gpd), which is higher than the TS 3.4.13d total allowable leak rate of 150 gpd per SG. The licensee modeled the primary-to-secondary leakage for all the SGs, 0.75 gpm, is into the three intact SGs.

RG 1.183, Appendix F, Regulatory Position 5.2, states:

> The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of the surveillance tests used to show compliance with leak rate technical specifications. These tests are typically based on cool liquid. Facility instrumentation used to determine leakage is typically located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).

The licensee's SGTR leak rate of 0.75 gpm corresponds to a leakage density of 62.4 lbm/ft^3 and is into the three intact SGs. RG 1.183, Appendix F, Regulatory Position 5.3, states:

> The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100 °C (212 °F). The release of radioactivity from the unaffected steam generators should be assumed to

continue until shutdown cooling is in operation and releases from the steam generators have been terminated.

In accordance with RG 1.183, the licensee assumes that the release of radioactivity from the ruptured SG continues for 1.63 hours and the unaffected SGs continues for 10.73 hours at which time shutdown cooling is initiated using the RHR system, and steam releases from the SGs have been terminated.

The licensee evaluated the dose consequences from discharges of steam from the intact SGs for a period of 10.73 hours, until the primary system has cooled sufficiently to allow an alignment to the RHR system. At this point in the accident sequence, steaming is no longer required for cooldown and releases from the intact SGs are terminated.

The licensee assumes that the source term resulting from the radionuclides in the primary system coolant, including the contribution from iodine spiking, is transported to the ruptured SG by the break flow. A portion of the break flow is assumed to flash to steam because of the higher enthalpy in the RCS relative to the secondary system. The licensee assumes that the flashed portion of the break flow will ascend through the bulk water in the SG, enter the steam space of the affected generator, and be immediately available for release to the environment with no credit taken for scrubbing. Although RG 1.183 allows the use of the methodologies, described in NUREG-0409, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident," May 1985, to determine the amount of scrubbing credit applied to the flashed portion of the break flow, the licensee did not credit scrubbing of the activity in the flashed break flow in the ruptured SG.

During the first 179 seconds of the event, prior to the reactor trip and the concurrent loss of offsite power, the licensee assumes that all of the SG flow is routed to the condenser. The licensee applied an additional DF of 100, to the condenser releases. Therefore, the steam released from the condenser during the first 179 seconds following the SGTR has a total iodine DF of 10,000, which includes a partition coefficient of 100 as a result of changing phase in the SG and an additional partitioning factor of 100 exiting through the condenser. The licensee assumes that iodine that remains in the main condenser enters the condensate and returns to both the intact SGs and the ruptured SG and is available for future steam releases. After 179 seconds, the condenser is no longer available due to the loss of offsite power. Therefore, the additional condenser partitioning factor of 100 is only applied to the flashed flow for the first 179 seconds of the event.

In accordance with RG 1.183, the licensee assumes that all noble gas radionuclides released from the primary system are released through the SGs to the environment without reduction or mitigation. In the ruptured SG, the licensee assumes the iodine in the flashed portion of the break flow is immediately available for release without reduction or mitigation.

The licensee has determined that for the SGTR accident, all SGs effectively maintain tube coverage. In accordance with RG 1.183, Appendix E, Regulatory Position 5.5.1, the licensee assumes that for the ruptured SG, and the unaffected SGs used for plant cooldown, the primary-to-secondary leakage mixes with the secondary water without flashing due to the total submergence of the SG tubes. The iodine in the primary-to-secondary leakage flow is assumed to mix uniformly with the SG liquid mass and be released to the environment in direct proportion to the steaming rate and in inverse proportion to the applicable partition coefficient. The NRC staff finds the SGTR transport assumptions to be consistent with RG 1.183 and, therefore, acceptable.

3.3.7.3 CR Habitability for the SGTR

The licensee evaluated CR habitability for the SGTR assuming that the CRVS automatically transfers to the pressurization mode of operation after the initiation of SI. The licensee determined that the time to generate the SI signal will be 219 seconds. The licensee assumes the CRVS normal intake dampers of the accident unit start to close after a 28.2 second delay due to diesel generator loading onto the emergency buses, and 10 seconds for damper closure giving an overall delay time of 257.2 seconds for CRVS pressurization.

The CRVS is designed to maintain the CRE at a positive pressure relative to the surrounding area, following a postulated SGTR. Upon CRVS Mode 4 initiation, the normal outside air supply to the CR is automatically routed through the less contaminated pressurization air intake of Unit 1 or Unit 2. Pressurization flow, which ranges between 650 to 900 cfm and is drawn from one of the two intakes on the north or south sides of the turbine building. The CR pressurization flow is routed through charcoal and HEPA filters. The CR charcoal filter efficiency for elemental and organic iodine is 93 percent and the HEPA filter efficiency for particulates is 98 percent. Additionally, during postulated accident conditions, the air in the CR is recirculated and a portion of the recirculated flow is filtered through the same filtration unit as the pressurization flow, at a flow rate of 1250 cfm. The CRVS parameters used in the AST analyses are shown in Section 3.3.2.5.1 of this SE.

The licensee evaluated the radiological consequences resulting from the postulated SGTR and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 50.67 and accident specific dose criteria specified in SRP Section 15.0.1. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the NRC staff are presented in Table 3.3-10 and the licensee's calculated dose results are given in Table 3.3-1 of this SE. The NRC staff performed independent confirmatory dose evaluations, as necessary, to ensure a thorough understanding of the licensee's methods. The EAB, LPZ, and CR doses estimated by the licensee for the SGTR were found to meet the applicable accident dose criteria, and are therefore, acceptable.

3.3.7.4 TSC Habitability for SGTR

The DCPP onsite TSC is common to Units 1 and 2. During normal plant operation, the TSC ventilation system intake flow rate of 500 cfm is processed through a HEPA filter. Unfiltered in-leakage during normal operation and Mode 4 operation is 60 cfm, which includes 10 cfm for CR ingress and egress.

In the LAR, the licensee performed a simplified analysis of the TSC dose during a SGTR and concluded that the TSC dose is bounded by the TSC LOCA analysis. The simplified approach conservatively estimates the 30-day integrated inhalation and submersion dose in the TSC for each non-LOCA event. The analysis utilized the RADTRAD LPZ dose model and adjusted the X/Q's and breathing rates for the TSC. Specifically, the X/Q values used for the SGTR analysis are applicable to the TSC roof and the RG 1.183 CR breathing rates are used. The licensee did not take credit for either the TSC ventilation systems or the TSC structure. Essentially, the analysis is reflective of an operator located on the roof of the TSC during a SGTR.

The NRC staff's review has found that the licensee used conservative analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The NRC staff performed independent confirmatory calculations of the dose consequences of the postulated SGTR, using the licensee's assumptions for input to the RADTRAD computer code, to ensure a thorough understanding of the licensee's methods. The NRC staff's calculations confirmed the results of the dose analysis performed by the licensee. Therefore, the NRC staff finds the TSC dose to be bounded by the AST LOCA analysis.

3.3.7.5 SGTR Accident Radiological Consequence Conclusion

The licensee evaluated the radiological consequences resulting from the postulated SGTR accident and concluded that the radiological consequences at the EAB, LPZ, and CR are within dose guidelines provided by 10 CFR 50.67 and accident specific-criteria specified by SRP Section 15.0.1. Also, based on the results of the independent confirmatory calculations, the NRC staff concluded that TSC dose is bounded by the LOCA AST analysis. The NRC staff finds that the EAB, LPZ, CR, and TSC radiological doses, estimated by the licensee for the SGTR accident, meet the applicable accident does criteria, and therefore, acceptable.

3.3.8 Loss-of Load-Event

None of the Condition II faults stated in DCPP's FSARU, Section 15.2, "Condition II – Faults of Moderate Frequency," are expected to cause breach of any of the barriers preventing fission product release from the core or plant. However, under some conditions, small amounts of radioactive isotopes could be released to the atmosphere following Condition II events as a result of atmospheric steam dumps required for plant cooldown. Condition II events that are expected to result in atmospheric steam releases are:

- Loss of electrical load and/or turbine trip
- Loss of normal feed water
- Loss of offsite power to the station auxiliaries
- Accidental depressurization of the main steam system

The amount of steam released following these events depends on the time relief valves remain open and the availability of condenser bypass cooling capacity. The mass of environmental steam release during the LOL event bounds all of the Condition II events and encompasses the LRA and CREA. The licensee has analyzed the LOL event based on the conservative assumptions in RG 1.183 for the MSLB.

3.3.8.1 Source Term

RG 1.183, Appendix E, Regulatory Position 2, states that if no or minimal fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by TS including the effects of pre-accident and concurrent iodine spiking. As stated above, the licensee's evaluation indicates that no fuel damage would occur as a result of a LOL event.

Therefore, the licensee considered the two radioiodine spiking cases described in RG 1.183 Appendix E. The first case is referred to as a pre-accident iodine spike and assumes that a reactor transient has occurred prior to the postulated LOL that has raised the primary coolant iodine concentration to the maximum value permitted by the TS for a spiking condition. For

DCPP, the maximum iodine concentration allowed by TS 3.4.16 as the result of an iodine spike is 60 µCi/gm DEI.

The second case assumes that the primary system transient associated with the LOL causes an iodine spike in the primary system. This case is referred to as a concurrent iodine spike. The increase in primary coolant iodine concentration for the concurrent iodine spike case is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the TS limit for normal operation. For DCPP the RCS TS 3.4.16 limit for normal operation is 1.0 µCi/gm of Dose Equivalent I-131. The duration of the concurrent iodine spike is assumed to be 8 hours in accordance with RG 1.183.

For the LOL event, the licensee evaluated the radiological dose contribution from the release of secondary-side activity using the equilibrium secondary-side specific activity found in DCPP TS 3.7.18 as 0.1 µCi/gm DEI. The licensee assumes that the chemical form of iodine released from the SGs to the environment is 97 percent elemental and 3 percent organic consistent with RG 1.183, Appendix E, Regulatory Position 4. The NRC staff finds the LOL source term assumptions to be consistent with RG 1.183 and, therefore, acceptable.

3.3.8.2 Release Transport

The licensee followed the guidance as described in Regulatory Position 5 of RG 1.183, Appendix E, in all aspects of the transport analysis for the LOL. For additional conservatism the licensee assumes a total primary-to-secondary leak rate equal to 0.75 gpm (1080 gpd), which is higher than the TS 3.4.13d total allowable leak rate of 150 gpd per SG. The licensee modeled the primary-to-secondary leakage for all SGs 0.75 gpm into an effective SG.

RG 1.183, Appendix E, Regulatory Position 5.2, states:

> The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of the parameter being converted. The ARC leak rate correlations are generally based on the collection of cooled liquid. Surveillance tests and facility instrumentation used to show compliance with leak rate technical specifications are typically based on cooled liquid. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).

The licensee's total primary-to-secondary leak rate of 0.75 gpm corresponds to a leakage density of 62.4 lbm/ft^3. RG 1.183, Appendix E, Regulatory Position 5.3, states:

> The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100 °C (212 °F). The release of radioactivity from unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.

In the LOL event the condenser is assumed to be unavailable due to a loss of offsite power coincident with reactor trip. The plant is cooled down by releasing steam to the environment via the MSSVs and 10 percent ADVs. In accordance with RG 1.183, the licensee assumes that steam releases continue for 10.73 hours, at which time shutdown cooling is initiated using the RHR system.

In accordance with RG 1.183, the licensee assumes that all noble gas radionuclides released from the primary system are released through the SG to the environment without reduction or mitigation. Following the guidance from RG 1.183, Appendix E, Regulatory Position 5.5, the licensee assumes that all of the primary-to-secondary leakage into the SG will mix with the secondary water without flashing during periods of total tube submergence. RG 1.183, Appendix E, Regulatory Position 5.5.4, states:

> The radioactivity in the bulk water is assumed to become vapor at a rate that is the function of the steaming rate and the partition coefficient. A partition coefficient for iodine of 100 may be assumed. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators.

