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

July 31, 2023

Mr. R. Keith Brown
Regulatory Affairs Director
Southern Nuclear Operating Co., Inc.
3535 Colonnade Parkway
Birmingham, AL 35243

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2, ISSUANCE OF AMENDMENTS NOS. 219 AND 202, REGARDING ALTERNATE SOURCE TERM, TSTF-51, TSTF-471, AND TSTF-490 (EPID L-2022-LLA-0096)

Dear Mr. Brown:

The Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 219 to Renewed Facility Operating License NPF-68 and Amendment No. 202 to Renewed Facility Operating License NPF-81 for the Vogtle Electric Generating Plant (Vogtle), Units 1 and 2, respectively. The amendments consist of changes to the License and technical specifications (TSs) in response to your application dated June 30, 2022, as supplemented by letters dated February 6 and March 24, 2023.

The amendments revise the licensing basis to support a selective scope application of the Alternative Source Term radiological analysis methodology and modifies TS 1.1, "Definitions," TS 3.3.6, "Containment Ventilation Isolation Instrumentation;" TS 3.4.16, "RCS [Reactor Coolant System] Specific Activity," TS 3.9.1, "Boron Concentration," TS 3.9.2, "Unborated Water Source Isolation Valves;" TS 3.9.3, "Nuclear Instrumentation," and TS 3.9.4, "Containment Penetrations," consistent with Technical Specifications Task Force (TSTF) Travelers TSTF-51-A, Revision 2, "Revise containment requirements during handling irradiated fuel and core alterations," TSTF-471-A, Revision 1, "Eliminate use of term CORE ALTERATIONS in ACTIONS and Notes," and TSTF-490-A, Revision 0, "Deletion of E Bar Definition and Revision to RCS Specific Activity Tech Spec."

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

Sincerely,

/RA/

John G. Lamb, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-424 and 50-425

Enclosures:
1. Amendment No. 219 to NPF-68
2. Amendment No. 202 to NPF-81
3. Safety Evaluation

cc: Listserv

UNITED STATES
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SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-424

VOGTLE ELECTRIC GENERATING PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 219
Renewed License No. NPF-68

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

   A. The application for amendment to the Vogtle Electric Generating Plant, Unit 1 (the facility) Renewed Facility Operating License No. NPF-68 filed by the Southern Nuclear Operating Company, Inc. (the licensee), acting for itself, Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the owners), dated June 30, 2022, as supplemented by letters dated February 6 and March 24, 2023, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 set forth in 10 CFR Chapter I;

   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 hereby amended by page 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-68 is hereby amended to read as follows:

   Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 219, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

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Date: 2023.07.31 12:51:46 -04'00'

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to License No. NPF-68
and the Technical Specifications

Date of Issuance: July 31, 2023

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

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-425

VOGTLE ELECTRIC GENERATING PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 202
Renewed License No. NPF-81

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

   A. The application for amendment to the Vogtle Electric Generating Plant, Unit 2 (the facility) Renewed Facility Operating License No. NPF-81 filed by the Southern Nuclear Operating Company, Inc. (the licensee), acting for itself, Georgia Power Company Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the owners), dated June 30, 2022, as supplemented by letters dated February 6 and March 24, 2023, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 set forth in 10 CFR Chapter I;

   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 hereby amended by page 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-81 is hereby amended to read as follows:

   <u>Technical Specifications and Environmental Protection Plan</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 202, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

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Date: 2023.07.31 12:52:52 -04'00'

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to License No. NPF-81
and the Technical Specifications

Date of Issuance: July 31, 2023

ATTACHMENT

VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2

TO LICENSE AMENDMENT NO. 219

RENEWED FACILITY OPERATING LICENSE NO. NPF-68

DOCKET NO. 50-424

AND

TO LICENSE AMENDMENT NO. 202

RENEWED FACILITY OPERATING LICENSE NO. NPF-81

DOCKET NO. 50-425

Replace the following pages of the Licenses 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.

Remove Pages	Insert Pages
License	License
License No. NPF-68, page 4	License No. NPF-68, page 4
License No. NPF-81, page 3	License No. NPF-81, page 3
TSs	TSs
1.1-2	1.1-2
1.1-3	1.1-3
1.1-4	1.1-4
1.1-5	1.1-5
1.1-6	1.1-6
1.1-7	1.1-7
------	1.1-8
3.3.6-3	3.3.6-3
3.3.6-6	3.3.6-6
3.4.16-1	3.4.16-1
3.4.16-2	3.4.16-2
3.4.16-3	-----------
3.4.16-4	-----------
3.9.1-1	3.9.1-1
3.9.2-1	3.9.2-1
3.9.3-1	3.9.3-1
3.9.4-1	3.9.4-1

(1) Maximum Power Level

Southern Nuclear is authorized to operate the facility at reactor core power levels not in excess of 3625.6 megawatts thermal (100 percent 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. 219, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Southern Nuclear Operating Company shall be capable of establishing containment hydrogen monitoring within 90 minutes of initiating safety injection following a loss of coolant accident.

(4) Deleted

(5) Deleted

(6) Deleted

(7) Deleted

(8) Deleted

(9) Deleted

(10) Mitigation Strategy License Condition

The licensee shall develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:
1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training and response personnel

(b) Operations to mitigate fuel damage considering the following:
1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for Implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy

(2) Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia, pursuant to the Act and 10 CFR Part 50, to possess but not operate the facility at the designated location in Burke County, Georgia, in accordance with the procedures and limitations set forth in this license;

(3) Southern Nuclear, pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) Southern Nuclear, 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;

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

(6) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as my be produced by the operation of the facility authorized herein.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter 1 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) <u>Maximum Power Level</u>

Southern Nuclear is authorized to operate the facility at reactor core power levels not in excess of 3625.6 megawatts thermal (100 percent power) in accordance with the conditions specified herein.

(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 202 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

The Surveillance requirements (SRs) contained in the Appendix A Technical Specifications and listed below are not required to be performed immediately upon implementation of Amendment No. 74. The SRs listed below shall be

CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.
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 the OPERABILITY of required alarm, interlock, and trip functions. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy. The COT may be performed by means of any series of sequential, overlapping, or total channel steps, and each step must be performed within the Frequency in the Surveillance Frequency Control Program for the devices included in the step.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or other 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. Unit 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 Committed Dose Equivalent (CDE) or Committed Effective Dose Equivalent (CEDE) dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11.

(continued)

DOSE EQUIVALENT XE-133	DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."
ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME	The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC (including transmitters in the Online Monitoring Program), or the components have been evaluated in accordance with an NRC approved methodology.
INSERVICE TESTING PROGRAM	The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).

(continued)

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known to not interfere with the operation of leakage detection systems; or

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System primary to secondary LEAKAGE);

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE;

c. Pressure Boundary LEAKAGE

LEAKAGE (except primary to secondary LEAKAGE) through a fault in an RCS component body, pipe wall, or vessel wall. LEAKAGE past seals, packing, and gaskets is not pressure boundary LEAKAGE.

MASTER RELAY TEST

A MASTER RELAY TEST shall consist of energizing each master relay and verifying the OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

MODE

A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

(continued)

OPERABLE — OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 14 of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, Cold Overpressure Protection System (COPS) arming temperature and the nominal PORV setpoints for the COPS, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Unit operation within these operating limits is addressed in individual specifications.
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3625.6 MWt.

(continued)

REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC (including transmitters in the Online Monitoring Program), or the components have been evaluated in accordance with an NRC approved methodology.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all RCCAs verified fully inserted by two independent means, it is not necessary to account for a stuck rod in the SDM calculation. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the hot zero power temperatures.
SLAVE RELAY TEST	A SLAVE RELAY TEST shall consist of energizing each slave relay and verifying the OPERABILITY of each slave relay. The SLAVE RELAY TEST shall include, as a minimum, a continuity check of associated testable actuation devices.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.

(continued)

1.1 Definitions (continued)

THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.
TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)	A TADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of required alarm, interlock, and trip functions. The TADOT shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the required accuracy. The TADOT may be performed by means of any series of sequential, overlapping, or total channel steps, and each step must be performed within the Frequency in the Surveillance Frequency Control Program for the devices included in the step.

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[a]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 350
4	Hot Shutdown[b]	< 0.99	NA	350 > T_{avg} > 200
5	Cold Shutdown[b]	< 0.99	NA	≤ 200
6	Refueling[c]	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. ------------NOTE-------------- Only applicable during movement of recently irradiated fuel assemblies within containment. ---------------------------------- No radiation monitoring channels OPERABLE. OR Required Action and associated Completion Time for Condition A not met.	C.1	Place and maintain containment purge and exhaust valves in closed position.	Immediately
	OR		
	C.2	Enter applicable Conditions and Required Actions of LCO 3.9.4, "Containment Penetrations," for containment purge supply and exhaust isolation penetrations not in required status.	Immediately

Table 3.3.6-1 (page 1 of 1)
Containment Ventilation Isolation Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1.	Manual Initiation	1,2,3,4	2	SR 3.3.6.6	NA
2.	Automatic Actuation Logic and Actuation Relays	1,2,3,4	2	SR 3.3.6.2 SR 3.3.6.3 SR 3.3.6.5	NA
3.	Containment Radiation	1,2,3,4,6[c]	2[a]	SR 3.3.6.1 SR 3.3.6.4 SR 3.3.6.7 SR 3.3.6.8	
	a. Gaseous (RE-2565C)				(b)
	b. Particulate (RE-2565A)				(b)
	c. Iodine (RE-2565B)				(b)
	d. Area Low Range (RE-0002, RE-0003)				≤ 15 mr/h[c] ≤ 50x background[d]
4.	Safety Injection[d]	1,2,3,4	Refer to LCO 3.3.2, "ESFAS Instrumentation,"Function 1, for all initiation functions and requirements.		

(a) Containment ventilation radiation (RE-2565) is treated as one channel and is considered OPERABLE if the particulate (RE-2565A) and iodine monitors (RE-2565B) are OPERABLE or the noble gas monitor (RE-2565C) is OPERABLE.

(b) Setpoints will not exceed the limits of Specifications 5.5.4.h and 5.5.4.i of the Radioactive Effluent Controls Program.

(c) During movement of recently irradiated fuel assemblies within containment.

(d) During MODES 1, 2, 3, and 4.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 not within limit.	-------------------NOTE-------------------- LCO 3.0.4.c is applicable. ---	
	A.1 Verify DOSE EQUIVALENT I-131 ≤ 60 μCi/gm.	Once per 4 hours
	AND	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. DOSE EQUIVALENT XE-133 not within limit.	-------------------NOTE-------------------- LCO 3.0.4.c is applicable. ---	
	B.1 Restore DOSE EQUIVALENT XE-133 to within limit.	48 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met. OR DOSE EQUIVALENT I-131> 60 μCi/gm.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	---------------------------NOTE--------------------------- Only required to be performed in MODE 1. -- Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 280 μCi/gm.	In accordance with the Surveillance Frequency Control Program
SR 3.4.16.2	---------------------------NOTE--------------------------- Only required to be performed in MODE 1. -- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm.	In accordance with the Surveillance Frequency Control Program AND Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentrations of the Reactor Coolant System, the refueling canal, and the refueling cavity shall be maintained within the limit specified in the COLR.

APPLICABILITY: MODE 6.

---NOTE---
Only applicable to the refueling canal and refueling cavity when connected to the RCS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend positive reactivity additions.	Immediately
	AND	
	A.2 Initiate action to restore boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.1.1	Verify boron concentration is within the limit specified in the COLR.	In accordance with the Surveillance Frequency Control Program

3.9 REFUELING OPERATIONS

3.9.2 Unborated Water Source Isolation Valves

LCO 3.9.2 Each valve used to isolate unborated water sources shall be secured in the closed position.

---NOTE--
Valves in the flowpath from the RMWST, through the chemical mixing tank, to the suction of the charging pumps may be opened under administrative control provided the reactor coolant system is in compliance with Specification 3.9.1 and the high flux at shutdown alarm is OPERABLE.

APPLICABILITY: MODE 6.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each unborated water source isolation valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -------------NOTE-------------- Required Action A.2 must be completed whenever Condition A is entered. ---------------------------------- One or more valves not secured in closed position.	A.1 Initiate actions to secure valve in closed position.	Immediately
	AND	
	A.2 Perform SR 3.9.1.1 (verify boron concentration).	12 hours

3.9 REFUELING OPERATIONS

3.9.3 Nuclear Instrumentation

LCO 3.9.3 Two source range neutron flux monitors shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One source range neutron flux monitor inoperable.	----------NOTE------------- CORE ALTERATIONS may continue to restore an inoperable source range neutron flux monitor. --------------------------------	
	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend positive reactivity additions.	Immediately
B. --------------NOTE------------- Condition A entry is required when Condition B is entered. ----------------------------------- Two source range neutron flux monitors inoperable.	B.1 Initiate action to restore one source range neutron flux monitor to OPERABLE status.	Immediately
	AND	
	B.2 Perform SR 3.9.1.1 (verify boron concentration).	Once per 12 hours

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

LCO 3.9.4 The containment penetrations shall be in the following status:

a. The equipment hatch is capable of being closed and held in place by four bolts;

b. The emergency and personnel air locks are isolated by at least one air lock door, or if open, the emergency and personnel air locks are isolable by at least one air lock door with a designated individual available to close the open air lock door(s); and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by at least two OPERABLE Containment Ventilation Isolation valves

--NOTE--
Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

APPLICABILITY: During movement of recently irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend movement of recently irradiated fuel assemblies within containment.	Immediately

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 219 TO RENEWED FACILITY OPERATING LICENSE NPF-68

AND

AMENDMENT NO. 202 TO RENEWED FACILITY OPERATING LICENSE NPF-81

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-424 AND 50-425

1.0 INTRODUCTION

By application dated June 30, 2022 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML22181B066), as supplemented by letters dated February 6, and March 24, 2023 (ML23037A856 and ML23083B398), Southern Nuclear Operating Company, Inc. (SNC, the licensee), requested changes to the technical specifications (TSs) for the Vogtle Electric Generating Plant (Vogtle), Units 1 and 2.

The proposed amendments would revise the licensing basis to support full implementation of the Alternative Source Term (AST) radiological analysis methodology and modifies TS 1.1, “Definitions,” TS 3.3.6, “Containment Ventilation Isolation Instrumentation,” TS 3.4.16, “RCS [Reactor Coolant System] Specific Activity,” TS 3.9.1, “Boron Concentration;” TS 3.9.2, “Unborated Water Source Isolation Valves,” TS 3.9.3, “Nuclear Instrumentation;” and TS 3.9.4, “Containment Penetrations,” consistent with Technical Specification Task Force (TSTF) Travelers TSTF-51-A, Revision 2, “Revise containment requirements during handling irradiated fuel and core alterations,” (ML20217E091, TSTF-471-A, Revision 1, “Eliminate use of term CORE ALTERATIONS in ACTIONS and Notes,” (ML19101A215), and TSTF-490-A, Revision 0, “Deletion of E Bar Definition and Revision to RCS Specific Activity Tech Spec,” (ML052630462).

The proposed amendments would revise the licensing basis to use an AST in evaluating the offsite and Control Room (CR) radiological consequences of the Vogtle, Units 1 and 2, design basis accidents (DBAs) as allowed by Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67, "Accident source term," and described in Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.183, "Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000 ((ML003716792).

TSTF-51 would revise certain TSs to remove the requirements for certain engineered safety features (ESF) systems to operate after sufficient radioactive decay of irradiated fuel has occurred following a plant shutdown.

TSTF-471 eliminates the use of the defined term CORE ALTERATIONS from TS 3.9.1, “Boron Concentration,” TS 3.9.2, “Unborated Water Source Isolation Valves,” and TS 3.9.4, “Containment Penetrations.”

TSTF-490 would replace the current limits on primary coolant gross specific activity with limits on primary coolant noble gas activity.

The supplements dated February 6 and March 24, 2023, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff’s original proposed no significant hazards consideration determination as published the *Federal Register* on September 6, 2022 (87 FR 54553).

2.0 REGULATORY EVALUATION

2.1 Proposed Changes

The proposed amendments would revise the licensing basis to use a full implementation of AST in evaluating the offsite and CR radiological consequences of the Vogtle, Units 1 and 2, DBAs.

SNC proposed to revise TS 1.1, TS 3.3.6, TS 3.4.16, TS 3.9.1, TS 3.9.2, TS 3.9.3, and TS 3.9.4. The proposed TS pages can be found in the letter dated June 30, 2022. The additions and deletions can be found in the markup pages in Attachment 1 in the SNC letter dated June 30, 2022. The licensee is not proposing any variations from TSTF-490.

2.2 Variations from TSTF-51 and TSTF-471

SNC proposed the TS changes based on the Standard Technical Specification (STS) changes described in TSTF-51, Revision 2, and TSTF-471, Revision 1, but SNC proposes variations from the NUREG-1431, “Standard Technical Specification – Westinghouse Plants, Revision 5, Volume 1, “Specifications,” and Volume 2, “Bases,” dated September 2021 (ML21259A155 and ML21259A159, respectively). in TSTF-51 and TSTF-471, as identified below, and include differing TS numbers and TS titles, where applicable.

In its letter dated June 30, 2022, the licensee stated:

> 1. The definition of CORE ALTERATION is being retained in TS Section 1.1, “Definitions,” because this terminology continues to be used in a number of TSs, which are not being modified as a result of this amendment request. This is an administrative variation from TSTF-471.
>
> 2. The control room emergency filtration system (CREFS) actuation instrumentation and the CREFS continue to be assumed to provide isolation, pressurization, and filtration of the MCR in the event of an FHA. Since this system and associated isolation instrumentation are mitigation systems necessary to maintain dose to personnel in the MCR below the

regulatory and regulatory guidance limits for an FHA, the following TSs and support TSs and associated Bases are not modified:

- TS 3.3.7, “Control Room Emergency Filtration/Pressurization System (CREFS) Actuation Instrumentation,”
- TS 3.7.10, “Control Room Emergency Filtration/Pressurization System (CREFS) - Both Units Operating,”
- TS 3.7.11, “Control Room Emergency Filtration/Pressurization System (CREFS) - One Unit Operating,”
- TS 3.8.2, “AC Sources – Shutdown,”
- TS 3.8.5, “DC Sources – Shutdown,”
- TS 3.8.8, “Inverters – Shutdown,”
- TS 3.8.10, “Distribution Systems – Shutdown,” and
- TS 3.9.7, “Refueling Cavity Water Level.”

This is a plant-specific variation from TSTF-51 and 471.

3. The applicability requirements associated with the containment ventilation isolation instrumentation are shown in TS Table 3.3.6-1. This is a presentation difference from the applicability requirements shown in the NUREG-1431 TS 3.3.6 marked up pages in TSTF-51. However, the proposed changes to footnote (c) in TS Table 3.3.6-1 are consistent with those shown in TSTF-51. These proposed changes are administrative variations from TSTF-51.

4. TS 3.9.3, “Nuclear Instrumentation,” Required Actions were not modified in accordance with TSTF-471. However, proposed Note added to Required Action A.1 is consistent with the intent of the proposed Note in TSTF-571-T, “Revise Actions for Inoperable Source Range Neutron Flux Monitor” (Reference 6). TSTF-571-T was accepted for use by the NRC as documented in a letter to the TSTF dated October 4, 2018 (Reference 7). Movement of fuel sources and reactivity control components within the reactor vessel is currently covered by the Core Alteration definition. Since the VEGP TSs retain the definition of Core Alteration, the required action continues to require suspension of core alterations and the note was modified to use the term Core Alterations. These proposed changes are considered administrative variations from TSTF-471.

SNC considers the differences from TSTF-51 and TSTF-471 listed herein to be either: 1) necessary variations to maintain the requirements for required safety systems assumed in the VEGP FHA analysis; or 2) minor variations or deviations that are administrative in nature.

