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

September 20, 2023

Mr. Fadi Diya
Senior Vice President and
Chief Nuclear Officer
Ameren Missouri
Callaway Energy Center
8315 County Road 459
Steedman, MO 65077

SUBJECT: CALLAWAY PLANT, UNIT NO. 1 - ISSUANCE OF AMENDMENT NO. 233 FOR ADOPTION OF ALTERNATIVE SOURCE TERM AND REVISION OF TECHNICAL SPECIFICATIONS (EPID L-2021-LLA-0177)

Dear Mr. Diya:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 233 to Renewed Facility Operating License No. NPF-30 for the Callaway Plant, Unit No. 1. The amendment consists of changes to the technical specifications (TSs) in response to your application dated September 28, 2021, as supplemented by letters dated December 1, 2021; July 5, 2022; September 1, 2022; December 8, 2022; and May 9, 2023.

The amendment revises the Callaway TSs and authorizes changes to the Callaway Final Safety Analysis Report to support a full scope application of the regulations in Title 10 of the *Code of Federal Regulations* Section 50.67, "Accident source term," and described in NRC Regulatory Guide 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors."

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's monthly *Federal Register* notice.

Sincerely,

/RA/

Mahesh L. Chawla, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-483

Enclosures:
1. Amendment No. 233 to NPF-30
2. Safety Evaluation

cc: Listserv

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

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT NO. 1

DOCKET NO. 50-483

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 233
License No. NPF-30

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

   A. The application for amendment by Union Electric Company (UE, the licensee), dated September 28, 2021, as supplemented by letters dated December 1, 2021, July 5, 2022, September 1, 2022, December 8, 2022, and May 9, 2023, 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 Renewed Facility Operating License No. NPF-30 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 233 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective as of its date of issuance and shall be implemented within 120 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Jennifer L. Dixon-Herrity

Digitally signed by Jennifer L. Dixon-Herrity
Date: 2023.09.20 09:40:00 -04'00'

Jennifer L. Dixon-Herrity, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility Operating License No. NPF-30 and the Technical Specifications

Date of Issuance: September 20, 2023

ATTACHMENT TO LICENSE AMENDMENT NO. 233

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-30

CALLAWAY PLANT, UNIT NO. 1

DOCKET NO. 50-483

Replace the following pages of Renewed Facility Operating License No. NPF-30 and the Appendix A, Technical Specifications, with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Renewed Facility Operating License

REMOVE	INSERT
-3-	-3-

Technical Specifications

REMOVE	INSERT
1.1-2	1.1-2
3.7-28	3.7-28
3.7-29	3.7-29
3.7-30	3.7-30
3.7-31	3.7-31
5.0-15	5.0-15
5.0-16	5.0-16
5.0-21	5.0-21
5.0-22	5.0-22

(3) UE, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) UE, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source of 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) UE, 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 renewed 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

UE is authorized to operate the facility at reactor core power levels not in excess of 3565 megawatts thermal (100% power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan*

The Technical Specifications contained in Appendix A, as revised through Amendment No. 233 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Environmental Qualification (Section 3.11, SSER #3)**

Deleted per Amendment No. 169.

* Amendments 133, 134, & 135 were effective as of April 30, 2000 however these amendments were implemented on April 1, 2000.

** 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.

1.1 Definitions (continued)

CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy. The COT may be performed by means of any series of sequential, overlapping, or total channel steps.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
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 thyroid dose conversion factors from: 1. Table 2.1 of EPA Federal Guidance Report No. 11, EPA-520/1-88-020, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1988.

(continued)

3.7 PLANT SYSTEMS

3.7.10 Control Room Emergency Ventilation System (CREVS)

LCO 3.7.10 Two CREVS trains shall be OPERABLE.

-- NOTE --
The control room envelope (CRE), control building envelope (CBE) and equipment room envelope (ERE) boundaries may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, 3, and 4,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One CREVS train inoperable for reasons other than Condition B.	A.1	Restore CREVS train to OPERABLE status.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One or more CREVS trains inoperable due to an inoperable CRE, ERE or CBE boundary in MODE 1, 2, 3, or 4.	B.1	Initiate action to implement mitigating actions.	Immediately
	<u>AND</u>		
	B.2	Verify mitigating actions to ensure CRE occupant radiological exposure will not exceed limits and CRE occupants are protected from chemical and smoke hazards.	24 hours
	<u>AND</u>		
	B.3	Restore the CRE, ERE and CBE boundaries to OPERABLE status.	90 days
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.	C.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	C.2	Be in MODE 5.	36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies.	D.1	Place OPERABLE CREVS train in CRVIS mode.	Immediately
	OR		
	D.2.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	D.2.2	Suspend movement of irradiated fuel assemblies.	Immediately
E. Two CREVS trains inoperable during movement of irradiated fuel assemblies. OR One or more CREVS trains inoperable due to an inoperable CRE, CBE or ERE boundary during movement of irradiated fuel assemblies.	E.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	E.2	Suspend movement of irradiated fuel assemblies.	Immediately
F. Two CREVS trains inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B.	F.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Operate each CREVS train pressurization filter unit for $\geq$ 15 continuous minutes and each CREVS train filtration filter unit for $\geq$ 15 continuous minutes.	In accordance with the Surveillance Frequency Control Program
SR 3.7.10.2	Perform required CREVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.10.3	Verify each CREVS train actuates on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.10.4	Perform required unfiltered air inleakage testing of the CRE, CBE and ERE boundaries in accordance with the Control Room Envelope Habitability Program.	In accordance with the Control Room Envelope Habitability Program

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 1.0% when tested at the system flowrate specified below.

ESF Ventilation System	Flowrate
Control Room Filtration	2000 cfm, ± 200 cfm
Emergency Exhaust System	9000 cfm, ± 900 cfm

c. Demonstrate for each of the ESF systems within 31 days after removal that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of 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 the relative humidity specified below.

ESF Ventilation System	Penetration	RH
Control Room Filtration	2.0%	70%
Emergency Exhaust System	2.0%	70%

d. Demonstrate at least once per 18 months for each of the ESF systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers (as applicable) is less than the value specified below when tested at the system flowrate specified below.

ESF Ventilation System	Delta P	Flowrate
Control Room Filtration	5.4" WG	2000 cfm, ± 200 cfm
Control Room Pressurization*	5.4" WG	500 cfm, +500,- 50 cfm
Emergency Exhaust System	5.4" WG	9000 cfm, ± 900 cfm

* A charcoal adsorber is not required to be installed or included in either control room pressurization train.

(continued)

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

e. Demonstrate at least once per 18 months that the heaters for the ESF systems dissipate the value specified below when tested in accordance with ANSI 510-1975 and corrected to design nameplate voltage settings.

ESF Ventilation System	Wattage
Emergency Exhaust System	37 ± 3 KW

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 Gaseous Radwaste System, the quantity of radioactivity contained in gas storage tanks and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure, Revision 0". The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures, Revision 2".

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Gaseous Radwaste System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and

c. A surveillance program to ensure that the quantity of radioactivity contained in the outdoor liquid radwaste tanks listed below that are not

(continued)

5.5.17 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 Emergency Ventilation System (CREVS), 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 total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:

a. The definition of the CRE, CRE boundary, control building envelope (CBE), CBE boundary, equipment room envelope (ERE), and the ERE boundary.

b. Requirements for maintaining the CRE, CBE and ERE boundaries in their design condition, including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE, CBE and ERE boundaries 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.

   The following exception is taken to Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0:

   1. The Tracer Gas Test based on the Brookhaven National Laboratory Atmospheric Tracer Depletion (ATD) Method is used to determine the unfiltered air inleakage past the CRE, CBE and ERE boundaries. The ATD Method is described in AmerenUE letters dated December 15, 2004 (ULNRC-05104), June 6, 2006 (ULNRC-05298), July 16, 2007 (ULNRC-05427), and October 30, 2007 (ULNRC-05448).

(continued)

5.5.17 Control Room Envelope Habitability Program (continued)

d. Measurement, at designated locations, of the CRE pressure relative to the outside atmosphere during the pressurization mode of operation by one train of the CREVS, operating at the flow rate required by the VFTP, at a Frequency of 18 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the periodic assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into CRE, CBE and ERE. 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 leakage 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 for assessing CRE habitability, determining CRE, CBE and ERE unfiltered inleakage, and measuring CRE pressure and assessing the CRE, CBE and ERE as required by paragraphs c and d.

5.5.18 Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, “Risk-Informed Method for Control of Surveillance Frequencies,” Revision 1.

c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

(continued)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 233 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-30

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT NO. 1

DOCKET NO. 50-483

1.0 INTRODUCTION

By application dated September 28, 2021 (Reference 1), as supplemented by letters dated December 1, 2021; July 5, 2022; September 1, 2022; December 8, 2022; and May 9, 2023 (References 2, 3, 4, 5, and 6, respectively); Union Electric Company, doing business as Ameren Missouri (the licensee), pursuant to Title 10 of *Code of Federal Regulations* (10 CFR) Section 50.90, “Application for amendment of license, construction permit, or early site permit,” and 10 CFR 50.67, “Accident source term,” submitted a license amendment request (LAR) for the Callaway Plant, Unit No. 1 (Callaway), to incorporate the alternative source term (AST) dose analysis methodology into its licensing basis and to revise the technical specifications (TS) definition of Dose Equivalent (DE) Iodine-131 (I-131) (DE I131) and TS 3.7.10, “Control Room Emergency Ventilation System (CREVS)”; 5.5.11, “Ventilation Filter Testing Program (VFTP)”; and 5.5.17, “Control Room Envelope Habitability Program.”

The licensee’s application is intended to enhance alignment between the Final Safety Analysis Report (FSAR) Chapter 15, “Accident Analysis” (Reference 7), accident radiological dose evaluation models, and the control room design and operation. This includes increasing the maximum allowable unfiltered air inleakage values to provide additional margin to measured values and removing credit for the charcoal adsorbers in the control building pressurization unit in the AST dose analyses.

The licensee has determined that updating the current licensing basis (CLB) control room model from a two-zone to a more representative three-zone model will improve the dose consequence model for the control room. While the CLB models the interaction between the control room and the control building, the revised three-zone model used in the AST analysis includes the ventilation interactions with the control room, the control building, and the control room equipment room.

The supplemental letters dated July 5, 2022, September 1, 2022, December 8, 2022, and May 9, 2023, provided additional information that clarified input to the analyses in the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC, the Commission) staff’s original proposed no significant hazards consideration determination as published in the *Federal Register* on February 22, 2022 (87 FR 9654).

1.1 Background

An accident source term is intended to be representative of a major accident involving significant core damage and is typically postulated to occur in conjunction with a large loss-of-coolant accident (LOCA). Although the LOCA is typically the maximum credible accident, other accident sequences of lesser consequence but higher probability of occurrence need consideration.

Since the publication of Technical Information Document (TID) - 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," dated March 23, 1962 (Reference 8), significant advances have been made in understanding the timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents. In 1995, the NRC published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," dated February 1995 (Reference 9). NUREG-1465 used this research to provide estimates of the accident source term that were more physically based and that could be applied to the design of future light-water nuclear power reactors. NUREG-1465 presents a representative accident source term for a pressurized-water reactor (PWR). Callaway is a PWR.

These source terms are characterized by the composition and magnitude of the radioactive material, the chemical and physical properties of the material, and the timing of the release to the containment. The NRC staff considered the applicability of the revised source terms to operating reactors and determined that the current analytical approach based on the TID-14844 source term would continue to be adequate to protect public health and safety. Operating reactors licensed under that approach would not be required to reanalyze accidents using the revised source terms. The NRC staff also determined that some licensees may use an AST in analyses to support cost-beneficial licensing actions.

The NRC staff, therefore, initiated several actions to provide a regulatory basis for operating reactors to use an AST in design-basis analyses. These initiatives resulted in the development and issuance of 10 CFR 50.67 and Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000 (Reference 10). The NRC's traditional methods for calculating the radiological consequences of design-basis accidents (DBAs) were described in earlier revisions of a series of RGs and various NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition" (SRP) sections. The above guidance was developed to be consistent with the TID-14844 source term and the whole body and thyroid dose guidelines stated in 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance."

Many of those analysis assumptions and methods are inconsistent with the AST methodology and with the total effective dose equivalent (TEDE) criteria provided in 10 CFR 50.67. RG 1.183 provides assumptions and methods that are acceptable to the NRC staff for performing design -basis radiological analyses using an AST. The guidance in RG 1.183 supersedes the corresponding radiological analysis assumptions provided in the earlier revisions to these RGs and SRP sections when used in conjunction with an approved AST and the TEDE criteria provided in 10 CFR 50.67.

2.0 REGULATORY EVALUATION

2.1 Proposed Changes

The licensee's AST evaluation supports the following proposed changes to the licensing basis for Callaway as stated in its LAR as supplemented:

Unfiltered Inleakage and Modeling:

- The proposed change to the Control Room unfiltered inleakage is from 25.62 cfm [cubic feet per minute], in the current dose consequence analyses, to 60 cfm in the AST analyses.
- The proposed change to the Control Building unfiltered inleakage is from 2785 cfm, in the current dose consequence analyses, to 6000 cfm in the AST analyses.
- The proposed change to the inleakage modeling is to explicitly account for Control Room Equipment Room unfiltered inleakage. The AST analyses support an in-leakage of 300 cfm.
- The AST Control Room radiological consequence analysis updates the current two-zone model to a three-zone model. The modeling change enhances alignment between the FSAR Chapter 15 accident dose evaluation models and the actual control room design and operation.
- The measured unfiltered inleakage from these tests was < [less than] 20 cfm for the Control Room, < 170 cfm for the Control Building, and < 100 cfm for the Control Room Equipment Room.
- Upon implementation of the AST, the maximum allowable unfiltered air inleakage values would be increased to provide additional margin to future measured values and to facilitate potential equipment out-of-service/Control Room habitability/operability determinations. The Control Room and Control Building inleakage values would be approximately doubled relative to the current licensing basis assumptions, and the Control Room HVAC [heating, ventilation, and air conditioning] Filtration Equipment Room inleakage value would be increased approximately three times the historically measured value.

Charcoal Adsorbers:

- Credit for the charcoal adsorbers in the Control Building Pressurization Unit is removed in the AST dose analyses.

Proposed TS Changes

In enclosure 3 to the LAR, as supplemented by letter dated December 1, 2021, the licensee proposed:

- Deleting items 1, 2 and 3 from the list of references in the definition of DE I-131 for thyroid dose conversion factors and renumbering item 4 to item 1.

- Changing TS 3.7.10 by adding the equipment room envelope (ERE) to the text of the NOTE in Limiting Condition for Operation (LCO) 3.7.10, to Conditions B and E and Required Action B.3 descriptions in the ACTIONS table, and Surveillance Requirement (SR) 3.7.10.4. The proposed changes include editorial and typographical changes to add the ERE boundary to the existing control room and control building envelope requirements.

- Deleting the term “with the heaters operating” from the existing text of SR 3.7.10.1.

- Deleting the “Control Room Pressurization” text and acceptance criteria from existing TSs 5.5.11.b, 5.5.11.c and 5.5.11.e. The licensee proposed adding an asterisk after the instance of “Control Room Pressurization” in TS 5.5.11.d and adding a footnote which would read: “A charcoal adsorber is not required to be installed or included in either control room pressurization train.”

- Deleting the text “whole body or its equivalent to any part of the body” from the program description in TS 5.5.17 and replacing it with “total effective dose equivalent (TEDE).”

- Making editorial and typographical changes to add the ERE boundary to the existing Control Room and Control Building envelope requirements in TS 5.5.17.

The licensee stated that the current TID -14844 accident source term will remain the licensing basis for equipment qualification. The licensee also stated that for calculations other than those for the control room habitability envelope and Technical Support Center (TSC) doses, the licensee will continue using evaluations consistent with NUREG-0737, “Clarification of TMI [Three Mile Island] Action Plan Requirements,” and “Clarification of TMI Action Plan Requirements – Requirements for Emergency Response Capability,” Supplement No. 1 (Reference 11).

2.2 AST Regulatory Evaluation

Per 10 CFR 50.67(b)(1), a licensee who seeks to revise its current accident source term in design basis radiological consequence analyses shall apply for a license amendment under 10 CFR 50.90, and the application shall contain an evaluation of the consequences of applicable design basis accidents previously analyzed in the safety analysis report. The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of design analyses or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products. Per 10 CFR 50.67(b)(2), the NRC may issue the amendment only if the applicant'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 (25 rem) 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) TEDE.

(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) TEDE for the duration of the accident.

The regulation,10 CFR 50.36(b), states in part: “The technical specifications will be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto, submitted pursuant to [10 CFR] 50.34 [“Contents of applications; technical information”].”

The regulation, 10 CFR 50.36(c)(2), states, in part, that “Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation 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.”

The regulation, 10 CFR 50.36(c)(3), states that “Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.”

The regulation, 10 CFR 50.36(c)(5), “Administrative controls,” states, in part, that “Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review, and audit, and reporting necessary to assure operation of the facility in a safe manner.”

The regulations in 10 CFR 50.49, “Environmental qualification of electric equipment important to safety for nuclear power plants,” identify requirements for establishing a program for qualifying electric equipment that is important to safety as defined in 10 CFR 50.49(b).