Accordingly, the licensee assumes that the radioactivity in the initial bulk water of the SG becomes vapor at a rate that is a function of the steaming rate and the inverse of the partition coefficient. The licensee used a partition coefficient of 100 for iodine released from the SG. The licensee assumes that noble gases are released freely to the environment without retention in the SGs.

The licensee determined that the LOL event does not initiate any signal, which could automatically start CRVS Mode 4. Therefore, the CRVS remains in normal operation mode and the licensee does not credit CR pressurization or any other safety functions of the CRVS in the LOL analysis.

The licensee evaluated the radiological consequences resulting from the postulated LOL, and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 50.67 and accident dose criteria specified in SRP Section 15.0.1. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the NRC staff are presented in Table 3.3-12 and the licensee's calculated dose results are given in Table 3.3-1 of this SE. The NRC staff performed independent confirmatory dose evaluations, as necessary, to ensure a thorough understanding of the licensee's methods. The EAB, LPZ, and CR doses estimated by the licensee for the LOL were found to meet the applicable accident dose criteria, and are therefore, acceptable.

3.3.8.3 TSC Habitability for LOL

The DCPP onsite TSC is common to Units 1 and 2. During normal plant operation, the TSC ventilation system intake flow rate of 500 cfm is processed through a HEPA filter. Unfiltered in-leakage during normal operation and Mode 4 operation is 60 cfm, which includes 10 cfm for CR ingress and egress.

In the LAR, the licensee performed a simplified analysis of the TSC dose during a LOL event and concluded that the TSC dose is bounded by the TSC LOCA analysis. The simplified approach conservatively estimates the 30-day integrated inhalation and submersion dose in the TSC for each non-LOCA event. The analysis utilized the RADTRAD LPZ dose model and adjusted the X/Q's and breathing rates for the TSC. Specifically, the X/Q values used for the LOL analysis are applicable to the TSC roof and the RG 1.183 CR breathing rates are used.

The licensee did not take credit for either the TSC ventilation systems or the TSC structure. Essentially, the analysis is reflective of an operator located on the roof of the TSC during a LOL event.

The NRC staff's review has found that the licensee used conservative analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The NRC staff performed independent confirmatory calculations of the dose consequences of the postulated LOL event, using the licensee's assumptions for input to the RADTRAD computer code, to ensure a thorough understanding of the licensee's methods. The NRC staff's calculations confirmed the licensee's dose results, and therefore, the NRC staff finds the TSC dose to be bounded by the AST LOCA analysis.

3.3.8.4 LOL Event Radiological Consequence Conclusion

The licensee determined that LOL event does not initiate any signal that could automatically start CRVS Mode 4. Therefore, the CRVS remains in normal operation mode and the licensee does not credit CR pressurization or any other safety functions of the CRVS in the LOL analysis.

Hence, the licensee evaluated the radiological consequences resulting from the postulated LOL event and concluded that the radiological consequences at the EAB, LPZ, and CR are within dose guidelines provided by 10 CFR 50.67 and accident specific-criteria specified by SRP Section 15.0.1. Also, based on the results of the independent confirmatory calculations, the NRC staff concluded that TSC dose is bounded by the LOCA AST analysis. The NRC staff finds that the EAB, LPZ, CR, and TSC radiological doses, estimated by the licensee for the LOL event, meet the applicable accident does criteria, and therefore, acceptable.

3.3.9 NRC Staff Evaluation of the Proposed TS Changes Described in Section 3.1

As part of implementing the AST, the licensee proposed changes to the DCPP TSs. These changes are described in Section 3.1 and include the following:

- Revise the definition of Dose Equivalent I-131 in TS Section 1.1.
- Revise the noble gas activity limit in SR 3.4.16.1 (TS 3.4.16).
- Revise Note 1 for LCO 3.6.3 and SRs 3.6.3.1, 3.6.3.2, and 3.6.3.7 in (TS 3.6.3).
- Revise the allowable methyl iodide penetration testing criteria in TS 5.5.11.
- Revise the CRE habitability program in TS 5.5.19.

3.3.9.1 Revise the Definition of Dose Equivalent I-131 in TS Section 1.1

The licensee has proposed to revise the dose conversion factors (DCFs) used to calculate the radiological dose from the DEI concentration to those listed in Table 2.1, "Exposure-to-Dose Conversion Factors for Inhalation," of the EPA's FGR 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion," September 1988 (ADAMS Accession No. ML111990404); instead of Table III of TID-14844, or Table E-7 of RG 1.109, Revision 1, "Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," October 1977 (ADAMS Accession No. ML003740384) or International Commission on Radiological Protection Publication 30, Supplement to Part 1, pages 192-212, Table titled, "Committed Dose Equivalent in Target Organs of Tissues per Intake of Unit Activity," 1979.

The intent of the TS on RCS specific activity is to ensure that assumptions made in the DBA radiological consequence analyses remain bounding. As such, the DEI TS definition should have a basis consistent with the basis of the dose analyses. The licensee currently calculates DEI using thyroid DCFs, since the limiting analysis result was the thyroid dose. The AST analyses, however, determine the TEDE, rather than the whole body dose and thyroid dose as done previously. Therefore, the licensee proposes to use the inhalation committed effective dose equivalent DCFs from FGR No. 11, to calculate DEI. The NRC staff has evaluated the proposed definition of DEI and has determined that the incorporation of the committed effective dose equivalent DCFs from FGR No. 11 in the DEI definition is acceptable because this change will allow the licensee to calculate DEI using the same DCFs as are used in the dose consequence analyses.

3.3.9.2 Revise the Noble Gas Activity Limit in SR 3.4.16.1 (TS 3.4.16)

The licensee has proposed to revise the noble gas activity limit of 600 μCi/gm Dose Equivalent Xe-133 (DEX) to less than or equal to 270 μCi/gm DEX in TS 3.4.16.

The current limit of 600 μCi/gm DEX corresponds to 1 percent fuel defects. The new proposed limit of 270 μCi/gm DEX corresponds to 0.5 percent fuel defects, which is more restrictive than the current limit. In addition, the new AST radiological consequence analyses that utilize this limit has shown that the offsite and CR accident dose criteria in 10 CFR 50.67 are met, therefore the NRC staff finds the proposed change to be acceptable.

3.3.9.3 Revise Note 1 for LCO 3.6.3 and SRs 3.6.3.1, 3.6.3.2, and 3.6.3.7 (TS 3.6.3)

The licensee has proposed to include a new SR that verifies the 48-inch purge valves are sealed closed during modes 1, 2, 3, and 4, removes the 48-inch purge valves from SR 3.6.3.2, and modifies the frequency for performing SR 3.6.3.7. In the application the licensee stated that this change will eliminate a potential dose contribution due to an open containment purge pathway at the initiation of a LOCA.

Sealing closed the containment purge valves is designed to ensure that a breach of containment is not caused by an inadvertent or spurious opening of a containment purge valve and therefore, removes the likelihood of a possible radioactivity release to the environment through these pathways. The NRC staff has determined that with these requirements in place that there is reasonable assurance that the potential dose contribution from the containment purge pathways has been eliminated, and it is not necessary to analyze these pathways in the radiological consequence analysis. Therefore, the NRC staff finds the proposed change to be acceptable.

3.3.9.4 Revise the Allowable Methyl Iodide Penetration Testing Criteria in TS 5.5.11.

The licensee has proposed to revise the allowable methyl iodide penetration testing criteria for the auxiliary building ventilation system charcoal filter from 15 percent to 5 percent in TS 5.5.11, "Ventilation Filter Testing Program (VFTP)." The allowable methyl iodide penetration is used to determine charcoal filter efficiency for removing iodine from radioactive atmospheric release. The licensee credits the auxiliary building ventilation system for filtration of the RHR system pump seal failure release as discussed above in Section 3.3.2.3.3 of this SE. The NRC staff has determined that the change in the auxiliary building ventilation system charcoal filter efficiency is accurately reflected in the radiological consequence analyses and that the AST radiological consequence analyses that utilize this limit has shown that the offsite and CR room accident dose criteria in 10 CFR 50.67 are met. Therefore, the NRC staff finds the proposed change to be acceptable.

3.3.9.5 Revise the CRE Habitability Program in TS 5.5.19.

One of the requirements in TS 5.5.19, "Control Room Envelope Habitability Program," is that the program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under DBA conditions without personnel receiving radiation exposures in excess of 5 rem whole body or its equivalent to any part of the body for the duration of the accident.

The licensee has proposed to revise this requirement to the dose criteria specified in 10 CFR 50.67. Specifically, the licensee is replacing the phrase "whole body or its equivalent to any part of the body" with "TEDE." The AST analyses, determine the TEDE, rather than the whole body dose and thyroid dose as done previously. The NRC staff has evaluated this proposed change to TS 5.5.19 and has determined that the incorporation of the TEDE from 10 CFR 50.67 is acceptable.

3.3.10 Proposed License Condition

In its letter dated August 31, 2015, as modified by letter dated June 9, 2016, the licensee proposed the following license condition in support of this LAR:

> Implementation of the amendment adopting the alternative source term shall include the following plant modifications:
>
> > Install shielding material, equivalent to that provided by the Control Room outer walls, at the external concrete west wall of the Control Room briefing room.
> >
> > Install a high efficiency particulate air filter in the Technical Support Center normal ventilation system.
> >
> > Re-classify a portion of the 40-inch Containment Penetration Area (GE/GW) Ventilation line from PG&E Design Class II to PG&E Design Class I and upgrade the damper actuators, pressure switches, and the damper solenoid valves to PG&E Design Class I.
> >
> > Update setpoints for the redundant safety related gamma sensitive area radiation monitors (1-RE 25/26, 2-RE 25/26).

The AST analysis assumptions proposed by the licensee, as discussed throughout Section 3.3 of this SE, are based on the installation and completion of the above stated modifications at DCPP. In the LAR, the licensee's provided an evaluation of the DBAs radiological consequences, which reflects the installation and completion of the modifications at DCPP. Hence, the licensee's estimates of the radiological doses of the DBAs reflect these modifications. The NRC staff reviewed the proposed license condition while reviewing the AST implementation proposed by the licensee for DCPP and determined that AST analysis assumptions are conservative and consistent with RG 1.183. Therefore, the NRC staff finds the proposed license conditions to be acceptable from a radiological dose consequences perspective. Section 3.5 of this safety evaluation includes additional technical analysis for acceptability of the proposed license conditions.

3.3.11 Radiological Consequences Analysis Conclusion

The NRC staff has reviewed the AST implementation proposed by the licensee for DCPP. The NRC staff also reviewed the plant modifications associated with this proposed implementation. In performing this review, the NRC staff relied upon information submitted by the licensee in support of the LAR and, where deemed necessary, on NRC staff confirmatory calculations.

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological impacts of the proposed plant modifications in the context of the proposed AST. The NRC staff finds that the licensee used analysis methods and assumptions consistent with the conservative guidance of RG 1.183, with the exceptions discussed and accepted earlier in the SE. The NRC staff finds the methods and assumptions used by the licensee to be in compliance with applicable requirements. The NRC staff compared the doses estimated by the licensee to the applicable acceptance criteria and to the results estimated by the NRC staff in its confirmatory calculations. The NRC staff finds with reasonable assurance that the licensee's estimates of the TEDE due to DBAs will comply with the requirements of 10 CFR 50.67 and the guidance of RG 1.183.

The NRC staff finds reasonable assurance that DCPP, as modified by the LAR for AST implementation, will continue to provide sufficient safety margins with adequate defense in depth to address unanticipated events and to compensate for uncertainties. Therefore the NRC staff finds that the proposed AST implementation and the associated plant modifications are acceptable.

This licensing action is considered a full implementation of the AST. With this approval, the previous accident source term in the DCPP design basis is superseded by the AST proposed by the licensee under this LAR. The previous offsite and CR accident dose criteria expressed in terms of whole body, thyroid, and skin doses are superseded by the TEDE criteria of 10 CFR 50.67 or small fractions thereof, as defined in RG 1.183. All future radiological analyses performed to demonstrate compliance with regulatory requirements shall address all characteristics of the AST and the TEDE criteria as described in DCPP's design basis.