2.3 Alternate Source Term (AST) Regulatory Evaluation

SNC’s request was pursuant to 10 CFR 50.67, “Accident source term,” which provides a mechanism for licensed power reactors to replace the traditional source term used in the radiological consequence analyses of DBAs discussed in Section 3.0 below, as described in NRC Regulatory Guide (RG) 1.183, “Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” July 2000 (ML003716792). Vogtle, Units 1 and 2, current DBA radiological consequence analyses are based on the source term from U.S. Atomic Energy Commission Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," dated March 23, 1962.

The NRC staff evaluated SNC's analysis of the radiological consequences of the affected DBAs for implementation of the AST methodology, and the associated changes to the TSs proposed by the licensee, against the radiological dose requirements specified in 10 CFR 50.67(b)(2), and dose limits specified in 10 CFR Part 50, Appendix A, "General Design Criteria (GDC) for Nuclear Power Plants," Criterion 19, "Control Room," Section 50.67(b)(2) states:
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)[2] total effective dose equivalent (TEDE).
>
> (ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).
>
> (iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.
>
> [2] The use of 0.25 SV (25 rem) TEDE is not intended to imply that this value constitutes an acceptable limit for emergency doses to the public under accident conditions. Rather, this 0.25 Sv (25 rem) TEDE value has been stated in this section as a Reference value, which can be used in the evaluation of proposed design basis changes with respect to potential reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation.

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

- 10 CFR 50.67, "Accident source term,"
- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants": GDC 19, "Control room,"
- NRC RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000 (ML003716792),
- NRC RG 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors," Revision 1, January 2007 (ML063560144),
- NUREG-0800, Standard Review Plan (SRP) Section 6.4, "Control Room Habitability System," Revision 3, March 2007 (ML070550069):

- Section 6.5.2, “Containment Spray as a Fission Product Cleanup System,” Revision 4, March 2007 (ML070190178), and
- Section 15.0.1, “Radiological Consequence Analyses Using Alternative Source Terms,” Revision 0, July 2000 (ML003734190).

This safety evaluation (SE) addresses the impact of the proposed changes on previously analyzed DBA’s radiological consequences and the acceptability of the revised analysis results.

The regulatory requirements from which the NRC staff based its acceptance are the accident radiation dose values in 10 CFR 50.67, and the accident specific guideline values in RG position 4.4 of RG 1.183 and Table 1 of SRP Section 15.0.1. The licensee has not proposed any significant deviation or departure from the guidance provided in RG 1.183.

2.4 TSTF-51, TSTF-471, and TSTF-490 Regulatory Evaluation

The NRC staff's TSTF-51, TSTF-471, and TSTF-490 evaluation is based upon the following regulations, RGs, and standards:

- 10 CFR 50.36, “Technical specifications.”

- 10 CFR 50.67.

- 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” GDC 13, “Instrumentation and control,” GDC 16, “Containment design,” GDC 19, “Control room,” GDC 20, “Protection system functions,” GDC 21, “Protection system reliability and testability,” GDC 22, “Protection system independence,” GDC 23, “Protection system failure modes,” GDC 24, “Separation of protection and control systems,” and GDC 64, “Monitoring radioactivity releases.”

2.5 Atmospheric Dispersion Factors Regulatory Evaluation

The NRC staff's evaluation of the proposed atmospheric dispersion factors is based upon the following RGs, codes, and standards:

- 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” Criterion 19, “Control room,”

- NRC NUREG-0800, SRP Section 2.3.4, “Short-Term Atmospheric Dispersion Estimates for Accident” (ML070730398),

- NRC RG 1.23, “Meteorological Monitoring Programs for Nuclear Power Plants” (ML070350028),

- NRC RG 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants” (ML003740205),

- NRC RG 1.183, “Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors” (ML003716792),

- NRC RG 1.194, “Atmospheric Relative Concentrations for Control Room

Radiological Habitability Assessments at Nuclear Power Plants" (ML031530505).

3.0 TECHNICAL EVALUATION

3.1 Radiological Consequences of Design Basis Accidents (DBAs)

SNC has proposed a licensing basis change for its offsite and CR DBA dose consequence analysis for Vogtle, Units 1 and 2. The proposed change will implement an AST methodology for determining DBAs offsite and CR doses. For full implementation of the AST DBAs analysis methodology, the dose acceptance criteria specified in 10 CFR 50.67 provides an alternative to the previous whole body and thyroid dose guidelines stated in 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance," and GDC 19.

As discussed in RG 1.183, Regulatory Position 1.2.1, "Full Implementation," states, in part, that:

> Full implementation is a modification of the facility design basis that addresses all characteristics of the AST, including composition, and magnitude of the radioactive material, its chemical and physical form, and the timing of its release. Full implementation revises the plant licensing basis to specify the AST in place of the previous accident source term and establishes the TEDE dose as the new acceptance criteria.

This applies not only to the analyses performed in this license amendment request (LAR), but also to all future design basis dose consequence analyses at Vogtle, Units 1 and 2. At a minimum for full implementation of the AST, the DBA loss-of-coolant accident (LOCA) must be reanalyzed. Since, upon issuance of this LAR, the AST and TEDE criteria will become part of the design basis for Vogtle, Units 1 and 2, new LARs of the AST would not require prior NRC approval unless stipulated by 10 CFR 50.59, "Changes, tests, and experiments," or unless the new LAR involved a change to a TS. However, a change from an approved AST to a different AST that is not approved for use at Vogtle, Units 1 and 2, would require a LAR under 10 CFR 50.67.

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

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

SNC has performed its evaluation based on full implementation of the AST as defined in RG 1.183, with the exception of the equipment qualification (EQ). The licensee has determined that the current TID-14844 AST will remain the licensing basis for EQ. Regulatory Position 6 of RG 1.183 states that the NRC staff is assessing the effect of increased cesium releases on EQ doses to determine whether licensee action is warranted and that until such time as this generic issue is resolved, licensees may use either the AST or the TID-14844 assumptions for performing the required EQ analyses. This issue has been resolved as documented in a memorandum dated April 30, 2001, “Initial Screening of Candidate Generic Issue 187, ’The Potential Impact of Postulated Cesium Concentration on Equipment Qualification in the Containment Sump,’” (ML011210348), and in NUREG-0933, Supplement 25, June 2001, “A Prioritization of Generic Safety Issues” (ML012190402). The conclusion of Generic Issue 187 states the following:

> [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.

Based on the above, the NRC staff finds that it is acceptable for the TID-14844 accident source term to remain the licensing basis for EQ at Vogtle, Units 1 and 2. Guidance in RG 1.183, Regulatory Position 4.3, “Other Dose Consequences,” states, in part, that:

> The guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737 (Ref. 2) [ML051400209 and ML102560009]. Design envelope source terms provided in NUREG-0737 should be updated for consistency with the AST. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be expressed in terms of TEDE.

As stated in its letter dated February 6, 2023, the SNC LAR is a full scope application of the AST methodology for full implementation. SNC’s LAR seeks AST implementation for radiological consequences of major DBAs, specifically: LOCA, FHA, Main Steam Line Break (MSLB), Steam Generator Tube Rupture (SGTR), Control Rod Ejection (CRE), and Locked Rotor Analysis (LRA). It also seeks to implement TSTF-51, TSTF-471, and TSTF-490. The licensee is not proposing physical changes to the plant.RG 1.183, Regulatory Position 1.3.2, “Re-Analysis Guidance,” states, in part, that:

> The NRC staff does not expect a complete recalculation of all facility radiological analyses, but does expect licensees to evaluate all impacts of the proposed changes and to update the affected analyses and the design bases appropriately. An analysis is considered to be affected if the proposed modification changes one or more assumptions or inputs used in that analysis such that the results, or the conclusions drawn on those results, are no longer valid.

The only changes of assumptions or inputs are for the DBAs contained in the LAR, and those analyses’ results have been submitted. All other existing analyses remain the same and, as mentioned above, there are no physical changes to the plant being proposed.

A full implementation of the AST is proposed for Vogtle, Units 1 and 2. Therefore, to support the licensing, and plant operation changes discussed in the LAR, SNC analyzed the following accidents employing the AST as described in RG 1.183.

- LOCA;
- FHA;
- MSLB accident;
- SGTR accident;
- CRE accident (CREA);
- LRA.

The DBA dose consequence analyses evaluated the integrated TEDE dose at the exclusion area boundary (EAB) for the worst 2-hour period following the onset of the accident. The integrated TEDE doses at the outer boundary of the low population zone (LPZ) during the entire period of the passage of the radioactive cloud resulting from postulated release of fission products, and the integrated dose to a Vogtle, Units 1 and 2, CR operator were evaluated for the duration of the accident. The LOCA, FHA, MSLB, SGTR, CREA, and LRA dose consequence analyses was performed by SNC using the "RADTRAD: Simplified Model for RADionuclide Transport and Removal and Dose Estimation" (RADTRAD) Version 3.10. The development of the RADTRAD radiological consequence computer code was sponsored by the NRC, as described in NUREG/CR-6604 (ML15092A284 and ML081850607) and was developed by Sandia National Laboratories for the NRC. The code estimates transport and removal of radionuclides and radiological consequence doses at selected receptors. The NRC staff performs independent confirmatory dose evaluations using the RADTRAD version 5.0.3 computer code.

Each DBA radiological source term used in the AST analyses was developed based on a core power level of 3,636 megawatts thermal (MWt) including measurement uncertainties. The use of 3,636 MWt for the AST DBA radiological source term analyses bounds the current licensed core thermal power level of 3,625.6 MWt and is, therefore, acceptable to the NRC staff for use in the full scope implementation of the AST at Vogtle, Units 1 and 2. In addition, to account for potential cycle-to-cycle variations, SNC applied margin factors to the core inventory. The margin factors for the various isotopes in each of the accidents analyzed vary from analysis to analysis. The margin factors add conservatism and margin for isotopes that are critical to the calculation of dose consequences for each DBA. Each margin factor adds extra margin above the reactor concentrations for an equilibrium core.

Guidance in RG 1.183, Regulatory Position 3.1, "Fission Product Inventory," states, in part, that:

> The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power [RTP] times the emergency core cooling system (ECCS) evaluation uncertainty.[8] The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values[9]. The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 (Ref. 17) [Croff, A.G. "*A User's Manual for the OR/GEN 2 Computer Code,"* ORNL/TM-7175, Oak

Ridge National Laboratory, July 1980] or ORIGEN-ARP (Ref. 18) [Bowman, S. M., Leal, L.C., "The ORIGNARP Input Processor for ORIGEN-ARP," Appendix F7.A in *SCALE: A Modular Code System for Performing Standardized Analyses for Licensing Evaluation, NUREG/CR-0200, USNRC, March 1997*].

[8] The uncertainty factor used in determining the core inventory should be that value provided in Appendix K to 10 CFR Part 50, typically 1.02.

[9] Note that for some radionuclides, such as Cs-137, equilibrium will not be reached prior to fuel offload. Thus, the maximum inventory at the end of life should be used.

In accordance with RG 1.183, SNC developed the equilibrium core activity inventory using the ORIGEN-ARP. The determination of core inventory is dependent upon full power, core average conditions. The nominal inventory was based on an equilibrium cycle modeled with lead rod burnup of 62 gigawatt-days per metric ton of uranium (GWd/MTU). The NRC staff finds this approach to be consistent with current regulatory guidance and is, therefore, acceptable.

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

3.2 Atmospheric Dispersion and Meteorology

The atmospheric dispersion values (χ/Qs) for the EAB and the LPZ used in this analysis for Vogtle, Units 1 and 2, are consistent with the current licensing basis (CLB). In this LAR, SNC proposes new and revised χ/Q values for the CR to address potential leakage from the Refueling Water Storage Tank (RWST) vent and releases from the secondary side for evaluation of non-LOCA radiological consequences. The determination of CR χ/Q values were made using the ARCON96 atmospheric dispersion model (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes," (ML17213A187)) pursuant to the guidance of RG 1.194. The NRC staff reviewed SNC's new atmospheric dispersion analyses as described below.

3.2.1 Meteorological Data

In support of the LAR, SNC provided the hourly onsite meteorological data from calendar years 1998 through 2000, that was used in the analysis. The meteorological data was formatted for the ARCON96 atmospheric dispersion code to calculate updated χ/Q values for the CR. This format contained hourly data on wind speed, wind direction, and atmospheric stability class. The NRC staff previously completed a detailed review related to the acceptability and representativeness of the meteorological data for the VEGP site location in the 2009 Safety Evaluation Report for an Early Site Permit (ESP) (Accession No. ML092290630). Based on this review, the staff considers the onsite meteorological data acceptable for use in making new calculations for the CR atmospheric dispersion analyses used to support this LAR.

3.2.2 Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) Atmospheric Dispersion Factors

The χ/Q values used in the radiological consequence analyses for the EAB and the LPZ are consistent with the CLB in Vogtle, Units 1 and 2, Updated Final Safety Analysis Report (UFSAR) Table 2.3.4-1. RG 1.183, Section 5.3, "Meteorology Assumptions," states that:

> Atmospheric dispersion values (χ/Q) for the EAB, the LPZ, and the control room [CR] that were approved by the staff during the initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses identified by this guide.

The χ/Q values for the EAB and the LPZ used in the radiological consequence analyses are shown in Table 3.1 of the LAR.

3.2.3 Control Room (CR) Atmospheric Dispersion Estimates

In support of the LAR, SNC used the computer code ARCON96 to estimate new χ/Q values for the CR for potential accidental releases of radioactive material. The existing χ/Q values included releases from the Containment, Containment Hatch Door, and Fuel Handling Building. The new χ/Q values were developed for the RWST release points for the LOCA as well as the North and South Main Steam Valve Rooms for secondary side releases in non-LOCA events such as the CREA, LRA, SGTR, and MSLB.

Figure 3.1, "Air Intake Locations and Release Points," of the LAR dated June 30, 2022, provides a sketch of the general layout of Vogtle, Units 1 and 2, that has been annotated to highlight the onsite release and receptor point locations. All releases are taken as ground level releases. Table 3.2, "Distance and Geometry of Release and Receptor Locations," of the LAR dated June 30, 2022, provides information related to the relative elevations of the release-receptor combinations, the straight-line horizontal distance between the release point and the receptor location, and the direction from the receptor location to the release points. Finally, Table 3.3 of the LAR dated June 30, 2022, lists the χ/Q values calculated with ARCON96 from the various release points to the CR intakes for the 0-2 hours, 2-8 hours, 8-24 hours, 1-4 days, and 4-30 day time intervals.

The NRC staff confirmed SNC's CR atmospheric dispersion estimates by running the ARCON96 computer model and obtaining similar results. The NRC staff reviewed the inputs and assumptions provided by SNC for each of the release pathway-receptor combinations including the source-receptor distances, directions, heights, and area values. The NRC staff also confirmed that the analysis and assumptions were consistent with the guidance of RG 1.194. Based on the results of its confirmatory analysis and the fact that SNC followed NRC guidance, the NRC staff finds SNC's CR χ/Q values acceptable for use in the radiological consequence assessments.

3.2.4 Meteorology and Atmospheric Dispersion Conclusion

The NRC staff reviewed the guidance, assumptions, and methodology used by SNC to assess the χ/Q values associated with postulated releases from the potential release pathways. The NRC staff found that the licensee used methods consistent with regulatory guidance identified in Section 2.0 of this SE. SNC used onsite meteorological data that complied with the guidance of RG 1.23. The inputs and assumptions used to calculate the CR χ/Q values were also consistent with the guidance of RG 1.194. Therefore, based on this review of the licensee's atmospheric dispersion analysis and the NRC staff's own confirmatory analysis, the NRC staff finds the SNC's proposed χ/Q values acceptable for use in calculating the radiological consequences assessments associated with this LAR.

3.3 Control Room (CR) Habitability for All Design Basis Accidents (DBAs)

The Vogtle CR is common to Units 1 and 2. The CREFS is designed to maintain the CR envelope (CRE) at a positive pressure relative to the surrounding area, following postulated accidents. The CREFS is activated on a safety injection (SI) signal and/or high radiation in the normal outside air intakes. CREFS is designed to automatically isolate the CR and start one train of the emergency air filtration system upon a valid signal.

Also, initiation of the CR isolation (CRI) closes the isolation dampers between the normal and emergency systems. Two redundant and physically separated air handling unit trains with a moisture eliminator, an electric preheater, high-efficiency particulate air (HEPA) filters, and charcoal adsorbers are provided for each unit to process intake airflow and recirculated airflow in the combined CR. In emergency operation recirculation mode, the CR maintains a positive pressure of 1/8-inches water gauge (wg) relative to all adjacent areas.

SNC evaluated CR habitability for each DBA assuming that the CREFS automatically transfers to the pressurization mode of operation upon SI or high radiation in the CR signal. The CREFS automatically transfers to the pressurization mode of operation after the CRI signal. For all events which result in a valid SI Signal (SIS) or High Radiation (Hi Rad) signal, the time to achieve CRI is 8.0 seconds after the signal. The 8 seconds to achieve CRI includes 2 seconds of signal delay and 6 seconds for damper repositioning. After CRI, pressurization mode is achieved in an additional 88 seconds in the CR. CR pressurization is achieved 96 seconds after a valid signal. The 96 seconds from SIS / Hi Rad signal generated to CR pressurization mode achieved includes 36 seconds from SIS / Hi Rad generated to lead fan start signal and an additional 30 seconds from lead fan start signal to system low flow detection and another 30 seconds from lead fan low flow detection / lag fan start signal to CR pressure established (i.e., all events conservatively assume failure of lead fan to pressurize the CR).

During normal operation, CR unfiltered make-up flow rate is maintained less than or equal to 2575 cfm with the normal CR HVAC [heating, ventilation, and air conditioning] in service. On detection of a valid CRI signal, the normal outside air intakes for the CR are automatically isolated and the emergency operation/recirculation mode is initiated with 31,000 cubic feet per minute (cfm) filtered recirculation flow rate, and 1,800 cfm of filtered make-up flow rate, which helps maintain a positive pressure in the CR. Air within the CR is recirculated continuously through the emergency air-conditioning units, which contain upstream HEPA filters, charcoal adsorbers, downstream HEPA filters, cooling coil, and fan, to control the room temperature and airborne radioactivity. The outside air required for pressurization is mixed with the return air before it enters the filtration unit. The CR unfiltered in-leakage is 190 cfm, which contains 10 cfm for CR ingress and egress. The licensee increased the assumption of 140 cfm unfiltered in-leakage to a more conservative value of 190 cfm unfiltered in-leakage. This increase in assumed unfiltered in-leakage is more conservative than the CLB and is adopted by SNC to provide operational margin.

The CREFS is designed to maintain the CRE at a positive pressure relative to the surrounding area, following a valid CRI signal. The CR pressurization flow is routed through charcoal and HEPA filters. The CR pressurization charcoal filter efficiency for all iodine species is 99 percent (%) and the HEPA filter efficiency for particulates is also 99 %. The supply and recirculation fans use the same filters with 99% efficiency for all iodine species and particulates. The breathing rates and occupancy factors for the CR are in accordance with the values in Section C, Regulatory Position 4.2.6 of RG 1.183.

The occupancy factors and breathing rates for all accident analysis used in this LAR are consistent with RG 1.183, 4.2.6

> The dose receptor for these analyses is the hypothetical maximum exposed individual who is present in the control room for 100% of the time during the first 24 hours after the event, 60% of the time between 1 and 4 days, and 40% of the time from 4 days to 30 days.[16] For the duration of the event, the breathing rate of this individual should be assumed to be 3.5×10^{-4} cubic meters per second.
>
> [16] This occupancy is modeled in the χ/Q values determined in Reference 22 [*K.G. Murphy and K.W. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19," published in Proceedings of 13th AEC Air Cleaning Conference, Atomic Energy Commission (now USNRC), August 1974*] and should not be credited twice. The ARCON96 Code (Ref. 26) [*J.V. Ramsdell and C.A. Simonen, "Atmospheric Relative Concentrations in Building Wakes, NUREG-6331, Revision 1, USNRC, May 1997*] does not incorporate these occupancy assumptions, making it necessary to apply this correction in the dose calculations.