This safety evaluation (SE) addresses the impact of the proposed changes on previously analyzed DBA radiological consequences, the related offsite and onsite atmospheric dispersion modeling analyses, and the acceptability of those revised dose analysis results. The NRC staff's evaluation is based upon the following regulatory codes, guides, and standards:

- RG 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants,” Revision 1, November 1982 (Reissued February 1983) (Reference 12), provides guidance to determine relative concentrations for assessing the potential offsite radiological consequences for a range of postulated accidental releases of radioactive material to the atmosphere.

- 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 establishes an acceptable AST and identifies the significant attributes of other ASTs that may be found acceptable. Section 1.3.2, "Re-Analysis Guidance," of RG 1.183 identifies that the ability of the damper to close against increased containment pressure may need to be evaluated or the ability of ductwork downstream of the dampers to withstand increased stresses due to AST implementation. RG 1.183, Regulatory Position 4.4, "Acceptance Criteria," provides guidance concerning 50.67(b)(2)(i)-(iii). The licensee has not proposed any significant deviation or departure from the guidance provided in RG 1.183.

- RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," Revision 0, June 2003 (Reference 13), provides guidance on determining atmospheric relative concentration (χ/Q) values in support of design basis control room radiological habitability assessments at nuclear power plants.

- RG 1.23, "Meteorological Monitoring Programs for Nuclear Power Plants," Revision 1, March 2007 (Reference 14), provides improved guidance concerning criteria for an acceptable onsite meteorological measurements program to support plant licensing and operation (e.g., tower and instrument siting, instrument accuracy and range, maintenance and servicing schedules, data reduction and compilation).

- Safety Guide 23, "Onsite Meteorological Programs," dated February 17, 1972 (Reference 15), the predecessor to RG 1.23 (often referred to as Revision 0 of that guidance).

- NUREG-0696, "Functional Criteria for Emergency Response Facilities, Final Report," February 1981 (Reference 16).

- NUREG--0737

- The following sections of NUREG-0800 (SRP):

    - Section 2.3.3, "Onsite Meteorological Measurements Program," Revision 3, March 2007 (Reference 17.a.).

    - Section 2.3.4, "Short-Term Atmospheric Dispersion Estimates for Accident Releases," Revision 3, March 2007 (Reference 17.b.).

    - Section 6.5.2, "Containment Spray as a Fission Product Cleanup System," Revision 4, March 2007 (Reference 17.c.).

    - Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision. 0, July 2000 (Reference 17.d.), assigns responsibility to the Operator Licensing and Human Factors Branch for the review of issues related to emergency operating procedures (EOPs) and human factors engineering design. This section also states, in part, that "An acceptable implementation of an AST should demonstrate compliance with plant- specific licensing commitments made in response to the NUREG-0737…." Specific provisions of interest within the context of this SRP section include "NUREG-0737 III.D.3.4, Control-Room Habitability, as it relates to maintaining the control room in a safe, habitable condition under accident conditions by providing adequate protection against radiation and toxic gases." This section provides guidance concerning 50.67(b)(2)(i)-(iii).

  - Section 15.6.2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment," Revision 2, July 1981 (Reference 17.e.).

- NUREG-1465

- NUREG/CR-2260, "Technical Basis for Regulatory Guide 1.145, Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," October 1981 (Reference 18).

- NUREG/CR-2858, "PAVAN – An Atmospheric-Dispersion Program for Evaluating Design-Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations," November 1982 (Reference 19).

- NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Reactors," February 1988 (Reference 20).

- NUREG/CR-5950, "Iodine Evolution and pH Control," December 1992 (Reference 21).

- NUREG/CR-6331, "Atmospheric Relative Concentrations in Building Wakes," Revision 1, May 1997 (Reference 22).

- Pacific Northwest National Laboratory (PNNL-28667), "ARCON 2.0 User's Guide," prepared by PNNL for the U.S. Department of Energy under Contract DE-AC05-76RL01830, October 2021 (Reference 23)

- NRC Regulatory Issue Summary (RIS) 2006-04, "Experience with Implementation of Alternative Source Terms," dated March 7, 2006 (Reference 24).

- NRC Information Notice (IN) 2012-01, "Seismic Considerations-Principally Issues Involving Tanks," dated January 26, 2012 (Reference 25).

3.0 TECHNICAL EVALUATION

3.1 Radiological Consequences of DBAs

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. The licensee performed AST analyses for the PWR DBAs identified in RG 1.183 that could potentially result in significant Control Room, TSC, and offsite doses. These include the LOCA, the fuel handing accident (FHA), the main steam line break accident (MSLB), the steam generator tube rupture accident (SGTR), the control rod ejection accident (REA), and the locked rotor accident (LRA). The licensee also evaluated other DBAs not discussed in RG 1.183 including the loss of non-emergency alternating current (AC) power (LOAC) and the letdown line break (LLB).

The DBA radiological source term used in the AST analyses was developed based on a licensed core power level of 3,565 megawatts thermal (MWt) with an additional 2 percent to account for calorimetric measurement uncertainties yielding a total of 3,636 MWt for use in determining the inventory of radionuclides in the reactor fuel. The use of 3,636 MWt for the AST DBA radiological source term analyses bounds the current licensed core thermal power level of 3,565 MWt and is therefore acceptable to the NRC staff for use in the full implementation of the AST.

The licensee has determined that the accident source term as described in TID-14844 will remain the licensing basis for equipment qualification (EQ). RG 1.183, Regulatory Position 6, “Assumptions for Evaluating the Radiation doses for Equipment Qualification,” states:

> The 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 (Reference 26) and in NUREG-0933, Supplement 25, “A Prioritization of Generic Safety Issues,” dated June 2001 (Reference 27).

The conclusion to Generic Issue 187 in NUREG-0933, Supplement 25 states, in part:

> The staff concluded that there was 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 cited references, the NRC staff finds that continued usage of TID-14844, as an accident source term for EQ does not impact the licensee’s adoption of 10 CFR 50.67.

The licensee used the methodology contained in TID-14844 to determine the radiation doses in the existing EQ analyses for Callaway. The use of this methodology is consistent with the guidance contained in RG 1.183. The NRC staff confirmed that the licensee will continue to use the TID-14844 methodology, as a result of the proposed LAR. Based on this, the NRC staff finds that no new electrical equipment needs to be added to the licensee’s 10 CFR 50.49 EQ program and that the EQ of electrical equipment should remain bounded due to full-scope implementation of the AST.

The licensee has determined that the full implementation of the AST has no impact on the assumptions or inputs to the current TID-14844 based analyses for NUREG-0737 evaluations. This conclusion is consistent with the NRC staff findings. RG 1.183, Regulatory Position 1.3.2, “Re-Analysis Guidance,” states, in part, that “radiological analysis results based on the TID-14844 source term assumptions and the whole body and thyroid methodology generally bound the results from analyses based on the AST and TEDE methodology.”

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 and the integrated dose to a control room operator were evaluated for the duration of the accident. The dose consequence analyses were performed by the licensee using the RADTRAD (Radionuclide Transport and Removal and Dose)-NAI code developed by Zachry Nuclear Engineering, Inc. (formerly Numerical Applications, Inc.). RADTRAD-NAI is a descendant of the NRC RADTRAD computer code. NRC sponsored the development of the RADTRAD radiological consequence computer code. The RADTRAD code 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 on an as needed basis using the RADTRAD computer code. The results of the evaluations performed by the licensee, as well as the applicable dose acceptance criteria from RG 1.183, are shown in table 1 in section 4.0 of this SE.

The licensee, following the guidance in RG 1.183, generated the core and worst-case fuel assembly radionuclide inventories for use in determining source term inventories using the ORIGEN-S computer code from the Oak Ridge National Laboratory SCALE 6.1.3 code package. The licensee assumed a period of irradiation that was sufficient to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values.

The licensee used committed effective dose equivalent (CEDE) and effective DE dose conversion factors (DCFs) from the U.S. Environmental Protection Agency (EPA) Federal Guidance Report (FGR) No. 11, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” dated September 1988 and FGR No. 12, “External Exposure to Radionuclides in Air, Water, and Soil,” dated September 1993 (References 28 and 29, respectively), to determine the TEDE dose as is required for AST evaluations. The use of ORIGEN and DCFs from FGR No. 11 and FGR No. 12 is in accordance with RG 1.183 guidance and is therefore acceptable to the NRC staff.

3.1.1 MSLB Accident

The postulated MSLB accident assumes a complete severance of one main steam line outside the primary containment. The affected steam generator (SG) will rapidly depressurize, and release radionuclides initially contained in the secondary coolant to the outside atmosphere. Primary coolant activity transferred to the affected SG via tube leakage will also be released to the outside atmosphere. The licensee conservatively assumed that, after the break, all the allowable tube leakage is to the affected SG. This leads to an uncontrolled release of steam from the steam system. The resultant depressurization of the steam system causes a reactor trip and a turbine trip. For the MSLB DBA radiological consequence analysis, a loss of offsite power (LOOP) is assumed to occur shortly after the turbine trip signal. The plant is cooled down by releasing steam to the environment because the LOOP renders the main condenser unavailable.

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 regarding radiological consequences. The affected SG rapidly depressurizes and releases the initial contents of the SG to the environment. The MSLB accident is described in section 15.1.5, “Steam System Piping Failure,” of the Callaway FSAR. RG 1.183, appendix E, “Assumptions for Evaluating the Radiological Consequences of a PWR Main Steam Line Break Accident,” identifies acceptable radiological analysis assumptions for a PWR MSLB.

3.1.1.1 Source Term

RG 1.183, Appendix E, Regulatory Position 2 states that “[i]f no or minimal fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by the technical specifications” including the effects of pre-accident and concurrent iodine spiking. The licensee’s evaluation indicates that no fuel damage is predicted for the 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. The licensee’s maximum iodine concentration allowed by TS as the result of an iodine spike is 60 micro curies per gram (µCi/gm) DE I131.

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 for the concurrent or accident-initiated 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. The licensee’s reactor coolant system (RCS) TS limit for normal operation is 1.0 µCi/gm DE I-131. The duration of the concurrent iodine spike is assumed to be 8 hours.

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 TS LCO of 0.1 µCi/gm DE I-131.

3.1.1.2 Transport

The licensee followed the guidance in RG 1.183, Appendix E, Regulatory Position 5 in all other aspects of the transport analysis for the MSLB.

In accordance with RG 1.183 and the existing licensing basis, the licensee assumed that primary-to-secondary leakage continues for approximately 8 hours after the MSLB at which time the residual heat removal (RHR) system is assumed to begin operation.

The licensee, following the guidance in RG 1.183, assumed 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, Regulatory Positions 5.5.1, 5.5.2 and 5.5.3, the licensee assumed that all the primary-to-secondary leakage into the faulted SG will flash to vapor and be released to the environment with no mitigation. The licensee assumed that the primary-to-secondary leakage mixes with the secondary water without flashing for the unaffected SGs that are used for plant cooldown.

In accordance with RG 1.183, Appendix E, Regulatory Position 5.5.4, the licensee assumed that the radioactivity in the bulk water of the unaffected SGs becomes vapor at a rate that is a function of the steaming rate and the partition coefficient. The licensee used a partition coefficient of 100 for elemental iodine and other particulate radionuclides released from the intact SGs.

3.1.1.3 Control Room Ventilation Assumptions for the MSLB

The low steamline pressure safety injection (SI) set point will be reached almost immediately because of the assumed MSLB. The licensee's analysis assumes the SI signal occurs 2 seconds post-accident. The SI signal causes the control room to switch from the normal operation mode to the emergency operation mode. The licensee conservatively modeled the switchover to occur at 62 seconds following event initiation, which includes a 60 second delay from the initiating signal. The control room ventilation isolation signal starts both trains of the control room filtration system. For the MSLB and all accidents that include a control room isolation signal, the licensee assumed a failure of one of the filtration fans to start resulting in a larger unfiltered inflow to the control room since only half of the makeup flow to the control room can pass through a filter. After 30 minutes, operator action is credited to isolate the failed train reducing the unfiltered inflow to the control room. The control room parameters used in the MSLB AST analysis are shown in table 4 in section 4.0 of this SE.

3.1.1.4 MSLB Accident Conclusion

The licensee evaluated the radiological consequences resulting from the postulated MSLB accident and concluded that the radiological consequences at the EAB, LPZ, and control room are within the requirements of 10 CFR 50.67(b)(2)(i)-(iii). The NRC staff finds 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 5, and the licensee's calculated dose results are given in table 1 in section 4.0 of this SE. The NRC staff finds that the EAB, LPZ, and control room doses estimated by the licensee for the MSLB meet the applicable accident dose criteria and are therefore acceptable.

3.1.2 LOAC

The licensee assumed that a LOAC occurs that trips the reactor coolant pumps (RCPs) and decreases forced flow through the reactor core. The licensee's evaluation assumes that no fuel cladding damage or fuel melting occurs because of this event. RCS activity passes from the primary into the secondary system due to the pressure differential between the primary and secondary systems and assumed SG tube leakage. A portion of this radioactivity is released to the outside atmosphere through the atmospheric steam dumps (ASDs) and/or main steam safety valves (MSSVs). In addition, iodine and alkali metal activity is contained in the secondary coolant before the accident and some of this activity is released to the atmosphere due to steaming from the SGs following the LOAC.

3.1.2.1 Source Term

Since no fuel failure results from the accident, the licensee included an accident-initiated iodine spike in the LOAC scenario based on the assumption that the reactor trip associated with the LOAC creates an iodine spike. The licensee assumed that the spike would increase the iodine release rate from the fuel to the RCS to a value 500 times greater than the appearance rate corresponding to the maximum equilibrium RCS concentration of 1.0 μCi/gm DE I-131. The licensee conservatively calculated the appearance rates based on assuming the maximum letdown flow with 100 percent cleanup. Maximizing the removal rate from the RCS will maximize the equilibrium appearance rate that is then multiplied by a factor of 500 to produce the iodine spike source term. The licensee assumed the duration of the accident-initiated iodine spike to be 8 hours.

In addition, the licensee included the noble gas activity concentration in the RCS based on the TS value of 225 µCi/gm of DE Xenon-133 (Xe-133) and the alkali metal activity concentration in the RCS at a 1 percent fuel defect level. The licensee, following the general assumptions in RG 1.183, assumed that the iodine available for release to the atmosphere is 97 percent elemental and 3 percent organic.

3.1.2.2 Transport

The analysis of the LOAC event is not discussed in RG 1.183 since this event is not considered an event that has the potential for fuel damage. The licensee states and the NRC staff agrees that the release pathway for this analysis is similar, to the LRA, and the accident-initiated iodine spike is similar, to the MSLB event. Therefore, the licensee incorporated release pathway models consistent with RG 1.183, Appendix G, “Assumptions for Evaluating the Radiological Consequences of a PWR Locked Rotor Accident,” and accident-initiated iodine spiking models consistent with RG 1.183, Appendix E, to evaluate the LOAC event.

3.1.2.3 Control Room Ventilation Assumptions for the LOAC

The radiological release associated with the LOAC event are not sufficient to cause the control room to isolate. Therefore, the licensee evaluated the control room dose consequences assuming that the control room ventilation system remained in the normal mode for the duration of the event.

3.1.2.4 LOAC Conclusion

The NRC staff reviewed the licensee’s evaluation of the radiological consequences resulting from the postulated LOAC event and concluded that the radiological consequences at the EAB, LPZ, and control room are within the requirements of 10 CFR 50.67(b)(2)(i)-(iii). The NRC staff finds 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 6, and the licensee’s calculated dose results are given in table 1 in section 4.0 of this SE. The NRC staff finds that the EAB, LPZ, and control room doses estimated by the licensee for the LOAC meet the applicable accident dose criteria and are therefore acceptable.

3.1.3 Primary Coolant Pump LRA

The LRA begins with the instantaneous seizure of an 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 low coolant flow causes a degradation of core heat transfer resulting in localized temperature and pressure changes in the core. The licensee’s evaluation indicates that the fuel will experience a departure from nucleate boiling which results in fuel cladding damage. Activity from the fuel cladding damage is transported to the secondary side due to primary-to-secondary side leakage.

3.1.3.1 Source Term

The licensee assumed that the instantaneous seizure of the RCP rotor associated with the LRA results in a small percentage of fuel cladding damage. The dose analysis for this event conservatively assumes 5 percent of the fuel corresponding to 9.65 fuel assemblies experience fuel cladding 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.

The licensee assumed that each of the failed assemblies contain 35 fuel rods with burnups exceeding 54 gigawatt-days per metric ton of uranium (GWD/MTU) and powers above 6.3 kilowatts per foot (kw/ft). Therefore, the licensee increased the assumed gap release fractions for these higher burnup assemblies with the release fractions from NUREG/CR-5009 for I-131 and the Alkali metals. The licensee increased the gap release fraction for Krypton (Kr)-85 to 0.30 and the gap release fractions for the remaining noble gasses and halogens to 0.10 to account for the effects of increased burnup. For the remainder of the fuel rods in the failed assemblies, the licensee used the gap release fractions from table 3, “Non-LOCA Fraction of Fission Product Inventory in Gap,” of RG 1.183.