The following Tables (3.3-1 through 3.3-12) are references in Section 3.3 of this evaluation and support the NRC staff evaluations and conclusions.

Table 3.3-1
Total Effective Dose Equivalent per Accident in Roentgen Equivalent Man (rem)

Accident	EAB	LPZ	SRP 15.0.1 and RG 1.183 Limit	Control Room	10 CFR 50.67 and GDC 19 Limit
Loss-of-Coolant Accident	5.6	1	25	4.437[3]	5
Fuel Handling Accident in Fuel Handling Building	1.0	0.1	6.3	1.0	5
Fuel Handling Accident in Containment	1.0	0.1	6.3	4.3	5
Locked Rotor Accident	0.5	0.1	2.5	1.7	5
Control Rod Ejection Accident					
Containment Release	0.7	0.3	6.3	3.4	5
Secondary Release	0.7	0.2	6.3	0.5	5
Main Steam Line Break					
Pre-incident Iodine Spike	0.1	<0.1	25	2.0	5
Accident Initiated Iodine Spike	0.7	0.2	2.5	4.1	5
Steam Generator Tube Rupture					
Pre-incident Iodine Spike	1.3	0.1	25	0.6	5
Accident Initiated Iodine Spike	0.7	<0.1	2.5	0.3	5
Loss of Load					
Pre-incident Iodine Spike	<0.1	<0.1	2.5	<0.1	5
Accident Initiated Iodine Spike	<0.1	<0.1	2.5	<0.1	5

[3] Dose due to occupancy is 3.7 rem and dose due to direct shine is 0.7 rem, dose due to ingress/egress to control room is 0.037 rem

<table>
<tr><th colspan="6">Table 3.3-2
LOCA Total Elemental Iodine and Aerosol Removal Coefficients</th></tr>
<tr><th rowspan="2">From Time (seconds)</th><th rowspan="2">To Time (seconds)</th><th colspan="2">Elemental Iodine Removal Coefficient (hour^{-1})</th><th colspan="2">Aerosol Removal Coefficient (hour^{-1})</th></tr>
<tr><th>Sprayed Region</th><th>Unsprayed Region</th><th>Sprayed Region</th><th>Unsprayed Region</th></tr>
<tr><td>0</td><td>30</td><td rowspan="2">2.74</td><td rowspan="2">2.74</td><td>Not Available</td><td>Not Available</td></tr>
<tr><td>30</td><td>111</td><td>5.89</td><td>0.0062</td></tr>
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<tr><td>22,518</td><td>2,592,000</td><td>0.00</td><td>0.00</td></tr>
</table>

Table 3.3-3 LOCA RWST Iodine Release Fraction and Gas Venting Rate to Atmosphere			
From Time	To Time	Iodine Release Fraction to Atmosphere	Average Interval Weighted Gas Space Venting Rate to Atmosphere
Seconds	Seconds	Fraction $I_{released}/I_{entering}$[4]	Fraction V_{RWST}/ day[5]
829	7,200	9.451E-05	2.610E+00
7,200	28,800	6.357E-05	7.291E-01
28,800	86,400	8.796E-06	7.375E-02
86,400	345,600	4.560E-07	9.955E-03
345,600	471,600	6.347E-07	1.311E-02
471,600	1,011,600	8.231E-07	1.489E-02
1,011,600	2,048,400	1.114E-06	1.547E-02
2,048,400	2,592,000	1.483E-06	1.702E-02

[4] $I_{released}$ is the total iodine mass released to atmosphere during the specified time interval in grams. $I_{entering}$ is the total iodine mass entering the RWST during the specified time interval in grams

[5] Fraction V_{RWST} is the rate of fractional RWST gas volume vented during the specified time interval.

Table 3.3-4 LOCA MEDT Iodine Release Fraction and Gas Venting Rate to Atmosphere			
From Time	To Time	Iodine Release Fraction to Atmosphere	Average Interval Weighted Gas Space Venting Rate to Atmosphere
Seconds	Seconds	Fraction $I_{released}/I_{entering}$[6]	Fraction V_{MEDT}/ day[7]
829	7,200	4.521E-07	5.024E+00
7,200	28,800	1.386E-08	3.024E-01
28,800	86,400	2.362E-07	3.324E-02
86,400	183,289	3.950E-07	6.497E-03
183,289	345,600	1.236E-02[9]	Foot Note 8
345,600	752,400	2.028E-02[9]	Foot Note 8
752,400	1,530,000	2.390E-02[9]	Foot Note 8
1,530,000	2,592,000	2.166E-02[9]	Foot Note 8

Table 3.3-5 LOCA Assumptions	
Parameter	**Value**
Core power level (105% of rated power of 3411 MWt)	3580 MWt
Fuel release fractions	Per RG 1.183
Fuel release timing	Gap Onset: 30 seconds Gap Duration: 0.5 hours Early in Vessel Onset: 0.5 hours Early in Vessel Duration: 1.3 hours
Chemical form of Iodine released from fuel to containment atmosphere	4.85% Elemental 95% Particulate 0.15% Organic
Chemical form of Iodine released from RCS and sump water	97% Elemental 3% Organic
Containment Vacuum/Pressure Relief Parameters	
Minimum containment free volume	2.550E+06 ft^3
Chemical form of iodine released	97% Elemental 3% Organic
Maximum RCS flash fraction after LOCA	Noble Gases 100% Halogens 40%
Maximum Containment Pressure Relief Line Air Flow Rate	218 actual cfs
Maximum duration of release via containment pressure relief line	13 seconds
Release Point	Plant Vent
Containment Leakage Parameters	
Containment spray volume	Sprayed: 2.103E+06 ft^3 Unsprayed: 4.470E+5 ft^3

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[6] $I_{released}$ is the total iodine mass released to atmosphere during the specified time interval in grams. $I_{entering}$ is the total iodine mass entering the MEDT during the specified time interval in grams.

[7] Fraction V_{MEDT} is the rate of fractional MEDT gas volume vented during the specified time interval.

Table 3.3-5
LOCA Assumptions

Parameter	**Value**
Minimum mixing flow rate from unsprayed to sprayed region	Before CFCU actuation: 2 unsprayed regions per hour After CFCU actuation: 9.13 unsprayed regions per hour
CFCU initiation and duration	Start: 86 seconds End: 30 days
Containment spray in injection mode	Initiation time: 111 seconds Termination time: 3798 seconds
Maximum delay between end of injection spray and initiation of recirculation spray	12 minutes (manual operator action)
Containment spray in recirculation mode	Initiation time: 4518 seconds Termination time: 22,518 seconds
Long term sump water pH	≥ 7.5
Maximum allowable DF for fission product removal	Elemental Iodine: 200
Containment Leak Rate	0 to 24 hours: 0.1% weight per day 1 to 30 days: 0.05% weight per day
ESF System Leakage Parameters	
Minimum post LOCA containment water volume	480,015 gallons
Minimum time after LOCA when recirculation is initiated	829 seconds
Leakage duration	30 days
Maximum ECCS fluid temperature after initiation of recirculation	259.9 °F
Maximum ECCS leak rate (including safety factor of 2)	Unfiltered via Plant Vent: 240 cc/min Unfiltered via containment penetration areas: 12 cc/min
RHR pump seal failure	Filtered via plant vent starting at 24 hours post LOCA: 50 gpm for 30 minutes
Iodine airborne release fraction	10%
Auxiliary building ESF ventilation system filter efficiency	Elemental iodine: 88% Organic iodine: 88%
Refueling Water Storage Tank (RWST) Back Leakage Parameters	
Earliest initiation time of RWST back leakage	829 seconds
Maximum RWST inflow rate (includes safety factor of 2)	2 gpm
Miscellaneous Equipment Drain Tank (MEDT) Leakage Parameters	
Maximum MEDT inflow rate (includes safety factor of 2)	1900 cc/min

Table 3.3-5 LOCA Assumptions	
Parameter	**Value**
Control Room Emergency Ventilation	
Initiation time	Safety Injection signal generated: 6 seconds Nonaffected unit normal intake isolated: 18 seconds Affected unit normal intake isolated and mode 4 in full operation: 44.2 seconds

Table 3.3-6
FHA Assumptions

Parameter	**Value**
Core power level (105% of rated power of 3411 MWt)	3580 MWt
Number of damaged fuel assemblies	1
Total number of damaged fuel rods in the assembly	264
Decay time prior to fuel movement	72 hours
Radial peaking factor	1.65
Fraction of Core Inventory in gap	I-131: 8% I-132: 23% Kr-85: 35% Other Noble Gases: 4% Other Halides: 5% Alkali Metals: 46%
Iodine form of gap release before scrubbing	99.85% Elemental 0.15% Organic
Iodine form of gap release after scrubbing	57% Elemental 43% Organic
Water decontamination factors	Iodine: 200 Noble Gas: 1 Particulates: infinite
Release rate to the environment	2 hours or less
Environments Release Points and Rates	
FHA in the spent fuel pool in FHB release flow rate	Plant Vent: 46,000 cfm FHB out leakage: Ingress/egress: 30 cfm Miscellaneous gaps/openings: 470 cfm
Minimum free volume in FHB above spent fuel pool	317,000 ft^3
FHA in containment release point and timing	Open Equipment Hatch Within 2 hours
Minimum Free volume in containment above operating floor	2,013,000 ft^3
Control Room Emergency Ventilation	
Initiation time	Radiation monitor response time: 20 seconds Radiation monitor signal processing time: 2 seconds Control room damper closure time : 10 seconds Total Initiation time: 32 seconds

Table 3.3-7
LRA Assumptions

Parameter	Value
Core power level (105% of rated power of 3411 MWt)	3580 MWt
Reactor coolant mass	446,486 lbm
Primary-to-Secondary SG tube leakage	0.75 gpm (total for all 4 SGs) and leakage density 62.4 lbm/ft^3
Failed fuel percentage	10%
Radial peaking factor	1.65
Fraction of Core Inventory in fuel gap	I-131: 8% I-132: 23% Kr-85: 35% Other Noble Gases: 4% Other Halides: 5% Alkali Metals: 46%
Iodine form of gap release	95% Particulate 4.85% Elemental 0.15% Organic
Secondary Side Parameters	
Initial and minimum SG liquid mass	92,301 lbm per SG
Iodine species released to environment	97% Elemental 3% Organic
Steam releases	0 to 2 hours: 651,000 lbm 2 to 8 hours: 1,023,000 lbm 8 to 10.73 hours: same release rate as that for 2 to 8 hours
Iodine partition coefficient in SGs	100
Particulate carry over fraction in SG	0.0005 by weight
Fraction of noble gas released	1.0 (released without hold up)
Termination of release from SGs	10.73 hours
Environmental release point	MSSVs/10% ADV
Control Room Emergency Ventilation	
Initiation time	Does not initiate, remains in normal mode

Table 3.3-8
CREA Assumptions

Parameter	**Value**
Core power level (105% of rated power of 3411 MWt)	3580 MWt
Failed fuel percentage	10%
Percentage of core inventory in fuel gap	10% noble gases and halogens
Containment Leakage Pathway	
Containment free volume	2.550E+06 ft^3
Containment leak rate	0 to 24 hours: 0.1% volume fraction per day 1 to 30 days: 0.05% volume fraction per day
Chemical form of iodine in failed fuel	95% Particulate 4.85% Elemental 0.15% Organic
Form of iodine in the containment atmosphere	97% Elemental 3% Organic
Termination of containment release	30 days
Secondary Side Pathway	
Reactor coolant mass	446,486 lbm
Primary-to-Secondary leak rate	0.75 gpm (total for all 4 SGs) and leakage density 62.4 lbm/ft^3
Minimum post-accident SG liquid mass	92,301 lbm per SG
Iodine species released to environment	97% Elemental 3% Organic
Steam releases	0 to 2 hours: 651,000 lbm 2 to 8 hours: 1,023,000 lbm 8 to 10.73 hours: same release rate as that for 2 to 8 hours
Iodine partition coefficient in SGs	100
Fraction of noble gas released	1.0 (released without hold up)
Termination of release from SGs	10.73 hours
Environmental release point	MSSVs/10% ADV
Control Room Emergency Ventilation	
Initiation time	Safety Injection Signal generated: 300 seconds Nonaffected unit normal intake isolated: 312 seconds Affected unit normal intake isolated and mode 4 in full operation: 338.2 seconds