The proposed LAR does not alter the CLB of the design or operation of the CRE, or the CREFS. In support of the LAR, some of the input assumptions for the radiological dose consequence analysis have been revised. This section on CR habitability is common to and applicable to all DBAs. Inputs, assumptions, and initial conditions that are unique to DBAs analyzed in this LAR are discussed in their respective sections. Section 6.4, Habitability Systems, of the Vogtle, Units 1 and 2, UFSAR describes the CLB CR normal and emergency ventilation systems.

The NRC staff's review determined that SNC 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 2: Control Room Inputs and Assumptions". The NRC staff finds that the inputs, analyses, and assumptions used in determining the hypothetical maximum exposed individual who is present in the CR meet the applicable accident dose criteria and are, therefore, acceptable.

3.4 Loss-of-Coolant Accident (LOCA)

A DBA LOCA is a failure of the RCS that results in the loss of reactor coolant which, if not mitigated, could result in fuel damage including core melt. Analyses are performed using a spectrum of RCS break sizes to evaluate fuel and ECCS performance. A large break LOCA is postulated as the failure of the largest pipe in the RCS. RG 1.183 establishes the large break LOCA as the licensing basis LOCA with regards to radiological consequences since this represents the larger challenge to plant safety features designed to mitigate the release of radionuclides to the environment in the unlikely event that ECCS is not effective. Evaluation of the effectiveness of plant safety features, such as ECCS, has shown that core melt is unlikely. The objective of this DBA is to evaluate the ability of the plant design to mitigate the release of radionuclides to the environment in the unlikely event that ECCS is not effective in preventing core damage. Section 15.6.5 "Loss of Coolant Accidents," of the Vogtle, Units 1 and 2, UFSAR describes the CLB DBA.

The fission product release is assumed to occur in phases over a 2-hour period. When using the AST for the evaluation of a design basis LOCA for a PWR, it is assumed that the initial fission product release to the containment will last for 30 seconds and will consist of the radioactive materials dissolved or suspended in the RCS liquid. After 30 seconds, fuel damage is assumed to begin and is characterized by clad damage that releases the fission product inventory assumed to reside in the fuel gap. The fuel gap release phase is assumed to continue until

30 minutes after the initial breach of the RCS. As core damage continues, the gap release phase ends and the early in-vessel release phase begins. The early in-vessel release phase continues for the next 1.3 hours. SNC used the LOCA source term release fractions, timing characteristics, and radionuclide grouping as specified in RG 1.183 for evaluation of the AST.

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

- Containment leakage directly to the atmosphere,
- Release from the containment mini-purge,
- ESF systems leakage outside of containment, and
- RWST leakage to the atmosphere.

3.4.1 Loss-of-Coolant Accident (LOCA) Source Term

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

The guidance from RG 1.183 specifies that the iodine deposited in the sump water can be assumed to remain in solution as long as the containment sump pH is maintained at or above 7.0. SNC notes in Table A, Regulatory Section A-2 that the containment sump pH has been evaluated for the impact of the alternate source term and confirms that the sump pH remains greater than 7.0. In addition, Vogtle, Units 1 and 2, use trisodium phosphate to create a buffered sump solution that is resistant to change in pH. The NRC staff verified that this LAR did not impact CLB containment sump pH analysis and determined that it is applicable to the AST.

3.4.2 Assumptions on Transport in the Primary Containment

3.4.2.1 Containment Mixing, Natural Deposition, and Leak Rate

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

SNC credited the reduction of airborne radioactivity in the containment by natural deposition. RG 1.183 Appendix A position 3.2 states:

> Reduction in airborne radioactivity in the containment by natural deposition within the containment may be credited. Acceptable models for removal of iodine and aerosols are described in Chapter 6.5.2, "Containment Spray as a Fission Product Cleanup System," of the Standard Review Plan (SRP), NUREG-0800 (Ref. A-1 - ML070190178) and in NUREG/CR-6189, "A Simplified Model of

> Aerosol Removal by Natural Processes in Reactor Containments" (Reference 20 - ML100130305). The latter model is incorporated into the analysis code RADTRAD (Reference 21) *(K.F. Eckerman and J.C. Ryman, "External Exposure to Radionuclides in Air, Water, and Soil," Federal Guidance Report 12, EPA-402-R-93-081, Environmental Protection Agency, 1993*). The prior practice of deterministically assuming that a 50% plateout of iodine is released from the fuel is no longer acceptable to the NRC staff as it is inconsistent with the characteristics of the revised source terms.

In its LAR dated June 30, 2022, Table B of Attachment 4 states SNC's conformance with RG 1.183 Appendix A. Table B states that SNC's analysis for RG 1.183 Regulatory Position 3.2 is, "Conforms–An aerosol natural deposition rate of 0.1 hr^{-1} is assumed based upon values presented in Section VI of NUREG/CR-6189."

SNC incorporated an aerosol natural deposition removal coefficient of 0.1 hr^{-1} into the dose analysis after the termination of the containment spray system and continuing for the duration of the accident. RG 1.183, Regulatory Position 3.7 states that the primary containment should be assumed to leak at the peak pressure TS leak rate for the first 24 hours and that for PWRs, the leak rate may be reduced after the first 24 hours to 50% of the TS leak rate. Accordingly, the licensee assumed a containment leak rate of 0.21% per day for the first 24 hours, after which the containment leak rate is reduced to 0.105% per day for the duration of the accident. The licensee included an additional 5% margin for conservatism not to exceed SNC's Containment Leakage TS values. The licensee assumes the leakage is from both the sprayed and unsprayed regions of the containment to the environment.

3.4.2.2 Containment Spray Assumptions

RG 1.183, Appendix A, Regulatory Position 3.3 states, in part, that:

> The evaluation of the containment sprays should address areas within the primary containment that are not covered by the spray drops. The mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment building, provided that adequate flow exists between these regions, is assumed to be two turnovers of the unsprayed regions per hour, unless other rates are justified. The containment building atmosphere may be considered a single, well-mixed volume if the spray covers at least 90% [percent] of the volume and if adequate mixing of unsprayed compartments can be shown.

For SNC, the volume of the sprayed region is 2.30×10^6 cubic feet (ft^3) and the volume of the unsprayed region is 6.30×10^5 ft^3. A flow rate of 21,000 ft^3 per minute is used between the sprayed and unsprayed volume. This correlates to two turnovers of the unsprayed region volume per hour. In accordance with RG 1.183, Appendix A, Section 3.3, SNC used the mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment building of two turnovers of the unsprayed regions per hour. Since the sprayed region is less than 90 percent of the total containment volume, the licensee used a two-volume model to represent the sprayed and unsprayed regions of the containment. For SNC, the containment spray is initiated at 10 seconds after the LOCA initiation and terminates at 2 hours.

Using the guidance from SRP 6.5.2, “Containment Spray as a Fission Product Cleanup System,” and NUREG/CR-5966, “A Simplified Model of Aerosol Removal by Containment Sprays,” (ML063480542), SNC determined the aerosol removal rate from the effects of the containment spray system is 5.34 per hour.

Section 3.3 of Appendix A of RG 1.183 states:

> The SRP sets forth a maximum decontamination factor (DF) for elemental iodine based on the maximum iodine activity in the primary containment atmosphere when the sprays actuate, divided by the activity of iodine remaining at some time after decontamination. The SRP also states that the particulate iodine removal rate should be reduced by a factor of 10 when a DF of 50 is reached. The reduction in the removal rate is not required if the removal rate is based on the calculated time-dependent airborne aerosol mass. There is no specified maximum DF for aerosol removal by sprays.

Using the guidance from SRP 6.5.2, SNC determined the elemental iodine removal rate from the effects of the containment spray system when in operation is 13.7 per hour. However, in accordance with the guidance in SRP 6.5.2, the licensee limited the removal rate constant for elemental iodine to zero when the elemental iodine DF reaches a value of 200. The aerosol removal coefficient is reduced by a factor of 10 when the aerosol DF reaches 50. SNC applied the removal rates in the radiological dose analysis from the time of spray actuation until 2 hours after actuation. No credit is taken for organic iodine removal in the containment.

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

3.4.3 Assumptions on Engineered Safety Features (ESF) System Leakage

To evaluate the radiological consequences of ESF leakage, SNC used the deterministic approach as described in RG 1.183. This approach assumes, with the exception of noble gases, all the fission products released from the fuel to the containment instantaneously and homogeneously mix in the containment sump water at the time of release from the core. Except for iodine, all of the radioactive materials in the containment sump are assumed to be retained in the liquid phase. This source term assumption is conservative in that 100% of the radioiodine released from the fuel is assumed to reside in both the containment atmosphere and in the containment sump concurrently. ECCS leakage develops when ESF systems circulate containment sump water outside containment and leaks develop through packing glands, pump shaft seals, and flanged connections.

RG 1.183, Appendix A, Regulatory Position 5.5, states, in part, that:

If the temperature of the leakage is less than 212 degrees Fahrenheit (°F) or the calculated flash fraction is less than 10%, the amount of iodine that becomes airborne should be assumed to be 10% of the total iodine activity in the leaked fluid,unless a smaller amount can be justified based on the actual sump pH history and area ventilation rates.
In its submittal dated June 30, 2022, the licensee states that “Since the calculated flash fraction is less than 10 percent, and without a basis for justifying a smaller value, 10% of the iodine in the ECCS leakage is assumed to be released.” As such, SNC conforms with Regulatory Position 5.5.

For the LOCA analysis of ESF leakage, SNC assumed ECCS leakage is 4 gallons per minute (gpm) for leakage of sump water outside of containment into the auxiliary building, which represents two times the maximum permitted leakage of 2 gpm, as specified in RG 1.183, Appendix A, Item 5.2. ECCS leakage starts when the recirculation phase of the accident begins.

3.4.3.1 Assumptions on Engineered Safety Features (ESF) System Back Leakage to the Refueling Water Storage Tank (RWST)

Although the RWST is isolated during recirculation, design leakage through ECCS valves provides a pathway for back leakage of the containment sump water to the RWST. The RWST is vented to the atmosphere. Since this release path represents a bypass of the containment, the radiological dose consequences are considered. The concentration of radionuclides in the containment sump water is as modeled above for ESF leakage. SNC assumes that containment sump water leaks into the RWST at a rate of 14 gpm representing two times the maximum permitted back leakage of 7 gpm, as specified in RG 1.183, Appendix A, Regulatory Position 5.2. This is a change from their CLB, in that this leakage pathway was not previously modeled. The back leakage to the RWST starts when recirculation occurs in the ECCS systems and continues for the 30-day duration of the event. SNC assumed that the chemical form of the iodine released is 97 percent elemental and 3 percent organic.

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

3.4.4 Assumptions on Containment Purging

SNC evaluated the radiological consequences of containment leakage via the mini-purge system, which are assumed to be open to the extent allowed by SNC TS at the initiation of the LOCA and are terminated as part of the containment isolation. The assumed volumetric flow rate from the purge system is 5,000 cfm and is released directly to the environment until terminated by the containment isolation at 30 seconds post-LOCA.

During this time period of 30 seconds following accident onset, SNC assumes that fuel failure has not occurred. This assumption follows the guidance in Table 4 of RG 1.183 which indicates that the initial release of the RCS into containment for a PWR would occur within the first 30 seconds of the accident prior to the onset of fuel damage. Consistent with RG 1.183, RCS radionuclide concentrations for the AST analysis is based on the TS RCS equilibrium activity which includes 1 percent fuel defects. Therefore, this conservative approach for the evaluation of the radiological dose consequence is acceptable to the NRC staff.SNC used conservative assumptions to evaluate the containment purge contribution to the LOCA dose, and, therefore, the NRC staff finds this evaluation to be acceptable for the AST LOCA analysis.

3.4.5 Control Room (CR) Habitability for Loss-of-Coolant Accident (LOCA)

All inputs, assumptions, and initial conditions discussed in Section 3.3, “Control Room Habitability for All DBAs,” are applicable to the LOCA DBA. SNC evaluated CR habitability for the CRE assuming that the CREFS automatically transfers to the isolation and pressurization mode of operation upon SI or high radiation in the CR. The licensee determined that following initiation of a LOCA, the time to generate the CRI signal will be 3.3 seconds. This is 1.77 seconds from the initiation of the LOCA to the SIS setpoint reached, and 1.5 second signal delay (3.27 seconds rounded up to 3.3 seconds). The LOCA time to generate an SIS of 3.3 seconds is consistent with the CLB. During a LOCA, the CR becomes pressurized in 88 seconds from the CRI. The CRI occurs automatically upon the signal within 11.3 seconds, and CR Pressurization Mode Initiation is achieved at 99.3 seconds. These times are identical to CLB values.

SNC's analysis of CR habitability for the LOCA is consistent with applicable Appendix A of RG 1.183, which identifies acceptable radiological analysis assumptions for a LOCA. However, the timing associated with the CRI was inconsistent with other DBAs. Therefore, the NRC staff requested that the licensee further explain the operation of the CREFS during a LOCA. In its letter dated February 6, 2023, RAI response No. 6, SNC stated, in part, that:

> The LOCA time to generate an SIS of 3.3 seconds is consistent with the current licensing basis (CLB). The CLB FHA assumes that a CR Hi Rad signal is generated in 40 seconds, compared to the bounding value of 600 seconds used in the AST analysis. Control Room doses are only calculated for the LOCA and FHA in the CLB.
>
> The time from SIS or Hi Rad signal generation to achieve CRI and CR Pressurization is generally not event-specific and is as given in the following table. The 8 seconds to achieve CRI includes 2 seconds of signal delay and 6 seconds for damper re-positioning. The 96 seconds from SIS / Hi Rad signal generated to CR Pressurization mode achieved includes 36 seconds from SIS / Hi Rad generated to lead fan start signal + 30 seconds from lead fan start signal to system low flow detection + 30 seconds from lead fan low flow detection / lag fan start signal to CR pressure established (i.e., all events implicitly assume failure of lead fan to pressurize the CR).
>
> The CRI and CR Pressurization timing shown is consistent with the CLB LOCA. The CLB FHA conservatively assumes an additional 2 seconds for Pressurization mode to be achieved (total of 98 seconds from Hi Rad signal).

The NRC staff reviewed the licensee's initial conditions, inputs and assumptions for control room habitability, identified the corresponding values in licensee's provided calculations, and performed confirmatory calculations where appropriate and find SNC's response acceptable.

3.4.5.1 Direct Shine Dose Evaluations

RG 1.183 Regulatory Position 4.2.1 states:

> The TEDE analysis should consider all sources of radiation that will cause exposure to control room personnel. The applicable sources will vary from facility to facility, but typically will include:
>
> - Contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility,
>
> - Contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope [CRE],

- Radiation shine from the external radioactive plume released from the facility,
- Radiation shine from radioactive material in the reactor containment,
- Radiation shine from radioactive material in systems and components inside or external to the control room envelope [CRE], e.g., radioactive material buildup in recirculation filters.

In its LAR of June 30, 2022, Table A of Enclosure 5 states SNC's conformance with RG 1.183 Section C. Table A states that SNC analysis for RG 1.183 Regulatory Position 4.2.1 is:

> Conforms–The analyses consider the applicable sources of contamination to the control room atmosphere for each event.
>
> With respect to external and containment shine sources and their impact on control room doses, the physical design of the control room envelope and the surrounding auxiliary building provide more than 18" of concrete shielding between the operators and shine sources in all directions around the control room.
>
> The Control Room Emergency Filtration System filters are located outside of and above the control room envelope. The control room ceiling is approximately 18" thick. Accordingly, shielding from the walls, and the filter unit casings prevents an appreciable dose to the operators during the accident.

SNC considered the potential impact of the shine sources mentioned above to the CR operator doses. However, due to the shielding, and the distance from the source, SNC made the determination that the shine would not contribute significantly to the dose. The CR at Vogtle, Units 1 and 2, is located within the auxiliary building of the plant. It is not adjacent to the containment, which consists of a concrete wall 3 feet (ft) 9 inches (in) thick. The floors and ceiling of the CR consist of 18 in of concrete. Radiation releases in the auxiliary building during a LOCA are assumed to occur in the penetration room filtration system envelope, which is well away from the CR and does not communicate with it. Section 12.3.2.2.1 "Containment Shielding Design" of the UFSAR (ML20155B302) states that the containment walls and dome are a minimum of 3 ft 9 in thick reinforced, prestressed concrete. The penetration room walls are 2 ft thick. Therefore, the radiation shine from the containment, from any releases in the auxiliary building, and from the radioactive plume would be an insignificant contributor to operator dose.

The CREFS intake and recirculation filters are located in rooms above the CR and are separated by at least 18 in of concrete from the occupied spaces in the CR. The radioactive sources are not in the CRE. As such, there is no significant contribution from these sources to operator dose.

SNC considered the potential impact of the shine sources mentioned above to the CR operator doses. However, due to the shielding, and the distance from the source (containment, plume, radiation filters), SNC made the determination that the shine would not contribute significantly to the dose. This determination is consistent with its CLB.

3.4.6 Conclusion

SNC evaluated the radiological consequences resulting from the postulated LOCA and concluded that the radiological consequences at the EAB, LPZ, and CR are within the radiation dose reference values provided in 10 CFR 50.67 and the accident specific dose criteria specified in SRP Section 15.0.1. The NRC staff finds that SNC used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the NRC staff are presented in "Table 3: LOCA Inputs and Assumptions" and SNC's calculated dose results are given in Table 1: Total Effective Dose Equivalent per Accident in roentgen equivalent man (rem). The NRC staff performed independent review of all inputs, assumptions and initial conditions in the SNC dose assessment files, and performed independent confirmatory calculations using RADTRAD Version 5.0.3, as necessary, to ensure a thorough understanding of the licensee's methods and to verify that values used in the dose assessment code were in line with values provided in the LAR dated June 30, 2022. Based on the above, the NRC staff finds that the EAB, LPZ, and CR radiological doses for the LOCA meet the applicable accident dose criteria and are, therefore, acceptable.

3.5 Fuel Handling Accident (FHA)

The FHA involves the drop of a fuel assembly 70 hours after shutdown, onto another fuel assembly in the fuel building. The 70-hour decay time is less than, and more conservative than the CLB 90-hour decay time in the CLB. This limiting case analysis of the accident occurrence in the fuel building bounds an FHA in containment and is constant with the CLB. In its submittal dated June 30, 2022, SNC states that the significant differences in the atmospheric dispersion factors of the fuel building and the containment provided in Table 3.5a demonstrate that the accident occurring in the fuel building is the bounding case with respect to radiological consequences in the three receptor areas (CR, EAB, LPZ). The radiological consequences of an FHA in containment are not limiting regardless of the containment configuration compared to a release from the fuel building.

The mechanical part of SNC's analysis remains unchanged from the CLB. It assumes that the total number of failed fuel rods is 314 due to the accident. This includes 100 percent of the 264 rods in the dropped fuel assembly and 50 rods in a second assembly which was struck in the drop for a total of 314 damaged fuel rods. A radial peaking factor of 1.7 is applied to the fission product inventory of the damaged rods. The water above damaged fuel is not less than 23 feet and is controlled by TS 3.7.15 and TS 3.9.7. These values remain unchanged from the CLB. Following reactor shutdown, decay of short-lived fission products greatly reduces the fission product inventory present in irradiated fuel. The proposed AST amendment takes credit for the normal decay of irradiated fuel. Section 15.7.4, "Fuel Handling Accidents," of the Vogtle, Units 1 and 2, UFSAR describes the CLB DBA.