The licensee applied the maximum radial peaking factor of 1.65 to calculate the activity releases from the damaged fuel. The licensee’s LRA source term model also includes the maximum TS equilibrium secondary coolant activity concentration of 0.1 μCi/gm DE I-131.

3.1.3.2 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 assumed the TS maximum design-basis leak rate of 1.0 gallon per minute (gpm). A LOOP 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. The licensee conservatively modeled the inventory in the secondary side of the SGs using the minimum water mass to maximize the iodine concentration and thereby maximizing the dose consequence. The licensee modeled the primary-to-secondary leakage with a density of 8.33 pounds mass (lbm) per gallon based on cooled liquid.

The licensee, utilizing the assumptions in RG 1.183, reduced 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. A partition coefficient of 100 was applied to the iodines and alkali metals. Because of their volatility, 100 percent of the noble gases are assumed to be released to the outside atmosphere. The licensee’s analysis assumes that the RHR system is placed into service 7.2565 hours after the accident at which time there are no further steam releases to the atmosphere.

3.1.3.3 Control Room Ventilation Assumptions for the LRA

The radiological releases calculated by the licensee for the LRA are sufficient to reach the control room intake radiation monitor setpoint isolating the control room. The NRC staff finds the control room assumptions used in the dose analysis acceptable and these assumptions are shown in table 4 in section 4.0 of this SE.

3.1.3.4 LRA Conclusion

The NRC staff reviewed the licensee's evaluation of the radiological consequences resulting from the postulated LRA and concluded that the radiological consequences at the EAB, LPZ, and control room are within the requirements of 10 CFR 50.67(b)(2)(i)-(iii). The NRC staff finds 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 7 and the licensee's calculated dose results are given in table 1 in section 4.0 of this SE. The NRC staff finds that the EAB, LPZ, and control room doses estimated by the licensee for the LRA meet the applicable accident dose criteria and are therefore acceptable.

3.1.4 REA

Section 15.4.8.3 of the Callaway FSAR describes the REA as a mechanical failure of a control rod mechanism pressure housing resulting in the ejection of a rod cluster assembly and drive shaft. The consequences of this mechanical failure are a rapid reactivity insertion together with an adverse core power distribution possibly leading to localized fuel rod damage.

The licensee evaluated two separate release scenarios for the REA. In the first case, the rod ejection (RE) is assumed to induce a LOCA resulting in a release of fission products into the containment atmosphere and a subsequent release to the environment from the containment leakage pathway.

For the second case, the radiological consequences from an RE are 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. The REA is analyzed with the assumption of a concurrent LOOP which causes steam releases from the secondary system to occur through the SG ASDs and safety valves to the environment.

3.1.4.1 Source Term

The source term for the REA is assumed to result in fuel damage consisting of localized damage to fuel cladding with a limited amount of fuel melt occurring in the damaged rods. The source term for the REA is described in RG 1.183, Appendix H, "Assumptions for Evaluating the Radiological Consequences of a PWR Rod Ejection Accident," Regulatory Position 1.

The licensee conservatively assumed that 10 percent of the fuel corresponding to 19.3 fuel assemblies experience fuel clad damage that results in the release of all their gap activity. The licensee assumed that each of the assemblies with fuel clad damage contain 35 fuel rods with burnups exceeding 54 GWD/MTU and powers above 6.3 kw/ft. Therefore, the licensee increased the assumed gap release fractions for these higher burnup rods. To account for the effects of increased burnup, the licensee increased the gap release fraction for Kr-85 to 0.30, I-131 to 0.12, and the Alkali metals to 0.17. For the remainder of the fuel rods in the failed assemblies that are not high burnup rods, consistent with the guidance provided in appendix H of RG 1.183, the licensee assumed that 10 percent of the core inventory of noble gases and iodine reside in the fuel gap and 12 percent of the core inventory of Alkali metals reside in the fuel gap, as specified in table 3 of RG 1.183. The licensee applied the maximum radial peaking factor of 1.65 to calculate the activity releases from the damaged fuel.

The licensee also assumed that 0.25 percent of the core reaches fuel centerline melt (FCM) as a result of the REA. In accordance with RG 1.183, in the REA induced LOCA scenario, for the 0.25 percent of fuel experiencing FCM, the licensee assumed that 100 percent of the noble gases, 25 percent of the iodines, and 12 percent of the Alkali metals in the affected fuel will be available for release into the containment. In the secondary side RE release scenario, for the 0.25 percent of fuel experiencing FCM, the licensee assumed that 100 percent of the noble gases, 50 percent of the iodines, and 12 percent of the Alkali metals in the affected fuel will be available for release to the RCS.

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 RE induced LOCA; 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 RE secondary side release scenario.

The licensee, utilizing the guidance in RG 1.183, Appendix H, Regulatory Position 4, assumed that the chemical form of radioiodine released to the containment atmosphere from both the fuel and the RCS consist of 95 percent CsI, 4.85 percent elemental iodine, and 0.15 percent organic iodide.

The licensee included the radiological dose contribution from the release of secondary coolant iodine activity at the TS limit of 0.1 µCi/gm DE I-131.

3.1.4.2 Transport from Containment

The licensee used the minimum containment free air volume to conservatively maximize the radioactive concentration in containment. In contrast, the licensee used the maximum containment free air volume for determining the containment leakage to conservatively maximize the value for containment leakage.

The licensee assumed 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 in the containment due to containment sprays or natural deposition. The licensee assumed that all containment leakage is at the TS limit of 0.2 percent per day for the first 24 hours and 0.1 percent per day thereafter.

3.1.4.3 Transport from Secondary System

The licensee, utilizing the guidance in RG 1.183, Appendix H, Regulatory Position 7, evaluated the transport of activity from the RCS to the SGs secondary side assuming a total primary to secondary leak rate equal to 1 gpm. The licensee assumed that this leak rate persists for a period of 8 hours until the RHR system is in operation 7.29 hours after the accident, and the RCS and SG pressures have equalized.

The licensee's leak rate testing results are adjusted so that the allowable leakage corresponds to a density of 8.33 lbm per gallon and accordingly this density was used to convert the volumetric leak rate to a total mass flow rate due to SG tube leakage in the REA dose consequence analysis.

The licensee, utilizing the guidance in RG 1.183, assumed 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, Regulatory Positions 5.5.1, 5.5.2, and 5.5.3, the licensee assumed that all the primary-to-secondary leakage in both SGs mixes with the secondary water without flashing.

The licensee applied the guidance in RG 1.183, Appendix E, Regulatory Position 4, to the release of iodine from the SGs incorporating a partition coefficient of 100 for particulate and elemental iodine as well as the alkali metals. The licensee, following the guidance in RG 1.183, assumed the speciation of the released iodines from a SG release to be 97 percent elemental iodine and 3 percent organic iodide.

3.1.4.4 Control Room Ventilation Assumptions for the RE

The radiological releases calculated by the licensee for the RE are sufficient to reach the control room intake radiation monitor setpoint isolating the control room. The NRC staff finds the control room assumptions used in the dose analysis acceptable, and these assumptions are shown in table 4 in section 4.0 of this SE.

3.1.4.5 Rod Ejection Accident Conclusion

The licensee evaluated the radiological consequences resulting from the postulated RE event and concluded that the radiological consequences at the EAB, LPZ, and control room are within the requirements of 10 CFR 50.67(b)(2)(i)-(iii). The NRC staff finds 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 8 and the licensee's calculated dose results are given in table 1 in section 4.0 of this SE. The NRC staff finds that the EAB, LPZ, and control room doses estimated by the licensee for the REA meet the applicable accident dose criteria and therefore are acceptable.

3.1.5 LLB

As described in section 15.6.2.1 of the Callaway FSAR, this analysis models the complete severance of the letdown line outside of containment, which would result in a loss of reactor coolant at the rate of 158.9 gpm. This release continues until operations personnel isolate the letdown line. Isolation is initiated at 30 minutes and valve closure takes 10 seconds. The total reactor coolant inventory lost is 39,958 lbm over 1,810 seconds. The LLB is used as a representative scenario for all small lines transporting reactor coolant inventory outside of containment since it poses the most severe consequences regarding radioactivity release based upon break size.

The LLB event is not discussed in RG 1.183 since this event is not assumed to result in fuel damage. The licensee used the guidance in RG 1.183 in conjunction with event guidance from section 15.6.2 of the SRP, Revision 2.

3.1.5.1 Source Term

Since no fuel failure results from the accident, consistent with SRP section 15.6.2, the licensee considered an accident-initiated spike in the RCS. The licensee assumed that the iodine spike increases the iodine release rate from the fuel to the RCS to a value 500 times greater than the appearance rate corresponding to the maximum equilibrium RCS concentration of 1.0 μCi/gm of

DE I-131. In addition, the licensee included the noble gas activity concentration in the RCS based on the TS value of 225 μCi/gm of DE Xe-133 and the alkali metal activity concentration in the RCS at a 1 percent fuel defect level. The licensee, following the general assumptions in RG 1.183, assumed that the iodine available for release to the atmosphere is 97 percent elemental and 3 percent organic.

3.1.5.2 Transport

The licensee assumed that 20 percent of the leaking coolant flashes to steam based on the temperature and pressure conditions of the letdown line flow. The licensee assumed that the iodine and alkali metal in this steam become airborne and available for release to the atmosphere and all the noble gases contained in the leaking primary coolant are released to the atmosphere with no mitigation or reduction.

3.1.5.3 Control Room Ventilation Assumptions for the LLB

The licensee determined that the control room intake radiation monitor setpoint is not reached for this event. Therefore, the licensee assumed that the control room remains in the normal mode of operation.

3.1.5.4 LLB Conclusion

The licensee evaluated the radiological consequences resulting from the postulated LLB and concluded that the radiological consequences at the EAB, LPZ, and control room are within the requirements of 10 CFR 50.67(b)(2)(i)-(iii). The NRC staff finds 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 9, and the licensee's calculated dose results are given in table 1 in section 4.0 of this SE. The NRC staff finds that the EAB, LPZ, and control room doses estimated by the licensee for the LLB meet the applicable accident dose criteria and therefore are acceptable.

3.1.6 SGTR with Failed ASD

The licensee evaluated the radiological consequences of an SGTR accident as a part of the full implementation of an AST. The SGTR accident is evaluated based on the assumption of an instantaneous and complete severance of a single SG tube. The postulated break allows primary coolant liquid to leak to the secondary side of the ruptured SG. Integrity of the barrier between the RCS and the main steam system is significant from a radiological release standpoint. The radioactivity from the leaking SG tube mixes with the shell-side water in the affected SG.

For the SGTR DBA radiological consequence analysis, a LOOP is assumed to occur shortly after the trip signal. As a result of low pressurizer pressure, the reactor trips followed by a turbine trip. The turbine trip is assumed to cause a LOOP which renders the main condenser unavailable. With the condenser unavailable the plant is cooled down by releasing steam to the environment through the SG ASDs or the MSSVs.

The licensee evaluated two cases of concern with respect to a radioactive release associated with a SGTR. The first case evaluated is the failure of an ASD on the ruptured SG in the open position leading to continued release of SG fluid and contained radioactivity. The second case evaluated the potential for overfill of the ruptured SG with water entering the main steam line resulting in water relief through an ASD valve and/or an MSSV.

For the case of an SGTR with postulated stuck open ASD on the ruptured SG, the licensee maximized the radioactive releases by assuming the ruptured SG ASD is stuck-open for 20 minutes. For the case of an SGTR with postulated failure of the ruptured SG auxiliary feedwater (AFW) flow control valve, the licensee maximized AFW flow to increase the probability for the ruptured SG to overfill and to maximize subsequent liquid relief from its safety valve. The licensee maximized radioactive releases by assuming that the safety valve is stuck-open following liquid relief with an effective flow area equal to 5 percent of the total safety valve flow area. The details of the overfill case are discussed in section 3.1.7 of this SE.

For the failed ASD case, the SG blowdown will automatically be terminated by the SG blowdown and sample isolation signal AFW actuation signal which is initiated by the SI signal. The assumed coincident LOOP will cause closure of the condenser steam dump valves to protect the condenser. The SG pressure will then increase rapidly resulting in steam discharge as well as activity release through the SG ASD valves.

The licensee conservatively assumed that, in addition to the assumed failed open ASD on the ruptured SG, an ASD valve on one of the unaffected SGs fails to open and will therefore be unavailable to support the RCS cooldown. This assumption has the effect of increasing the time it takes to reduce the RCS temperature to below the ruptured SG saturation temperature. The licensee notes and the NRC staff agrees that this additional failure is beyond the required single failure criterion. Venting from the SG, which experiences the tube rupture, will continue until the manual block valve is closed isolating the stuck-open ASD valve on the ruptured SG thereby terminating the release from the ruptured SG. The remaining unaffected SGs remove core decay heat by venting steam through the ASD valves until the controlled cooldown is terminated.

3.1.6.1 Source Term

The licensee determined that the SGTR will not result in fuel damage. Therefore, the licensee based the source term for the SGTR on the two iodine spike cases from RG 1.183, which are described in section 3.1.1 of this SE for the MSLB.

3.1.6.2 Transport

The licensee, in accordance with the CLB for the SGTR, took credit for the ability of the large area in the condenser to provide a significant amount of iodine plateout. The licensee applied a decontamination factor (DF) of 100 to the iodine releases for the first 10 minutes of the accident prior to the reactor trip.

As discussed for the MSLB, the licensee applied the guidance in RG 1.183, Appendix E, Regulatory Position 4, to the release of iodine from the SGs incorporating a partition coefficient of 100 for iodine and the alkali metals. The licensee, utilizing the guidance in RG 1.183, assumed that the iodine speciation for a SG release would be 97 percent elemental iodine and 3 percent organic iodide. For the SGTR, the licensee included the radiological dose contribution from the release of secondary coolant iodine activity at the TS limit of 0.1 μCi/gm DE I-131.

3.1.6.3 Control Room Ventilation Assumptions for the SGTR with Failed ASD

The licensee's evaluation of the SGTR indicates that the control room intake radiation monitor setpoint will be reached isolating the control room and initiating the control building and control room emergency mode of operation. For the dose consequence analysis, the licensee credited the control room emergency mode to begin 120 seconds after the event based on 60 seconds for the control room isolation with an additional 60 seconds for conservative margin.

3.1.6.4 SGTR with Failed ASD Conclusion

The licensee evaluated the radiological consequences resulting from the postulated SGTR accident with a failed ASD and concluded that the radiological consequences at the EAB, LPZ, and control room are within the requirements of 10 CFR 50.67(b)(2)(i)-(iii). The NRC staff finds 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 10, and the licensee's calculated dose results are given in table 1 in section 4.0 of this SE. The NRC staff finds that the EAB, LPZ, and control room doses estimated by the licensee for the SGTR accident with a failed ASD meet the applicable accident dose criteria and are therefore acceptable.

3.1.7 SGTR with SG Overfill

The licensee evaluated the radiological consequences of an SGTR accident as a part of the full implementation of an AST. The SGTR accident is evaluated based on the assumption of an instantaneous and complete severance of a single SG tube. The postulated break allows primary coolant liquid to leak to the secondary side of the ruptured SG. Integrity of the barrier between the RCS and the main steam system is significant from a radiological release standpoint. The radioactivity from the leaking SG tube mixes with the shell-side water in the affected SG. For the SGTR DBA radiological consequence analysis, a LOOP is assumed to occur shortly after the trip signal. As a result of low pressurizer pressure, the reactor trips followed by a turbine trip. The turbine trip is assumed to cause a LOOP which renders the main condenser unavailable. With the condenser unavailable, the plant is cooled down by releasing steam to the environment through the SG ASDs or the MSSVs.

The licensee evaluated two cases of concern with respect to radioactive release associated with an SGTR. The first case evaluated is the failure of an ASD in the open position leading to continued release of SG fluid and contained radioactivity. The second case evaluated the potential for overfill of the ruptured SG with water entering the main steam line resulting in water relief through an ASD valve and/or an MSSV.

For the case of an SGTR with postulated stuck open ASD on the ruptured SG, the licensee maximized the radioactive releases by assuming the ruptured SG ASD is stuck-open for 20 minutes. For the case of an SGTR with postulated failure of the ruptured SG AFW flow control valve, the licensee maximized AFW flow in order to increase the probability for ruptured SG overfill and to maximize subsequent liquid relief from its safety valve. The licensee maximized radioactive releases by assuming that the safety valve is stuck-open following liquid relief with an effective flow area equal to 5 percent of the total safety valve flow area. The details of the failed ASD case are discussed in section 3.1.6 of this SE.

The licensee evaluated the potential effects of an SG overfill for a complete investigation of the potential effects of an SGTR. The analysis assumes a complete severance of a single SG tube while the reactor is operating at full rated power, a coincident LOOP, and the failure of the AFW control valve on the discharge side of the motor-driven AFW pump feeding the ruptured SG. The ASD on the ruptured SG is not assumed to fail open. Rather the ASD on the ruptured SG fails to open and all liquid relief is considered through a MSSV. The AFW control valve is assumed to fail in the wide-open position which maximizes the flow to the ruptured SG. For the overfill scenario, the licensee conservatively assumed that the reactor trip and SI actuation occurs at SGTR initiation, which maximizes the AFW addition to the ruptured SG. This assumption also results in the assumed LOOP occurring at event initiation eliminating the ability to credit condenser releases prior to reactor trip as in the failed ASD case.