Table 3.3-9 MSLB Assumptions	
Parameter	**Value**
Core power level (105% of rated power of 3411 MWt)	3580 MWt
Reactor coolant mass	446,486 lbm
Leak rate to faulted SG	0.75 gpm (total for all 4 SGs) and leakage density 62.4 lbm/ft^3
Leak rate to intact SGs	0 gpm
Failed fuel percentage	0% or None
RCS TS 3.4.16 iodine concentration	1.0 micro curies per gram (DEI)
RCS TS 3.4.16 noble gas concentration	270.0 micro curies per gram (DEX)
RCS equilibrium iodine appearance rates	Rate associated with an RCS iodine concentration of 1.0 micro curies per gram (DEI)
Pre-accident iodine spike concentration	60.0 micro curies per gram (DEI)
Accident initiated iodine spike appearance rate	500 times equilibrium appearance rate
Duration of accident initiated iodine spike	8 hours
Initial secondary coolant iodine concentration	0.1 micro curies per gram (DEI)
Secondary System Release Parameters	
Iodine species released to environment	97% Elemental 3% Organic
Fraction of iodine released from faulted SG	1.0 (released without hold up)
Fraction of Noble Gas released from faulted SG	1.0 (released without hold up)
Liquid mass in each SG	Faulted: 182,544 lbm maximum Intact: 92,301 lbm minimum and initial
Release rate of SG liquid activity from faulted SG	Dry out within 10 seconds
Steam releases from intact SGs	0 to 2 hours: 384,000 lbm 2 to 8 hours: 893,000 lbm 8 to 10.73 hours: same release rate as that for 2 to 8 hours
Iodine Partition Coefficient in intact SGs	100
Termination of release: faulted SG	30 hours when RCS reaches 212 °F
Termination of release: intact SGs	10.73 hours
Environmental release point: faulted SGs	Outside containment at the steam line break location
Environmental release point: Intact SGs	MSSVs/10% ADV
Control Room Emergency Ventilation	
Initiation time	Safety Injection Signal generated Nonaffected unit normal intake isolated: 12.6 seconds Affected unit normal intake isolated and mode 4 in full operation: 38.8 seconds

Table 3.3-10
SGTR Assumptions

Parameter	**Value**
Core power level (105% of rated power of 3411 MWt)	3580 MWt
Reactor coolant mass	446,486 lbm
Time of reactor trip	179.0 seconds
Time of isolation of stuck open 10% ADV on the ruptured SG	2653 seconds
Termination of break flow from ruptured SG that flashes	3402 seconds
Termination of break flow from ruptured SG	5872 seconds
Time of manual depressurization of the ruptured SG	2 hours
Tube leakage rate to intact SGs	0.75 gpm (total for all 4 SGs) and leakage density 62.4 lbm/ft^3
RCS TS 3.4.16 iodine concentration	1.0 micro curies per gram (DEI)
RCS TS 3.4.16 noble gas concentration	270.0 micro curies per gram (DEX)
RCS equilibrium iodine appearance rates	Rate associated with an RCS iodine concentration of 1.0 micro curies per gram (DEI)
Pre-accident iodine spike concentration	60.0 micro curies per gram (DEI)
Accident initiated iodine spike appearance rate	335 times equilibrium appearance rate
Duration of accident initiated iodine spike	8 hours
Initial secondary coolant iodine concentration	0.1 micro curies per gram (DEI)
Secondary System Release Parameters	
Initial SG Liquid mass	89,707 lbm per SG
Iodine species released to environment	97% Elemental 3% Organic
Steam flow rate to condenser from ruptured SG before trip	63,000 lbm per minute
Steam flow rate to condenser from intact SGs before trip	189,000 lbm per minute
Partition factor in main condenser	0.01 Elemental iodine 1.0 Organic iodine and Noble Gases
Post-accident minimum SG liquid mass for ruptured SG	89,707 lbm
Post-accident minimum SG liquid mass for intact SGs	89,707 lbm per SG
Fraction of iodine released (flashed portion)	1.0 (released without hold up)
Fraction of noble gas released from all SG	1.0 (released without hold up)
Iodine Partition Coefficient	100

Table 3.3-10 SGTR Assumptions	
Parameter	**Value**
Back flow and steam releases from intact and faulted SGs	Table 3.3-11
Environmental release point: faulted SGs	Plant Vent: 0 to 179 seconds MSSVs/10% ADVs: 179 seconds to 10.73 hours
Control Room Emergency Ventilation	
Initiation time	Safety Injection Signal generated: 219 seconds Nonaffected unit normal intake isolated: 231 seconds Affected unit normal intake isolated and mode 4 in full operation: 257.2 seconds

Table 3.3-11 Steam Generator Tube Rupture Break Flows and Steam Releases				
	Break Flow and Steam Release within each Time Interval			
Time from Break (sec)	**Flashed Break Flow (lbm)**	**Un-flashed Break Flow (lbm)**	**Ruptured SG Steam Releases (lbm)**	**Intact SGs Steam Releases (lbm)**
0	1678	8422	187822	563100
179	2217	30003	10527	42565
853	12121	90754	113657	118
2653	1355	15906	0	146
2953	779	23177	0	85467
3402	0	45026	0	97164
4324	0	16870	0	9237
4739	0	23892	0	29103
5872	0	0	0	103300
7200	0	0	270000	1,342,400
38628	0	0	0	0

Table 3.3-12 LOL Assumptions	
Parameter	**Value**
Core power level (105% of rated power of 3411 MWt)	3580 MWt
Reactor coolant mass	446,486 lbm
Primary-to-Secondary SG tube leakage	0.75 gpm (total for all 4 SGs) and leakage density 62.4 lbm/.ft^3
RCS TS 3.4.16 iodine concentration	1.0 micro curies per gram (DEI)
RCS TS 3.4.16 noble gas concentration	270.0 micro curies per gram (DEX)

Table 3.3-12 LOL Assumptions	
Parameter	**Value**
RCS equilibrium iodine appearance rates	Rate associated with an RCS iodine concentration of 1.0 micro curies per gram (DEI)
Pre-accident iodine spike concentration	60.0 micro curies per gram (DEI)
Accident initiated iodine spike appearance rate	500 times equilibrium appearance rate
Duration of accident initiated iodine spike	8 hours
Initial secondary coolant iodine concentration	0.1 micro curies per gram (DEI)
Initial and minimum SG Liquid mass	92,301 lbm per SG
Steam releases	0 to 2 hours: 651,000 lbm 2 to 8 hours: 1,023,000 lbm 8 to 10.73 hours: same release rate as that for 2 to 8 hours
Iodine partition coefficient in intact SGs	100
Iodine species released to environment	97% Elemental 3% Organic
Fraction of noble gas released	1.0 (released without hold up)
Termination of release from SGs	10.73 hours
Environmental release point	MSSVs/10% ADV
Control Room Emergency Ventilation	
Initiation time	Does not initiate, remains in normal mode

3.4 Post-Accident Containment Sump pH Evaluation – Potential for Re-Evolution of Iodine

3.4.1 Background

Implementation of the AST by the licensee required reanalyzing several DBAs using new source terms. The licensee performed these tasks by following the requirements of 10 CFR 50.67. It also applied for a license amendment under 10 CFR 50.90. An acceptable accident source term is a permissible amount of radioactive material that could be released to the containment from the damaged core following an accident. As a result of improved understanding of the mechanisms of the release of radioactivity, 10 CFR 50.67 permits licensees to voluntarily replace their current TID 14844 accident source term with the AST. However, this replacement is subject to performing a successful reevaluation of the major DBAs. The guidance for implementation of an AST is provided in RG 1.183. According to RG 1.183, maintaining a pH basic will minimize re-evolution of iodine from the suppression pool water.

3.4.2 Evaluation

According to NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," February 1995 (ADAMS Accession No. ML041040063), iodine released from the damaged core to the containment after a LOCA is composed of 95 percent CsI, which is a highly ionized salt soluble in water. Iodine in this form does not present any radiological problems since it remains dissolved in the sump water and does not enter the containment atmosphere. However, in the radiation field existing in the containment, some of this iodine could be transformed from the ionic to the elemental form, which is scarcely soluble in water and can be, therefore, released to the containment atmosphere. Conversion of iodine to the elemental form depends on several parameters, of which pH is very important. Maintaining pH basic in the sump water will ensure that this conversion will be minimized. The pH of the sump water at DCPP is controlled by the addition of sodium hydroxide (NaOH) from the spray additive tank (SAT) to the boric acid (H_3BO_3) dissolved in the sump water after a LOCA. The calculation takes into account H_3BO_3 from the RCS, SI accumulators, and RWST. The NaOH is made available during containment spray (CS) injection mode post-accident. At DCPP, CS in the injection mode is exhausted within approximately an hour after accident initiation, or earlier if full safeguards are available. The CS is switched to recirculation mode within minutes of termination of injection spray. After a LOCA, several acids are generated in the containment. Relative amounts of these acids and that of NaOH determine the pH reached by the containment sump water.

The licensee used available experimental data with respect to H_3BO_3 and NaOH concentrations in a solution and the resultant pH to determine the ultimate sump pH at DCPP. The experimental data and information from a literature search were used to develop a graph of NaOH versus H_3BO_3 concentrations, and constant pH curves were drawn through the data. The analysis assumes minimum volume and concentration values for NaOH, in combination with maximum volume and concentration values for H_3BO_3 for conservatism. The parameters the licensee used in the sump pH analysis and described in Table 1 of the letter dated October 22, 2015, are listed in Table 3.4-1 below.

Table 3.4-1: Design Parameters Used in Minimum Sump pH Analyses

Component	Volume	Concentration	Weight Percent
	cubic feet (ft^3)	parts per million (ppm)	-
RWST	61,266	2500 Boron	-
RCS	11,455	1900 Boron	-
SI Accumulators	3,544	2500 Boron	-
SAT	207	-	30 NaOH

Using the parameters in Table 3.4-1, the licensee determined the boron and NaOH concentrations for the sump pH analysis to be approximately 0.2162 and 0.0267 gram-moles per liter (g-mol/L), respectively. These concentrations are based on temperature and pressure assumptions at the beginning of the accident scenario. In addition, the sump volume was determined to be 76,456 cubic feet (ft^3) (2.165×10^6 liters (L)) and is reflective of containment conditions at the end of the CS recirculation phase.

The licensee considered the effects of strong acid generation on the post-LOCA sump pH. Per guidance NUREG/CR-5950, "Iodine Evolution and pH Control," December 1992 (ADAMS Accession No. ML063460464), the licensee calculated the mass of the hydrochloric acid (HCl) and nitric acid (HNO_3). HCl is formed from decomposition of chlorinated polymer cable insulation by radiation and nitric acid is generated by the radiolysis of air and water inside containment.

To establish the cable inventory inside containment, the licensee used an upper bound approach by taking into consideration: (a) cable insulation data provided in NUREG/CR-5950, specifically, the amounts of ethylene propylene rubber/Hypalon cable from PWRs listed in Table 2.2 of the guidance document; and (b) the mass/type of cable installed in containments constructed by Stone and Webster. The licensee applied a safety factor of 1.5 to the largest mass of electrical cable identified (was determined to be a 4-loop PWR with a power level slightly greater than DCPP) to estimate an upper bound value for the electrical cable installed inside the DCPP containments. The electrical cable mass was determined to be approximately 123,090 lbm (5.5832×10^4 kilogram). The licensee conservatively applied the estimated electrical cable mass of both insulation and conductors, as applicable to insulation available in the DCPP containment despite the fact that only cable insulation contributes to acid generation.