3.5.1 Source Term

The fission product inventory that constitutes the source term for this event is the gap activity in the fuel rods assumed to be damaged because of the postulated design basis FHA. Volatile constituents of the core fission product inventory migrate from the fuel pellets to the gap between the pellets and the fuel rod cladding during normal power operations. The fission product inventory in the fuel rod gap of the damaged fuel rods is assumed to be instantaneously released to the surrounding water because of the accident.

SNC performed a detailed analysis to ensure that the most restrictive case would be considered for the FHA dose consequence analysis.

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

As corrected by item 8 of Regulatory Issue Summary 2006-04 (ML053460347), RG 1.183, Appendix B, Regulatory Position 2, should read as follows:

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

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

SNC's analysis of the source term for an FHA is consistent with applicable Appendix B of RG 1.183, which identifies acceptable radiological analysis assumptions for a fuel handling accident. However, the licensee did not account for all applicable radionuclides in Regulatory Position 3.2, which states that "Radionuclides that should be considered include xenons, kryptons, halogens, cesiums, and rubidiums.

In its letter dated February 6, 2023, SNC stated, in part, to the RAI response No. 7:

> SNC's fuel handling accident (FHA) analysis complies with RG 1.183.
>
> The RG 1.183 Appendix B Regulatory Position 1.2 identifies radionuclides (xenons, kryptons, halogens, cesiums, and rubidiums) that should be considered for the gap activity. The verb "consider" is defined as *to think carefully about, especially in order to make a decision*.
>
> All radionuclides listed in RG 1.183 Section C Regulatory Position 3.4, including those above, were considered. Those that would not contribute to the dose consequences were identified and eliminated from further evaluation. This was determined via a one-step screening process.

The screening process was based on the element's state. Per RG 1.183 Appendix B Regulatory Position 3, particulates, (i.e., solids), will be retained by the water in the spent fuel pool or refueling cavity. This is because Vogtle [Units 1 and 2], Technical Specifications 3.7.15 and 3.9.7 require a minimum level of 23 feet above the top of active fuel in the refueling cavity and spent fuel pool. As a result, only the noble gases (xenons and kryptons) and halogens (iodine and bromine) screen in as a part of the accident specific FHA source term.

This screening process resulted in the following radionuclides selected for determining the fuel handling accident dose consequences:

- Halogens: I-130, I-131, I-132, I-133, I-134, I-135, Br-82, Br-83, Br-84
- Noble Gases: Kr-83m, Kr-85, Kr-85m, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, and Xe-135, Xe-135m, Xe-138

The NRC staff reviewed the licensee's provided source term and identified the corresponding values in licensee's provided calculations and find SNC's response acceptable.

3.5.1.1 Gap Release Fractions for Alternate Source Term (AST)

RG 1.183 provides guidance on AST implementation. Footnote 11 in Section 3.2 of RG 1.183, Revision 0, notes that the non-LOCA fuel rod gap fractions listed in Table 3 have been found acceptable for light-water reactor (LWR) fuel with peak burnup of 62 GWd/MTU, provided that the maximum linear heat generation rate (LHGR) does not exceed 6.3 kW/ft at burnups greater than 54 GWD/MTU.

SNC requests an exception to the Footnote 11 LHGR limit for 40 percent of the rods in an assembly. The licensee states that this exception only impacts the FHA analysis. Additionally, in response to RAI-1 in letter dated February 6, 2023, SNC stated that there is no limit on the number of assemblies that this exception would apply. An evaluation of the gap fractions and this exception are evaluated below.

3.5.1.2 Gap Fractions for Fuel Handling Accidents (FHAs)

Section 3.5 of the Enclosure of its submittal dated June 30, 2022, discusses the gap fractions assumed for the analysis. SNC requests an exception to the RG 1.183, Rev. 0, Footnote 11, for 6.3 kilowatt per foot (kW/ft) LHGR limit between 54 and 62 GWd/MTU for 40 percent of the rods in any assembly. Specifically, SNC requested that 40% of the rods be allowed to exceed the 6.3kW/ft limit and those 40% of rods be approved for a LHGR limit of 7.4 kW/ft. Rather than utilizing the gap fractions in RG 1.183, Rev. 0, Table 3, SNC stated that the FHA analysis will employ the gap fractions presented in Table 2.9 of Pacific Northwest National Laboratory (PNNL) report, PNNL-18212 Rev. 1, "Update of Gap Release Fractions for Non-LOCA Events Utilizing the Revised ANS [American Nuclear Society] 5.4 Standard ["Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel"]" (ML112070118). RG 1.183, Rev. 0, is based on the 1982 ANS-5.4 standard release model, while PNNL-18212, Rev. 1, published in 2011, utilizes the updated 2011 ANS-5.4 standard release model. The Table 2.9 gap fractions in PNNL-18212 Rev. 1 for Kr-85, I-132, other noble gases, and alkali metals are greater than the RG 1.183, Rev. 0, Table 3 gap fractions. PNNL-18212, Rev. 1, states that the applicability of the gap fractions presented in the report for PWRs are the power and burnup bounds provided in Figure A.1. Figure A.1 of PNNL-18212, Rev. 1, defines a bounding rod-average power history that begins at 12.2 kW/ft at the beginning of life, then at 35 GWd/MTU begins to decrease until 65 GWd/MTU, where the rod-average power is 7.0 kW/ft. In other words, for the PNNL-18212, Rev. 1, gap fractions to be fully applicable to PWRs, operation must be equal to or below the bounding power history presented in Figure A.1 of the report.

SNC's proposed LHGR limit between 54 and 62 GWd/MTU in Section 3.5 of its submittal dated June 30, 2022, to be used meets the PNNL-18212, Rev. 1, Figure A.1 limits, but the LAR did not address whether the Figure A.1 limits would be met for burnups outside of the 54 to 62 GWd/MTU range. On March 3, 2023 (ML23065A061), the NRC staff issued RAI-14 concerning the rod-average power outside of this range in relationship to the power history in PNNL-18212, Rev. 1, Figure A.1. By letter dated March 24, 2023 (ML23083B398), SNC responded to RAI-14, the licensee stated:

> Adherence to the entirety of the curve presented in Figure A.1 will be ensured. An upper limit for rods exceeding the Footnote 11 applicability limits of 40% will be validated on a cycle-by-cycle basis as part of the Reload Safety Analysis Checklist (RSAC). The linear heat rates and burnups of that 40% will be within the PNNL-18212, Rev. 1, Table 2.9 gap fraction applicability limits (12.2 kw/ft up to 35 GWD/MTU, decreasing to 7.5 kw/ft at 62 GWD/MTU). This will also be validated on a cycle-by-cycle basis as part of the RSAC.

The NRC staff finds the application of the PNNL-18212, Rev. 1, gap fractions to the FHA analysis acceptable because the gap fractions utilize the updated ANS-5.4 (2011) standard, and SNC confirmed in response to RAI-14 that plant operation will meet the PNNL-18212, Rev. 1, Figure A.1 power history. Similarly, the NRC staff finds the exception to the LHGR limit in Footnote 11 of RG 1.183 Rev. 0, to be acceptable, because the licensee will use the PNNL-18212, Rev. 1, gap fractions in place of the RG 1.183, Rev. 0, gap fractions for the FHA dose analysis and plant operation will remain within the bounds of the PNNL-18212, Rev. 1, Figure A.1 power history. The NRC has approved use of the 2011 ANS-5.4 standard and the PNNL-18212 Rev. 1 gap fractions in previous safety evaluations including at Wolf Creek Generating Station (ML19100A122) and Surry Power Station (ML19028A384).

The NRC staff notes that there is an inconsistency in the submittal regarding the LHGR limit that SNC intends to employ in place of the RG 1.183, Rev. 0, Footnote 11 limit of 6.3 kW/ft limit (i.e., Section 3.5 of the Enclosure to the LAR dated June 30, 2022, states that the Footnote 11 LHGR limit will be raised to 7.4 kW/ft, while Attachment 4 to the Enclosure to the LAR dated June 30, 2022, states that the Footnote 11 limit will be raised to 7.5 kW/ft). Considering the SNC's response the RAI-14, the NRC staff finds the specific LHGR limit for the gap release fractions to be inconsequential to the analysis. The NRC staff finds the use of the PNNL-18212, Rev. 1, gap fractions acceptable for operation that is less than or equal to the bounding power history in Figure A.1 of PNNL-18212, Rev. 1 because they are conservative and, therefore, provide reasonable assurance of adequate protection of public health and safety.

Overall, the NRC staff finds the gap fractions employed in the FHA dose analysis to be acceptable, because gap fractions based on the 2011 ANS-5.4 standard from PNNL-18212, Rev. 1, are employed and operation below the PNNL-18212, Rev. 1, Figure A.1 bounding power history will be ensured on a cycle-specific basis. The gap fractions are acceptable for use in the FHA analysis up to the PNNL-18212, Rev. 1, Figure A.1 bounding power history.

3.5.2 Transport

3.5.2.1 Fuel Handling Accidents (FHA) in Spend Fuel Pool (SFP) area of Auxiliary building

Releases from the FHA in SFP are via the plant vent stack. During normal operation, the system is in normal mode with two process radiation detectors monitoring the effluent. On alarm signal, the SFP air supply and exhaust dampers close and the ESF emergency filtration system is placed into service as the system is placed into emergency mode. In emergency mode, the fuel building ventilation automatically reconfigures and exhausts through ESF emergency filtration system charcoal and HEPA filters to remove halogens and particulates prior to discharging to the atmosphere via the plant vent. Although the ESF emergency filtration system will remove halogens and particulates, no is credit taken for filtration from the Fuel Handling Building (FHB) post-accident exhaust filters. Analysis of the FHA in the fuel building takes no credit for either filtration, or holdup in the fuel building. The FHB post-accident exhaust system is designed to maintain a slightly negative pressure within the FHB following an FHA. Consistent with RG 1.183, the FHA in the SFP is released over a two-hour period.

3.5.2.2 Fuel Handling Accidents (FHA) Atmospheric Dispersion values (χ/Q)

The licensee’s submittal differs from the CLB in that it does not analyze the case of FHA release in containment. This is due to the fact that the dose consequence of an accident in the FHB building bounds the dose consequences of an accident in the containment. The CLB value of χ/Q for containment during the duration of an FHA is 1.04×10^{-3}. The AST LAR value for the FHB χ/Q is 6.01×10^{-3}. This value is greater than five times the CLB χ/Q for containment. The NRC staff finds that use of the FHB χ/Q value during the analysis of an FHA is conservative and bounding for all FHA accidents and is, therefore, acceptable.

3.5.3 Control Room (CR) Habitability for Fuel Handling Accidents (FHA)

All inputs, assumptions, and initial conditions discussed in Section 3.3, “Control Room Habitability for All DBAs,” are applicable to the FHA DBA. SNC evaluated CR habitability for the CRE assuming that the CREFS automatically transfers to the isolation and pressurization mode of operation upon SI or high radiation in the CR signal.

Radioactive material from the accident will reach the radiation monitors at the CR intakes, the radiation monitor signal will initiate the automatic isolation of the CR, and then system will automatically bring the CR to a positive pressure. The CREFS automatically transfers to the pressurization mode of operation after the CRI signal. SNC determined that the time to generate the CRI signal, from a High Radiation alarm will be 600 seconds following FHA, 8 seconds to achieve CRI (2 seconds of signal delay and 6 seconds for damper repositioning), and 90 seconds for CR Pressurization Mode Initiation is 90 seconds. Total delay time 698 seconds after transient initiation to CR Emergency Ventilation Mode. These values are more conservative than the CLB values of 48 seconds from the event initiation until CRI and 138 seconds from event initiation until CR pressurization. These values are more conservative than the CLB and are, therefore, acceptable for use in the FHA DBA analysis.

3.5.4 Conclusion

SNC evaluated the radiological consequences resulting from a postulated FHA and concluded that the radiological consequences at the EAB, LPZ, and CR are within the radiological dose guidelines provided in 10 CFR 50.67 and accident specific dose criteria specified in SRP Section 15.0.1. The NRC staff finds that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions are presented in Table 4, "FHA Inputs and Assumptions," and SNC's calculated dose results are given in Table 1, "Total Effective Dose Equivalent per Accident in roentgen equivalent man (rem)." The NRC staff performed independent review of all inputs, assumptions and initial conditions in the licensee dose assessment files and performed independent confirmatory calculations using RADTRAD Version 5.0.3, as necessary, to ensure a thorough understanding of SNC's methods, and to verify that values used in the dose assessment code were in line with values provided in the SNC submittal dated June 30, 2022, and supplemented by letters dated February 6, and March 24, 2023. Based on the above, the NRC staff finds that the EAB, LPZ, and CR doses for the FHA meet the applicable accident dose criteria and are, therefore, acceptable.

3.6 Main Steam Line Break (MSLB) Accident

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

The radiological consequences of a MSLB outside containment will bound the consequences of a break inside containment. Therefore, only the MSLB outside of containment is considered regarding the radiological consequences. The affected SG, hereafter referred to as the faulted SG, rapidly depressurizes, and releases its initial contents to the environment. The MSLB accident is described in Vogtle, Units 1 and 2, UFSAR Section 15.1.5 "Steam System Piping Failure" (ML20155B336). RG 1.183, Appendix E, identifies acceptable radiological analysis assumptions for a PWR MSLB.

As stated above, the steam release from a rupture of a main steam line would result in an initial increase in steam flow, which decreases during the accident as the steam pressure decreases. The increased energy removal from the RCS causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity. If the most reactive rod cluster control assembly is stuck in its fully withdrawn position after the reactor trip, there is an increased possibility that the core will become critical and return to power. The core is ultimately shut down by the boric acid delivered by the SI system. Section 15.1.5, Steam System Piping Failure, of the Vogtle, Units 1 and 2, UFSAR describes the CLB DBA.

3.6.1 Source Term

Appendix E of RG 1.183 identifies acceptable radiological analysis assumptions for a PWR MSLB accident. RG 1.183, Appendix E, Regulatory Position 2, states that:

> If no, or minimal[2] 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]. Two cases of iodine spiking should be assumed.
>
> [2] The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum technical specification values, whichever maximizes the radiological consequences. In determining dose equivalent I-131 (DE I-131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.

SNC's evaluation indicates that no fuel damage would occur as a result of a MSLB accident.

Therefore, the licensee considered the two radioiodine spiking cases described in RG 1.183. The first case is referred to as a pre-accident iodine spike and assumes that a reactor transient has occurred prior to the postulated MSLB that has raised the primary coolant iodine concentration to the maximum value permitted by the TS for a spiking condition. For Vogtle, Units 1 and 2, the maximum iodine concentration allowed by TS 3.4.16 as the result of an iodine spike is ≤ 60 micro curies per gram (μCi/gm) of dose equivalent I-131 (DEI).

The second case assumes that the primary system transient associated with the MSLB causes an iodine spike in the primary system. This case is referred to as a concurrent iodine spike. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value specified in Vogtle, Units 1 and 2, TS. For Vogtle, Units 1 and 2, the RCS TS 3.4.16 limit for equilibrium or normal operation is ≤ 1.0 μCi/gm DEI. The duration of the concurrent iodine spike is assumed to be 8 hours in accordance with RG 1.183. No fuel damage is postulated with an MSLB. These values are consistent with the CLB.

For the MSLB accident, SNC evaluated the radiological dose contribution from the release of secondary side activity using the equilibrium secondary side specific activity found in Vogtle, Units 1 and 2, TS 3.7.16 as 0.1 μCi/gm DEI. The alkali metals in the secondary coolant are assumed to be 10 percent of those in the RCS corresponding to operation with 1 percent failed fuel. The feedwater system flows into the SG are modeled as a source of radioiodine in this analysis. The licensee assumes that the chemical form of iodine released from the SGs to the environment is 97 percent elemental and 3 percent organic consistent with RG 1.183, Appendix E, Regulatory Position 4.

3.6.2 Release Transport

SNC followed the guidance as described in RG 1.183, Appendix E, Regulatory Position 5 in all aspects of the transport analysis for the MSLB. For additional conservatism, the licensee assumes a total primary-to-secondary leak rate equal to 1 gpm (1,440 gallons per day (gpd)), which is higher than the TS 3.4.13 d. limits primary to secondary LEAKAGE to 150 gpd through any one SG, which is a total of 600 gpd from all four SGs. SNC modeled the assumed primary-to-secondary leakage of 0.35 gpm into the faulted SG and 0.65 gpm total into the remaining three intact SGs.

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

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

SNC assumes a leakage density of 62.4 pounds mass per cubic feet (lbm/ft^3). RG 1.183, Appendix E, Regulatory Position 5.3, states:

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

In accordance with RG 1.183, SNC assumes that primary-to-secondary leakage from the intact SGs continues until the RCS reaches 212°F, which is 20 hours after the MSLB, at which time shutdown cooling is initiated. Primary-to-secondary leakage in the faulted SG also ceases when residual heat removal (RHR) is placed in service 20 hours after initiation of the event. This is consistent with the CLB. In accordance with RG 1.183, the licensee assumes that all noble gas radionuclides released from the primary system are released to the environment without reduction or mitigation. Following the guidance from RG 1.183, Appendix E, Regulatory Positions 5.5.1, 5.5.2, 5.5.3, and 5.5.4, SNC assumes that all of the primary-to-secondary leakage into the faulted SG will flow directly from the RCS to the environment with no partitioning. For the unaffected SGs that are used for plant cooldown, the licensee assumes a partition factor of 100 is applied to the iodine nuclides. The iodine releases to the environment from the unaffected SGs are assumed to be 97 percent elemental and 3 percent organic, which is consistent with Regulatory Position 4 in RG 1.183, Appendix E.

Primary-to-Secondary leakage (consistent with CLB) is assumed to be 0.35 gpm to the “faulted” SG, and 0.65 gpm (total) going to the intact SGs. It is postulated that the MSLB causes the associated “faulted” SG to blow dry, releasing activity directly to the environment through the broken main steam line. Activity from three intact SGs released to the environment via steaming until the RCS is placed on RHR cooling (assumed at 20 hours). The secondary system volumetric releases are 10 percent greater than the CLB, adding conservatismto the dose consequence analysis and therefore provides reasonable assurance of adequate protection.

3.6.3 Control Room (CR) Habitability for Main Steam Line Break (MSLB)

All inputs, assumptions, and initial conditions discussed in section 3.3 “Control Room Habitability for All DBAs” are applicable to the MSLB DBA. SNC evaluated CR habitability for the CRE assuming that the CREFS automatically transfers to the isolation and pressurization mode of operation upon SI or high radiation in the CR signal. The licensee stated the time for CRI is 11.3 seconds after a MSLB and subsequent SI signal, and CR Pressurization Mode

Initiation occurs at 99.3 seconds, 88 seconds after a valid control room isolation signal. These values are identical to the CLB.

3.6.4 Main Steam Line Break (MSLB) Conclusion

SNC evaluated the radiological consequences resulting from the postulated MSLB and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 50.67 and accident specific dose criteria specified in SRP Section 15.0.1. The NRC staff finds that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions are presented in Table 5, “MSLB Inputs and Assumptions,” and SNC’s calculated dose results are given in Table 1, “Total Effective Dose Equivalent per Accident in roentgen equivalent man (rem).” The NRC staff performed independent review of all inputs, assumptions and initial conditions in the licensee dose assessment files as necessary to ensure a thorough understanding of SNC’s methods, and to verify that values used in the dose assessment code were in line with values provided in the submittal dated June 30, 2022. The EAB, LPZ, and CR doses for the MSLB were found to meet the applicable accident dose criteria and are, therefore, acceptable.