3.1.7.1 Source Term

The licensee determined that the SGTR will not result in fuel damage. Therefore, the licensee based the source term for the SGTR on the two iodine spike cases from RG 1.183, which are described in section 3.1.1 of this SE for the MSLB.

3.1.7.2 Transport

As discussed for the MSLB, the licensee applied the guidance in RG 1.183, Appendix E, Regulatory Position 4, to the release of iodine from the SGs incorporating a partition coefficient of 100 for iodine and the alkali metals. The licensee, in accordance with the guidance in RG 1.183, assumed that the iodine speciation for a SG release would be 97 percent elemental iodine and 3 percent organic iodide. For the SGTR, the licensee included the radiological dose contribution from the release of secondary coolant iodine activity at the TS limit of 0.1 μCi/gm DE I-131.

The licensee evaluated the dose consequences for a SGTR occurring while the plant is at 100 percent thermal power. The licensee conservatively assumed that the reactor trip and coincident turbine trip and LOOP occur concurrent with the SGTR making the condenser unavailable for plant cooldown. As an additional conservativism for the overfill case, the licensee assumed that the AFW also starts immediately and that the AFW control valve fails open on the ruptured SG. The reactor trip generates an SI signal to initiate SI and normal feedwater stops and the main steam isolation valves (MSIVs) close.

The licensee assumed that the operators would terminate turbine-driven AFW flow at approximately 10 minutes and the motor-driven AFW flow at 20 minutes both in response to a high and increasing SG level. The licensee assumed that before the AFW is completely isolated, the combination of AFW and ruptured tube flow causes the level in the ruptured SG to overfill which also results in filling of the steam line up to the MSIV. This causes the pressure to spike and water containing the assumed source term activity is released through an MSSV while the ASD on the ruptured loop steam line fails closed. Water discharging through the MSSV causes it to fail open 5 percent and the liquid release continues until approximately 84 minutes after the event after which only steam is released through the failed open MSSV for the remainder of the event.

After the rupture flow is terminated (about 64 minutes), a cooldown to RHR conditions is initiated by opening the intact ASDs. Steam release from the intact SGs terminates when RHR conditions are reached at approximately 6.4 hours post-accident and the RHR system can be activated to provide cooling capacity to the plant in lieu of steam release through the ASDs. The steam release from the ruptured SG through the failed open MSSV continues until the SG inventory has cooled to 212°F at approximately 9.4 hours post-accident.

Break flow is transferred to the ruptured SG beginning at event initiation and terminating at approximately 1.1 hours. The ruptured SG tubes do not experience uncover, so flashing does not occur. The entire 1 gpm primary-to-secondary leakage allowed by the TS is assumed to be leaking into the intact SGs with a density based on cooled liquid. This leakage begins at event initiation and continues until RHR conditions are reached and the ASDs are closed at approximately 6.4 hours post-accident. Since the intact or unaffected SG tubes are uncovered briefly, a portion of this flow is assumed to be released directly to the environment through flashing, which is calculated based on the primary side hot leg and SG secondary side fluid enthalpies. Flashed break flow begins at approximately 1.7 hours and terminates at 2.5 hours post-accident.

Liquid release to the environment due to overfill through the failed open MSSV on the faulted SG starts at approximately 0.26 hours and is terminated at approximately 1.4 hours post-accident. The licensee conservatively assumed that 50 percent of the activity in the liquid release becomes airborne and contributes to the offsite and control room doses. The licensee states, and the NRC staff agrees, that this is conservative relative to the value of 10 percent from RG 1.183, Appendix A, "Assumptions for Evaluating the Radiological Consequences of a LWR Loss-of-Coolant-Accident," Regulatory Position 5.5, for the evaluation of the airborne release from ECCS leakage.

3.1.7.3 Control Room Ventilation Assumptions for the SGTR with SG Overfill

The licensee's evaluation of the SGTR indicates that the control room intake radiation monitor setpoint will be reached isolating the control room and initiating the control building and control room emergency mode of operation. For the dose consequence analysis, the licensee credited the control room emergency mode to begin 120 seconds after the event based on 60 seconds for the control room isolation with an addition 60 seconds for conservative margin.

3.1.7.4 SGTR with SG Overfill Conclusion

The licensee evaluated the radiological consequences resulting from the postulated SGTR accident with SG overfill and concluded that the radiological consequences at the EAB, LPZ, and control room are within the requirements of 10 CFR 50.67(b)(2)(i)-(iii). The NRC staff finds 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 11, and the licensee's calculated dose results are given in table 1 in section 4.0 of this SE. The NRC staff finds that the EAB, LPZ, and control room doses estimated by the licensee for the SGTR accident with SG overfill meet the applicable accident dose criteria and are therefore acceptable.

3.1.8 LOCA

The radiological consequence design-basis LOCA analysis is a deterministic evaluation based on the assumption of a major accident resulting in a substantial core melt with the release of a substantial amount of radioactivity into the containment. A possible cause for this accident could be the rupture of the primary RCS piping. The accident scenario, as specified in RG 1.183, assumes the deterministic failure of the ECCS to provide adequate core cooling which results in a significant amount of core damage. This general scenario does not represent any specific accident sequence but is representative of a class of severe damage incidents that were evaluated in the development of the RG 1.183 source term characteristics. Such a scenario would be expected to require multiple failures of systems and equipment and lies beyond the severity of incidents evaluated for design-basis transient analyses.

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.

The licensee assumed that an abrupt failure of the main reactor coolant pipe occurred allowing the radioactivity in the RCS to be released into containment. The licensee also assumed that a portion of this activity is released to the atmosphere via the mini-purge system prior to containment isolation. It is assumed that the emergency core cooling features fail to prevent the core from experiencing significant fuel melting. This sequence cannot occur unless there are multiple failures and thus goes beyond the typical DBA that considers a single active failure. Activity from the core is released to the containment and from there is released to the environment by means of containment leakage. In addition, once recirculation of the ECCS is established, iodine activity in the sump solution may be released to the environment by means of leakage from engineered safety feature (ESF) equipment outside containment in the auxiliary building and by means of back-leakage from the equipment to the reactor water storage tank (RWST) with subsequent leaking or venting of the RWST. The total offsite doses are the sum of the doses resulting from the four postulated release paths. The total onsite doses for the control room are the sum of the doses resulting from the four postulated release paths plus the dose due to external shine and operator access dose.

3.1.8.1 Containment Leakage

For the containment leakage pathway, the licensee assumed that all activity released from the fuel migrates into the unsprayed portion of containment before being mixed with the sprayed portion of the containment. The licensee evaluated the time-dependent removal of elemental iodine and particulates from the containment atmosphere by crediting removal by containment spray, radioactive decay, and leakage from containment. The licensee only credited noble gases and organic iodine removal by radioactive decay and leakage from containment.

The licensee modeled the maximum free volume of the containment for the containment leakage pathway as 2,000,000 cubic feet (ft^3). Since the licensee determined that the volume of the sprayed region is less than 90 percent of the containment free volume the licensee modeled the region covered by spray drops (85 percent) and the unsprayed region (15 percent) separately. The licensee assumed a mixing rate between the sprayed and unsprayed regions to be two turnovers of the unsprayed regions per hour or 13,500 cfm. This assumption follows the guidance in RG 1.183, Appendix A, Assumption 3.3. The licensee conservatively assumed that the mixing would be delayed for 2 minutes but continue for the remainder of the event.

The licensee, utilizing the guidance in RG 1.183, assumed that the containment leaks at the design leak rate of 0.2 percent per day for the first 24 hours of the accident and then leaks at half that rate (0.1 percent per day) for the remainder of the 30-day period considered in the analysis. The licensee credited containment spray actuation at 2 minutes following accident initiation and termination of containment sprays at 4 hours following accident initiation. The licensee followed the methodology described in section 6.5.2 of the SRP for the determination of containment spray removal of particulates and elemental iodine. The NRC staff has reviewed the licensee's application of credit for iodine removal from the operation of the containment spay system and found the analysis acceptable because it is conservative and follows the applicable regulatory guidance.

3.1.8.2 ECCS Leakage

For the ECCS leakage pathway, the licensee assumed that all iodine activity released from the fuel is in the sump solution immediately. The only removal of activity from the sump is by radioactive decay or leakage to the auxiliary building. The sump volume is 428,000 gallons. When ECCS recirculation is established following the LOCA, leakage is assumed to occur from ESF equipment in the auxiliary building. Recirculation is modeled to initiate at the time of control room isolation (62 seconds) and continues throughout the event. Normally, licensees model the ECCS leakage contribution to the LOCA dose analysis to start at the time the recirculation phase begins which is 11.8 minutes for this facility. The NRC staff notes that by modeling the recirculation mode to start at 62 seconds post-accident, the licensee added an additional conservatism in the dose consequence analysis beyond the conservative assumptions detailed in RG 1.183.

The leakage to the auxiliary building is modeled at a rate of 2 gpm. The leakage value was doubled in accordance with RG 1.183. The analysis assumes that 10 percent of the iodine activity in the leakage becomes airborne and is available for release to the environment. The activity of the airborne leakage is further reduced as it is released through the auxiliary building vent filters with 90 percent efficiency for all forms of iodine.

According to NUREG-1465, iodine released from the damaged core to the containment after a LOCA is composed of 95 percent CsI which is a highly ionized salt that is 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. Trisodium phosphate dodecahydrate (TSP) is used as the buffering agent at Callaway. The TSP is introduced into the containment sump fluid via two stationary baskets following a postulated LOCA. After a LOCA, several acids are either generated or are added to the containment. Relative amounts of these acids and that of TSP determine the value of pH reached by the containment sump water. The buffering action provided by the TSP is intended to maintain basic pH in the suppression pool despite the presence of strong acids.

In order to verify that the dissolution of TSP will provide sufficient buffering action to maintain pH above 7, the licensee performed calculations using the methodology developed from NRC research contained in NUREG/CR-5950. Assumptions made by the licensee with respect to input parameters to the calculation were designed to produce the lowest possible calculated pH. By demonstrating that the suppression pool pH remains above 7 with all parameters conservatively adjusted to create the lowest possible pH condition, the licensee ensures that under realistic plant conditions the pH threshold will not be challenged.

The licensee considered the effects of strong acid generation on the post-LOCA sump pH. Generation of strong acids inside containment is the primary phenomena that will lower suppression pool pH following a LOCA. Per the guidance in NUREG/CR-5950, the licensee calculated the mass of nitric acid (HNO_3) generated by the radiolysis of air and water inside containment and the mass of hydrochloric acid (HCL) generated by the radiolysis of electrical cable insulation inside containment. The NRC staff verified that the licensee used conservative assumptions when calculating strong acid generation. For example, the quantity of HCL generated was based on a conservatively large mass of electrical cable insulation (65,820 lbs.) versus the quantity present in containment. The NRC staff referenced data from similar plant designs to confirm the conservatism in the assumed cable mass.

Sources of boron include the RWST and the SI accumulators. The maximum TS concentrations of boron for RWST and SI accumulator sources (2500 parts per minute (ppm)) was used in the calculations in order to minimize the calculated sump pH. Similarly, a maximum RCS boron concentration of 1500 ppm was assumed. Calculations were performed for both maximum and minimum fluid volume since the sump volume determines the level of water interacting with the TSP baskets and therefore impacts dissolution rate for the TSP. Calculations demonstrated that both the minimum and maximum volume cases met the long-term-pH acceptance criteria of a pH greater than 7.1 consistent with the guidance in NUREG/CR-5950. The limiting case which took the longest time to achieve pH greater than 7.0 was determined to be the minimum volume and level calculation. For this scenario, the pH was greater than 7.0 in approximately 78 minutes and greater than 7.1 in 103 minutes. The minimum long term (30 day) equilibrium sump pH was calculated to be between 7.2 and 7.4 for the scenarios analyzed.

The NRC staff has independently verified the licensee's calculations and finds that by using TSP as a buffer, the pH of the suppression pool will remain above a pH of 7 for 30 days post-LOCA.

3.1.8.3 RWST Back-Leakage

For the RWST back-leakage pathway, the licensee assumed that a portion of the ECCS recirculation leaks into the RWST. All iodine activity released from the fuel is assumed to be in the sump solution immediately. The only removal of activity from the sump is by radioactive decay or leakage to the RWST. The sump volume is 428,000 gallons. Recirculation is modeled to initiate at the time of control room isolation (62 seconds) and continues throughout the event.

Leakage to the RWST is modeled at a rate of 4 gpm (3 gpm below the water line and 1 gpm above the water line). The activity is modeled to be delivered directly to the gas filled portion of the RWST; however, only 10 percent of the activity in the 1 gpm becomes airborne for the first 24 hours of the event and 8 percent thereafter is available for release to the environment. The vast majority of the radioiodine in the 3 gpm delivered below the water line is retained in the liquid remaining in the RWST. This retention in the liquid is supported by a calculation performed in accordance with NUREG/CR-5950, accounting for gradual changes in pH and iodine concentration in the RWST liquid.

The release rate from the RWST to the environment is based on the volume displacement from the incoming leakage. An adjustment is made to account for a reduction in the RWST gas volume available for dilution as the leakage into the RWST increases the water level. The RWST gas volume is decreased by the rate of leakage into the RWST.

3.1.8.4 Containment Purge Contribution to the LOCA Dose

For the containment purge system release pathway, all of the initial primary coolant activity is instantly released from the RCS and is evenly distributed throughout the containment volume. The total initial mass in containment modeled in this pathway is 723,290 lbm (551,068 lbm from the RCS and 172,222 lbm as air). The only removal of activity from containment is by radioactive decay or the purge flow. The maximum air flow rate from containment to the environment as a function of time is modeled until the purge line is isolated at 11 seconds.

Instead of simply using the normal mini-purge flow rate of 4,000 cfm, the air flow that was utilized reflects additional flow resulting from the increase in containment pressure resulting from the mass and energy release from the RCS. The assumed iodine chemical fractions do not impact the analysis results since spray removal is not credited for removal of RCS activity in containment.

The assumed iodine chemical fractions for the containment purge pathway do not impact the analysis results since neither spray nor filtration is credited for removal of the RCS activity in the release path. Also, since control room isolation does not occur until after mini-purge is isolated, no filtration of air flow to the control room is modeled.

3.1.8.5 Control Room Ventilation Assumptions for the LOCA

In the event of a LOCA, the low pressurizer pressure SI setpoint will be reached almost immediately following the break. The SI signal causes the control room to switch from the normal operation mode to the emergency operation mode. The switch from the normal operation mode to the emergency operation mode is conservatively modeled at 62 seconds following event initiation, which includes a 60-second delay from the initiating signal.

The licensee determined that with the significant amount of control room shielding accounted for, the dose to control room personnel from external sources would be 0.012 rem TEDE. These external sources include the activity remaining in containment following the LOCA, the activity cloud outside of the control room in the environment, and the activity buildup on recirculation filters. This is added to the dose calculated from the four release paths discussed above.

The licensee made several changes to the CLB control room unfiltered inleakage assumptions to account for the unfiltered inleakage into the control room equipment room. The licensee updated the CLB two-zone model to a more representative three-zone model to evaluate the potential contribution to control room dose from unfiltered inleakage into the control room equipment room. While the CLB models the interaction between the control room and the control building, the revised three-zone model used in the AST analysis includes the ventilation interactions with the control room, the control building, and the control room equipment room.

To measure the control room unfiltered inleakage, the licensee conducted inleakage measurement tracer gas tests using the Brookhaven National Laboratory (BNL) Atmospheric Tracer Depletion (ATD) Method in 2004, 2011, and 2018. Reviews performed by BNL personnel of the CREVS design and operation determined that although the plant's CLB is based on two interacting zones for CREVS (i.e., the control room and control building), there are three zones involved for proper inleakage and dose analyses. The noted testing provides results applicable to the three zones (the control room, the control building, and the control room HVAC filtration equipment rooms).

The NRC staff notes that results from these tests indicate unfiltered inleakage of less than 20 cfm for the control room, less than 170 cfm for the control building, and less than 100 cfm for the control room equipment room. To provide additional margin, the licensee assumed unfiltered inleakage values in the dose consequence AST analysis, as shown in table 4 in section 4.0 of this SE, that are significantly higher than the test results.

3.1.8.6 LOCA Control Room Transit Dose Evaluation

The licensee calculated the dose to control room personnel due to transit to and from the control room. The transit dose calculations included contributions from inhalation and immersion (cloud shine) as well as the direct shine dose from ground deposition along the operator's path between the control room and the parking lot. The licensee, using conservative assumptions, determined that the transit dose for control room access would be 0.81 rem TEDE. The licensee added the transit dose to the dose calculated from the four release paths discussed above to calculate the total control room dose for the LOCA.

The licensee conservatively assessed a transit dose due to ground shine from the deposition of radionuclides along the control room access pathway. The dispersion factors for DBA dose consequence analyses typically do not credit deposition to maximize the dose from inhalation and immersion. Calculating a ground shine dose from deposition adds conservatism to the analysis since the inhalation and immersion doses are based on atmospheric dispersion coefficients that include all released activity including the radionuclides that are assumed to be deposited in the ground shine calculation. Further, since the control room dose analysis covers a 30-day period, realistic emergency provisions such as decontaminating the control room access pathway and providing respirators would be expected to significantly reduce both the external and internal dose contributions during control room access and egress. The NRC staff confirmed that the licensee used conservative assumptions in their assessment and notes that the use of more realistic assumptions based on expected emergency response actions would significantly reduce the licensee's control room transit dose estimation.