The licensee considered gamma radiation exposure information from a 4-loop PWR to develop the source term for HNO_3 generated in the DCPP containment sump. The l4-loop PWR radiation exposure of 40×10^6 rad (40×10^4 Gray (Gy)) was corrected for differences in sump volume and power level, and used as the source term for determining the concentration of HNO_3 in the DCPP containment sump. Similar to the calculation used to determine the containment insulation, the licensee used a safety factor of 1.5 in its calculation for determining the quantity of HNO_3 generated.

In accordance with guidance identified in NUREG/CR-5175, "Beta and Gamma Dose Calculations for PWR and BWR [Boiling Water Reactor] Containments," the licensee conservatively assumed the airborne LOCA radiation dose to be approximately 2×10^8 rad (2×10^6 Gy) as an upper bound value for the beta dose inside containment for PWRs, while gamma dose estimate was a decade lower. Utilizing the information derived from the stated NUREG/CRs, the licensee determined the total amount of HCl and HNO_3 as shown in Table 3.4-2 below (information derived from Table 2 of letter dated October 22, 2015).

Table 3.4-2: Net Acid Addition at 30 Days

	Acid (g-mol)	Concentration (g-mol/L)
HCl	11,324.3	0.00523
HNO_3	1,997.4	0.00092
Total	13,321.7	0.00615

Although the licensee did not provide the pH values at various times of interest (i.e., time-dependent pH values), it did provide the concentrations for boron, NaOH and other elements consistent with the beginning and at the end of 30 days, post-LOCA. The licensee stated that as a result of recirculation, the sump water will be well mixed by the time interval of 0 – 16 hours post-accident. After acid addition, the licensee calculated the net free caustic available ($C_{NaOH\ free}$). By subtracting the g-mols of strong acid generated from the g-mols of NaOH, the net free NaOH available was estimated to be 44,514 g-mol or equal to $C_{NaOH\ free}$ of 0.0206 g-mol/L (as shown in Table 3 of letter dated October 22, 2015).

Table 3.4-3: Sump Composition at 30 Days

	Concentration (g-mol/L)
C_{Boron}	0.2162
$C_{NaOH\ free}$	0.0206
C_{Cl}	0.00523
C_{NO3}	0.00092
pH	>7.5

In order to neutralize the H_3BO_3, HCl, and HNO_3, the licensee chose to buffer the sump pool water by using a NaOH buffer. Such buffering action is intended to maintain basic pH in the sump pool despite the presence of the acids. By completing the containment spray chemical injection mode and switching to the recirculation mode within the first 16 hours post-accident, the sump pH is expected to remain above 7 before vapor phase elemental iodine can occur. The licensee has calculated that by adding approximately 0.0267 g-mol/L of NaOH from the SAT, the pH in the sump water will remain basic for 30 days. Per the analysis, the licensee determined that the sump pH post-accident will be greater than 7.5 at the end of 30 days, using conservative parameters.

The NRC staff has reviewed the information the licensee provided and determined that by using NaOH as the buffer in the quantity specified, the pH of the sump will remain above 7 for 30 days post-LOCA. The NRC staff has verified that the assumptions and methodology applied by the licensee are consistent with parameters and methodologies other similarly operating nuclear power plants have used to request implementation of the AST and received approval by the NRC.

3.4.3 Conclusion

The NRC staff reviewed the licensee's assumptions, methodology, and conclusions regarding the pH of sump water and the corresponding fraction of the dissolved iodine in the sump water that is converted into the elemental form. The methodology relies on using buffering actions of NaOH. Based on the NRC staff review, the NRC staff determined that the assumptions are appropriate and consistent with the methods accepted by the NRC staff for the calculation of post-accident containment sump pH. The calculations were made for the 30 day period following a LOCA. The NRC staff also verified that the post-accident containment sump pH will be maintained above 7.0 for 30 days following a LOCA.

3.5 Safety-Related Electrical Systems and Environment Qualification of Electrical Components

3.5.1 Background

GDC 17 defines the requirements to provide onsite and offsite electric power systems to permit functioning of structures, systems, and components important to safety. Section 50.49 of 10 CFR requires that the electrical equipment important to safety, which are relied upon to remain functional during and following design-basis events be qualified for accident (harsh) environment. Section 50.67 of 10 CFR provides a provision for licensees to revise the AST used in design basis radiological analyses.

RG 1.183 provides guidance to licensees of operating power reactors on acceptable applications of ASTs; the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on analyzed risk; and content of submittals. RG 1.183 provides guidance on an acceptable AST and identifies the significant attributes of other ASTs that may be found acceptable by the NRC staff. This RG also identifies acceptable radiological analysis assumptions for use in conjunction with the accepted AST. RG 1.183 states that the licensees may use the AST or the TID-14844 assumptions for performing the required equipment qualification (EQ) analyses to show that the equipment remains bounding. RG 1.183 further states that no plant modifications are required to address the impact of the difference in source term characteristics (i.e., AST versus TID-14844) on EQ doses.

3.5.2 Evaluation

The licensee proposed the implementation of the AST methodology for radiological dose consequence analyses to calculate the offsite and onsite radiological consequences for DCPP. DCPP previously received NRC approval for selective implementation of an AST radiological dose calculation for the FHA in the FHB. In accordance with current licensing basis (CLB), an FHA is assumed to occur in the spent fuel pool located in the FHB, or in the Containment. On page 1 of the Enclosure to letter dated June 17, 2015, the licensee stated that they performed new calculations to support full implementation of an AST for each of the applicable DBAs addressed by RG 1.183. The following DBAs are addressed: LOCA, FHA in the Containment, FHA in the FHB, LRA, CREA, MSLB, SGTR, and LOL event. Acceptable criteria consistent with those required by 10 CFR 50.67 and RG 1.183 were used to replace the current design basis source term acceptance criteria.

On page 2 of the Enclosure to letter dated June 17, 2015, the licensee stated that the full implementation of AST for DCPP does not include revising the source term used for EQ of safety-related equipment associated with shielding and vital area access. The licensee further stated that, the licensee does not propose to modify the EQ result basis nor the shielding and vital area access dose rate to adopt AST, since the results of the analyses based on TID-14844 would be more limiting for a period up to 1 to 4 months after which the AST results would be more limiting. Based on the above, the NRC staff evaluated the impact to the safety-related electrical systems and the EQ of electrical equipment due to the full implementation of the AST, and issued an RAI on October 7, 2015 (ADAMS Accession No. ML15280A443), to determine the acceptability of the proposed implementation of the AST methodology for radiological dose consequence analyses.

The NRC staff requested the licensee to address whether any nonsafety-related systems and components are credited in the AST analyses and describe their impact on the safety-related electrical systems. In its letter dated November 6, 2015, the licensee stated that a portion of the 40-inch containment penetration area (GE/GW) ventilation line and associated damper actuators, pressure switches, and the damper solenoid valves and a 2-inch gaseous radwaste system-line (originally designed as PG&E Design Class I safety-related, but currently classified as nonsafety-related PG&E Design Class II) that connect to the Plant Vent (safety-related) will have to go through design modifications due to the implementation of the AST. In its letter dated June 17, 2015 (Attachment 7), the licensee made a regulatory commitment to reclassify these lines to PG&E Design Class I and upgrade the isolating damper solenoid valves, the associated damper actuators, and the pressure switches to PG&E Design Class I prior to implementation of the AST. In its letter dated August 31, 2015, the licensee removed the commitments and changed the commitments to license conditions.

In its letter dated June 9, 2016, the licensee stated that it performed an assessment and concluded that there is no need to upgrade the design classification of the 2-inch gaseous radwaste system line that connects to the plant vent to PG&E Design Class I. The licensee stated that classifying the piping as PG&E Design Class I would not have resulted in any physical changes to the piping and that the dose impact in the CR of a potential break location at the interface of the 2-inch gaseous radwaste system line with the plant vent is bounded by the current dose consequence analysis. Therefore the licensee updated Section 2.4 of the LAR to delete the reclassification of the 2-inch gaseous radwaste system line as PG&E Design Class I. The NRC staff accepts this change as the connecting radwaste system line with the plant vent is bounded by the current dose analysis. However, since the licensee is crediting a portion of the 40-inch containment penetration area (GE/GW) ventilation line and associated damper actuators, pressure switches, and the damper solenoid valves in the AST analysis, the licensee provided a license condition to re-classify these components to PG&E Design Class I before implementation of AST. The proposed re-classification of these components is required for crediting the non-safety components in the AST analysis and, therefore, the proposed license condition is acceptable to the NRC staff.

In the RAI responses dated November 6, 2015, the licensee also stated that credit is taken for pressure boundary integrity of the containment pressure/vacuum relief system ductwork, which is seismically qualified. The NRC staff reviewed Section 2.4 of the LAR, "Plant Changes," and the responses to the NRC staff RAIs and finds that with the exception of the 2-inch gaseous radwaste system line that connects to the plant vent, no nonsafety-related systems and components are credited in the AST analysis. As such the NRC staff determined that the licensee proposed implementation of the AST methodology for radiological dose consequence analyses does not add the possibility of a fault between the nonsafety-related and the safety-related systems. Therefore the independence (electrical and physical) of the nonsafety-related systems from the safety-related systems will be maintained, consistent with the requirements of GDC 17.

On page 19 of the Enclosure to letter dated June 17, 2015, the licensee stated, in part, that "there are no additional or new emergency diesel generator (EDG) loads and the timing of the EDG loads did not change as a result of the AST." Furthermore on page 7 the licensee stated, in part, that for the CS system "no changes in operation are being proposed, other than requiring its operation within 12 minutes following terminating injection spray, instead of being optional in accordance with EOPs [emergency operating procedures] or at the discretion of the TSC." In the RAI dated October 7, 2015 (ADAMS Accession No. ML15280A443), the NRC staff requested the licensee to address whether this will impact the EDG loading sequence. In its letter dated November 6, 2015, the licensee stated that the change in operation of the CS does not impact EDG loading sequence. The change affects the recirculation phase, where motive force is provided by an RHR pump. The NRC staff notes that the RHR pumps are manually restarted in the recirculation phase and are not part of the EDG loading and as such, the change in operation of the CS does not impact the EDG loading sequence. Therefore the NRC staff determined that the proposed full implementation of an AST will not affect the capability of the EDG.

Based on the above, the NRC staff concludes that there is no impact on safety-related electrical systems as a result of full implementation of the AST.

In the RAI dated October 7, 2015, the NRC staff also requested the licensee to address if there are any changes to the EQ profile on temperature and pressure due to the implementation of the AST. In its letter dated November 6, 2015, the licensee stated, in part, that the "containment integrity analysis performed by Westinghouse in support of implementation of AST at DCPP determined that there has been no impact on the EQ envelope post-LOCA peak pressure and temperature inside containment. The containment analysis also concluded that the use of containment spray in the recirculation mode for the minimum safeguards case... has a minimal effect on the current long term pressure and temperature envelopes used for equipment qualification." On page 19 of the Enclosure to letter dated June 17, 2015, the licensee stated, in part, that the containment spray system will now be credited during sump water recirculation following a LOCA for dose mitigation, but DCPP is already licensed for recirculation containment spray operation. Therefore, there are no additions to the EQ list..." Based on its review of the information provided by the licensee, the NRC staff confirmed that there will be no impact on the EQ envelope post-LOCA peak pressure and temperature inside containment, and no additions will be necessary to the EQ list due to the implementation of the AST. The NRC staff determined that the safety-related electrical equipment, which is relied upon to remain functional during and following design-basis events will continue to be qualified for accident (harsh) environment as required in 10 CFR 50.49.