3.7 Steam Generator Tube Rupture (SGTR) Accident

The SGTR accident assumes an instantaneous and complete severance of a single SG tube. The postulated break allows RCS to leak to the secondary side of the ruptured SG. The radioactivity from the leaking SG tube mixes with the shell-side water in the affected SG. The release location from the faulted SG is conservatively assumed to originate from the main steamline room closest to the CR. The SGTR assumes a concurrent LOOP to maximize the release to the environment. Section 15.6.3, “Steam Generator Tube Failure,” of the Vogtle, Units 1 and 2, UFSAR describes the CLB DBA.

After the initiation of the accident, the faulted SG is manually isolated in 20 minutes. This is accomplished by isolating steam flow from and stopping feedwater flow to the faulted SG. With the isolation of the faulted SG, the associated atmospheric relief valve (ARV) fails open. With the ARV failed open, release RCS to the environment via the faulted SG continues until isolation of the ARV.

It is assumed manual operator action to locally close the block valve associated with the failed open ARV is completed at 16 minutes. ARV isolation occurs 36 minutes after initiation of the event. For the SGTR DBA, radiological consequence analysis, mass transfer from the primary to the secondary in the faulted SG continues until the break flow is terminated. Break flow from the primary side of the SG to the secondary side is terminated in 92 minutes.

Leakage into the intact SGs continues with activity released to the environment through steaming until the RCS is cooled to cold shutdown conditions and placed on RHR after 20 hours. The above assumptions and inputs are consistent with the CLB.

3.7.1 Source Term

Appendix F of RG 1.183 identifies acceptable radiological analysis assumptions for a SGTR accident. RG 1.183, Appendix F, Regulatory Position 2, states:

> If no, or minimal[2] 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]. Two cases of iodine spiking should be assumed.
>
> [2] The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum technical specification values, whichever maximizes the radiological consequences. In determining dose equivalent I-131 (DE I-131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.

SNC's evaluation indicates that no fuel damage would occur as a result of a SGTR accident. Therefore, consistent with RG 1.183, the licensee performed the SGTR accident analyses for two radioiodine spiking cases. The first case is referred to as a pre-accident iodine spike and assumes that a reactor transient has occurred prior to the postulated SGTR that has raised the primary coolant iodine concentration to the maximum value permitted by the TS for a spiking condition. For Vogtle, Units 1 and 2, the maximum iodine concentration allowed by TS 3.4.16 as a result of an iodine spike is 60 μCi/gm DEI.

The second case assumes that the primary system transient associated with the SGTR causes an iodine spike in the primary system. This case is referred to as a concurrent iodine spike. Initially, the plant is assumed to be operating with the RCS iodine activity at the TS limit for normal operation. For Vogtle, Units 1 and 2, the RCS TS 3.4.16 limit for normal operation is 1.0 μCi/gm DEI. The increase in primary coolant iodine concentration for the concurrent iodine spike case is estimated using a spiking model that assumes that as a result of the accident, iodine is released from the fuel rods to the primary coolant at a rate that is 335 times greater than the iodine equilibrium release rate. The CLB value of iodine release rate of 500 times greater than the release rate corresponding to the iodine concentration at equilibrium, was revised to align with RG 1.183, Regulatory Position F-2.2 of a value of 335 times greater than the release rate corresponding to the iodine concentration at equilibrium.

The concurrent iodine spike duration is assumed to be a period of 8 hours. SNC assumes that the activity released from the iodine spiking mixes instantaneously and homogeneously throughout the RCS.

In accordance with RG 1.183, Appendix F, Regulatory Position 4, SNC assumes the speciation for iodine release from the SGs is 97 percent elemental and 3 percent organic. In addition, SNC included the radiological dose contribution from the release of secondary coolant iodine activity at the TS 3.7.16 limit of 0.1 μCi/gm DEI. In both the pre-accident iodine spike and the concurrent spike, the RCS activity includes equilibrium noble gas concentration of 280 μCi/g dose equivalent Xe-133 and assumption of 1 percent of the fuel rods having cladding defects. Although a LOOP is assumed, the licensee modeled continued feedwater system flows into the SG, until failed SG isolation, as a source of radioiodine in this analysis for conservatism.

3.7.2 Release Transport

SNC followed the guidance as described in RG 1.183, Appendix F, Regulatory Position 5 in all aspects of the transport analysis for the SGTR. For additional conservatism, SNC assumes a total primary-to-secondary leak rate equal to 1 gpm (1,440 gpd), which is higher than the TS 3.4.13 d. limits primary to secondary LEAKAGE to 150 gpd through any one SG. The licensee modeled the assumed primary-to-secondary leakage of 0.35 gpm into the faulted SG and 0.65 gpm into the three intact SGs.

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

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

SNC's SGTR leak rate of 1 gpm corresponds to a leakage density of 62.4 lbm/ft^3 and is into the four SGs. RG 1.183, Appendix F, Regulatory Position 5.3, states:

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

SNC assumes that the release of radioactivity from the faulted SG continues for 36 minutes at which time the failed open ARV for the faulted SG is manually blocked, and the unaffected SGs release continues for 20 hours at which time shutdown cooling is initiated, and steam releases from the SGs have been terminated. At this point in the accident sequence, steaming is no longer required for cool down and releases from the intact SGs are terminated. The analysis models an exposure duration of 30 days for the CR and LPZ, and the worst 2 hours for the EAB.

SNC assumes that the source term resulting from the radionuclides in the RCS, including the contribution from iodine spiking, is transported to the ruptured SG by the break flow. A portion of the break flow is assumed to flash to steam based upon the thermodynamic conditions in the RCS relative to the secondary system. The licensee assumes that the flashed portion of the break flow will ascend through the bulk water in the SG, enter the steam space of the affected SG, and be immediately available for release to the environment with no credit taken for scrubbing. Although RG 1.183 allows the use of the methodologies described in NUREG-0409, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident," January 1978 (ML19269F014), to determine the amount of scrubbing credit applied to the flashed portion of the break flow, SNC did not credit scrubbing of the activity in the flashed break flow in the ruptured SG.

In accordance with RG 1.183, the licensee assumes that all noble gas radionuclides released from the RCS are released through the SGs to the environment without reduction or mitigation. In the ruptured SG, SNC assumes the iodine in the flashed portion of the break flow is immediately available for release without reduction or mitigation. The break and leakage flow that does not flash mixes uniformly with the SG liquid mass and activity is released to the environment in direct proportion to the steaming rate and the partition coefficient, in accordance with RG 1.183, Appendix F, Regulatory Position 5.6. A SG partition coefficient for iodine nuclides of 100 is assumed.

3.7.3 Control Room (CR) Habitability for Steam Generator Tube Rupture (SGTR)

All inputs, assumptions, and initial conditions discussed above in section 3.3, "Control Room Habitability for All DBAs," are applicable to the SGTR DBA. SNC evaluated CR habitability for the CRE assuming that the CREFS automatically transfers to the isolation and pressurization mode of operation upon SI or high radiation in the CR signal. The licensee determined that CRI would occur at 121 seconds after initiation of a SGTR and subsequent SI signal, and pressurization mode would be reached at 211 seconds.

3.7.4 Steam Generator Tube Rupture (SGTR) Conclusion

SNC evaluated the radiological consequences resulting from the postulated SGTR and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 50.67 and accident specific dose criteria specified in SRP Section 15.0.1. The NRC staff finds that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions are presented in Table 6, "SGTR Inputs and Assumptions," and SNC's calculated dose results are given in Table 1, "Total Effective Dose Equivalent per Accident in roentgen equivalent man (rem)." The NRC staff performed independent review of all inputs, assumptions and initial conditions in the licensee dose assessment files and performed independent confirmatory calculations using RADTRAD Version 5.0.3, as necessary, to ensure a thorough understanding of SNC's methods, and to verify that values used in the dose assessment code were in line with values provided in the LAR dated June 30, 2022. The EAB, LPZ, and CR doses for the SGTR were found to meet the applicable accident dose criteria and are, therefore, acceptable.

3.8 Control Rod Ejection Accident (CREA)

Vogtle, Units 1 and 2, UFSAR Section 15.4.8, "Spectrum of Rod Cluster Control Assembly Ejection Accidents," (ML20155B336) describes the CREA as the mechanical failure of a control rod drive mechanism (CRDM) pressure housing resulting in the ejection of a rod cluster control assembly (RCCA) and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

Following the applicable guidance, SNC evaluated two separate release scenarios for the CREA. In the first case, the failed fuel resulting from the CREA is released in its entirety into the containment via the ruptured CRDM housing, is mixed in the free volume of the containment, and then released to the environment at the containment TS leak rate, plus 5 percent for conservatism, for the first 24 hours and at half that value for the remaining 29 days. For the second case, the radiological consequence from a CREA is evaluated assuming that the RCS boundary remains intact and that fission products are released to the environment from the secondary system. In this case, fission products from the damaged fuel are assumed to be released to the RCS and transported to the secondary system through primary-to-secondary leakage in the SGs. Section 15.4.8, "Spectrum of Rod Cluster Control Assembly Ejection Accidents," of the Vogtle, Units 1 and 2, UFSAR describes the CLB DBA.

3.8.1 Source Term

The source term for the CREA is assumed to result in fuel damage consisting of localized damage to fuel cladding with some fuel melt occurring in the damaged rods. The source term for the CREA is described in RG 1.183, Appendix H, Regulatory Position 1, which states that:

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

SNC assumed that as a result of the CREA, 10% of the fuel experiences departure from nucleate boiling (DNB) resulting in cladding damage. A peaking factor of 1.7 was applied to the fission product inventory of the damaged rods. Consistent with the guidance provided in RG 1.183, Appendix H, the licensee assumed that 10 percent of the core inventory of noble gases and iodine reside in the fuel gap and will be available for release in both the containment and the secondary side release scenarios. SNC assumes 0.25 percent of fuel rods experience fuel melting for both scenarios.

In accordance with RG 1.183, Appendix H, Regulatory Position 3, 100 percent of the released activity is assumed to be released instantaneously and mixed homogeneously throughout the containment atmosphere for the ruptured CRDM housing, and 100 percent of the released activity is assumed to be released instantaneously and completely dissolved in the RCS and available for release to the secondary containment in the secondary side release scenario. SNC assumed that 100 percent of the noble gases and 25 percent of the iodine isotopes within the melting rods are available for release from the containment pathway and 100 percent of the noble gases and 50 percent of the iodine isotopes within the melting rods are available for release from the RCS through the secondary system pathway. In addition, the licensee included the radiological dose contribution from the release of secondary coolant iodine activity at the TS 3.7.16 limit of 0.1 μCi/gm DEI. The alkali metals in the secondary coolant are assumed to be 10 percent of those in the RCS corresponding to 1 percent failed fuel. The value of 10 percent of the alkali metals in the secondary coolant is based upon the ratio of DEI in the RCS, 1.0 μCi/gm DEI, vs the secondary concentration of 0.1 μCi/gm DEI.

3.8.1.1 Gap Release Fractions for Alternate Source Term (AST)

RG 1.183 provides guidance on AST implementation. Footnote 11 in Section 3.2 of RG 1.183, Revision 0, notes that the non-LOCA fuel rod gap fractions listed in Table 3 have been found acceptable for LWR fuel with peak burnup of 62 GWd/MTU, provided that the maximum LHGR does not exceed 6.3 kW/ft at burnups greater than 54 GWD/MTU. SNC requests an exception to the Footnote 11 LHGR limit for 40 percent of the rods in an assembly. The licensee states that this exception only impacts the FHA analysis. Additionally, in response to RAI-1 dated February 6, 2023, the licensee stated that there is no limit on the number of assemblies that this exception applies to. An evaluation of the gap fractions and this exception are evaluated below.

3.8.1.2 Gap Fractions for Control Rod Ejection Accident (CREA)

The gap fractions stated in Section 3.8 of the Enclosure to the LAR dated June 30, 2022, and Attachment 9 to the Enclosure for CREA are consistent with the gap fractions provided in Footnote 11 of RG 1.183, Rev. 0. In response to RAI-13 dated February 6, 2023, SNC stated that the CREA gap fractions are independent of the Table 3 gap fractions, and thus the Footnote 11 limitations on LHGR and burnup, so do not need to be altered if the Footnote 11 limits are exceeded. The NRC staff disagrees with this statement, it may not be appropriate to use the Footnote 11 for CREA gap fractions for operation outside of the Footnote 11 limits. Reactivity initiated accident testing has demonstrated that the Footnote 11 gap fractions may not conservatively capture the release of transient fission gas that occurs in the fuel rod during a CREA, as documented in Section 3 of PNNL-18212, Rev. 1 (ML112070118). Despite this, the NRC staff finds the licensee's use of the Footnote 11 control rod ejection gap fractions to be acceptable though due to a reasonable assurance of adequate protection of public health and safety as a result of conservatisms in the dose calculation and the margin to the dose acceptance criteria, which are discussed below.

One example of a conservatism in the control rod ejection dose calculation is that SNC applied the maximum core radial power peaking factor for all rods that failed, as stated in Attachment 9 to the Enclosure, which is consistent with Section 3.1 of RG 1.183, Rev. 0. This assumption is conservative because not every rod that fails will have the maximum peaking factor. The more rods that fail, the more conservative this assumption becomes. Additionally, the licensee assumes that 10 percent of the rods fail during the CREA in Attachment 9 to the Enclosure. A CREA is also typically a localized event with effects generally limited to the fuel assemblies near the ejected control element. The value of 10 percent is an assumed value rather than a calculated value and is intended to bound the percentage of the core that is actually expected to fail with some degree of margin as to allow for operational flexibility. If the 10 percent value is challenged due to operational changes, SNC would be required to submit a new CREA dose analysis. Therefore, the assumption that 10 percent of the core fails during a CREA is generally conservative.

The results of the CREA dose analysis are presented in Table 3.8 in Section 3.8 of the Enclosure to the LAR dated June 30, 2022. The results in this table indicate that the EAB, LPZ, and CR doses for the CREA are no more than 29 percent of the acceptance criteria for each area. Therefore, there is ample margin in the calculation that is expected to be able to accommodate any increases in the fission product release from the fuel rods due to transient fission gas release.

Overall, the NRC staff finds the licensee's use of the Footnote 11 control rod ejection gap fractions to be acceptable because it provides a reasonable assurance of adequate protection of health and safety due to conservatisms in the dose calculation and the ample margin to the acceptance criteria in the results.

3.8.2 Transport from Containment

SNC assumes that the activity released to the containment through the rupture in the reactor vessel head mixes instantaneously throughout the containment with no credit assumed for removal of elemental iodine or noble gas in the containment due to containment sprays or for natural deposition of elemental iodine. The licensee is taking credit for natural deposition of aerosols in containment and a removal rate of 3.005×10^{-2} per hour is used. This value varies from the CLB value of 50 percent plateout, and is consistent with RG 1.183, Appendix H. SNC assumes that all containment leakage is 0.21 percent per day, which equals the CLB TS limit of 0.20 percent per day with an added 5 percent for conservatism, for the first 24 hours and half of that value, 0.105 percent per day thereafter.

The licensee assumes that the iodine released to the containment from the fuel consists of 95 percent particulate, 4.85 percent elemental, and 0.15 percent organic per RG 1.183, Appendix H, Regulatory Position 4.

Because SNC stated in its submittal June 30, 2022, that containment sprays will not necessarily be activated during a CREA, no credit is taken for pH being controlled at values of 7 or greater, the NRC staff asked the licensee to provide the plant-specific evaluation that determined that the chemical form of radioiodine released to the containment atmosphere of 95 percent cesium iodide, 4.85 percent elemental iodine, and 0.15 percent organic iodide is conservative at Vogtle, Units 1 and 2, and that the iodine does not re-evolve. In the February 6, 2023, response SNC stated:

> During the CREA, pressure in containment is not expected to reach the containment spray actuation set point. However, the Plant Vogtle containment sump houses baskets of trisodium phosphate (TSP).
>
> Plant Vogtle Technical Specification 3.5.6 controls the amount of TSP. SNC assumes the event introducing water into the sump causes dissolution of TSP, thereby maintaining the pH of the sump water at 7 or greater. Given that the pH condition of RG 1.183 is met, it may be assumed that iodine in the containment atmosphere is 95 percent particulate, 4.85 percent elemental, and 0.15 percent organic.
>
> In conclusion, the sump pH is controlled above 7 using large quantities of TSP additive. As a result, sensitivities on Iodine composition resulting from re-evolution are not required and the chemical form of Iodine specified in Regulatory Guide 1.183 Section C, Regulatory Position 3.5 and Appendix H, Regulatory Position 4 is appropriate and conservative.
>
> Attachment 4, Table G: Conformance with Regulatory Guide 1.183, Appendix H (Rod Ejection Accident), Regulatory Position H-4 has been updated accordingly.

The NRC staff finds that the licensee that if pH is controlled at 7 or greater than consistent with RG 1.183, Appendix H, Regulatory Position 4 the chemical form of radioiodine released to the containment atmosphere may be assumed to be 95 percent cesium iodide, 4.85 percent elemental iodine, and 0.15 percent organic iodide.

3.8.3 Transport from Secondary System

In accordance with RG 1.183, Appendix H, Regulatory Position 7, SNC 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 (1440 gallons per day (gpd)) from all four SGs, which is higher than the TS 3.4.13 d. total allowable leak rate of 150 gpd through any one SG (600 gpd for all SGs), to account for any accident induced leakage.

In accordance with RG 1.183, SNC assumes that all noble gas radionuclides released from the RCS are released to the environment without reduction or mitigation. Following the guidance from RG 1.183, Appendix E, Regulatory Position 5.5.1, the licensee believes that all of the primary-to-secondary leakage in the SGs mix with the secondary water without flashing. For iodine, because the SG tubes remain covered for the duration of the CREA, the partition coefficient of 100 was taken directly from RG 1.183. The retention of particulate radionuclides in the SG is limited by the moisture carryover from the SG. The licensee modeled the transport of particulates and iodines using a maximum moisture carryover value, which is 0.32 percent in its submittal dated June 30, 2022. This value is greater than, and therefore more conservative than the CLB value of 0.25 percent. SNC assumed for the secondary side release that the chemical form of iodine released from the SGs to the environment is 97 percent elemental and 3 percent organic.

3.8.4 Control Room (CR) Habitability for Control Rod Ejection Accident (CREA)

All inputs, assumptions, and initial conditions discussed above in section 3.3 are applicable to the CREA DBA. SNC evaluated CR habitability for the CRE assuming that the CREFS automatically transfers to the isolation and pressurization mode of operation upon SI or high radiation in the CR signal. The licensee determined that the time to reach a SIS setpoint (121 seconds based on a 2 in^2 CREA-induced small break LOCA, and a 1.5 second signal delay (rounded to 2 seconds)) would be 123 seconds with CR pressurization achieved at 219 seconds.

SNC's analysis of CR habitability for the CREA is consistent with applicable Appendix H of RG 1.183, which identifies acceptable radiological analysis assumptions for rod ejection accident. However, the timing associated with the CRI was inconsistent with other DBAs. Therefore, the NRC staff requested that the licensee further explain the operation of the CREFS during a CREA. In the letter dated February 6, 2023, RAI response No. 6, SNC stated, in part, that:

> Attachment 9 to Enclosure, "Control Rod Ejection Accident Analysis," (page A9-9) lists an incorrect value for the CR Pressurization Mode Initiation time. The correct value used in the CREA analysis is 219 seconds (vs. 211 seconds given in Attachment 9 of the LAR), which represents a time difference of 88 seconds between CR Isolation (CRI) and CR Pressurization (versus the 80 seconds value indicated in the RAI), consistent with the LOCA analysis.

The NRC reviewed the licensee's initial conditions, inputs and assumptions for CREA, identified the corresponding values in licensee's provided calculations, and performed independent confirmatory calculations where appropriate and finds SNC's response acceptable.