3.1.8.7 LOCA Conclusion

The licensee evaluated the radiological consequences resulting from the postulated LOCA and concluded that the radiological consequences at the EAB, LPZ, and control room are within the requirements of 10 CFR 50.67(b)(2)(i)-(iii). The NRC staff finds 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 12, and the licensee's calculated dose results are given in table 1 in section 4.0 of this SE. The NRC staff finds that the EAB, LPZ, and control room doses estimated by the licensee for the LOCA meet the applicable accident dose criteria and are therefore acceptable.

3.1.9 FHA

FHAs are described in section 15.7.4.5 of the FSAR. The licensee evaluated the postulated FHA for two cases: (1) a FHA outside the containment in the fuel handling building (FHB), and (2) a FHA inside the containment. The accident is defined as the dropping of a spent fuel assembly onto the fuel storage area floor, refueling pool floor, or cask loading pool. All the fuel rods contained in the dropped assembly are assumed to be damaged. Additionally, the dropped assembly is assumed to damage 20 percent of the rods of an additional assembly for the accident in the containment.

3.1.9.1 Source Term

The licensee used the analytical methods and assumptions outlined in RG 1.183, Appendix B, "Assumptions for Evaluating the Radiological Consequences of a Fuel Handling Accident," for the FHA radiological consequences. The licensee assumed that all the fuel rods in the dropped assembly are damaged to the extent that all their gap activity is released. For the FHA in the FHB, the licensee assumed that the dropped assembly contains 32 fuel rods with burnups exceeding 54 GWD/MTU and powers above 6.3 kw/ft. For the FHA in the containment, the licensee assumed that the dropped assembly as well as the additional assembly impacted by the drop contain 64 fuel rods with burnups exceeding 54 GWD/MTU and powers above 6.3 kw/ft. Accordingly, the licensee adjusted the gap fractions in a similar manner to the adjustments made to the gap fractions used in the LRA evaluation described in section 3.1.3 of this SE. The licensee applied a radial peaking factor of 1.65 to calculate the activity released from the damaged fuel.

The licensee assumed a decay time of 72 hours in determining the inventory of the damaged rods which supports the TS Bases B 3.9.7 assumption of 72 hours decay time prior to movement of irradiated fuel.

3.1.9.2 Transport

The licensee, utilizing the guidance in RG 1.183, took credit for an overall pool DF for iodine of 200 for a pool depth of 23 feet. As described in RIS-2006-04, the overall iodine DF of 200 is based on a DF of 285 for elemental iodine and a DF of 1 for organic iodine, as well as chemical fractions of 99.85 percent for elemental iodine and 0.15 percent for organic iodine. Thus, the normalized split between elemental and organic iodine leaving the pool is 70 percent for elemental iodine and 30 percent for organic iodine.

The licensee, utilizing the guidance in RG 1.183, assumed that all activity released from the fuel pool is released to the atmosphere in 2 hours and the licensee took no credit for mixing or holdup in the FHB atmosphere. The licensee also took no credit for filtration by the ESF emergency filtration system.

The licensee assumed that all the gap activity available for release is released over 2 hours for the FHA in the containment. The licensee took no credit for mixing or holdup in the containment atmosphere. No credit is taken for isolation of containment. The accident analysis assumes that a direct pathway exists between containment and the atmosphere for the duration of the release. No credit is taken for filtration.

3.1.9.3 Control Room Ventilation Assumptions for the FHA

The licensee's TS does not require the radiation monitors at the control room air intakes to be operable during movement of irradiated fuel assemblies in the FHB. Instead, the high gaseous fuel building exhaust radiation channels GG-RE-27 and GG-RE-28 actuate both the emergency exhaust system (EES) and control room isolation. For these channels, the Hi Alarm setpoint is 3.2×10^{-3} micro curies per cubic centimeter (μCi/cc) Xe-133.

The high gaseous fuel building exhaust radiation Hi Alarm setpoint is reached in the analysis. Since the analysis assumes no delays in the release of activity and no delays in the transport from the release point to the detector, the Xe-133 activity concentration at the detector immediately following event initiation (72 hours after shutdown) is 0.63 μCi/cc. This value is significantly greater than the setpoint. Therefore, an instantaneous generation of the high radiation signal may be assumed.

However, control room isolation is assumed to occur 120 seconds after event initiation; a 60 second delay is allowed for the detector to reach its setpoint, and an additional 60 seconds is allowed for control room isolation once the setpoint has been reached.

For irradiated fuel movements in the containment, the licensee's TS require that the control room intake radiation monitors be operable. The licensee determined that the control room intake radiation monitor signal setpoint is reached in the analysis of the FHA in the containment. Since the analysis assumes no delays in the release of activity and no delays in the transport of activity through the primary and secondary systems or in the transport from the release point to the air intake, the activity concentration at the detector immediately following event initiation is 9.65×10^{-2} μCi/cc Xe-133. The analysis models a detector setpoint of 2.2×10^{-3} μCi/cc Xe-133. Therefore, an instantaneous generation of the high radiation signal could be assumed. For conservatism, the licensee assumed the control room isolation occurs 120 seconds after event initiation which includes a 60 second delay for the detector to reach its setpoint and an additional 60 seconds for control room isolation once the setpoint has been reached.

3.1.9.4 Fuel Handling Accident Conclusion

The licensee evaluated the radiological consequences resulting from a postulated FHA and concluded that the radiological consequences at the EAB, LPZ, and control room are within the dose guidelines provided in 10 CFR 50.67 and accident specific dose criteria specified in section 15.0.1 of the SRP. The NRC staff finds 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 tables 13 and 14 in section 4.0 of this SE. The licensee's calculated dose results are given in table 1 of this SE. The NRC finds that the EAB, LPZ, and control room doses estimated by the licensee for the FHA meet the applicable accident dose criteria and are therefore acceptable.

3.1.10 Safety-Related Piping

In enclosure 1 of section 2.2.2 of the LAR, the licensee addresses CREVS design and operation. This part of the technical evaluation addresses the effect of AST adoption on Callaway's safety-related piping, HVAC system, dampers, and ductwork.

In response to request for additional information (RAI) No. 4.a by supplemental letter dated July 5, 2022, on whether the adoption of the AST affects any of the safety-related piping, the licensee provided the following information. The licensee stated that "Callaway's adoption of the Alternative Source Term (AST) relies on the seismic qualification of the safety related piping connecting the containment recirculation sump to the Refueling Water Storage Tank (RWST) for consideration of RWST back-leakage in the event of a Loss of Coolant Accident (LOCA)."

In accordance with NRC IN 2012-01, all flow paths above and below the normal water level of the RWST are: designed, installed, maintained, and qualified to seismic Category I criteria in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code), Section III, Class 2. Those flow paths are also isolated from non-seismic Category I piping by redundant automatic isolation valves which close on a SI signal and fail closed on loss of power, or isolated from non-seismic Category I piping by a locked closed isolation valve.

The design and isolation capability of Callaway safety-related piping make them acceptable for AST adoption. Based on a review of the information provided by the licensee, the NRC staff concludes that the affected safety-related piping is adequately designed for AST adoption.

HVAC Systems, Dampers, Piping, and Ductwork

In response to RAI No. 4.b by supplemental letter dated July 5, 2022, on whether any safety-related HVAC system, dampers, and ductwork are credited in the AST adoption, the licensee stated, in part, that as part of the AST adoption at Callaway, credit is taken for certain safety-related HVAC systems:

> During a Loss of Coolant Accident (LOCA), the Auxiliary Building (AB) emergency exhaust filtration is credited for the ECCS leakage case during recirculation.
>
> During a Fuel Handling Accident (FHA) in the Fuel Handling Building (FHB), Emergency Exhaust from the FHB is credited without filtration.
>
> Safety-grade filtration of outside makeup air and recirculated air is credited during emergency mode operation in the AST dose analyses.
>
> Actuation of emergency control room HVAC mode is not credited for certain accidents. … Acceptable control room doses were calculated with a maximum unfiltered inleakage of 6000 cfm to the control building, 60 cfm to the control room, and 300 cfm to the equipment room.

The CREVS consists of the supporting piping, electrical supply, instrumentation, ductwork, and dampers. Each of these components is designed to seismic Category I criteria (RG 1.29, "Seismic Design Classification," (Reference 30)) and qualified either by test, analysis, or a combination thereof. All the power supplies and control functions necessary for safe functioning of the control room air-conditioning system at Callaway are Class IE and designed, installed, and qualified to seismic Category I criteria.

Control Room Filter Adsorber Units (FGK01A/B), and Control Room Pressurization Filter Adsorber Units (FGK02A/B) were qualified by analysis and test. The filter adsorber units were qualified by finite element analysis with discrete components qualified by test in accordance with the Institute of Electrical and Electronics Engineers (IEEE) 344-1975, "IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations." The licensee stated that Control Room Air Conditioning Units (SGK04A/B) were qualified by test in accordance with IEEE 344-1975.

Regarding the Control Room Filtration Fans (CGK03A/B) and Control Room Pressurization Fans (CGK04A/B), the licensee stated that the fans were qualified by analysis in accordance with IEEE 344-1975 and the motors were qualified by analysis/test in accordance with IEEE 323-1974, "IEEE standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations" and IEEE 344-1975.

The licensee stated that the "Isolation Dampers are categorized as safety related seismic Category I components and qualified in accordance with IEEE 344-1975 by a combination of test and analysis."

The licensee stated that the Pressure Piping servicing these equipment items and pipe supports are designed, analyzed, and installed in accordance with the ASME Code, Section III, Class 3.

The licensee also stated that the ductwork was designed, analyzed, and installed as safety related, seismic Category I items and supported based on ASME/ANSI [ASME/American National Standards Institute] standards for HVAC systems following the guidance of RG 1.52, "Design, Inspection, Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered -Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants" (Reference 31).

The NRC reviewed the licensee's response related to the impact of Callaway's AST adoption on safety-related piping, control room HVAC system, safety-related piping and supports, ductwork and dampers. Based on its review, the NRC finds that these components are adequately designed to the applicable standards to support Callaway's adoption of the AST.

<u>Summary</u>

Based on its review, the NRC finds that affected HVAC system, safety-related piping and supports, dampers, and ductwork are adequately designed to the applicable standards to support Callaway's adoption of the AST.

3.2 <u>Atmospheric Dispersion Estimates</u>

The NRC staff evaluated the licensee's offsite and onsite, accident-related, atmospheric dispersion modeling analyses, related meteorological data, and other model inputs. This dispersion modeling occurs upstream of the dose analyses and provides direct inputs to the determination of the accident-related doses based on the licensee's proposed implementation of the AST in RG 1.183.

In general, the NRC staff reviewed the meteorological monitoring program, its data, and the atmospheric dispersion modeling portions of the LAR application in accordance with SRP sections 2.3.3 and 2.3.4. The licensee's offsite and onsite dispersion analyses used the PAVAN-NAI and the ARCON96-NAI codes, respectively. These models are adapted versions of two NRC-approved dispersion models (i.e., PAVAN and ARCON96). Additional information specific to each of the licensee's models and the codes used by the NRC staff in performing their confirmatory analyses are discussed in their respective sections below.

3.2.1 Meteorological Measurements and Dispersion Model Input Data

According to the FSAR, the meteorological monitoring program was established at the present location in 1973 to support the facility's construction permit phase. It was sited and based on the monitoring guidance in Safety Guide 23. This monitoring program continues during the plant's operational phase. Instrumentation and related features have been periodically upgraded over the years. However, Safety Guide 23 remains the CLB for the onsite meteorological monitoring program as is the case for other facilities in the existing fleet.

For this LAR, the licensee used meteorological data from its onsite monitoring program for a 4-year period of record (POR) from January 1, 2013, through December 31, 2016. Data recovery was well above the 90 percent criterion for the individual weather elements of wind speed, wind direction, and atmospheric stability, as well as the joint (concurrent) recovery of these model input parameters for the individual years and the composite POR.

Atmospheric stability is determined based on ambient temperature difference measurements (or the delta-T) between the 60-meter (m) minus the 10-m levels of the Callaway meteorological tower. Stability classes (A through G) are defined in table 2, "Classification of Atmospheric Stability," of Safety Guide 23 as the equivalent delta-T over a 100-m vertical difference of separation. However, the FSAR text as written is somewhat inconsistent, as is the output of the PAVAN-NAI model. A literal reading of both suggests that the temperature difference for a given hour may have been determined between the concurrent values at 10-m minus the 60-m measurement levels. The NRC staff's review confirmed that the delta-T values were calculated in accordance with the guidance in Safety Guide 23.

Scalar (or arithmetic) averaging was used to determine hourly wind speed values based on the valid 1-minute averages obtained during each period. Hourly wind directions (relative to True North), although a vector quantity, were determined appropriately based on the average of the valid 1-minute east-west (U) and north-south (V) components during each hourly period.

The wind speed categories used to develop joint frequency distributions (JFDs) of wind speed, wind direction, and atmospheric stability meet the intent of the current meteorological monitoring guidance in table 3, "Example Joint Frequency Distribution of Wind Direction, Wind Speed, and Stability Class," to RG 1.23, Revision 1. This guidance provides enhanced resolution in the lower wind speed categories and the overall (i.e., greater) number of wind speed classes compared to those specified in table 1 of Safety Guide 23. This approach is conservative and acceptable to the NRC staff. Note that meteorological input to the PAVAN-NAI dispersion model can be either in the form of JFDs or hourly data. The licensee opted for the former with the wind speed classes defined in meters per second (m/sec).

Meteorological data was input to the ARCON96-NAI dispersion model as hourly averages of wind speed, wind direction, and atmospheric stability. This approach is consistent with the guidance in RG 1.194. Note that wind speed measurements were in units of miles per hour. This differs from the units of measure used for the JFDs input to the PAVAN-NAI model and as reported in the files of hourly data (i.e., in m/sec). The NRC staff spot-checked the unit's conversion for this weather element and found them to be correct. All meteorological parameters input to the ARCON96-NAI model are considered to be acceptable.

3.2.2 Offsite Atmospheric Dispersion Factors Modeling Analysis

To demonstrate meeting 10 CFR 50.67(b)(2)(i) and (ii), the licensee performed a dispersion modeling analysis to estimate DBA dispersion factors (X/Q values) at the EAB and the outer boundary of the LPZ. The licensee used the PAVAN-NAI dispersion model to calculate these values. PAVAN-NAI is an adaptation of the NRC-approved PAVAN dispersion model and appears to the NRC staff to be essentially the same in terms of its accident-related dispersion calculations.

The PAVAN model implements RG 1.145, the associated user's guidance in NUREG/CR-2858, and the technical basis document for the regulatory guide in NUREG/CR-2260. However, slight differences between the input to and output of the two codes had to be accounted for in the NRC staff's review. Enclosure 14 to the supplemental letter dated December 1, 2021, discusses the differences between PAVAN-NAI and PAVAN.

The licensee initially provided a number of PAVAN-NAI model input and output files to evaluate accident releases from a variety of potential sources. The licensee eventually withdrew all but two pairs - runs designated as “RB” and “RWST” that modeled releases from the vent atop the reactor building (RB) and from the RWST, respectively.

The NRC staff confirmed that the distances to the EAB and LPZ are consistent with the distances from the midpoint between the Unit 1 RB and the cancelled Unit 2 RB to each offsite boundary as given in the FSAR (i.e., 1,200 meters (m) and 4,023 m, respectively). PAVAN-NAI was configured to account for and to exclude enhanced building wake effects on plume dispersion, as available in the PAVAN model. Only the PAVAN-NAI output corresponding to the enhanced building wake effects configuration was used by the licensee. The NRC staff considers this approach to be appropriate.

The dispersion modeling results input to the NRC staff’s confirmation of the licensee’s offsite dose calculations are listed in table 3 of this SE. Those X/Qs are based on the results listed by the licensee in table 3-18 (EAB X/Q Factors from PAVAN Run for RB Release), table 3-19 (LPZ X/Q Factors from PAVAN Run for RB Release), table 3-20 (EAB X/Q Factors from PAVAN Run for RWST Leakage), table 3-21 (LPZ X/Q Factors from PAVAN Run for RWST Leakage), and table 3-22 (Offsite Boundary X/Q Factors for Dose Analysis), and further consolidated in table 3-23 (Bounding Offsite Boundary X/Q Factors for Dose Analysis) of enclosure 1 to the supplemental letter dated December 1, 2021. The summaries in tables 3-18 to 3-21 are used to identify the highest of either the sector-specific or the 5 percent overall site X/Q values for each of the five averaging intervals at the EAB and LPZ and for potential accident releases from the RB and the RWST. This is consistent with the intent of the regulatory guidance. The results of those evaluations are compiled in table 3-22 of the supplemental letter for the EAB and LPZ.

Table 3-23 of the supplemental letter is a further consolidation that lists the bounding (i.e., highest) X/Qs for each averaging interval input to the dose analyses for the two boundaries. The only difference between tables 3-23 and 3-22 is that the licensee conservatively assigned the modeled 0- to 2-hour X/Q to all five averaging intervals for the EAB. This is acceptable to the NRC staff.