The NRC staff also reviewed the EQ portion of the LAR. On page 2 of the Enclosure to letter dated June 17, 2015, the licensee stated that DCPP will not revise the source terms used for EQ of safety-related equipment nor the shielding and vital area access dose rates in order to adopt AST. In its RAI, the NRC staff requested the licensee if the source terms and the shielding and vital area access dose rates will continue to be based on assumptions in TID-14844. In its letter dated November 6, 2015, the licensee confirmed that the radiation source terms used for the EQ of safety-related equipment and the shielding and vital area access dose rates will continue to be based on assumptions in TID-14844, and will remain unchanged. As stated in RG 1.183, Regulatory Position 6, "Assumptions for Evaluating the Radiation Doses for Equipment Qualification," the licensee may use either the AST or the TID-14844 assumptions for performing the required EQ analyses until such time as a generic issue related to the effect of increased cesium releases on EQ doses is resolved. This generic issue has been resolved (dropped) in a NRC staff Memorandum dated April 30, 2001 (ADAMS Accession No. ML011210348) and in Supplement 25 to NUREG-0933, "A Prioritization of Generic Safety Issues," Generic Issue 187, "The Potential Impact of Postulated Cesium Concentration on Equipment Qualification in the Containment Sump in Nuclear Power Plants," June 2001 (ADAMS Accession No. ML012190402). The NRC staff concluded in the memorandum and NUREG-0933 that there was no clear basis for backfitting the requirement to modify the design basis for EQ to adopt the AST and there would be no discernable risk reduction associated with such a requirement. Based on the above, the NRC staff determined that it is acceptable for the

TID-14844 based assumptions to remain the licensing basis for equipment EQ analyses for DCPP and will remain unaffected for the implementation of the AST.

As an added assurance, in its letter dated November 6, 2015, the licensee also made a regulatory commitment that existing environmentally qualified components will be evaluated per the EQ program to confirm acceptability as part of AST implementation. Also, as required by 10 CFR 50.49, , the list of the components associated with the 10 CFR 50.49 EQ Program will be updated after completing the evaluation per the EQ Program, as part of the AST implementation.

3.5.3 Conclusion

The NRC staff has evaluated the information provided by the licensee related to the proposed implementation of the AST methodology for radiological dose consequence analyses for DCPP. Based on its review, the NRC staff determined that, with the exception of the 2-inch gaseous radwaste system line and the 40-inch containment penetration area ventilation line, no nonsafety-related systems and components are credited in the AST analysis. The licensee has provided a license condition to reclassify a portion of the 40-inch containment penetration area ventilation line from PG&E Class II to Class I and upgrade the damper actuators and, pressure switches, and damper solenoid valves to PG&E Class I. By letter dated June 9, 2016, the licensee confirmed that based on its assessment there is no need to upgrade the design classification of the 2-inch gaseous radwaste system line that connects to the plant vent to PG&E Design Class I. As such the NRC staff determined that independence (electrical and physical) of the nonsafety-related systems from the safety-related systems will be maintained consistent with the requirements of Criterion 17. The NRC staff also reviewed the EQ portion of the LAR and determined that the safety-related electrical equipment, which are relied upon to remain functional during and following design basis events, will continue to be qualified for accident (harsh) environments as required in 10 CFR 50.49. The NRC staff notes that as stated in RG 1.183, the licensee may use either the AST or the TID-14844 assumptions for performing the required EQ, and it is acceptable for the TID-14844 based assumptions to remain the licensing basis for equipment EQ analyses for DCPP and will remain unaffected for the implementation of the AST. Therefore, the NRC staff concludes that the proposed changes are acceptable.

3.6 Containment and Ventilation Systems and Leak Test Program Evaluation

3.6.1 Background

Appendix J to 10 CFR 50, provides containment leakage test requirements to ensure that (a) leakage through containments or systems and components, penetrating containments, does not exceed allowable leakage rates specified in the TS; and (b) integrity of the containment structure is maintained during the service life of the containment. RG 1.52 provides information on acceptable maximum allowable methyl iodide penetration and filter efficiency for the CR Emergency Ventilation charcoal adsorber.

3.6.2 Evaluation

The adoption of the AST would allow the licensee to update its accident analyses with new dose calculations associated with the accident offsite and CR dose consequences. The scope of review for this section of the SE are changes to the TSs related to containment isolation valves and the ventilation filter testing program. Additionally, the licensee made changes to the SG leakage rate and filter and procedure changes in support.

TS Limiting Conditions for Operation (LCO) 3.6.3, "Containment Isolation Valves," Note 1 will be revised to read as follows:

> Penetration flow path(s) except for 48-inch purge valve flow paths, may be unisolated intermittently under administrative controls.

On page 8 of the Enclosure to its letter dated June 17, 2015, the licensee stated that the 48-inch containment purge valves are to be sealed closed by removing motive power during Modes 1, 2, 3, and 4. The purpose of this change is to eliminate a potential dose contribution due to an open containment purge pathway at the initiation of a LOCA. The proposed revision is also consistent with NUREG-1431, Volume 1, Revision 4.0, "Standard Technical Specifications Westinghouse Plants," April 2012 (ADAMS Accession No. ML12100A222).

SR 3.6.3.1, which is currently not used, has been revised to state:

> Verify each 48 inch purge valve is sealed closed, except for one purge valve in a penetration flow path while in Condition D of this LCO.

In the existing TSs, SR 3.6.3.2 addresses the SRs for the 48-inch containment purge supply/exhaust and 12-inch vacuum/pressure relief valves. The proposed SR 3.6.3.1 addresses the SRs for the 48-inch containment purge supply/exhaust valve and SR 3.6.3.2 addresses SRs for 12-inch vacuum/pressure relief valve. The change is consistent with NUREG-1431 and does not need additional evaluation because the SRs for these valves have not changed.

SR 3.6.3.1 will have a FREQUENCY in accordance with the Surveillance Frequency Control Program (SFCP). The LAR did not specify the initial frequency for performing this SR. In response to the NRC staff RAI dated January 11, 2016 (ADAMS Accession No. ML16011A317), in its letter dated February 10, 2016, the licensee stated that the initial frequency will be determined prior to implementation. The licensee also stated that the initial frequency will be selected in accordance with its SFCP and will be evaluated by a licensee expert panel to consider both the quantitative and qualitative factors before it is approved by the Plant Staff Review Committee. The valves are being sealed closed because they may be unable to close in the environment following a LOCA in sufficient time to support the DBA acceptance criteria. The NRC staff finds this approach acceptable.

In SR 3.6.3.7, the licensee proposed to remove the surveillance within 92 days after opening the containment purge supply and exhaust lines during the performance of leakage rate testing for containment purge supply and exhaust and vacuum/pressure relief valves with resilient seals. In a letter dated February 10, 2016, the licensee stated that the change is requested because the leakage testing described is required following the opening of the 48-inch containment supply and exhaust valves in Modes 1, 2, 3, and 4, which are to be sealed closed. The NRC staff finds this change acceptable because there will be no condition where it would require the testing since the valves are sealed closed.

The proposed change in TS 5.5.11, "Ventilation Filter Testing Program (VFTP)," revises the auxiliary building ventilation system charcoal filter maximum allowable methyl iodide penetration testing criteria from 15 percent to 5 percent. In its LAR, the licensee states, in part, that the "allowable methyl iodide penetration is used to determine charcoal filter efficiency for removing iodine from atmospheric releases." The dose analyses prompted the change of the penetration testing criteria. The results of the review of the radiological consequences analyses are discussed in Section 3.3 of this SE. Based on the results of the review in Section 3.3, and the fact that the proposed change is conservative, the change in the auxiliary building ventilation system charcoal filter maximum allowable methyl iodide penetration testing criteria is acceptable and is in accordance with the applicable regulatory requirements.

3.6.3 Changes to Current Licensing Basis

The licensee described the licensing bases changes incorporated into the revised dose analyses in Section 2.1 of the Enclosure to letter dated June 17, 2015, as part of the AST implementation. This section addresses the following licensing basis changes:

- Installation of new back-draft damper in the CR emergency filter recirculation lines
- Use of CS in the recirculation mode following a LOCA for fission product cleanup

Back-draft dampers in the CR emergency filter recirculation lines were installed to prevent reverse unfiltered flow into the CR. In an RAI dated January 11, 2016, the NRC staff requested the licensee to address the impact the back-draft dampers have on the CR ventilation analyses assumptions or the unfiltered in-leakage testing methods/results, assuming a 100 cfm damper leakage. In the letter dated February 10, 2016, the licensee stated that the back-draft dampers were installed prior to the 2012 DCPP CR Ventilation System Tracer Gas Test and that the assumed unfiltered in-leakage conservatively encompasses the tracer gas test results. The NRC staff finds that the impact of the installation of the back-draft dampers has been adequately evaluated because the assumed in-leakage conservatively bounds the test results.

The revised dose analyses credits operation of the CS in recirculation mode following a LOCA for fission product cleanup. In the RAI dated January 11, 2016, the NRC staff requested the licensee to address the impact of this change, if any, on the bounding containment analyses.

In its letter dated February 10, 2016, the licensee states, in part;

> At DCPP, containment spray in the injection mode is exhausted within approximately one hour after accident initiation, or earlier if full safeguards are available. Thus in order for the containment spray to continue to be effective as a fission product removal mechanism, the sprays have to be made available beyond the injection mode and continue in the recirculation mode.
>
> PG&E has performed time and motion studies to estimate the delay time between termination of injection spray and the operator's manual initiation of recirculation spray for a double ended pump suction break with failure of one train of the solid-state protection system. The results of the referenced time and motion studies have indicated that the 12-minute delay assumed in the LOCA dose consequence analysis bounds, with significant margin, the recorded time delay experienced to initiate recirculation spray.

The licensee further confirmed that there is no effect on the existing peak pressure and temperature inside of containment with this change and that there is minimal effect on the long-term pressure and temperature envelopes. The licensee states that the limiting DCPP licensing basis containment integrity analyses remain unchanged and conservatively bound operation of

CS in the recirculation mode. Based on its review, the NRC staff concludes that the change is acceptable because the licensing basis is not impacted.

3.6.4 Conclusion

The NRC staff determined that since there have been no changes to the leak testing program, the proposed change meets the requirements of 10 CFR Part 50, Appendix J. The proposed change meets the guidance described in RG 1.52 because the change to the auxiliary building ventilation system charcoal filter maximum allowable methyl iodide penetration testing is conservative.

The NRC staff concludes that as described above, the licensee has adequately addressed the proposed changes to TSs 3.6.3 and 5.5.11. The licensee has also adequately addressed the licensing basis changes due to installation of back-draft damper in the CR emergency filter recirculation lines and crediting the operation of the CS in recirculation mode following a LOCA.

Based on the above, the NRC staff concludes that the proposed TS changes are acceptable.

3.7 <u>Reactor Systems – Proposed Gap Fraction Evaluation</u>

3.7.1 Background

Section 50.34 of 10 CFR defines the content requirements for the FSARU, including evaluations required to show that accident dose criteria are met. Section 50.36 of 10 CFR provides requirements for inclusion of LCOs in technical specifications. Section 50.67 of 10 CFR requires the application to contain an evaluation of the consequences of applicable DBAs previously analyzed in the safety analysis report.

RG 1.183 provides guidance to licensees of operating power reactors on acceptable applications of ASTs; the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on analyzed risk; and content of submittals. RG 1.183 Appendices A, B, E, F, G, and H provide licensee guidance for the evaluating the radiological consequences of PWR accidents of concern for AST. As specifically cited by RG 1.183, Section 15.0.1 of the SRP applies for the assessment of the AST. This SRP section provides, in part, guidance to the NRC staff for the review of the models, assumptions, and parameter inputs used by the licensee for the calculation of the AST radiological consequences. Regulatory Position C.1.3.2, "Re-Analysis Guidance," specified in RG 1.183 states, in part, that "[a]n analysis is considered to be affected if the proposed modification changes one or more assumptions or inputs used in that analysis such that the results, or the conclusions drawn on those results, are no longer valid." Additionally, Regulatory Position C.5.1.3, "Assignment of Numeric Input Values," in RG 1.183 states, in part, that the "numeric values that are chosen as inputs to the analyses...should be selected with the objective of determining a conservative postulated dose."