3.8.5 Control Rod Ejection Accident (CREA) Conclusion

SNC evaluated the radiological consequences resulting from the postulated CREA and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines provided in 10 CFR 50.67 and accident specific dose criteria specified in SRP Section 15.0.1. The NRC staff finds that SNC used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions are presented in Table 7, "CRE Inputs and Assumptions," and SNC's calculated dose results are given in Table 1. The NRC staff performed independent review of all inputs, assumptions and initial conditions in the licensee dose assessment files and performed independent confirmatory calculations using RADTRAD Version 5.0.3, as necessary, to ensure a thorough understanding of SNC's methods, and to verify that values used in the dose assessment code were in line with values provided in the LAR. The EAB, LPZ, and CR doses for the CREA were found to meet the applicable accident dose criteria, provides reasonable assurance of adequate protection and are therefore, acceptable.

3.9 Locked Rotor Accident (LRA)

The LRA considers the instantaneous seizure of a reactor coolant pump (RCP) rotor, which causes a rapid reduction in the flow through the affected RCS loop. The sudden decrease in core coolant flow causes a reactor trip. SNC's evaluation indicates that fuel cladding damage will occur because of this accident. Activity from the fuel cladding damage is transported to the secondary side due to primary-to-secondary side leakage. Radioactivity is released to the outside atmosphere from the secondary coolant system via steaming until cold shutdown conditions are established in the RCS. Following reactor trip and based on a coincident assumption of LOOP, the condenser is unavailable, and reactor cooldown is achieved using steam releases from the SGs until initiation of shutdown cooling. For conservatism, the licensee assumes a total primary-to-secondary leak rate equal to 1 gpm from all four SGs, which is higher than the TS total allowable leak rate of 150 gpd through any one SG. Section 15.3.3, "Reactor Coolant Pump Shaft Seizure (Locked Rotor)," of the Vogtle, Units 1 and 2, UFSAR describes the CLB DBA.

3.9.1 Source Term

SNC assumed that the instantaneous seizure of the RCP rotor associated with the LRA results in a small percentage of fuel clad damage. As in the Vogtle, Units 1 and 2, CLB, radiological dose analysis for this event assumes 5 percent fuel clad damage with no fuel melt predicted. Therefore, the source term available for release is associated with this fraction of damaged fuel cladding and the fraction of core activity existing in the gap. A radial peaking factor of 1.7 was applied to the fission product inventory of the damaged rods. The activity released from the fuel is assumed to be released instantaneously and homogeneously through the RCS.

Following the guidance in RG 1.183, Appendix G, Regulatory Position 4, SNC assumes that the chemical form of radioiodine released from the fuel to the reactor coolant consists of 95 percent CsI, 4.85 percent elemental iodine, and 0.15 percent organic iodide, and that the iodine releases from the SGs to the environment is 97 percent elemental iodine and 3 percent organic iodine.

In addition, the licensee included the radiological dose contribution from the release of secondary coolant iodine activity at the TS 3.7.16 limit of 0.1 μCi/gm DEI. The alkali metals in the secondary coolant are assumed to be 10 percent of those in the RCS, based on the ratio of the secondary to RCS DE I-131 concentrations, corresponding to 1 percent failed fuel. SNC assumes a LOOP to maximize the release to the environment; however, continued feedwater flow is modeled into the SG as a source of radioiodine in this analysis for conservatism.

3.9.1.1 Gap Release Fractions for Alternate Source Term (AST)

RG 1.183 provides guidance on AST implementation. Footnote 11 in Section 3.2 of RG 1.183 Revision 0 notes that the non-LOCA fuel rod gap fractions listed in Table 3 have been found acceptable for LWR fuel with peak burnup of 62 GWd/MTU, provided that the maximum LHGR does not exceed 6.3 kW/ft at burnups greater than 54 GWD/MTU. SNC requests an exception to the Footnote 11 LHGR limit for 40% of the rods in an assembly. The licensee states that this exception only impacts the FHA analysis. Additionally, in response to RAI-1, dated February 6, 2023, SNC stated that there is no limit on the number of assemblies that this exception applies

3.9.1.2 Gap Fractions for Locked Rotor Accident (LRA)

The gap fractions employed in the LRA dose analysis are presented in Attachment 10 to the Enclosure to the submittal dated June 30, 2022. Despite the exception requested to the RG 1.183 Rev. 0, Footnote 11, LHGR limit for 40 percent of the rods in any assembly, SNC stated that gap fractions presented in RG 1.183, Rev. 0, Table 3 are assumed for this event. In Table 7 of Attachment 11 to the Enclosure to the LAR dated June 30, 2022, the licensee justifies the use of the RG 1.183, Rev. 0, gap fractions despite the LHGR exception, stating that “Per reload requirements assemblies that could exceed 6.3 kW/ft are not loaded in locations where DNB could occur.” In other words, SNC will ensure, on a cycle-specific basis, that no rods above the Footnote 11 LHGR limit will fail as a result of DNB. Therefore, any rods between 54 and 62 GWd/MTU that are predicted to fail, and thus contribute to the radiological consequences, during the LRA event will meet the Footnote 11 LHGR limits. As such, the NRC staff finds the use of the gap fractions assumed in the LRA analysis to be acceptable and provide a reasonable assurance of adequate protection of public health and safety.

3.9.2 Release Transport

The activity that originates in the RCS is released to the secondary coolant by means of the primary-to-secondary coolant leak rate. SNC assumes a design basis leak rate of 1 gpm from all four SGs. This equates to a total of 1,440 gpd, which is greater than the maximum allowable operational leakage of 150 gpd for any one SG imposed in TS 3.4.13 d. 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.

Because of the release dynamic of the activity from the SGs, RG 1.183 allows for a reduction in the amount of activity released to the environment based on partitioning of nuclides between the liquid and gas states of water for this release path. For iodine, because the SG tubes remain covered for the duration of the LRA, the partition coefficient of 100 was taken directly from the suggested guidance. The retention of particulate radionuclides in the SG is limited by the moisture carryover from the SG which is 0.32 percent at Vogtle, Units 1 and 2. Because of its volatility, 100 percent of the noble gases are assumed to be released directly to the environment. All remaining isotopes are transported to the SGs at the rate of 1 gpm. The release continues for 20 hours by which time the RCS is placed on RHR system cooling.

The total mass released from the SG to the environment is 5.12×10^6 lbm over a 20-hour time period.

3.9.3 Control Room (CR) Habitability for Locked Rotor Accident (LRA)

All inputs, assumptions, and initial conditions discussed above in Section 3.3 of this SE are applicable to the LRA DBA. SNC evaluated CR habitability for the CRE assuming that the CREFS automatically transfers to the isolation and pressurization mode of operation upon SI or high radiation in the CR signal. The licensee determined that CRI would occur at 608 seconds after an LRA. This would be 600 seconds to generate the SIS, and 8 seconds for isolation. CR pressurization would be achieved at 698 seconds.

3.9.4 Locked Rotor Accident (LRA) Conclusion

SNC evaluated the radiological consequences resulting from the postulated LRA and concluded that the radiological consequences at the EAB, LPZ, and CR are within the radiation dose reference values provided in 10 CFR 50.67 and the accident specific dose criteria specified in SRP Section 15.0.1. The NRC staff 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: LRA Inputs and Assumptions," and SNC's calculated dose results are given in Table 1, "Total Effective Dose Equivalent per Accident in roentgen equivalent man (rem)." The NRC staff performed independent review of all inputs, assumptions and initial conditions in the licensee dose assessment files as necessary to ensure a thorough understanding of SNC's methods, and to verify that values used in the dose assessment code were in line with values provided in the submittal dated June 30, 2022. The NRC staff finds that the EAB, LPZ, and control room radiological doses for the LRA meet the applicable accident dose criteria, provides reasonable assurance of adequate protection and are, therefore, acceptable.

3.10 Alternate Source Term (AST) Methodology Technical Conclusion

The NRC staff has reviewed the AST implementation proposed by SNC for Vogtle, Units 1 and 2. In performing its review, the NRC staff relied upon information placed on the docket by SNC and NRC staff review of all inputs, assumptions and initial conditions contained in the licensee dose calculations.

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by SNC to assess the radiological impacts of the proposed AST. The NRC staff finds that the licensee used analysis methods and assumptions consistent with the guidance of RG 1.183, with differences discussed and reviewed in applicable portions of this SE. The NRC staff finds the methods and assumptions used by the licensee to be in compliance with applicable requirements. The NRC staff finds with reasonable assurance that SNC's estimates of the TEDE due to DBAs will comply with the requirements of 10 CFR 50.67 and meet the guidance of RG 1.183.

The NRC staff finds reasonable assurance that Vogtle, Units 1 and 2, will continue to provide sufficient safety margins with adequate defense in depth to address unanticipated events and to compensate for uncertainties in accident progression and analysis assumptions and parameters. The NRC staff concludes that the proposed AST implementation is acceptable.

This licensing action is considered full implementation of the AST. With this approval, the previous accident source term in the Vogtle, Units 1 and 2, design basis is superseded by the AST proposed by SNC. The previous offsite and CR accident dose criteria expressed in terms of whole body, thyroid, and skin doses are superseded by the TEDE criteria of 10 CFR 50.67 or small fractions thereof, as defined in RG 1.183. All future radiological analyses performed to demonstrate compliance with regulatory requirements shall address all characteristics of the AST and the TEDE criteria as described in the Vogtle, Units 1 and 2, design basis.

3.11 Heavy Load Drop Analysis

In its submittal dated February 6, 2023, SNC provided the following information with regarding the heavy load drop analysis:

> The Vogtle Units 1 and 2 FSAR Section 9.1.5.3.1.1.3 discusses the controls in place to prevent the drop of a heavy load onto exposed irradiated assemblies. Due to identification and use of safe load paths, the only heavy load that is available to be dropped on irradiated assemblies is that of the polar crane main hoist load block. Section 9.1.5.3.1.1.3 discusses that administrative controls are in place which prevent the main hoist from functioning during movement over exposed irradiated fuel. This eliminates the potential for a load drop that could result in damage to irradiated fuel assemblies in the core.

The NRC staff's review of SNC's explanation as above determined that there is no apparent impact on the heavy load drop analysis, and is, therefore, acceptable.

3.12 Removal of the Core Alterations TS Applicability

In a letter dated October 4, 2018 (ML17346A587), from the NRC staff to TSTF, the NRC staff states, in part:

> After considerable review and analysis, the NRC staff concludes that for certain facilities, LARs adopting TSTF-51 and TSTF-471 could result in exceeding the bounding licensing basis Fuel Handling Accident (FHA) analysis of record dose for the control room and is therefore considered an unanalyzed condition.

After approval of LARs adopting TSTF-51 and TSTF-471, the containment purge and exhaust isolation instrumentation and the containment penetrations are no longer required to be operable during core alterations. The NRC staff identified that dropping a source, fuel assembly, or component during core alterations could damage a recently irradiated fuel assembly creating a radioactive source term that may result in exceeding the resultant radiological doses calculated by the licensing basis FHA analysis of record. Therefore, the NRC staff recommended that when adopting TSTF-51 and TSTF-471, the licensees provide one of the following discussions to remove the defined term "CORE ALTERATIONS" from the TS Applicability:

- Confirm that the length of time defined as "recently" is less than the time required to remove the reactor vessel head and internals and expose the irradiated fuel after a shutdown.

- Provide an analysis that demonstrates that the dropping of any unirradiated fuel assembly, sources, reactivity control component, or other component affecting reactivity within the reactor vessel onto irradiated fuel assemblies prior to the period of time defined as "recently" will not result in a radioactive release from the irradiated fuel.

- Describe the limitations or controls that would prevent movement of any unirradiated fuel assembly, source, reactivity control component, or other component affecting reactivity within the reactor vessel capable of damaging a fuel assembly prior to the time period defined as "recently"; or

- Provide an analysis that demonstrates that the dose consequences of a failure of a single irradiated fuel assembly with no TS-required mitigation systems available remain below the regulatory limits and the regulatory guidance limits for a fuel handling accident.

In its letter dated February 6, 2023, SNC states the following:

> SNC's "Refueling Operations (Mode 5 to Mode 6)" procedure (12007-C) provides limitations and controls on core alterations (movement of recently irradiated fuel) within the reactor vessel. This procedure also satisfies Plant Vogtle's Technical Requirements Manual 13.9.1 "Decay Time". This procedure:
>
> - Prevents core alterations (movement of recently irradiated fuel) in the reactor vessel prior to the decay time analyzed in the FHA event. This corresponds to 70 hours for SNC's submittal.
>
> - Requires containment equipment hatch and personnel airlock to either be closed or be able to be closed by designated personnel in a timely manner.
>
> - Requires confirmation that containment penetrations or other access pathways are confirmed or capable of being secured."

Based on the above, the NRC staff has determined that dropping a source, fuel assembly, or component during core alterations will not damage a recently irradiated fuel assembly. Without the creation of a radioactive source term from a core alterations event, the radiological doses will remain bounded by the proposed FHA with recently irradiated fuel analysis. Therefore, the NRC staff finds the proposed changes acceptable.

3.13 Technical Specification Task Force TSTF-51 and TSTF-471

3.13.1 Technical Evaluation of Vogtle Accident Analysis for the Adoption of TSTF-51 and TSTF-471

3.13.1.1 Radiological Consequences of Design Basis Accidents (DBAs)

The NRC staff's review focused on independently confirming the changes meet the NRC regulations and guidance regarding radiological dose consequences and ensuring any variations from the NRC-approved Travelers TSTF-51 and TSTF-471 are acceptable.
As described in the Reviewer's Note incorporated by TSTF-51 into the STSs, information is to be provided by the licensee describing the evaluation of recently irradiated fuel demonstrating that after sufficient radioactive decay has occurred (from the time of shutdown) the radiological doses resulting from an FHA remain below the regulatory limits specified in GDC 19, and well within the offsite radiation dose values specified in 10 CFR 50.67, without crediting the systems not required to be operable. Below describes SNC's radiological analysis assumptions, inputs, and methods for an FHA involving recently irradiated fuel. Section 3.1.2 of the submittal dated dated June 30, 2022, discusses the impact of the proposed changes on the licensee's analysis of dropping of a heavy load onto irradiated fuel assemblies and Section 3.1.3 describes the removal of the defined term "CORE ALTERATIONS " from the applicability section of the proposed TSs.

3.13.1.2 Evaluation of Recently Irradiated Fuel

The FHA involves the drop of a fuel assembly 70 hours after shutdown, onto another fuel assembly in the fuel building. The 70-hour decay time is less than, and more conservative than the CLB 90-hour decay time in the CLB. This limiting case analysis of the accident occurrence in the fuel building is constant with the CLB. SNC states that the significant differences in the atmospheric dispersion factors of the fuel building, and the containment provided in Table 3.5a of its submittal dated June 30, 2022, demonstrate that the accident occurring in the fuel building is the bounding case with respect to radiological consequences in the three receptor areas (CR, EAB, LPZ). The accident dose consequences of an accident in the fuel handling building exceed, and therefore bound, the dose consequences of an accident in containment. This includes all potential configurations of the containment building during refueling, such as doors or equipment hatches open.

SNC's analysis remains unchanged from the CLB in that it assumes that the total number of failed fuel rods is 314 due to the accident. This includes 100 percent of the rods in the dropped fuel assembly, 264 rods, and 50 rods in a second assembly which was struck in the drop. A peaking factor of 1.7 is applied to the fission product inventory of the damaged rods. The depth of water over the damaged fuel is not less than 23 feet and is controlled by TS 3.7.15 "Fuel Storage Pool Water Level". These values are consistent with the CLB. Following reactor shutdown, decay of short-lived fission products greatly reduces the fission product inventory present in irradiated fuel. The proposed AST submittal dated June 30, 2022, takes credit for the normal decay of irradiated fuel. Section 15.7.4, "Fuel Handling Accidents", of the Vogtle, Units 1 and 2, UFSAR describes the CLB DBA. The depth of water over the damaged fuel is not less than 23 feet and is controlled by TS 3.7.15. Following reactor shutdown, decay of short-lived fission products greatly reduces the fission product inventory present in irradiated fuel.

3.13.1.3 Source Term

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

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

As corrected by Item 8 of Regulatory Issue Summary 2006-04 (ML053460347), RG 1.183, Appendix B, Regulatory Position 2, should read as follows:

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

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

The licensee was not able to use the RG 1.183, Table 3, "Non-LOCA Fraction of Fission Product Inventory in Gap," due to exceeding footnote 11 for Table 3. SNC is requesting that 40 percent of the rods be allowed to exceed the 6.3 kW/ft LHGR and be approved for 7.4 kW/ft for Vogtle, Units 1 and 2. For burnup up to 54 GWD/MTW, the use of Table 3 would still be allowed. The nominal source term inventory for this LAR dated June 30, 2022, is based upon a lead rod burnup of 62 GWD/MTU (with variable enrichment regions) and 40 percent of the rods LGHR of 7.4 kW/ft. This exceeds the bounds of RG 1.183, Table 3 bounds as described in footnote 11.

Section 3.5 of the Enclosure to its submittal dated June 30, 2022, discusses the gap fractions assumed for the analysis. As stated previously, SNC requests an exception to the RG 1.183, Rev. 0, Footnote 11 6.3 kW/ft LHGR limit between 54 and 62 GWd/MTU for 40 percent of the rods in any assembly. Rather than utilizing the gap fractions in RG 1.183, Rev. 0, Table 3, the licensee stated that the FHA analysis will employ the gap fractions presented in Table 2.9 of PNNL report, PNNL-18212, Rev. 1, "Update of Gap Release Fractions for Non-LOCA Events Utilizing the Revised ANS 5.4 Standard ["Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel"]," (ML112070118).

RG 1.183, Rev. 0, is based on the 1982 ANS-5.4 standard release model, while PNNL-18212, Rev. 1, published in 2011, utilizes the updated 2011 ANS-5.4 standard release model. The Table 2.9 gap fractions in PNNL-18212, Rev. 1, for Kr-85, I-132, other noble gases, and alkali metals are greater than the RG 1.183, Rev. 0, Table 3 gap fractions. PNNL-18212, Rev. 1 states that the applicability of the gap fractions presented in the report for PWRs are the power and burnup bounds provided in Figure A.1. Figure A.1 of PNNL-18212, Rev. 1 defines a bounding rod-average power history that begins at 12.2 kW/ft at the beginning of life, then at 35 GWd/MTU begins to decrease until 65 GWd/MTU, where the rod-average power is 7.0 kW/ft. In other words, in order for the PNNL-18212, Rev. 1 gap fractions to be fully applicable to PWRs, operation must be equal to or below the bounding power history presented in Figure A.1 of the report.

SNC's proposed LHGR limit between 54 and 62 GWd/MTU in Section 3.5 of the LAR dated June 30, 2022, to be used meets the PNNL-18212, Rev. 1, Figure A.1 limits, but the LAR did not speak to whether the Figure A.1 limits would be met for burnups outside of the 54 to 62 GWd/MTU range. The NRC issued RAI-14 dated March 3, 2023 (ML23065A061), concerning the rod-average power outside of this range in relationship to the power history in PNNL-18212, Rev. 1, Figure A.1. In response to RAI-14 dated March 24, 2023, SNC stated that it would confirm that the PNNL-18212, Rev. 1, Figure A.1 power history would bound operation as part of its reload analysis.