Note, however, that the NRC staff identified an input file error common to all of the PAVAN-NAI model runs. The licensee acknowledged this error and that it resulted in erroneous X/Q estimates and, therefore, doses ranging between 0.3 to 8.3 percent higher than otherwise. The licensee opted to not revise the PAVAN-NAI modeling and related dose calculations reported in enclosure 1 of the December 1, 2021, supplemental letter and accepted these conservatively higher results.

3.2.3 Onsite Atmospheric Dispersion Factors Modeling Analysis

To demonstrate meeting10 CFR 50.67(b)(2)(iii), the licensee performed a dispersion modeling analysis to estimate DBA X/Q values in support of determining acceptable control room habitability doses. Although not specified in 10 CFR 50.67, the same dose criterion for habitability applies to occupants of the TSC as called for in NUREG-0696 and by section 8.2.1, item (f), in NUREG-0737.

The licensee used the ARCON96-NAI dispersion model to calculate onsite, accident-related X/Q values. ARCON96-NAI is an adaptation of the NRC-approved ARCON96 dispersion model. ARCON96 implements RG 1.194 and the associated user’s guidance in NUREG/CR-6331. ARCON96-NAI appears to the NRC staff to be essentially the same as ARCON96 in terms of its dispersion calculations. However, slight differences between the input to and output of ARCON96-NAI and ARCON96 had to be accounted for in the NRC staff's review. Enclosure 14 of the supplemental letter dated December 1, 2021, discusses the differences between the two codes.

Accident sources that could impact the normal air intake to the control room, the emergency air intakes to the control building (i.e., A, B, and the midpoint between them), and the air intake to the TSC were evaluated by the licensee. Model runs corresponding to the scenarios discussed in section 3.1 of this SE, with respect to the applicable source and bounding intake location, include potential releases from the:

- plant stack (vent) atop the containment building (or RB),
- RWST vent,
- closest point(s) on the FHB,
- closest atmospheric safety dump valve(s),
- closest MSSV(s),
- closest point(s) on the main steam line,
- closest point(s) on the feedwater line,
- containment maintenance hatch,
- closest point(s) of the steam jet air ejector,
- closest point(s) on the condenser,
- containment building (or RB) wall (as a diffuse source),
- turbine-driven AFW pump exhaust vent, and
- emergency personnel access hatch.

Enclosures 9, 10, and 11 of the supplemental letter dated December 1, 2021, comprised hard copy in pdf form for 65 ARCON96-NAI model runs corresponding to potential releases from the specific sources above. The hard copy listings included input file specifications, output summary logs, and cumulative frequency distributions of X/Qs, respectively. The same information for these 65 model runs was among the 78 electronic run files that had also been provided as part of the supplemental letter.

In response to RAI No. 2.c. by supplemental letter dated July 5, 2022, the licensee did not include 13 of the 78 electronic model run files. The licensee stated that these runs "were not intended to be transmitted and as such should not be considered as part of the License Amendment Request or the December 1, 2021, supplemental submittal."

Enclosures 9, 10, and 11 also included hard copy listings, as above, for 24 other ARCON96-NAI model runs. This set of runs was used by the licensee to estimate X/Qs at various points assumed to represent the path of ingress to or egress from the control building during the 30-day course to be evaluated for an accident event. Potential releases from the plant stack (vent) and the RWST vent due to a LOCA were considered. Modeled receptors were located at 2.0 m above ground (presumably representing the breathing level) and at ground level (i.e., 0.0 m). However, 32 electronic model run files had been initially provided as part of the supplemental letter.

The eight additional model runs not reflected in the enclosures or included in the licensee's response to RAI No. 2.c consisted of modeled releases from the RWST with receptor heights at ground level (i.e., 0.0 m). It appears to the NRC staff that the X/Qs in these eight model runs were about 40 to 50 percent lower than those based on potential releases from the plant stack (vent).

In RAI Questions 2-a and 2-b dated June 2, 2022 (Reference 32), the NRC staff identified issues with the onsite dispersion modeling analyses and called for clarification or correction of other information in the supplemental letter dated December 1, 2021. The issues pertained to key inputs to the analyses of the source-and receptor-specific and the control building ingress/egress model runs, respectively.

In attempting to set up its confirmatory ARCON96 model run files (using ARCON 2.0 (Reference 23) which now includes a user-friendly graphical user interface), the NRC staff evaluated the adequacy and resolution of figure 3.1, “Onsite Release/Receptor Location Sketch,” in the supplemental letter dated December 1, 2021. That figure did not allow the distances between many of the ARCON96-NAI source and receptor pairs and/or the receptor-to-source compass directions relative to True North (as opposed to Plant North), as listed in table 3-25 (Release/receptor pairs and inputs for on-site X/Q calculation) of the supplemental letter to be reasonably verified. In its July 5, 2022, response to RAI Question 2-a, the licensee provided two drawings adapted from plant drawing 8600-X-88100,”Property – Site Layout Owner Controlled Area and Surrounding Area (FSAR figure 1.2-44),” to be used in lieu of figure 3.1. The two drawings were annotated with more precise source and receptor locations and orientation information corresponding to the inputs to its ARCON96-NAI model runs. The NRC staff considered the clarified and corrected information acceptable for use in its source- and receptor-specific confirmatory model runs.

The NRC staff also evaluated a sketch of the assumed ingress/egress pathway provided as figure 3.2, “Control Room Operator Access Path Sketch,” of the supplemental letter dated December 1, 2021, in setting up those confirmatory model run files. Figure 3.2, like figure 3.1, lacked adequate information related to the distance scale, orientations relative to True North, and the locations of the receptors and potential release points as modeled with ARCON96-NAI. In its July 5, 2022, response to RAI Question 2-b, the licensee provided a revised drawing adapted, again, from plant drawing 8600-X-88100 to be used in lieu of figure 3.2. The drawing was annotated with more precise information corresponding to the licensee’s model runs. The NRC staff considered the clarified information acceptable for use in its ingress/egress confirmatory model runs.

The NRC staff used this clarified and corrected input information to complete the setup of its confirmatory ARCON96 / ARCON 2.0 modeling runs. The NRC staff then compared its X/Q results to the corresponding ARCON96-NAI X/Qs listed in enclosure 1 of the supplemental letter dated December 1, 2021, (i.e., table 3-26 (X/Q Factors for Each Release/Receptor Pair) for the source- and receptor-specific accident releases, tables 3-27 (Bounding Stack/Plant Vent X/Q Factors for Control Room Access Dose Analysis) and 3-28 (Bounding RWST Vent X/Q Factors for Control Room Access Dose Analysis) for the ingress/egress accident releases from the plant stack (vent) and the RWST vent, respectively, and the controlling source- and receptor-specific accident scenarios summarized in table 3-30 (Control Room X/Q Factors for Dose Analysis) as input to the control room dose analyses).

In general, the accident durations were consistent with the time intervals in RG 1.194. For modeling the source- and receptor-specific accident releases, durations from 0-2 hours, 2-8 hours, 8-24 hours, 24-96 hours (i.e., between 1 and 4 days), and 96-720 hours (i.e., between 4 and 30 days) were used. Similar intervals also applied to the modeling of ingress/egress accident releases from the plant stack (vent) and the RWST vent except that the two shortest time intervals were from 0-2 hours and 0-8 hours.

With respect to the controlling source- and receptor-specific accident scenarios in table 3-30 of the supplemental letter dated December 1, 2021, the shortest time interval for X/Q estimates varied depending on the accident scenario - either from 0-2 hours when control room isolation was not necessary or more conservatively from time 0 to control room isolation and from isolation to 2 hours. Table 3-29 (Accident Release Sources) in enclosure 1 of the supplemental letter, lists the accident scenarios, transport paths, potential release points, and the limiting release points for the dispersion results in table 3-30, which also includes additional footnotes.

The NRC staff notes that the time 0 to isolation X/Qs for three of the potential accident scenarios (i.e., an FHA in containment, and containment leakage due to a LOCA and an REA) are based on modeling a diffuse (rather than a point) source from the containment wall. The normal air intake to the control room is the applicable receptor prior to isolation. However, the distance between the normal air intake and the containment wall, at 9.3 m, is slightly less than the minimum limit of 10 m specified in RG 1.194, Regulatory Position 3.4, “Determination of Source-Receptor Distances and Directions” (paragraph 2), which states, in part, “will need to be addressed on a case-by-case basis.”

The licensee derived a multiplier of about 3.7 to account for this distance being slightly under 10 m (see the discussion of Regulatory Position 3.4 in Attachment C, “Regulatory Guide 1.194 Conformance Table,” of enclosure 1 of the supplemental letter dated December 1, 2021). The multiplier was applied to the 0-2 hours X/Q estimated by ARCON96-NAI for these three cases. The NRC staff finds this approach acceptable because the separation distance is only slightly less than the minimum stated in the cited regulatory guidance and a X/Q multiplier of nearly a factor of four appears reasonable on that basis. More importantly, the resulting doses, which use the X/Q values as direct inputs, are below the applicable limit.

In doing the confirmatory modeling comparisons, the NRC staff noted that the ARCON96-NAI X/Qs differed from the corresponding ARCON96 / ARCON 2.0 results only for the longest accident averaging interval (i.e., 96-720 hours). Comparisons between the X/Qs for the other averaging intervals showed only minor differences. However, for the 96-720-hour interval, X/Q underpredictions of up to about 22 percent and overpredictions of up to about 5 percent were observed while some X/Qs were fairly close between the two models. The NRC staff traced an input file error found to be common to all the ARCON96-NAI model runs. Specifically, the minimum number of hours required for a valid 96-720-hour average had been set to 720 hours rather than 648 hours in accordance with RG 1.194 and the ARCON96 model user’s guidance.

On October 14, 2022, the NRC staff issued an RAI (Reference 33) notifying the licensee of the observed input file error and the need for correction, if confirmed, of all ARCON96-NAI model runs and any affected tables and/or figures in the LAR submittal. In its December 8, 2022, response, the licensee acknowledged this modeling error and self-identified several other discrepancies in the information provided in previous submittals. As a result, the licensee revised the onsite dispersion modeling, resubmitted tables 3-26, 3-27, 3-28, and 3-30, and updated a number of subsections of the LAR that were affected (i.e., subsections 3.3.2.5, 3.3.6.2.3.2, 3.3.6.4, 3.3.8.1.5, 3.3.9.2.3, 3.3.9.4, and 3.3.10.2.3) in the supplemental letter.

The NRC staff compared the revised ARCON96-NAI onsite dispersion modeling results to its ARCON96 / ARCON 2.0 confirmatory modeling and finds the revised results to be the same or acceptably similar for all accident averaging intervals. The licensee's accident-related X/Qs used in determining control room doses are listed in table 2 in section 4.0 of this SE.

3.3 Steam Release Calculations

The licensee chose to develop new steam releases for each of the FSAR Chapter 15 steam release transient analyses requiring an AST dose analysis: MSLB (FSAR section 15.1.5.3), LOAC (FSAR section 15.2.6.3), LRA (FSAR section 15.3.3.3), REA (FSAR section 15.4.8.3), LLB (FSAR section 15.6.2.1), SGTR (FSAR section 15.6.3), LOCA (FSAR section 15.6.5.4) and FHA (FSAR section 15.7.4.5).

These secondary system steam releases were calculated using the RETRAN-3D thermal-hydraulic system code. The RETRAN-3D based steam releases for dose analyses were initially based on the pre-existing RETRAN-02 analyses, including the development of a RETRAN-3D based methodology to address the REA outside containment steam release. RETRAN-3D is classified by the NRC as safety-related software and is maintained and run under a 10 CFR Part 50 Appendix B, Quality Assurance (QA) program. An earlier version of RETRAN, RETRAN-02 was used to perform the licensing basis FSAR Chapter 15 accident analyses for all the scenarios analyzed.

Callaway maintained QA versions of RETRAN-3D input decks that had previously been converted from the licensing basis RETRAN-02 input. RETRAN-3D input decks for each of the various Chapter 15 analyses were provided by Callaway and then modified as appropriate to implement a steam release calculation in accordance with assumptions outlined in RG 1.183 for the following accident case: MSLB; LOAC; LRA; REA; SGTR, with a failed open ASDs; and SGTR Overfill. Based on its review, the NRC finds that these components are adequately designed to the applicable standards to support Callaway's adoption of the AST.

3.4 Human Factors Evaluation of EOPs and Credited Manual Actions

The NRC staff reviewed the LAR as supplemented with respect to human factors considerations. The NRC staff issued RAI No. 3 (RAI HFE-1) by email dated June 2, 2022 (Reference 32), requesting the licensee to describe whether Callaway would be modifying any EOPs as part of the LAR and, if so, to describe the procedural changes, any changes in the time constraints associated with the performance of procedurally driven actions, and any operator training associated with those changes. In response to RAI No. 3 in supplemental letter dated July 5, 2022, the licensee stated that the LAR would result in manual operator action being credited for initiating an EES within 10 minutes of accident initiation. The licensee supplemented this response in letter dated September 1, 2022, stating that further review of a supporting calculation had determined that the EES in question is not required to mitigate accident consequences and that operator action therefore did not need to be credited. Based upon the LAR, as supplemented, not requiring modification to either the EOPs or credited manual operator actions, the staff find that the LAR will not have an impact upon these areas and that, accordingly, further staff evaluation of associated human factors engineering considerations is not warranted.

3.5 Technical Specification Changes

3.5.1 TS Definitions Section 1.1, "Dose Equivalent I-131"

The licensee has proposed to restrict the list of dose conversion factors used to calculate DE I-131 to those listed in FGR No. 11.

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 specification should have a basis consistent with the basis of the dose analyses. The licensee currently calculates DE I-131 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.

Typically, changes to the DE I-131 definition are included as a part of the AST submittal. It is appropriate for those plants using the AST methodology to use a definition of DE I-131 based on the CEDE DCFs instead of thyroid DCFs. Some licensees using the AST methodology have chosen to reference the committed dose equivalent (CDE) thyroid dose in their TS definition of DE I-131. Although technically it is more accurate to reference the CEDE DCFs, the numerical difference in the calculated value of DE I-131 using either CEDE or CDE thyroid DCFs, for a given isotopic mixture, is not significant.

The licensee has proposed to maintain the current definition of DE I-131 which is based on the determination of the inhalation dose to the thyroid. The licensee has proposed to use the DCFs from FGR No. 11 to calculate DE I-131. The NRC staff has evaluated the proposed definition of DE I-131 and has determined that the incorporation of either the CDE thyroid DCFs or the CEDE DCFs from FGR No.11 in the DE I-131 definition is acceptable.

3.5.2 TS 3.7.10, "Control Room Emergency Ventilation System (CREVS)"

The licensee included the equipment room in the control room envelope for the TS dealing with the operability of the CREVS. The NRC staff reviewed these changes and finds that the licensee made the appropriate changes to reflect the revised control room dose consequence analyses and therefore the NRC staff finds that these TS changes are acceptable.

The licensee removed the requirement for the operation of the heaters while performing SR 3.7.10.1. The NRC staff finds that this change is acceptable since the AST dose consequence analyses no longer credits the charcoal adsorbers in the control building pressurization unit.

3.5.3 TS 5.5.11, "Ventilation Filter Testing Program (VFTP)"

The proposed AST dose analyses do not credit the charcoal adsorbers in the control building pressurization unit. The NRC staff finds the changes made to the VFTP acceptable since the charcoal adsorber is not required to be installed or included in either control room pressurization train.

3.5.4 TS 5.5.17, "Control Room Envelope Habitability Program"

The licensee changed the description of the control room dose acceptance criteria from 5 rem whole body or its equivalent to any part of the body to 5 rem TEDE in TS 5.5.17 to align with the guidance in RG 1.183. The licensee also revised the Control Room Envelope Habitability Program to define and include requirements for the ERE and the ERE boundary. The NRC staff finds these changes acceptable as they align the TS with the guidance of RG 1.183 and the revised AST analysis.

In addition, the NRC staff also reviewed the markups made to the Callaway FSAR Chapter 7.0, “Instrumentation and Controls”. The NRC staff found that the proposed changes in this LAR do not impact the existing instrumentation and control (I&C) design associated with the reactor trip system and ESF actuation system, including the setpoints and allowable values established for the I&C functions for accident mitigation or protection against violating the core fuel design limits or reactor coolant pressure boundary during anticipated operational occurrences. Therefore, the NRC staff finds from the I&C perspectives that the changes proposed in this LAR are acceptable.

3.5.5 TS Specifications Changes Conclusion

The NRC staff determined that the TS as amended by the proposed changes described in section 2.1 and illustrated in enclosure 3 of the September 28, 2022, LAR, as supplemented in the letter dated December 1, 2021, will continue to be derived from the analyses and evaluation included in the FSAR, and amendments thereto. The NRC staff also determined that the TS as amended by the proposed changes to the TS 3.7.10 Note, Conditions, and Required Actions will continue to provide the lowest functional capability or performance levels of equipment required for safe operation of the facility as well as provide permitted remedial actions for the instances when the LCO is not met. The NRC staff also determined the TS as amended by the proposed changes to TS 3.7.10 SRs will continue to provide assurance that the necessary quality of systems and components is maintained that facility operation will be within safety limits and that the limiting conditions for operation will be met. Finally, the NRC staff determined that the TS as amended by the proposed changes to TS 5.5.11 and 5.5.17 will continue to list provisions necessary to assure operation of the facility in a safe manner.