As discussed in Chapter 15 of the SRP, in order to establish a licensing basis, licensees must analyze transients and accidents in accordance with the requirements of 10 CFR 50.34, 10 CFR 50.46, and where applicable, per NUREG-0737. These accidents and transients are described in the SRP. Specifically, Section 15.0.2 of the SRP describes the NRC staff's review process and acceptance criteria for analytical models and computer codes used by licensees to analyze accident and transient behavior. The purpose of the NRC staff review for this SRP section is to verify that the evaluation model is adequate to simulate the accident under consideration.

Guidance to the industry for the analysis of transient behavior is set forth in RG 1.203 and, in particular, licensees must include a complete assessment of all code models against applicable experimental data and/or exact solutions in order to demonstrate that the code is adequate for analyzing the chosen scenario.

3.7.2 Evaluation

The changes to the CLB and key design input values for adoption of AST are described in Appendix B, "Changes to Key Design Input Values (By Accident): CLB vs AST," of the TR submitted by letter dated December 27, 2016. Also, the proposed gas gap fractions for non-LOCA events for the proposed AST can be found in the referenced document. In addition to the TS changes, the licensee has provided the proposed FSARU markup, resulting from the adoption of AST, in its letters dated June 17 and December 17, 2015 and April 21, June 9, and October 6, 2016. The FSARU markup is provided to assist in NRC staff review and is for information only.

3.7.2.1 Summary of Technical Information Related to Proposed Gap Fractions Provided by Licensee

In its TR submitted by letter dated December 27, 2016, prepared by WECTEC Global Project Services Inc., providing a summary of the dose analyses and results for AST implementation, the licensee states that the gas gap fractions for Non-LOCA events will be based on Draft RG DG-1199. Table 3 in Draft RG DG-1199, provides the acceptable gas gap fractions for Non-LOCA events, with the exception of the CREA, for AST applications. The AST gas gap fractions provided by the licensee are proposed to meet Note 11 of RG 1.183. Note 11 states that the release fractions listed in Table 3 of RG 1.183 are acceptable for use with currently approved LWR fuel with a peak burnup of 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kilowatt per foot peak rod average power for burnups exceeding 54 GWD/MTU.

In Appendix B of Attachment 2 to the Enclosure of the letter dated December 17, 2015, the licensee stated that the methodology/scenarios used in the DBA analyses discussed in the DCPP FSARU are being updated to reflect the AST guidance provided in RG 1.183 and the DBA analyses for LOCA, FHA, LRA, CREA, MSLB, SGTR, and LOL Events are being updated.

Appendix B, referenced above, provides a comparison between the design input values used in the CLB dose consequence analyses supporting DCPP, to those utilized in the AST analyses supporting this LAR. Appendix B provides a comparison of the CLB values to the proposed AST values, with a description for the change, for the DBA accident analyses listed above. It is noted that the DCPP CLB assesses CR habitability for the LOCA, MSLB, SGTR, and FHA. The methodology used to assess the CLB analyses supporting the CREA, LRA, and LOL event are DCPP specific with pre-NUREG-0800 assumptions. In addition, the CLB analyses for the CREA, LRA, and LOL only address offsite dose consequences.

3.7.2.2 NRC Staff Review of the Proposed Gap Fractions

The NRC staff conducted an audit from January 12-13, 2016 at the Westinghouse offices in Rockville, Maryland (Audit Plan dated December 24, 2015, ADAMS Accession

No. ML15355A157), to review the supporting documentation and calculation files for the gas gap fractions and FSARU Chapter 15 accident analyses thermal hydraulic parameters.

During the regulatory audit, the NRC staff reviewed calculations associated with the proposed gas gap fractions for the AST. The purpose of the regulatory audit was to gain understanding of the calculations, to verify information, and/or identify information that will require docketing to support the basis of the regulatory decision. The NRC staff did not identify any additional information necessary for docketing as a result of the regulatory audit.

The licensee has three major design basis non-LOCA events that are postulated to result in fuel damage: the LRA, FHA, and CREA. Since fuel damage is assumed for these design-basis events, the different isotopes used to determine the dose consequences are important. To determine the amount of the isotopes released, gas gap fractions and the number of failed fuel rods are used to determine the source term of the accident. Gas gap fractions are the fraction of a given isotope residing in the gap between the fuel pellet and the fuel cladding. When fuel damage is assumed to occur, the gas in the gap is released.

High burnup fuel requires special consideration for gas gap fractions, as the extended amount of time in the reactor coupled with the breakdown of the fuel pellet results in a higher gap fraction for certain isotopes. To address the concerns with higher burnup fuel, the licensee elected to use the fuel gap fractions provided in Table 3 of Draft RG DG-1199 for all non-LOCA events except for the CREA. This approach is acceptable provided the licensee stays below the maximum allowable power operating envelope for PWRs as shown in Figure 1 of DG-1199. The power envelope within DG-1199 provides a benchmark to demonstrate if the gap fractions are reasonable. The licensee demonstrated compliance with the maximum power envelope by adding an additional verification of core power peaking in the administrative procedure TS6.DC3, "Reload Core Design Process," to place a limit on peak rod linear heat generation rate in fuel assemblies during normal operations. The gap fractions used to assess the dose consequences for FHA and LRA are as follows:

Nuclide Group	FHA and LRA (based on DG-1199)
I-131	0.08
I-132	0.23
Kr-85	0.35
Other Noble Gases	0.04
Other Halogens	0.05
Alkali Metals	0.46

For the CREA, the licensee used 0.10 for Noble Gases and 0.10 for Halogens, which is consistent with Appendix H and Note 11 of Table 3 for RG 1.183.

The gap fractions for LRA and FHA are acceptable for assessing the dose consequences because of the use of the bounding power history outlined in DG-1199, and the reload design process that confirms the power history remains bounding. The bounding power histories for the gap fractions utilized in the dose consequences are bounding for the non-LOCA design basis events. The gap fractions for the CREA are acceptable because the values are consistent with RG 1.183.

Section 1 of the Enclosure to letter dated June 17, 2015 provides the following description of the revised source terms.

> The AST methodology as established in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000, [Reference 3 of the letter] is used to calculate the offsite and Control Room radiological consequences for DCPP Units 1 and 2. Attachment 4 [to the Enclosure of letter dated June 17, 2015] contains a summary of the analyses and results for the following events that are expected to produce the most limiting dose consequences. Conformance to RG 1.183 is provided in Attachment 5 [to the Enclosure of letter dated June 17, 2015].
>
> - Loss of Coolant Accident (LOCA)
> - FHA in the Containment
> - FHA in the FHB
> - Locked Rotor Accident (LRA)
> - Control Rod Ejection Accident (CREA)
> - Main Steam Line Break (MSLB)
> - Steam Generator Tube Rupture (SGTR)
> - Loss-of-Load (LOL) Event

During the audit, the NRC staff reviewed Westinghouse Electric Company LLC (WEC) topical report WCAP-16638-P, Revision 1, "Diablo Canyon Units 1 and 2 Replacement Steam Generator Program NSSS [Nuclear Steam Supply System] Licensing Report," dated January 2008[8] and WCAP-16985, Revision 2, "Diablo Canyon Units 1 and 2 T_{avg} and T_{feed} Ranges Program NSSS Engineering Report," dated April 2009[10] to gain a detailed understanding of the design parameters used for the thermal analysis for the AST. WCAP-16638-P documents the technical basis for the SG replacements for Unit 1 prior to Cycle 16 and Unit 2 prior to Cycle 15. WCAP-16638P, for the SG replacement was implemented under the 10 CFR 50.59 process. WCAP-16638P and WCAP-16985, for the SG replacement, provided a frame of reference for the changes to the assumed parameters for the thermal hydraulics of the AST. The assumptions and bases that were employed for the SG replacement analysis were reviewed during the regulatory audit (Regulatory Audit Report dated March 14, 2016; ADAMS Accession No. ML16063A170).

The NRC staff reviewed the FSARU Chapter 15 accident analyses contained in WCAP-16638P, which includes Large-Break LOCA, Small-Break LOCA, LOL, MSLB, LRA, and SGTR. The NRC staff reviewed these Chapter 15 accident analyses to determine the computer codes used for each analysis.

The NRC staff reviewed various calculation notes for different Chapter 15 events, such as MSLB, SGTR, LOL, LRA, and CREA. Various thermal hydraulic parameters were of interest for each event, and are listed below.

[8] Document reviewed during the audit from January 12-14, 2016, is a plant specific document and is not publicly available.

3.7.2.2.1 Main Steam Line Break

The thermal hydraulic parameters of interest for MSLB are minimum RCS mass following accident, leak rate to faulted SG, liquid mass in each SG, release rate of SG liquid activity from faulted SG, and steam releases from intact SG. Each parameter was compared from the CLB value to the value used in the dose analysis, was documented using the references that capture where and why the changes occurred were logged, and annotated appropriately.

Based on the TR submitted by letter dated December 27, 2016, the minimum RCS mass, following the accident, changed from 566,000 lbm to 446,486 lbm. The RCS mass is a calculated value and a lower value is considered conservative with respect to iodine dose consequences.

As indicated by the TR, the leak rate to the faulted SG changed from 10.5 gpm to 0.75 gpm. The leakage rate will be confirmed by measurement and is considered conservative for the event. The leak rate also uses the RG 1.183 leakage density assumption.

Based on the TR, the liquid mass in both the faulted and intact SGs changed in the AST analysis. The faulted mass changed from 162,784 lbm to 182,544 lbm. The intact SG changes from 81,500 lbm/SG to 92,301 lbm/SG. The mass was rounded up and a 10 percent uncertainty was added based on WEC assumptions to add conservatism. The calculated value was decreased by 10 percent per WEC's suggestion to add conservatism. The release rate of the SG liquid activity from the faulted SG changed from instantaneous to the dryout of SG liquid in 10 seconds. This information was verified an confirmed by review of the appropriate reference documents during the Regulatory Audit performed from January 14-16, 2016 as documented in the audit report dated March 14, 2016.

The steam releases from the intact SG changed for the different time periods. For 0 to 2 hours, the value changed 393,464 lbm to 384,000 lbm. For 2 to 8 hours, the value changed from 915,000 lbm to 893,000 lbm. Also, the timeframe was extended to 10.73 hr. It is important to note that blowdown is not a function of time and the steam release is assumed to be rapid at no-load conditions. An additional calculational note is referenced, and shows that the change is due to a lower feedwater temperature in the 0- to 2-hour timeframe. The time was increased out to 10.73 hours to address cooldown limitations during the accident.

The NRC staff has reviewed the calculational notes pertaining to the MSLB analysis. The NRC staff determined that the changes made to the MSLB analysis based on the proposed AST to be conservative, and thus acceptable.

3.7.2.2.2 Steam Generator Tube Rupture

The thermal hydraulic parameters of interest for SGTR are the RCS mass, initial SG liquid mass, steam flow rate to the condenser from the ruptured SG before trip and from the intact SGs before trip, steam releases from the ruptured SG and intact SGs, the post-accident minimum SG liquid mass for the ruptured SG and intact SGs, tube leakage rate, and break flow from RCS into ruptured SG.

The RCS mass changed from 499,500 lbm to 446,486 lbm and the change was documented in ASTAP-14-13, dated June 3, 2014, "Design Input Transmittal (DIT-50497328-5-0). Non-LOCA

Steam Releases and Thermal Hydraulic Input Parameters (MSLB/SGTR)[9]. The notes on why the change occurred are captured in the MSLB section.

The initial SG liquid mass changed for both the intact and faulted SG. The intact changed from 118,500 lbm to 89,707 lbm and the faulted changed from 106,000lbm to 89,707 lbm. The changes are captured in the specific calculational note. For the ruptured opened power operated relief valve phase, the average was taken from the RETRAN outputs and rounded down to increase the Iodine inventory in the SG liquid.

The steam flow rate to condenser from the ruptured SG and intact SG before trip did not change as the value is from the nominal full power steam flow rate (63,000 lbm/min).

The steam releases from ruptured SG and intact SGs vary over break time. The values change over the break time however there was no change from the CLB to the AST values.