The NRC staff finds the application of the PNNL-18212, Rev. 1 gap fractions to the FHA analysis acceptable, because the gap fractions utilize the updated ANS-5.4 (2011) standard and SNC confirmed in response to RAI-14 dated March 24, 2023, that Vogtle, Units 1 and 2, operation will meet the PNNL-18212, Rev. 1, Figure A.1 power history. Similarly, the NRC staff finds the exception to the LHGR limit in Footnote 11 of RG 1.183, Rev. 0 to be acceptable, because the licensee will use the PNNL-18212, Rev. 1 gap fractions in place of the RG 1.183, Rev. 0 gap fractions for the FHA dose analysis and plant operation will remain within the bounds of the PNNL-18212, Rev. 1, Figure A.1 power history. The NRC has approved use of the 2011 ANS-5.4 standard and the PNNL-18212 Rev. 1 gap fractions in previous safety evaluations including at Wolf Creek Generating Station (ML19100A122) and Surry Power Station (ML19028A384).

The NRC staff notes that there is an inconsistency in the submittal dated June 30, 2023, regarding the LHGR limit that the licensee intends to employ in place of the RG 1.183, Rev. 0, Footnote 11 limit of 6.3 kW/ft limit (i.e., Section 3.5 of the Enclosure to the LAR dated June 30, 2023, states that the Footnote 11 LHGR limit will be raised to 7.4 kW/ft, while Attachment 4 to the Enclosure to the LAR dated June 30, 2022, states that the Footnote 11 limit will be raised to 7.5 kW/ft). Considering the response to RAI-14 in the letter dated June 30, 2022, the NRC staff finds the specific LHGR limit for the gap release fractions to be immaterial. In other words, the NRC staff find the use of the PNNL-18212, Rev. 1 gap fractions acceptable for operation that is less than or equal to the bounding power history in Figure A.1 of PNNL-18212, Rev. 1 because they are conservative and, therefore, provide reasonable assurance of adequate protection of public health and safety.

Overall, the NRC staff find the gap fractions employed in the FHA dose analysis to be acceptable because gap fractions based on the 2011 ANS-5.4 standard from PNNL-18212, Rev. 1 are employed and operation below the PNNL-18212, Rev. 1 Figure A.1 bounding power history will be ensured on a cycle-specific basis. The gap fractions are acceptable for use in the FHA analysis up to the PNNL-18212, Rev. 1, Figure A.1 bounding power history.

3.13.1.4 Transport for the Fuel Handling Accident (FHA) in the Spent Fuel Pool (SFP) Area of Auxiliary Building

Releases are from the FHA in SFP are via the plant vent stack. During normal operation, the system is in normal mode with two process radiation detectors monitoring the effluent. On alarm signal, the SFP air supply and exhaust dampers close and the ESF emergency filtration system is placed into service as the system is placed into emergency mode. In emergency mode the fuel building ventilation automatically reconfigured and exhausts through ESF emergency filtration system charcoal and HEPA filters to remove halogens and particulates prior to discharging to the atmosphere via the plant vent. Although the ESF emergency filtration system will remove halogens and particulates, no credit is taken for filtration from the FHB post-accident exhaust filters. Analysis of the FHA in the fuel building takes no credit for either filtration, or holdup in the fuel building. The FHB post-accident exhaust system is designed to maintain a slightly negative pressure within the FHB following an FHA. Consistent with RG 1.183, the FHA in SFP is released over a two-hour period.

3.13.1.5 FHA Atmospheric Dispersion Values (χ/Q)

The LAR differs from the CLB in that it does not analyze the case of FHA release in containment. This is because the dose consequence of an accident in the FHB bounds the dose consequences of an accident in the containment. The CLB value of χ/Q for containment during the duration of an FHA is 1.04 x 10^{-3}. The AST LAR value for the FHB χ/Q is 6.01 x 10^{-3}. This value is greater than five times the CLB χ/Q for containment. The NRC staff finds that use of the FHB χ/Q value during the analysis of an FHA is bounding for all accidents and will provide reasonable assurance of adequate protection.
.

3.13.1.6 Control Room (CR) Habitability for the Recently Irradiated Fuel Handling Accident (FHA)

All inputs, assumptions, and initial conditions discussed above in Section 3.3 of this SE are applicable to the FHA DBA. The licensee evaluated CR habitability for the CRE assuming that the CREFS automatically transfers to the isolation and pressurization mode of operation upon SI or high radiation in the control room signal.

Radioactive material from the accident will reach the radiation monitors at the CR intakes, the radiation monitor signal will initiate the automatic isolation of the CR, and then system will automatically bring the CR to a positive pressure. The CREFS automatically transfers to the pressurization mode of operation after the CRI signal. SNC determined that the time to generate the CRI signal, from a High Radiation alarm, will be 600 seconds following FHA.

CR Pressurization Mode Initiation is achieved at 698 seconds after the FHA. These values bound the CLB values of 48 seconds from the event initiation until CRI and 138 seconds from event initiation until CR pressurization. Although the values to isolate and pressurize the CR differ from the CLB, it is more conservative and an increase in time to isolate the CR would result in a higher calculated dose., The revised analysis meets the dose acceptance criteria and will provide reasonable assurance of adequate protection, and therefore is acceptable for use in the FHA DBA analysis.

3.13.1.7 Recently Irradiated FHA Conclusion

SNC evaluated the radiological consequences resulting from an FHA with recently irradiated fuel and concluded that the radiation doses at the EAB, LPZ, and CR are within the radiation dose guidelines provided in 10 CFR 50.67 and accident specific dose criteria specified in SRP Section 15.0.1. 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. SNC's calculated dose results are given in Table 2, Section 3.5, "Fuel Handling Accident," of this SE. The NRC staff performed independent confirmatory dose evaluations and review of all inputs, assumptions and initial conditions in the licensee dose assessment files as necessary to ensure a thorough understanding of SNC's methods. The NRC staff finds that the EAB, LPZ, and CR doses for the FHA with recently irradiated fuel meets the applicable accident dose criteria and are, therefore, acceptable.

3.13.2 TS 3.3.6, "Containment Ventilation Isolation Instrumentation"

Containment Ventilation Isolation Instrumentation closes the containment isolation valves in the mini-purge and main purge systems. This action isolates the containment atmosphere from the environment to minimize releases of radioactivity in the event of an accident.

The "Applicable Modes or Other Specified Conditions," for Function 3, "Containment Radiation," as specified in Table 3.3.6-1, is currently required to be OPERABLE in are Modes 1, 2, 3, 4, 6(c).

Current Note (c) states:

> During CORE ALTERATIONS, and movement of irradiated fuel assemblies within containment.

SNC proposed the following changes:

> In Table 3.3.6-1, the "Applicable Modes or Other Specified Conditions" for the function Containment Radiation to be Modes 1, 2, 3, 4. The specified condition (c) "During CORE ALTERATIONS" is deleted, and the word *recently* is added.

Proposed new Note (c) states:

> (c) During movement of recently irradiated fuel assemblies within containment.

Current Note in TS 3.3.6, Condition C, is revised from:

> Only applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment.
>
> to:
>
> Only applicable during movement of recently irradiated fuel assemblies within containment.

3.13.2.1 Acceptability of TS 3.3.6

CORE ALTERATION is defined in TS 1.1, "Definitions," as follows:

> CORE ALTERATION shall be the movement of any fuel, sources, or other 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.

The proposed deletion of requirement for suspension of CORE ALTERATIONS operation does not affect the current Required Actions (RAs). These actions would continue to require that if no radiation monitoring channels are operable or the RA and associated COMPLETION TIME (CT) of Condition A are not met, operation may continue as long as the RA to place and maintain containment purge supply and exhaust isolation valves in their closed position is met or the applicable Conditions of the Limiting Conditions of Operation (LCO) 3.9.4, "Containment

Penetrations," are met for each penetration not in the required status. The CT for these RAs is Immediately. In addition, this is consistent with STS 3.9.2 in NUREG-1431, Revision 5 (Volume 1 -ML21259A155; Volume 2 – ML21259A159). Therefore, the NRC staff finds that the proposed TS 3.3.3.6 is acceptable**.**

3.13.3 TS 3.9.1, Boron Concentration

Maintaining boron concentration within limits in Mode 6 is required by TS 3.9.1 to assure that a core k_{eff} of 0.95 is maintained during refueling conditions. SNC proposed the following changes to RAs in TS 3.9.1:

> Required Action A.1, "Suspend CORE ALTERATIONS," and the associated Completion Time are deleted and Required Actions A.2 and A.3 are renumbered to A.1 and A.2 without changes to the associated requirements.

3.13.3.1 Acceptability of TS 3.9.1

With the proposed deletion of immediately suspending CORE ALTERATIONS, the proposed RA A.1 (current RA A.2) still requires immediate suspension of positive reactivity additions, which would prohibit diluting the boron concentration of the coolant in the reactor coolant system (RCS), adding fuel assemblies to the reactor vessel, or removing reactivity control components, and proposed RA A.2 (current RA A.3) continues to require the operator to initiate action to restore boron concentration to within its required limits. Therefore, the proposed RAs A.1 and A.2 would continue to assure that the requirements of boron concentration limits are met and is consistent with STS 3.9.1 in NUREG-1431, Revision 5. Therefore, the NRC staff finds that the proposed TS 3.9.1 is acceptable.

3.13.4 TS 3.9.2, Unborated Water Source Isolation Valves

Maintaining boron concentration within limits in Mode 6 is required by TS 3.9.2 which assures isolation of all sources of unborated water to the RCS during refueling conditions. SNC proposed the following changes to RAs in TS 3.9.2:

> Required Action A.1, "Suspend CORE ALTERATIONS," and the associated Completion Time are deleted and Required Actions A.2 and A.3 are renumbered to A.1 and A.2 without changes to the associated requirements.

3.13.4.1 Acceptability of TS 3.9.2

With the proposed deletion of immediately suspending CORE ALTERATIONS, the proposed RA A.1 (current RA A.2) still requires immediate isolation of unborated water sources valves to be secured in the closed position and proposed RA A.2 (current RA A.3) continues to require the operator to initiate action to restore boron concentration to within its required limits. Therefore, the proposed RAs A.1 and A.2 would continue to assure that the requirements of boron concentration limits are met and is consistent with STS 3.9.2 in NUREG-1431, Revision 5. Therefore, the NRC staff finds that the proposed TS 3.9.2 is acceptable.

3.13.5 TS 3.9.3, Nuclear Instrumentation

The source range nuclear flux monitors (SRNFMs) are used during operation in Mode 6 to monitor the core reactivity condition. The SRNFMs provide a signal to alert the operator to unexpected changes in core reactivity and the audible count rate from the SRNFMs provides prompt and definite indication to the operator to recognize the initiation of a boron dilution event. Current TS LCO 3.9.3 requires two operable SRNFMs and one operable channel of audible count rate during operation in Mode 6. With one SRNFM inoperable, RAs A.1 and A.2 require the operator to immediately suspend CORE ALTERATIONS, and suspend positive reactivity additions, respectively. With two SRNFMs inoperable, RAs B.1 and B.2 require the operator to immediately initiate action to restore one SRNFM to operable status, and perform Surveillance Requirement (SR) 3.9.1.1 once per 12 hours. SR 3.9.1.1 requires the operator to verify that the boron concentration is within the required limits within the TS specified frequency.

3.13.5.1 Changes to TS 3.9.3

SNC proposed addition of the following Note to TS 3.9.3 RA A.1:

-----------NOTE--------------------
CORE ALTERATIONS may
continue to restore an
inoperable source range
neutron flux monitor.

3.13.5.2 Acceptability of the Added Note to TS 3.9.3 RA A.1

TS 3.9.3 required two operable SRNFMs to provide a signal to alert the operator to unexpected changes in core reactivity. With one SRNFM inoperable, the existing RA A.1 requires immediate suspension of CORE ALTERATIONS. The proposed Note added to TS 3.9.3 RA A.1, allowing CORE ALTERATIONS to continue, would permit fuel assemblies, sources, and reactivity control components to be moved, if necessary, to restore an inoperable SRNFM.

While the proposed Note is added to TS 3.9.3 RA A.1, the existing RA A.2, which requires immediately suspending positive reactivity additions, remains unchanged. This action would prohibit the positive reactivity changes in the core, including the dilution of the boron concentration of the coolant in the RCS, the loading of fuel assemblies or sources into the core, or the removal of reactivity control components. The NRC staff finds that the proposed Note added to RA A.1 with the existing RA A.2, requiring immediately suspending positive reactivity additions, does not reduce the current TS 3.9.3 requirement to continue minimizing positive reactivity changes within the core, while providing the ability to safely restore the SRNFM capability, and therefore, the proposed Note is acceptable.

3.13.6 Variations for the Approved TSTF-51 and TSTF-471 Travelers

SNC is proposing the following variations from TSTF-51 and TSTF-471 to the Vogtle, Units 1 and 2, TSs changes. The NRC staff's review of the variations determined that the variations do not affect the applicability of TSTF-51 and TSTF-471 as explained below:

In its submittal dated June 30, 2022, states:

- The definition of CORE ALTERATION is being retained in TS Section 1.1, "Definitions," because this terminology continues to be used in a number of TSs, which are not being modified as a result of this amendment request. This is an administrative variation from TSTF-471 [and does not affect the applicability of TSTF-471].

The NRC staff finds this TS change acceptable because it is an administrative change.

SNC proposed the following variation:

- The control room emergency filtration system (CREFS) actuation instrumentation and the CREFS continue to be assumed to provide isolation, pressurization, and filtration of the MCR [main control room] in the event of an FHA. Since this system and associated isolation instrumentation are mitigation systems necessary to maintain dose to personnel in the MCR below the regulatory and regulatory guidance limits for an FHA, the following TSs and support TSs and associated TSs Bases are not modified:

    - TS 3.3.7, "Control Room Emergency Filtration/Pressurization System (CREFS) Actuation Instrumentation,"
    - TS 3.7.10, "Control Room Emergency Filtration/Pressurization System (CREFS) - Both Units Operating,"
    - TS 3.7.11, "Control Room Emergency Filtration/Pressurization System (CREFS) - One Unit Operating,"
    - TS 3.8.2, "AC Sources – Shutdown,"
    - TS 3.8.5, "DC Sources – Shutdown,"
    - TS 3.8.8, "Inverters – Shutdown,"
    - TS 3.8.10, "Distribution Systems – Shutdown," and
    - TS 3.9.7, "Refueling Cavity Water Level."

This is a plant-specific variation from TSTF-51 and TSTF-471 that is consistent with the proposed FHA licensing basis. Therefore, the NRC staff finds this variation acceptable. SNC proposed the following variation:

- The applicability requirements associated with the containment ventilation isolation instrumentation are shown in TS Table 3.3.6-1. This is a presentation difference from the applicability requirements shown in the NUREG-1431, TS 3.3.6 marked-up pages in TSTF-51. However, the proposed changes to footnote (c) in TS Table 3.3.6-1 are consistent with those shown in TSTF-51. These proposed changes are administrative variations from TSTF-51.

The NRC staff finds the proposed variation to be acceptable because it is an administrative change.

SNC proposed the following variation:

- TS 3.9.3, “Nuclear Instrumentation,” Required Actions were not modified in accordance with TSTF-471. However, [the] proposed Note added to Required Action A.1 is consistent with the intent of the proposed Note in TSTF-571-T, “Revise Actions for Inoperable Source Range Neutron Flux Monitor” (Reference 6 – [ML18221A561]). TSTF-571-T was accepted for use by the NRC as documented in a letter to the TSTF dated October 4, 2018 (Reference 7 – [ML17346A587]). Movement of fuel sources and reactivity control components within the reactor vessel is currently covered by the Core Alteration definition. Since the VEGP TSs retain the definition of Core Alteration, the required action continues to require suspension of core alterations and the note was modified to use the term Core Alterations. These proposed changes are considered administrative variations from TSTF-471.

The NRC staff finds the proposed variation to be acceptable because it is an administrative change.

In its submittal dated June 30, 2022, Section 3.14, item 4, SNC included the following statement,

> TSTF-571-T was accepted for use by the NRC as documented in a letter to the TSTF dated October 4, 2018.

This statement could be misleading, implying NRC staff approval. The NRC staff did not approve the Traveler TSTF-571-T. The NRC staff’s letter dated October 4, 2018 (ML17346A587), states in part, that “If a licensee includes the changes of traveler TSTF-571-T when adopting TSTF-286, “Operations Involving Positive Reactivity Additions,” (ML20106F133), the NRC staff’s technical concerns should be adequately addressed with regards to TSTF-286. As explained above, the NRC staff’s review of page E-1 of SNC’s letter dated February 6, 2023, finds a sufficient response to the NRC staff’s concerns stated in the NRC letter dated October 4, 2018.

3.13.7 Technical Evaluation Conclusion for TSTF-51 and TSTF-471

The NRC staff has reviewed the modified TS changes with the adoption of TSTF-51 and finds that, the specified ESF systems are no longer required during an FHA to ensure CR personnel dose remains below the 10 CFR 50.67(b)(2)(iii) dose limit and of-site dose remains below the accident dose limit specified in the NRC SRP.

The NRC staff has reviewed modified TS changes with the adoption of TSTF-471 and finds that elimination of the term CORE ALTERATIONS from the TSs will facilitate the refueling operations during Mode 6 and provide operational flexibility to operators. Since the requirements to suspend the movement of irradiated fuel assemblies within the containment will remain, the TS Action item, “Suspend CORE ALTERATIONS,” has no effect on the initial conditions or mitigation of any design accident or transient, and the licensee will eliminate the Action item from these TSs.

The NRC staff reviewed SNC changes to revise the Vogtle, Units 1 and 2, TSs to be consistent with TSTF-490 (ML052630462) and make associated changes, which include replacing the current specific activity of the reactor coolant limits with limits on RCS DEI and DEX specific activity. The NRC staff reviewed the proposed changes to (1) delete the reactor coolant gross activity limit, its associated conditions, RAs [Required Actions], and SRs [surveillance requirements], (2) delete the determination of Ē, and (3) add a new DEX limit, its associated conditions, RAs actions, and SRs, and has concluded that the changes are consistent with the methodology used to analyze the radiological consequences of the SGTR accident. The NRC staff finds that there is reasonable assurance that SNC's estimates of the TEDE from adoption of TSTF-51 and TSTF-471 comply with the acceptance criteria in RG 1.183 and the radiation dose limits in 10 CFR 50.67.

The NRC staff reviewed the proposed changes and determined that changes to the TSs meet the standards for TSs in 10 CFR 50.36(b) and 10 CFR 50.36(c). The proposed SRs assure that the necessary quality of systems and components is maintained that facility operation will be within safety limits, and that the LCOs will be met and satisfy 10 CFR 50.36(c)(3). Additionally, the changes to the TSs were reviewed for technical clarity and consistency with customary terminology and format in accordance with SRP Chapter 16. The NRC staff concludes that the proposed TS changes meet the requirements stated in 10 CFR 50.36 and are acceptable.

3.14 Technical Evaluation of Vogtle, Units 1 and 2, Accident Analysis for the Adoption of TSTF-490

The NRC staff evaluated SNC's submittal dated June 30, 2022, February 6, and March 24, 2023, to determine if the proposed changes are consistent with the regulations, guidance, and licensing information discussed in Section 2.0 of this SE and TSTF490, Revision 0.

3.14.1 TSTF-490 Background

The proposed changes would revise TS requirements relating to RCS primary coolant activity limits. The changes are consistent with TSTF Improved STS Change Traveler, TSTF-490, Revision 0, "Deletion of E-Bar Definition and Revision to RCS Specific Activity Tech Spec" (ML052630462). The proposed changes would replace the current TS limit on primary coolant gross specific activity with a new limit on primary coolant noble gas specific activity. The noble gas specific activity limit would be based on a new "Dose Equivalent Xenon (Xe)-133" (DE Xe-133) definition that would replace the current "Ē [E-Bar] Average Disintegration Energy" definition.