Therefore, the NRC staff determined that the changes are acceptable because the TS, as amended, will continue to meet the regulatory requirements of 10 CFR 50.36(c)(2), 50.36(c)(3), and 50.36(c)(5).

3.6 Technical Evaluation Conclusion

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological consequences of DBAs with full implementation of an AST. The NRC staff finds that the licensee used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in section 2.0 of this SE. The NRC staff compared the doses estimated by the licensee to the applicable criteria identified in section 2.0 of this SE. The NRC staff also finds, with reasonable assurance, that the licensee’s estimates of the EAB, LPZ, and control room doses will comply with these criteria. The NRC staff further finds reasonable assurance that sufficient safety margins with adequate defense-in-depth to address unanticipated events are maintained, and the ability to compensate for uncertainties in accident progression and analysis assumptions and parameters is continued. Therefore, the proposed license amendment is acceptable with respect to the radiological consequences of DBAs. The NRC staff finds that the licensee’s evaluation of the MSLB acceptable because it adhered to the guidance in RG 1.183.

This licensing action is considered a full implementation of the AST. With this approval, the licensee's CLB accident source term is superseded by the AST. The previous offsite and control room accident dose criteria expressed in terms of whole body, thyroid, and skin doses are superseded by the TEDE criteria of 10 CFR 50.67(b)(2)(iii), or fractions thereof, as defined in RG 1.183. All future radiological accident analyses performed to show compliance with regulatory requirements shall address all characteristics of the AST and the TEDE criteria as defined in the licensee's design basis and modified by the present LAR. The NRC staff reviewed the application in accordance with SRP, section 15.0.1 and NUREG-0737. The NRC staff finds the request acceptable because there are no substantive changes to the EOPs or credited manual actions from the current licensing basis.

The NRC staff reviewed the licensee's assumptions, methodology, and conclusions regarding the pH of suppression pool. The methodology relies on using buffering action of TSP. The assumptions are appropriate and consistent with the methods accepted by the NRC staff for the calculation of post-accident containment suppression pool pH. The calculations were made for the 30-day period following a LOCA. The NRC staff verified that the post-accident containment sump pH will be maintained above 7.0 for 30 days following a LOCA.

4.0 TABLES

Table 1

Callaway Plant Radiological Consequences Expressed as TEDE [1]
(rem)

Design-Basis Accidents	EAB [2]	LPZ [3]	CR	TSC
Main Steamline Break Accident [4]	0.11	0.10	2.16	0.26
Loss of Non-Emergency AC Power [4]	5.3×10^{-3}	2.9×10^{-3}	0.82	Not Reported[8]
Steam Generator Tube Rupture Overfill[4]	3.5	1.21	1.6	3.8
Steam Generator Tube Rupture Failed ASD[4]	2.00	0.72	2.44	2.39
Loss of Coolant Accident	5.64	6.24	4.36[9]	2.10
Dose Acceptance Criteria	25	25	5	5
Control Rod Ejection Accident [6]	1.43	3.24	4.09	1.56
Control Rod Ejection Accident [7]	1.34	0.72	4.13	0.43
Fuel Handling Accident in FHB	0.78	0.27	0.65	0.57
Fuel Handling Accident in Containment	0.97	0.34	1.58	0.75
Dose Acceptance Criteria	6.3	6.3	5	5
Main Steamline Break Accident [5]	0.55	0.5	2.81	0.93
Loss of Non-Emergency AC Power [5]	1.1×10^{-2}	5.6×10^{-3}	1.5	Not Reported[8]
Locked Rotor Accident	0.40	0.21	1.34	0.11
Letdown Line Break	0.39	0.13	1.93	0.17
Steam Generator Tube Rupture Overfill[5]	2.13	0.73	0.69	1.57
Steam Generator Tube Rupture Failed ASD[5]	1.38	0.50	0.59	1.06
Dose Acceptance Criteria	2.5	2.5	5	5

(1) Total effective dose equivalent
(2) Exclusion area boundary
(3) Low population zone
(4) Pre-accident iodine spike case
(5) Accident-initiated iodine spike case
(6) Assumes containment release
(7) Assumes secondary side release
(8) The licensee did not report TSC doses since the control room (CR) does not isolate in this event
(9) Dose due to occupancy is 3.37 rem and dose due to transit is 0.81 rem.

Table 2 (page 1 of 2)

Callaway Plant Control Room Atmospheric Dispersion Factors (sec/m^3)

Accident Specific Release Locations / Duration	χ/Q (sec/m^3)
Containment Leakage for the LOCA and REA	
0 - Isolation	7.12×10^{-3}
Isolation – 2 hours	7.49×10^{-4}
2 - 8 hours	5.32×10^{-4}
8 - 24 hours	2.29×10^{-4}
24 - 96 hours	1.50×10^{-4}
96 - 720 hours	1.02×10^{-4}
Unit Vent for the LOCA Mini-Purge & ECCS Leakage	
0 – Isolation for LOCA Mini-Purge & ECCS Leakage	1.90×10^{-3}
Isolation – 2 hours	6.86×10^{-4}
2 - 8 hours	5.72×10^{-4}
8 - 24 hours	2.32×10^{-4}
24 - 96 hours	1.42×10^{-4}
96 - 720 hours	1.00×10^{-4}
Unit Vent Exhaust to Normal CR Intake for the LLB	
0 - 2 hours (No CR isolation assumed)	1.90×10^{-3}
2 - 8 hours	1.58×10^{-3}
8 - 24 hours	6.67×10^{-4}
24 - 96 hours	3.90×10^{-4}
96 - 720 hours	2.65×10^{-4}
RWST Vent for LOCA RWST Back-leakage	
0 – Isolation	9.28×10^{-4}
Isolation – 2 hours	7.47×10^{-4}
2 - 8 hours	6.55×10^{-4}
8 - 24 hours	2.71×10^{-4}
24 - 96 hours	1.52×10^{-4}
96 - 720 hours	1.15×10^{-4}
FHA in FHB (FHB Closest Point)	
0 – Isolation	2.23×10^{-3}
Isolation – 2 hours	1.17×10^{-3}
2 - 8 hours	1.04×10^{-3}
8 - 24 hours	4.27×10^{-4}
24 - 96 hours	2.34×10^{-4}
96 - 720 hours	1.94×10^{-4}

Table 2 (page 2 of 2)

Callaway Plant Control Room Atmospheric Dispersion Factors (sec/m^3)

Accident Specific Release Locations / Duration	χ/Q (sec/m^3)
Emergency Personnel Access Hatch for the FHA in Containment	
0 – Diffuse containment wall leakage prior to Isolation	7.12×10^{-3}
Isolation – 2 hours	8.61×10^{-4}
2 - 8 hours	7.54×10^{-4}
8 - 24 hours	3.22×10^{-4}
24 - 96 hours	1.84×10^{-4}
96 - 720 hours	1.50×10^{-4}
Locked Rotor, SGTR and Rod Ejection (Closest ASD)	
0 – Closest MSSV used prior to Isolation	1.76×10^{-2}
Isolation – 2 hours	1.74×10^{-3}
2 - 8 hours	1.33×10^{-3}
8 - 24 hours	6.50×10^{-4}
24 - 96 hours	3.62×10^{-4}
96 - 720 hours	2.96×10^{-4}
MSSV to Normal CR Intake for the Loss of AC Power	
0 - 2 hours (No CR isolation assumed)	1.76×10^{-2}
2 - 8 hours	1.46×10^{-2}
8 - 24 hours	6.74×10^{-3}
24 - 96 hours	3.81×10^{-3}
96 - 720 hours	3.05×10^{-3}
Closest Main Steam Line for the MSLB	
0 – Closest MSSV used prior to Isolation	1.76×10^{-2}
Isolation – Closest ASD used for first 2 hours after isolation	1.74×10^{-3}
2 - 8 hours	1.56×10^{-3}
8 - 24 hours	6.61×10^{-4}
24 - 96 hours	3.83×10^{-4}
96 - 720 hours	3.23×10^{-4}

Table 3

Offsite Atmospheric Dispersion Factors (sec/m^3)

Time Period	EAB RB Release	LPZ RB Release	EAB RWST Release	LPZ RWST Release
0 - 2 hours	2.00×10^{-4}	6.87×10^{-5}	2.05×10^{-4}	6.87×10^{-5}
2 - 8 hours	2.00×10^{-4}	3.42×10^{-5}	2.05×10^{-4}	3.57×10^{-5}
8 - 24 hours	2.00×10^{-4}	2.42×10^{-5}	2.05×10^{-4}	2.57×10^{-5}
24 - 96 hours	2.00×10^{-4}	1.13×10^{-5}	2.05×10^{-4}	1.26×10^{-5}
96 - 720 hours	2.00×10^{-4}	3.83×10^{-6}	2.05×10^{-4}	4.54×10^{-6}

Table 4

Callaway Plant Control Room Data and Assumptions and Direct Shine Results

Control Building	
Mixing volume	148,000 ft^3
Filtered intake prior to operator action (0 – 30 minutes)	900 cfm
Filtered intake after operator action (30 minutes – 720 hours)	450 cfm
Particulate filter efficiency	95 percent
Allowable unfiltered inleakage values (Q_{cb}) are paired with corresponding CR values (Q_{cr}) (FSAR Figure 15A-2)[1]	Max 6,000 cfm $Q_{cb} \leq 6000\text{-}100{*}Q_{cr}$
Control Room	
Mixing volume	48,500 ft^3
Filtered inflow from control building	440 cfm
Unfiltered inflow from control building (0 – 30 minutes)	440 cfm
Unfiltered inflow from control building (30 min – 720 hours)	0 cfm
Filtered recirculation	1,030 cfm
Filter efficiency (all iodine species)	95 percent
Allowable unfiltered inleakage values (Q_{cr}) are paired with corresponding CR values (Q_{cb}) (FSAR Figure 15A-2)	Max 60 cfm $Q_{cb} \leq 6000\text{-}100{*}Q_{cr}$
Control Room Equipment Room Volume	
Volume	10,000 ft^3
Unfiltered inleakage	Max 300 cfm
Control room Operator Breathing Rate	
0 - 720 hours	3.5×10^{-4} m^3/sec
Control Room Occupancy Factors	
0 - 24 hours	1.0
24 - 96 hours	0.6
96 - 720 hours	0.4
LOCA CR Gamma Shine Component Doses (mrem)	
Containment	8.5
External Cloud	0.12
CR filter	3.7
Total gamma shine dose (mrem)	12.32

1. Q_{cb}:Control Building Inleakage, Q_{cr}: Control Building Room Inleakage:

Table 5

Callaway Plant Data and Assumptions for the MSLB Accident

Power level conservatively analyzed at Hot Zero Power	34 MWt
Initial RCS equilibrium activity	
Iodines	1.0 µCi/gm DE I-131
Nobel Gases	225 µCi/gm DE Xe-133
Initial secondary system activity	10% of Initial RCS Activity
Pre-accident spike iodine concentration	60 µCi/gm DE I-131
Accident-initiated iodine spike appearance rate multiplier	500
Letdown flow rate plus identified RCS leakage	140 gpm
Primary-to-secondary leakage	1 gpm
Duration of accident-initiated iodine spike	8 hours
RCS mass maximum	580,700 lbm
Equilibrium appearance rates (Ci/min)	
I-131	0.436
I-132	4.08
I-133	0.761
I-134	16.02
I-135	2.19
Chemical form of iodine releases	
Elemental	97 percent
Organic	3 percent
Particulate	0 percent
Event scenario timeline	
Safety injection signal	2 seconds
CR isolation (including delay)	62 seconds
Faulted SG releases all initial activity	2 minutes
Failed CR HVAC train isolated	30.033 minutes
RHR operation terminates releases from intact SGs	7.899 hours
RCS < 212ºF terminates releases form faulted SG	22 hours
Initial faulted SG release (0 – 2 minutes)	155,500 lbm
Primary-to-secondary leakage through faulted SG to atmosphere	1 gpm
Steam released from intact SGs	
0 to 2 hours	501,905 lbm
2 to 7.899 hours	1,025,601 lbm
RCS Mass minimum	551,068 lbm
Faulted SG secondary mass	155,500 lbm
Intact SGs total secondary mass	466,500 lbm
Iodine and particulate Partition Factor	100

Table 6

Callaway Plant Data and Assumptions for the Loss of Non-Emergency AC Power

Initial RCS equilibrium activity	
Iodines	1.0 µCi/gm DE I-131
Nobel Gases	225 µCi/gm DE Xe-133
Initial secondary system activity	10% of Initial RCS activity
Pre-accident spike iodine concentration	60 µCi/gm DE I-131
Accident-initiated iodine spike appearance rate multiplier	500
RCS Letdown flow rate	130 gpm
RCS leakage (Identified plus unidentified leakage)	10 + 1 = 11 gpm
Duration of accident-initiated iodine spike	8 hours
RCS mass maximum	580,700 lbm
Equilibrium appearance rates (Ci/min)	
I-131	0.44
I-132	4.08
I-133	0.76
I-134	16.02
I-135	2.19
Steam Generator Tube Leakage	500 lbm/hour
Chemical form of iodine releases	
Elemental	97 percent
Organic	3 percent
Maximum RCS Mass	580,700 lbm
Minimum RCS Mass	551,068 lbm
Total Steam Generator Secondary Side Mass	370,000 lbm
RHR operation terminates releases from intact SGs	7.275 Hours
Timing of Steam Releases	
0 to 2 hours	350,168 lbm
2 to 7.275 hours	935,834 lbm
Duration of Steam Generator Tube Uncovery	2.667 hours
Flashing Fraction during Uncovery	5 percent
Iodine and particulate Partition Factor	100
CR ventilation system remains in normal mode operation	Duration of event

Table 7

Callaway Plant Data and Assumptions for the LRA

Percentage of Failed Fuel (Cladding Failure)	5 percent of core - Equivalent to 9.65 fuel assemblies
Number of high burnup rods in failed assemblies	35
Radial peaking factor applied to damaged fuel	1.65
Gap Release Fractions	LAR Table 3-46
Steam Generator Tube Leakage	500 lbm/hour (1 gpm)
Minimum RCS Mass	551,068 lbm
Total Steam Generator Secondary Side Mass	370,000 lbm
Chemical form of iodine releases	
Elemental	97 percent
Organic	3 percent
RHR operation terminates releases from intact SGs	7.275 Hours
Timing of Steam Releases	
0 to 2 hours	422,208 lbm
2 to 7.275 hours	928,631 lbm
Duration of Steam Generator Tube Uncovery	2.48 hours
Flashing Fraction during Uncovery	5 percent
Iodine and particulate Partition Factor	100
Time delay to isolate and pressurize CR	2 minutes
Normal Control Building HVAC Isolation Set Point	2.2×10^{-3} μCi/cc Xe-133

Table 8

Callaway Plant Data and Assumptions for the REA

Percentage of Failed Fuel (Cladding Failure)	10 percent of core - Equivalent to 19.3 fuel assemblies
Number of high burnup rods in failed assemblies	35
Fraction of fuel assumed to melt	0.25 percent of core
Radial peaking factor applied to damaged fuel	1.65
Gap Release Fractions	LAR Table 3-48
Containment Leakage Scenario Assumptions	
Chemical form of iodine releases	
Aerosol	95 percent
Elemental	4.85 percent
Organic	0.15 percent
Containment Volume	2,700,000 ft^3
Containment Leakage	
0 – 24 hours	0.2 percent per day
24 – 720 hours	0.1 percent per day
Percentage of fuel melt available for release	
Noble gases	100 percent
Iodines	25 percent
Containment spray	Not credited
Natural deposition in containment	Not credited
Secondary Side Release Scenario Assumptions	
Percentage of fuel melt available for release	
Noble gases	100 percent
Iodines	50 percent
Steam Generator Tube Leakage	500 lbm/hour (1 gpm)
Minimum RCS Mass	551,068 lbm
Total Steam Generator Secondary Side Mass	370,000 lbm
Chemical form of iodine releases	
Elemental	97 percent
Organic	3 percent
RHR operation terminates releases from intact SGs	7.275 Hours
Timing of Steam Releases	
0 to 2 hours	519,087 lbm
2 to 7.29 hours	1,144,922 lbm
Duration of Steam Generator Tube Uncovery	2.622 hours
Flashing Fraction during Uncovery	5 percent
Iodine and particulate Partition Factor	100
Time delay to isolate and pressurize CR	2 minutes

Table 9

Callaway Plant Data and Assumptions for the LLB

Initial RCS equilibrium activity	
Iodines	1.0 µCi/gm DE I-131
Nobel Gases	225 µCi/gm DE Xe-133
Letdown Line Break Flow Rate	158.9 gpm
Accident-initiated iodine spike appearance rate multiplier	500
Duration of accident-initiated iodine spike	8 hours
RCS Identified Leakage	5000 lbm/hour (10 gpm)
RCS Unidentified Leakage	500 lbm/hour (1 gpm)
Minimum RCS Mass	551,068 lbm
Time to identify leakage and close letdown isolation valve	30 minutes and 10 seconds
Total reactor coolant inventory lost over 1810 seconds	39,958 lbm
Fraction of leaking coolant that flashes to steam	20 percent
Chemical form of iodine releases	
Elemental	97 percent
Organic	3 percent
CR ventilation system remains in normal mode operation	Duration of event

Table 10

Callaway Plant Data and Assumptions for the SGTR Accident
Stuck-Open ASD Scenario

RCS and secondary coolant system density	8.33 lbm/gallon
Steam Generator Tube Leakage Rate	8.33 lbm per minute
Minimum RCS mass	551,068 lbm
Maximum RCS mass	580,700 lbm
Ruptured Steam Generator Secondary Side Mass	92,500 lbm
Intact Steam Generator Secondary Side Mass	277,500 lbm
Total Steam Generator Secondary Side Mass	370,000 lbm
Initial RCS equilibrium activity	
Iodines	1.0 µCi/gm DE I-131
Nobel Gases	225 µCi/gm DE Xe-133
Initial secondary side equilibrium activity	0.1 µCi/gm DE I-131
Concurrent Iodine Spiking Factor	335
Concurrent Iodine Spike Duration	8 hours
Pre-accident Iodine Spike Concentration	60 µCi/gm DE I-131
Time to isolate stuck-open ASD(1)	20 minutes
Reactor trip with associated LOOP	0.1658 hours
RHR system in service ending releases to the atmosphere	5.86 hours
Mass release form Ruptured steam generator	
0 - 2 hours	114,107 lbm
2 - to RHR system in service @ 5.86 hours	111,380 lbm
Mass release form Intact steam generators	
0 - 2 hours	321,930 lbm
2 - to RHR system in service @ 5.86 hours	647,375 lbm
Break flow rate from Ruptured Steam Generator Tube	
0 – to termination @ 1.39 hours	60 lbm per second
Condenser Iodine Partition Factor	100
Steam Generator Iodine and particulate Partition Factor	100
Chemical Form of Iodine Release	
Elemental	97 percent
Organic	3 percent
Time delay to isolate and pressurize CR	2 minutes

(1) As an additional failure beyond the required single failure, the licensee assumed that an ASD valve on one of the unaffected steam generators does not open and will therefore be unavailable to support the RCS cooldown. This conservative assumption has the effect of increasing the time it takes to reduce the RCS temperature to below the ruptured steam generator saturation temperature thereby increasing the releases from the intact steam generators prior to reaching RHR conditions.