The post-accident minimum SG liquid mass changed for the ruptured SG and intact SG. For the ruptured SG, the liquid mass went from 106,000 lbm to 89,707 lbm. The CLB for ruptured SGs is an average of initial mass and initial stuck open ADV phase. The smaller mass is conservative to increase the iodine activity for the dose consequences. The intact SGs changed from 118,500 lbm/SG to 89,707 lbm/SG. These changes were found in the appropriate calculational notes that were reviewed during the audit. The minimum mass is the initial mass of the SG following the reactor trip.

The tube leakage rate is the same as the MSLB.

The break flow from the RCS into ruptured SG values are similar to the steam releases from the ruptured SG and intact SG. The results of the analysis were consistent between the calculational notes and license amendment.

The NRC staff has reviewed the calculational notes pertaining to the SGTR analysis. The NRC staff determined that the changes made to the SGTR analysis based on the proposed AST to be conservative, and thus acceptable.

3.7.2.2.3 Loss-of-Load

The thermal hydraulic parameters of interest for LOL are the RCS mass, primary to secondary SG tube leakage, initial and minimum SG liquid mass, and steam releases.

The RCS mass changed from 499,500 lbm to 446,486 lbm and the change was documented in the calculational note. The notes are captured in the MSLB notes.

The tube leakage rate is the same as the MSLB.

The intact SGs change from 81,500 lbm/SG to 92,301 lbm/SG. The faulted SG was captured in the calculational notes. The mass of the SGs are rounded up and an uncertainty was added based on WEC assumptions to add conservatisms. For the intact SGs, the masses were

[9] Document reviewed during the audit from January 12-14, 2016.

captured in the appropriate calculational notes. The calculated value was decreased per WEC suggestion to add conservatism.

The steam releases changed for the multiple time frames. From 0 to 2 hours, the steam releases changed from 656,000 lbm to 651,000 lbm. For 2 to 8 hours, the steam releases changed from 1,035,000 lbm to 1,023,000 lbm. Additionally, an 8- to10.73-hour timeframe was added with the same steam releases as the 2- to 8-hour timeframe. WCAP-16985, states that the mass of the environmental steam releases for the LOL event bound all Condition II events, as well as the steam releases following a LRA and CREA.

The NRC staff has reviewed the calculational notes pertaining to the LOL analysis. The NRC staff determined that the changes made to the LOL analysis based on the proposed AST to be conservative, and thus, acceptable.

3.7.2.2.4 Locked Rotor and Control Rod Ejection

The locked rotor and control rod ejection accidents were based on the LOL references and thermal hydraulic parameters. See the discussion for the LOL event for the thermal hydraulic parameters for LRA and CREA.

3.7.3 Results of the NRC Staff Review

In its LAR, PG&E proposed new gas gap fractions for the AST. The analytical technique, inputs, and assumptions used in the PG&E calculations were found to be conservative, appropriate, and consistent with RG 1.183 and DG 1199. Based on the above, the NRC staff concludes that the proposed multipliers on the RG 1.183 gap inventories to be acceptable.

The NRC staff reviewed the underlying PG&E and WEC engineering calculations during the regulatory audit to gain better understanding of the analysis performed in support of the proposed change. The analytical technique, inputs, and assumptions used in the PG&E and WEC calculations were determined to be conservative and appropriate for FSARU Chapter 15 accident thermal hydraulic analyses. Hence, the NRC staff determined that the proposed thermal hydraulic parameters for the FSARU Chapter 15 accident analyses to be acceptable.

3.8 Human Factors

3.8.1 Background

The NRC staff reviewed the licensee's overall request using the guidance contained in RG 1.183 and SRP Section 15.0.1. With regard to the proposed changes to manual operator actions, the NRC staff used the guidance contained in IN 97-78, ANSI/ANS 58.8-1994 and SRP Chapter 18.

3.8.2 Evaluation

3.8.2.1 Changes to Manual Operator Actions

To support its request to implement an AST at DCPP, PG&E reanalyzed selected DBAs, consistent with the requirements of 10 CFR 50.67 and NRC guidance documents (e.g., RG 1.183).

As described on page 7 of the Enclosure to letter dated June 17, 2015, contained within this reanalysis was one assumption regarding manual operator actions which differ from the CLB:

> The LOCA dose analysis also credits a time critical operator action (TCOA). A TCOA is a manual action or series of actions with a specified completion time limit to meet a plant licensing basis requirement. The LOCA dose analysis assumes that containment spray is realigned from the injection mode to the recirculation mode within 12 minutes of terminating injection spray to ensure that the duration of spray operation (injection + recirculation) exceeds 6.25 hours following the event. The TCOA will be implemented as part of AST implementation, using the guidelines provided in NRC Information Notice 97-78, "Crediting of Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times." The required actions for the new TCOA have been demonstrated on the simulator, showing that the 12 minute time requirement can be achieved with margin.

PG&E has requested for approval of this change in manual operator actions such that these changes become a part of the CLB for DCPP.

Based on the guidance contained in IN 97-78, ANSI/ANS 58.8-1994, and NUREG-0800, Chapter 18, the NRC staff determined that the proposed change to manual operator actions is acceptable with respect to human performance.

The proposed change, to realign CS from injection mode to recirculation mode within 12 minutes following termination of injection spray, is not a part of the CLB at DCPP. The NRC staff reviewed the change in the CLB for this new manual operator action. The NRC staff concludes that crediting this manual operator action is acceptable, based on the following four considerations.

1. The operator action is directed by plant procedures

Procedure step sequencing will cue the operator to align CS from RHR, even if only one RHR train has been successfully aligned for cold leg recirculation. In response to RAI, APHB-RAI-9, in its letter dated October 22, 2015, the licensee stated that the EOP E-1.3, "Transfer to Cold Leg Recirculation," will provide clear guidance on the conditions required to implement these actions, and as an approved plant procedure, provide appropriate plant status control.

2. The operator action to realign containment spray is a simple task

There is minimal increase in operator workload as a result of this change. The proposed license amendment does not change the circumstances under which the EOP E-1.3 is performed, rather only a sequence of actions within the procedure. The revision to EOP E-1.3 ensures that CS is aligned to RHR even when only a single train of RHR is able to be placed in the cold leg recirculation lineup. There are no additional field actions required by the procedure revision and no new system control methodologies required. A new TCOA will be required to implement the proposed amendment, however, simulator demonstrations of the procedure revision have proven that the new TCOA would be easily achievable.

3. Operators will be trained on the new required action.

In response to RAI APHB-RAI-5, in its letter dated October 22, 2015, the licensee stated that operators are routinely trained and evaluated on their ability to properly carry out actions specified in EOPs. Training coordination is required for revised EOPs. This coordination will determine the training timeline, the appropriate audience (licensed operators), and the venue for the training (classroom/simulator).

4. The change has been through a verification and validation (V&V) process to ensure the ability of the operators to accomplice the tasks required in the license amendment.

In response to RAIs APHB-RAI-3 and APHB-RAI-11, in its letter dated October 22, 2015, the licensee stated that demonstration of the requirements for verification and validation (V&V) per Administrative Procedure AD1.DC12," Writer's Guide for Emergency Operating Procedures and Abnormal Operating Procedures," Revision 9 and Interdepartmental Administrative Procedure (IDAP) OP1.ID2, "Time Critical Operator Action," Revision 8A, was performed by three operating crews for the proposed changes to EOP E-1.3. For the V&V process, crews had no prior knowledge of the event or any additional training on the proposed new steps to be added to the EOP. This demonstration was performed with a single Unit Shift Forman who provided procedural direction and a board operator who performed equipment manipulations. This lineup is representative of TSs minimum staffing. The operating crews demonstrated that they could perform the new steps within the 12 minutes timeframe.

3.8.3 Conclusion

The NRC staff has reviewed the proposed change to credited manual operator actions associated with implementing an AST at DCPP. Based on the above, the NRC staff has concluded that the TS, as revised, continues to meet the requirements of 10 CFR 50.34; 10 CFR 50.36; 10 CFR 50.49; 10 CFR 50.67; 10 CFR 50.90; 10 CFR Part 50, Appendix A, GDCs 17 and 19; and 10 CFR 50, Appendix J. The NRC staff also concludes that the TS changes are consistent with the guidance provided by documents listed in Section 2.0, "Regulatory Evaluation," and additional references included in the individual sections of this SE. Hence, the NRC concludes that 1) there is a reasonable assurance that the health and safety of the public will not be endangered by the proposed changes in manual operator actions, (2) such activities will be conducted in compliance with the Commission's regulations and guidance, and (3) the issuance of the license amendment will not be inimical to the common defense and security or to the health and safety of the public. Therefore, the NRC staff concludes that the request for changes in operator manual actions is acceptable.

4.0 <u>REGULATORY COMMITMENTS</u>

In its letter dated June 17, 2015, the licensee made several regulatory commitments, which were replaced by license conditions proposed by letter dated August 31, 2015, later revised by letter dated June 9, 2016, and are described in Section 5.0 of this SE.

In response to an RAI by the NRC staff by letter dated November 6, 2015, the licensee made the following regulatory commitment:

> Existing environmentally qualified components will be evaluated per the EQ program to confirm acceptability as part of AST implementation. As required, the list of the components associated with [10 CFR] 50.49 program will be updated

after completing the evaluation per the EQ program, as part of the AST implementation.

Based on the NRC staff review, there will be no impact on the EQ envelope post-LOCA peak pressure and temperature inside containment, and no additions will be necessary to the EQ list due to the implementation of the AST. As an added assurance, the licensee made the above regulatory commitment and the NRC staff did not rely on the commitment to reach its conclusion.

5.0 LICENSE CONDITIONS

In its letter dated June 9, 2016, the licensee proposed the following additional license conditions (Appendix D to the Operating Licenses for DCPP, Units 1 and 2):

> Implementation of the amendment adopting the alternative source term shall include the following plant modifications:
>
>> Install shielding material, equivalent to that provided by the Control Room outer walls, at the external concrete west wall of the Control Room briefing room.
>>
>> Install a high efficiency particulate air filter in the Technical Support Center normal ventilation system.
>>
>> Re-classify a portion of the 40-inch Containment Penetration Area (GE/GW) Ventilation line from PG&E Design Class II to PG&E Design Class I and upgrade the damper actuators, pressure switches, and the damper solenoid valves to PG&E Design Class I.
>>
>> Update setpoints for the redundant safety related gamma sensitive area radiation monitors (1-RE 25/26, 2-RE 25/26).

The AST analysis assumptions proposed by the licensee, as discussed throughout Sections 3.3 and 3.5 of this SE, are based on the installation and completion of the above stated modifications at DCPP. Based on the NRC staff review of the licensee's analyses, the NRC staff finds the proposed license conditions to be acceptable.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the California State official was notified of the proposed issuance of the amendments. In an e-mail dated January 19, 2017 (ADAMS Accession No. ML17025A400), the State official provided the following comment:

> At this time, the California State Liaison Officer Robert B. Weisenmiller would like to express support for the License Amendment as it may improve reactor safety in the event that certain postulated accidents occurred. The License Amendment Request: Pacific Gas and Electric Company (PG&E), Docket Nos. 50–275 and 50–323, Diablo Canyon Nuclear Power Plant (DCPP), Unit Nos. 1 and 2 is responsive to safety concerns. The adoption of the alternative source term (AST), which prompted the subject License Amendment, necessitates the revision of both the Updated Final Safety Analysis Report (UFSAR) and the

Technical Specifications (TS), which appear to strengthen measures that cope with the offsite and control room radiological consequences of certain postulated accident scenarios. For these reasons, the California State Liaison Officer supports the proposed License Amendment for its potential to improve facility operating safety while posing no additional significant hazards.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding published in the *Federal Register* on November 8, 2016 (81 FR 78664). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Date: April 27, 2017

SUBJECT: DIABLO CANYON POWER PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: REVISE LICENSING BASES TO ADOPT ALTERNATIVE SOURCE TERM (CAC NOS. MF6399 AND MF6400), DATED APRIL 27, 2017

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