Table 11

Callaway Plant Data and Assumptions for the SGTR Accident with Failure of Faulted SG AFW Control Valve Resulting in SG Overfill

Reactor/Turbine trip, LOOP and safety injection conservatively assumed at SGTR initiation	
Steam Generator Tube Leakage Rate	8.33 lbm per minute (1gpm)
Minimum RCS mass	551,068 lbm
Maximum RCS mass	580,700 lbm
Total Steam Generator Secondary Side Minimum mass	370,000 lbm
Ruptured Steam Generator Secondary Side Mass	
0 – 0.1944 hours	119,000 lbm
0.1944 – 1.6944 hours	265,000 lbm
1.6944 – 2.1944 hours	257,000 lbm
2.1944 hours to end of accident evaluation	245,000 lbm
Intact Steam Generator Secondary Side Mass	
0 – 0.1667 hours	357,000 lbm
0.1667 – 0.4861 hours	495,000 lbm
0.4861 – 0.75 hours	423,000 lbm
0.75 – 1.1944 hours	495,000 lbm
1.1944 hours to end of accident evaluation	360,000 lbm
Ruptured Steam Generator Release Rate	
0 – 0.264 hours	0 lbm/minute
0.264 – 1.361 hours (liquid release from overfill)	2,962 lbm per minute
1.361 – 2 hours	3.55 lob per minute
2 – 8 hours	1.89 lbm per minute
8 – 9.16 hours	2.24 lbm per minute
SG inventory cooled to 212 ºF at 9.16 hours ending release	
Airborne fraction of liquid release during SG overfill	50 percent
Intact Steam Generator Release Rate	
0 – 0.527 hours	0 lbm per minute
0.527 – 2 hours	44.05 per minute
2 – RHR at 6.397 hours ending release	22.03 per minute
RHR conditions reached at 6.397 hours ending release	
Ruptured Tube Break Flow Rate	
0 – 1.11 hours	60 lbm per second
Flashing Fraction for tube leakage during Intact SGs tube uncovery	
Between 6,000 and 9,000 seconds	4 percent
Steam Generator Iodine and particulate Partition Factor	100
Chemical Form of Iodine Release	
Elemental	97 percent
Organic	3 percent
Iodine Spike Assumptions and CR response	Same as Failed ASD case

Table 12 (page 1 of 2)

Callaway Plant Data and Assumptions for the LOCA

Licensed power level	3,565 MWt	
Power level for radiological source term	3,636 MWt	
Containment volume	2,700,000 ft^3	
Reactor coolant system TS activity limits		
Iodine	1.0 μCi/gm DE I-131	
Noble gases	225 μCi/gm DE Xe-133	
Remaining nuclide groups	1 percent fuel defect level	
Containment leak rate		
0 - 24 hours	0.2 percent per day	
24 - 720 hours	0.1 percent per day	
Average fall height of spray drops to operating deck	131.4 feet	
Percentage of containment volume that is sprayed	85 percent	
Unsprayed containment air Volume	405,000 ft^3	
Flow rate between sprayed and unsprayed regions[1]	13,500 cfm	
Time delay for mixing between sprayed and unsprayed	2 minutes	
Containment spray operation	From 2 min	
Initiation	T = 2 minutes	
Termination	T = 4 hours	
Net spray flow rate per train during injection phase	3,086 gpm	
Spray solution pH		
During injection phase	6.59	
Recirculation phase at equilibrium	≥ 7.1	
Elemental iodine spray removal coefficient	Credited	Calculated
Initially until DF = 200	20 hr^{-1}	37.3 hr^{-1}
After DF = 200	0 hr^{-1}	N/A
Particulate spray removal coefficients		
Initially	6.46 hr^{-1}	
Reduced by factor of 10 for DF > 50	0.646 hr^{-1}	
Organic iodide spray removal coefficient	0.0	
Iodine chemical form in containment atmosphere		
cesium iodide	95 percent	
elemental iodine	4.85 percent	
organic iodine	0.15 percent	
Minimum containment sump liquid volume	428,000 gallons	
Assumed ECCS recirculation leakage (two times TS limit)	2 gpm	
Assumed ECCS recirculation leakage start time	62 seconds	
ECCS Flashing fraction	10 percent	

[1] Based two turnovers of the unsprayed volume per hour (405,000/30 minutes = 13,500 cfm)

Table 12 (page 2 of 2)

Callaway Plant Data and Assumptions for the LOCA

Total Back leakage to the RWST	4 gpm
Leakage into RWST below the water line	3 gpm
Leakage into RWST above the water line	1 gpm
Chemical form of released iodine from ECCS leakage	
Elemental	97 percent
Organic	3 percent
Particulate	0 percent
Minimum equilibrium sump pH during recirculation	7.1

Table 13

Callaway Plant Data and Assumptions for the FHA in the FHB

Minimum post shutdown fuel decay time	72 hours	
Radial peaking factor	1.65	
Number of fuel assemblies affected	1.0	
Number of High Burnup fuel rods in dropped assembly	32	
FHA gap release fractions for rods in dropped assembly	>54 GWD/MTU and 6.3 kw/ft	<62 GWD/MTU
I-131	0.12	0.08
Kr-85	0.30	0.10
Other Noble Gases	0.10	0.05
Other Halogens	0.10	0.05
Alkali Metals	0.17	0.12
Percent of gap activity released	100 percent	
Minimum pool water depth	23 feet	
Pool DF		
Noble gases and organic iodine	1	
Aerosols	Infinite	
Elemental iodine (with 23 ft of water cover)	285	
Overall iodine (with 23 ft of water cover)	200 (effective DF)	
Chemical form of iodine	Pool Water	Released
Elemental	99.85 percent	70 percent
Organic	0.15 percent	30 percent
Duration of release to the environment	2 hours	
Time delay to isolate and pressurize CR Room	2 minutes	

Table 14

Callaway Plant Data and Assumptions for the FHA in the Containment

Minimum post shutdown fuel decay time	72 hours	
Radial peaking factor	1.65	
Number of fuel assemblies affected	1.2	
Number of High Burnup fuel rods	64	
FHA gap release fractions for rods in dropped assembly	>54 GWD/MTU and 6.3 kw/ft	<62 GWD/MTU
I-131	0.12	0.08
Kr-85	0.30	0.10
Other Noble Gases	0.10	0.05
Other Halogens	0.10	0.05
Alkali Metals	0.17	0.12
Percent of gap activity released	100 percent	
Minimum pool water depth	23 feet	
Pool DF		
Noble gases and organic iodine	1	
Aerosols	Infinite	
Elemental iodine (with 23 ft of water cover)	285	
Overall iodine (with 23 ft of water cover)	200 (effective DF)	
Chemical form of iodine	Pool Water	Released
Elemental	99.85 percent	70 percent
Organic	0.15 percent	30 percent
Duration of release to the environment	2 Hours	
Time delay to isolate and pressurize CR	2 minutes	

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Missouri State official was notified of the proposed issuance of the amendment on June 14, 2023. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes 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 and changes SRs. The NRC staff has determined that the amendment involves 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 amendment involves no significant hazards consideration published in *Federal Register* on February 22, 2022 (87 FR 9654), and there has been no public comment on such finding. Accordingly, the amendment meets 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 amendment.

7.0 CONCLUSION

The Commission has concluded that the applicant's analysis demonstrates with reasonable assurance that the criteria in 10 CFR 50.67(b)(2)(i)-(iii) are met. Accordingly, 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

8.0 REFERENCES

1. Banker, S. P., Ameren Missouri, letter to NRC, "Docket Number 50-483, Callaway Plant Unit 1, Union Electric Co., Renewed Facility Operating License NPF-30, License Amendment Request for Adoption of Alternate Source Term and Revision of Technical Specifications," dated September 28, 2021 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML21272A167).

2. Meyer, S. J., Ameren Missouri, letter to NRC, "Docket Number 50-483, Callaway Plant Unit 1, Union Electric Co., Renewed Facility Operating License NPF-30, Supplement to License Amendment Request for Adoption of Alternative Source Term and Revision of Technical Specifications," dated December 1, 2021 (Package ML21335A451).

3. Witt, T. A., Ameren Missouri, letter to NRC, "Docket Number 50-483, Callaway Plant, Unit 1, Union Electric Co., Renewed Facility Operating License NPF-30, Response to Request for Additional Information Regarding License Amendment Request for Adoption of Alternate Source Term and Revision of Technical Specification," dated July 5, 2022 (Package ML22186A103).

4. Witt, T. A., Ameren Missouri, letter to NRC, "Docket Number 50-483, Callaway Plant, Unit 1, Union Electric Co., Renewed Facility Operating License NPF-30, Revised Response to Request for Additional Information Regarding License Amendment Request for Adoption of Alternate Source Term and Revision of Technical Specification (LDCN 21-0015)," dated September 1, 2022 (Package ML22244A167).

5. Witt, T. A, Ameren Missouri, letter to NRC, "Docket Number 50-483, Callaway Plant, Unit 1, Union Electric Co., Renewed Facility Operating License NPF-30, Response to Request for Additional Information Regarding License Amendment Request for Adoption of Alternate Source Term and Revision of Technical Specifications (LDCN 21-0015)," dated December 8, 2022 (Package ML22342B264).

6.. Jungmann, B. L., Ameren Missouri, letter to NRC, "Docket Number 50-483, Callaway Plant, Unit 1, Union Electric Co., Renewed Facility Operating License NPF-30, Withdrawal of Previously Submitted Enclosures Regarding License Amendment Request for Adoption of Alternative Source Term and Revision of Technical Specifications (LDCN 21-0015)," dated May 9, 2023 (ML23129A830).

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9. NRC, “Accident Source Terms for Light-Water Nuclear Power Plants,” Final Report, NUREG-1465, dated February 1995 (ML041040063).

10. NRC, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” RG 1.183, dated July 2000 (ML003716792).

11. NRC, “Clarification of TMI Action Plan Requirements,” NUREG-0737, dated November 1980 and “Clarification of TMI Action Plan Requirements – Requirements for Emergency Response Capability,” NUREG-0737, Supplement No. 1, Reprinted February 1989 (ML102560051 and ML102560009, respectively).

12. NRC, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plant,” RG 1.145, Revision 1, dated November 1982 (Reissued February 1983) (ML003740205).

13. NRC, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants,” RG 1.194, Revision 0, dated June 2003 (ML031530505).

14. NRC, “Meteorological Monitoring Programs for Nuclear Power Plants, RG 1.23, Revision 1, dated March 2007 (ML070350028).

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16. NRC, “Functional Criteria for Emergency Response Facilities, Final Report, NUREG-0696, dated February 1981 (ML051390358).

17. NRC, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition,” NUREG-0800:

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    c. Section 6.5.2, “Containment Spray as a Fission Product Cleanup System,” Revision 4, dated March 2007 (ML070190174).

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32. Chawla, M., NRC, email to T. B. Elwood, Ameren Missouri, “Final – Request for Additional Information – Callaway Plant, Unit 1 – License Amendment Request for Adoption of Alternate Source Term and Revision of Technical Specifications – EPID L-2021-LLA-0177,” dated June 2, 2022 (ML22154A012).

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Attachment:
List of Abbreviations and Acronyms

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Date: September 20, 2023

LIST OF ABBREVIATIONS AND ACRONYMS

AB	auxiliary building
AC	alternating current
ADAMS	Agencywide Documents Access and Management System
AFW	auxiliary feedwater
ANSI	American National Standards Institute
ASD	atmospheric steam dump
ASME	American Society of Mechanical Engineers
AST	alternative source term
ATD	atmospheric tracer depletion
BNL	Brookhaven National Library
°C	degree Celsius
CDE	committed dose equivalent
CEDE	committed effective dose equivalent
cfm	cubic feet per minute
CFR	*Code of Federal Regulations*
Ci/min	curies per minute
CLB	current licensing basis
CR	control room
CREVS	control room emergency ventilation system
CsI	cesium iodide
DBA	design-basis accident
DCF	dose conversion factor
DE	Dose Equivalent
DE I-131	Dose Equivalent Iodine 131
DE Xe-133	Dose Equivalent Xenon 133
DF	decontamination factor
EAB	exclusion area boundary
ECCS	emergency core cooling system
EES	emergency exhaust system
EOP	emergency operating procedure
EPA	U.S. Environmental Protection Agency
EQ	equipment qualification
ERE	equipment room envelope
ESF	engineered safety feature
°F	degree Fahrenheit
FCM	fuel centerline melt
FGR	Federal Guidance Report
FHA	fuel handling accident
FHB	Fuel handling building
FSAR	Final Safety Analysis Report
ft^3	cubic feet
GDC	General Design Criterion/Criteria
gpm	gallons per minute
GWD/MTU	gigawatt days per metric ton of uranium
HCl	hydrochloric acid
HVAC	heating, ventilation, and air conditioning

HNO_3	nitric acid
I-131	Iodine-131
I&C	instrumentation and control
IEEE	Institute of Electrical and Electronics Engineers
IN	Information Notice
JFD	joint frequency distribution
Kr	krypton
kw/ft	kilowatt per foot
LAR	license amendment request
lbm	pounds mass
LCO	limiting condition for operation
LLB	letdown line break
LOAC	loss of non-emergency AC power
LOCA	loss-of-coolant accident
LOOP	loss of offsite poser
LPZ	low population zone
LRA	locked rotor accident
LWR	light-water reactor
m	meter
m/sec	meter per second
mrem	millirem
MSIV	main steam isolation valve
MSLB	main steam line break
MSSV	main steam safety valve
MWt	megawatt thermal
NRC	U.S. Nuclear Regulatory Commission
pH	scale of acidity
PNNL	Pacific Northwest National Laboratory
POR	period of record
ppm	parts per million
PWR	pressurized-water reactor
QA	quality assurance
Q_{cb}	control building inleakage
Q_{cr}	control room building inleakage
RAI	request for additional information
RB	reactor building
RCP	reactor coolant pump
RCS	reactor coolant system
RE	rod ejection
REA	rod ejection accident
rem	roentgen equivalent man
RG	Regulatory Guide
RHR	residual heat removal
RIS	Regulatory Issue Summary
RWST	refueling water storage tank
SE	safety evaluation
SG	steam generator
SGTR	steam generator tube rupture
SI	safety injection

SR	surveillance requirement
SRP	Standard Review Plan
Sv	sievert
TEDE	total effective dose equivalent
TID	Technical Information Document
TMI	Three Mile Island
TS	Technical Specification
TSC	technical support center
TSP	trisodium phosphate dodecahydrate
VFTP	ventilation filter testing program
XE-133	Xenon-133
µCi/cc	micro curies per cubic centimeter
µCi/gm	micro curies per gram

SUBJECT: CALLAWAY PLANT, UNIT NO. 1 - ISSUANCE OF AMENDMENT NO. 233 FOR ADOPTION OF ALTERNATIVE SOURCE TERM AND REVISION OF TECHNICAL SPECIFICATIONS (EPID L-2021-LLA-0177) DATED SEPTEMBER 20, 2023

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