The primary coolant specific activity level is used in DBA analyses to determine the radiological consequences of accidents that involve the release of primary coolant activity with no substantial amount of fuel damage. For events that also include significant amounts of fuel damage, the contribution from the initial activity in the primary coolant is considered insignificant and is not normally evaluated.

The maximum allowable primary coolant specific activity is governed by the TSs. Due to the importance of iodine in the dose consequence analyses, the TSs specify separate dose equivalent specific activity limits for the iodine isotopes and non-iodine isotopes. The limit for iodine isotopes is specified in units of Dose Equivalent I-131 (DEI), which is the normalized quantity of I-131 that would result in the same dose consequence as the combination of the major isotopes of iodine present in the primary coolant.

The TSs for DEI include both an equilibrium long-term limit as well as a higher maximum allowable short-term limit, to account for iodine spiking.

The limit for non-iodine isotopes has traditionally been based on an evaluation of the average beta and gamma disintegration energy of the total non-iodine activity in the RCS, which is referred to as Ē. The STS TSs define Ē as “the average (weighted in proportion to the concentration of each radionuclide in the sample) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the isotopes, other than iodines, with half-lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.” The RCS non-iodine specific activity limit is then expressed as the quantity 100 divided by Ē in units of µCi/gm. In DBA dose consequence analyses, based on releases from the RCS with no significant fuel damage, the concentration of noble gas activity in the coolant is derived from that level associated with 1 percent fuel clad defects. Based on operating experience, depending on the isotopes used to calculate Ē and the actual degree of fuel clad defects, the routinely calculated value of Ē may not effectively indicate the level of noble gas activity relative to the levels used in the DBA dose consequence analyses on which the limit is based.

3.14.2 Proposed TS Changes with the Adoption of TSTF-490

The following are the proposed changes to TS Section 1.0, “Definitions:”

Remove definition of Ē–Average Disintegration Energy which currently states:

> Ē shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half-lives > 14 minutes, making up at least 95% of the total non-iodine activity in the coolant.

Current definition of DOSE EQUIVALENT I-131 states:

> DOSE EQUIVALENT I-131 shall be that concentration of 1-131 (microcuries per gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in EPA Federal Guidance Report No. 11 [FGR 11], “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion,” EPA-520/1-88-020, September 1988.

Proposed definition of DOSE EQUIVALENT I-131 states:

> 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 Committed Dose Equivalent (CDE) or Committed Effective Dose Equivalent (CEDE) dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11.

Add a new definition of Dose Equivalent XE-133 as follows:

DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External exposure to Radionuclides in Air, Water, and Soil."

Proposed changes to LCO 3.4.16, "Reactor Coolant System (RCS) Specific Activity," are as follows:

(Note: The changes listed below are shown in bold text.)

LCO 3.4.16 ~~**The specific activity of the reactor coolant shall be within limits**~~. **RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits.**

APPLICABILITY: MODES 1, ~~**and**~~ **2**, 3, and 4
~~**MODE 3 with RCS average temperature (Tavg) ³ 500°F.**~~

ACTIONS

~~-------------------------**NOTE**--~~
~~**LCO 3.0.4.c is applicable.**~~

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 **not within limit**~~**> 1.0 mCi/gm.**~~	-------------------**NOTE------------** **LCO 3.0.4.c is applicable**. A.1 Verify DOSE EQUIVALENT I-131 ≤ 60 µCi/gm ~~**within the acceptable region of Figure 3.4.16-1.**~~ AND A.2 Restore DOSE EQUIVALENT I-131 to within limit.	 Once per 4 hours 48 hours
B. ~~**Gross specific activity of the reactor coolant**~~ **DOSE EQUIVALENT XE-133** not within limit.	------------------NOTE----------- **LCO 3.0.4.c is applicable.** 1 ~~**Be in MODE 3 with Tavg < 500°F**~~ **Restore DOSE EQUIVALENT XE-133 to within limit.**	 ~~**6**~~ **48 hours**

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A **or B** not met. OR DOSE EQUIVALENT I-131 **> 60 µCI/gm.** ~~**in the unacceptable region of Figure 3.4.16-1.**~~	C.1 Be in MODE 3. ~~**3 with Tavg < 500°F.**~~ **AND** **C.2 Be in MODE 5**	6 hours 36 hours
SURVEILLANCE REQUIREMENTS		
SURVEILLANCE		FREQUENCY
SR 3.4.16.1 -------------------------------**NOTE-------------------------** **Only required to be performed in MODE 1.** **---** **Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 280 µCi/gm.** ~~**Verify reactor coolant gross specific activity £ 100/Ē mCi/gm**~~		In accordance with the Surveillance Frequency Control Program
SR 3.4.16.2 ----------------------------NOTE---------------------------- Only required to be performed in MODE 1. ------------------------ Verify reactor coolant DOSE EQUIVALENT I-131 specific activity £ 1.0 mCi/gm.		In accordance with the Surveillance Frequency Control Program AND Between 2 and 6 hours after a THERMAL POWER change of ³ 15% RTP within a 1 hour period
~~**SR 3.4.16.3 ----------------------------NOTE---------------------------- Not required to be performed until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. --- Determine Đ from a sample taken in MODE 1 after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours**~~		~~**In accordance with the Surveillance Frequency Control Program**~~

In addition, the proposed changes delete TS Figure 3.4.16-1, “Reactor Coolant dose equivalent I-131 reactor coolant specific activity limit versus percent of Rated Thermal Power with the reactor coolant specific activity >1 µci/gram dose equivalent I-131.”

3.14.3Technical Evaluation of Modified TS Changes per TSTF-490

3.14.3.1 ‘Deletion of the Definition Ē and the Addition of a New Definition for DE XE-133

The new definition for DE XE-133 is similar to the definition for DEI. The determination of DE XE-133 will be performed in a similar manner to that currently used in determining DEI, except that the calculation of DE XE-133 is based on the acute dose to the whole body and considers the noble gases krypton (Kr)-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present, which are significant in terms of contribution to whole body dose. Some noble gas isotopes are not included due to low concentration, short half-life, or small DCF. The calculation of DE XE-133 would use the effective DCFs for air submersion from Table III.1 of the EPA Federal Guidance Report No. 12, 1993, “External Exposure to Radionuclides in Air, Water, and Soil.” Using this approach, the limit on the amount of noble gas activity in the primary coolant would not fluctuate with variations in the calculated values of Ē. If a specified noble gas nuclide is not detected, the new definition states that it should be assumed that the nuclide is present at the minimum detectable activity. This will result in a conservative calculation of DE XE-133.

When Ē is determined using a design basis approach in which it is assumed that 1.0 percent of the power is being generated by fuel rods having cladding defects, and it is also assumed that there is no removal of fission gases from the letdown flow, the value of Ē is dominated by Xe 133. The other nuclides have relatively small contributions. However, during normal plant operation, there are typically only a small amount of fuel clad defects, and the radioactive nuclide inventory can become dominated by tritium and corrosion and/or activation products, resulting in the determination of a value of Ē that is very different than would be calculated using the design basis approach. Because of this difference, the accident dose analyses become disconnected from plant operation, and the LCO becomes essentially meaningless. This difference also results in a TS limit that can vary during operation as different values for Ē are determined.

This proposed change will implement an LCO that is consistent with the whole-body radiological consequence analyses, which are sensitive to the noble gas activity in the primary coolant but not to other non-gaseous activity currently captured in the Ē definition. TS SR 3.4.16.1 specifies the limit for primary coolant specific activity of the RCS as 100/Ē µCi/gram. The current Ē definition includes radioisotopes that decay by the emission of both gamma and beta radiation. The current Required Action B.1 of TS LCO 3.4.16 would rarely, if ever, be entered for exceeding 100/Ē µCi/gram, since the calculated value is very high (the denominator is very low) if beta emitters such as tritium are included in the determination, as required by the Ē definition.

The following TS 1.11 definition “Ē-AVERAGE DISINTEGRATION ENERGY” is deleted:

> Ē shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half-lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

A new definition "DOSE EQUIVALENT XE-133," replaces the above Ē-AVERAGE DISINTEGRATION ENERGY definition. The added definition for DEX states:

> DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at a minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."

The NRC staff has reviewed the changes to the TS and finds that SNC's proposed deletion of the above-stated definition for Ē and addition of a new definition for DEX in TS 1.1 is acceptable from a radiological dose perspective since it will result in an LCO that more closely relates the non-iodine Reactor Coolant System (RCS) activity limits to the dose consequence analyses that form their bases.

3.14.3.2 Revision of TS LCO 3.4.16

Revision of TS LCO 3.4.16 is modified to specify that iodine specific activity in terms of DEI and noble gas specific activity in terms of DEX shall be within limits. Currently, the limiting indicators are not explicitly identified in the LCO, but instead defined in current Condition A and SR 3.4.16.1 for reactor coolant gross non-iodine activity and in current Condition A and SR 3.4.16.2 for iodine specific activity.

The proposed change replaces "The specific activity of the reactor coolant," with "RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity," such that the modified TS LCO 3.4.16 states, "RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits." This proposed change modifies the TS LCO so that it is easier to read because the limiting indicators are explicitly stated in the LCO. The NRC staff finds that this proposed change provides clarity and is acceptable.

3.14.3.3 TS 3.4.16 Applicability

Revision to TS 3.4.16 applicability is modified to include all of Mode 3 and Mode 4. It is necessary for the LCO to apply during Modes 1 through 4 to limit the potential radiological consequences of an SGTR that may occur during these modes. In Mode 5, with the RCS loops filled, the SGs are specified as a backup means of decay heat removal by natural circulation. In Mode 5, however, due to the reduced temperature of the RCS, the probability of a DBA involving the release of significant quantities of RCS inventory is greatly reduced. Therefore, monitoring of RCS specific activity is not required. In Mode 5, with the RCS loops not filled, and in Mode 6, the SGs are not used for decay heat removal, the RCS and SGs are depressurized, and primary-to-secondary leakage is minimal. Therefore, the monitoring of RCS specific activity is not required. The proposed change to modify the TS 3.4.16 applicability to include all of Mode 3 and Mode 4 is necessary to limit the potential radiological consequences of an SGTR that may occur during these modes, and is, therefore, acceptable from a radiological dose perspective and meets all applicable regulatory requirements.

3.14.3.4 Condition A Revision to Include Action for DOSE EQUIVALENT I-131 (DEI) Limit

Modification to TS 3.4.16, RA A.1 is revised to remove the reference to Figure 3.4.16-1, "Reactor Coolant DOSE EQUIVALENT I-131 Reactor Coolant Specific Activity Limit versus Percent of RATED THERMAL POWER With the Reactor Coolant Specific Activity ≥ 1.0 µCi/gm DOSE EQUIVALENT I-131," and insert a limit of less than or equal to the site-specific DEI spiking limit, which is 60 µCi/gm. The curve contained in Figure 3.4.16-1 was provided by the Atomic Energy Commission (AEC) in a letter dated June 12,1974 (Microfiche 35028: 117 – 118), on the subject entitled "Proposed Standard Technical Specifications for Primary Coolant Activity." The radiological dose consequence analysis for the SGTR accident takes into account the pre-accident iodine spike and does not consider the elevated RCS iodine specific activities permitted by Figure 3.4.16-1 for operation at power levels below 80 percent RTP. Instead, the pre-accident iodine spike analyses assume a DEI concentration 60 times higher than the corresponding long-term equilibrium value, which corresponds to the specific activity limit associated with 100 percent RTP operation. The NRC staff finds it acceptable that TS 3.4.16 RA A.1 is based on the short-term site-specific DEI spiking limit to be consistent with the assumptions contained in the radiological consequence analyses.

3.14.3.5 TS 3.4.16 Condition B Revision to Include Action for DOSE EQUIVALENT XE-133 (DEX) Limit

Condition B of TS 3.4.16 is revised for DEX not within limits. This change is made to be consistent with the change to the TS 3.4.16 LCO, which requires the DEX specific activity to be within limits, as discussed above in Section 3.2.2, item 2. The DEX limit is site-specific, and the numerical value of 280 µCi/gm is contained in revised SR 3.4.16.1. The site-specific limit of 280 µCi/gm DEX is below the maximum accident analysis RCS activity corresponding to 1% fuel clad defects and has sufficient margin to accommodate the exclusion of isotopes present in the RCS which have low concentration, short half-life, or small dose conversion factors. The primary purpose of TS 3.4.16 LCO on RCS specific activity and its associated conditions is to support the dose analyses for DBAs. The whole-body dose is primarily dependent on the noble gas activity, not the non-gaseous activity currently captured in the $\bar{E}$ definition.

The NRC staff notes that SNC elected to incorporate a more restrictive DE Xe-133 than would be calculated based on the nominal Noble Gas concentrations listed in UFSAR Table 11.1-2 and as updated on pages E-24 and E-25 of the AST submittal dated June 30, 2022. Using these concentrations, the calculated DE Xe-133 value would be approximately 659 micro curies (µCi) per gram. The licensee conservatively chose to propose a TS limit for DE Xe-133 of 280 µCi/gram providing additional margin to the dose acceptance criteria associated with the dose analyses performed using the AST methodology. The NRC staff finds that SNC made a conservative determination of the proposed DE Xe-133 TS limit. Therefore, the NRC staff finds the proposed adoption of a DE Xe-133 value of 280 µCi/gram is acceptable from a dose consequence perspective.

The Completion Time (CT) for the revised TS 3.4.16 RA B.1 requires restoration of DEX to within the limit in 48 hours. This is consistent with the proposed CT for TS 3.4.16 RA A.1 for DEI. The radiological consequences for the SGTR accident demonstrate that the calculated thyroid doses are generally a greater percentage of the applicable acceptance criteria than the calculated whole-body doses. It then follows that the CT for noble gas activity being out of specification in revised RA B.1 should be at least as great as the CT for iodine specific activity being out of specification in revised RA A.2. Therefore, the CT of 48 hours for revised RA B.1 is acceptable from a radiological dose perspective, because it is expected that if there were a xenon spike in the normal coolant, iodine concentration would be restored within this time period. In addition, there is a low probability of an SGTR occurring during this time period.

A Note is also added to the revised RA B.1 that states LCO 3.0.4.c is applicable. This Note would allow entry into a mode or other specified condition in the LCO applicability when LCO 3.4.15 is not being met and is the same note that is currently stated for Required Actions A.1 and A.2. The proposed note would allow entry into the applicable modes from Mode 4 (hot shutdown) to Mode 1 (power operation) while the DEX limit is exceeded, and the DEX is being restored to within its limit. This mode change is acceptable due to the significant conservatism incorporated into the DEX specific activity limit, the low probability of an event occurring that is limiting due to exceeding the DEX specific activity limit, and the ability to restore transient specific excursions while the plant remains at, or proceeds to, power operation.

3.14.3.6 TS 3.4.16 Condition C Revision to Include Condition B (DEX not within limit)

Condition C of TS 3.4.16 is revised to include revised Condition B (DEX not within limit) if the Required Action and associated CT of Condition B is not met. This is consistent with the revised Condition B, which now provides the same CT for both components of RCS specific activity, as discussed above in the revision to Condition C. The revision to Condition C also replaces the limit on DEI from the deleted Figure 3.4.16-1 with a site-specific value of > 60 μCi/gm. This change makes revised Condition C consistent with the changes made to TS 3.4.16 RA A.1.

The proposed change to revised TS 3.4.16 RA C.1 deletes the requirement for average RCS temperature to be less than 500 ^{o}F and adds a new RA C.2, which requires the plant to be in Mode 5 within 36 hours. These changes are consistent with the changes made to the TS 3.4.16 applicability. The revised LCO is applicable throughout all of Modes 1 through 4 to limit the potential radiological consequences of an SGTR that may occur during these modes. In Mode 5 (COLD SHUTDOWN), with the RCS loops filled, the SGs are specified as a backup means of decay heat removal by natural circulation. In Mode 5, however, due to the reduced temperature of the RCS, the probability of a DBA involving the release of significant quantities of RCS inventory is greatly reduced. Therefore, monitoring of RCS specific activity is not required. In Mode 5, with the RCS loops not filled, and Mode 6 (REFUELING), the SGs are not used for decay heat removal, the RCS and SGs are depressurized, and primary to secondary leakage is minimal. Therefore, the monitoring of RCS specific activity is not required.

A new CT of 36 hours is added in TS 3.4.16 RA C.2 for the plant to reach Mode 5. This CT is reasonable, based on operating experience, to reach Mode 5 from full power conditions in an orderly manner and without challenging plant systems, and the value of 36 hours is consistent with other TSs, which have a CT to reach Mode 5. Based on all the above, the NRC staff finds the new Condition C, its associated RAs, and its CTs, provide reasonable assurance of adequate protection and are, therefore, acceptable.

3.14.3.7 SR 3.4.16.1 DEX Surveillance

The change would replace the current SR 3.4.16.1 surveillance for RCS gross specific activity with a surveillance to verify that the site-specific reactor coolant DEX specific activity is ≤ 280 μCi/gm. This change provides a surveillance for the new LCO limit added to TS 3.4.16 for DEX. The revised SR 3.4.16.1 surveillance requires performing a gamma isotopic analysis as a measure of the noble gas specific activity of the RCS in accordance with the surveillance frequency control program (SFCP). This measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. The surveillance provides an indication of any increase in the noble gas specific activity. The results of the surveillance on DE Xe-133 allow proper remedial actions to be taken before reaching the LCO limit under normal operating conditions, and is, therefore, acceptable.

SNC proposes to add a Note to SR 3.4.16.1 that the surveillances be performed in MODE 1 which is consistent with the current SR 3.4.16.2 for the verification of the I-131 Dose Equivalent specific activity and the NRC staff's approved TSTF-490. Therefore, addition of the Note to SR 3.4.16.1 is acceptable.

3.14.3.8 SR 3.4.16.3 Deletion

The current SR 3.4.16.3 that required the determination of Ē is deleted. TS 3.4.16 LCO on RCS specific activity supports the dose analyses for DBAs in which the whole-body dose is primarily dependent on the noble gas concentration, not the non-gaseous activity currently captured in the Ē definition. With the elimination of the limit for RCS gross specific activity and the addition of the new LCO limit for noble gas specific activity, this SR to determine Ē is no longer required; therefore, the NRC staff finds the deletion of SR 3.4.16.3 is acceptable.

3.15 Variations from the Approved TSTF-490

In its letter dated June 30, 20222, states, the licensee stated that "SNC is not proposing any variations or deviations from the applicable parts of the NRC staff's model safety evaluation. SNC is proposing the following variations from the TS changes described in the TSTF-490, Revision 0.

> The VEGP TS include a Surveillance Frequency Control Program. Therefore, the periodic Surveillance Frequencies shown in TSTF-490 are replaced with the statement, "In accordance with the Surveillance Frequency Control Program."
>
> Additionally, SNC is proposing an administrative change which deletes the global NOTE regarding LCO 3.0.4c applicability. This NOTE is replaced by the same NOTE added to Action A (administrative change) and to Action B (TSTF-490). The NOTE does not apply to Action C (shutdown action)."

The NRC staff has reviewed Vogtle, Units 1 and 2, SFCP as specified in Vogtle, Units 1 and 2, TS 5.5.15. This program provides controls for Surveillance Frequencies. It states, "The program shall ensure that SRs specified in the TSs are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met." The Staff's review determined that the licensee's proposed deviation meets the requirements for implementation of the SFCP, and is, therefore, acceptable.

The NRC staff's review of SNC's variation regarding relocation of LCO 3.4.16 Note, "LCO 3.0.4c is applicable," to LCO 3.4.16 Required Actions for LCO 3.4.16 Conditions A and B is appropriate since the Note is construed to be part of the Required Actions, and therefore it meets the 10 CFR 50.36 regulation, additionally, the change is consistent with STS 3.4.16 in NUREG-1431, Revision 5. Therefore, the variation is acceptable.

4.0 TABLES

Table 1
Total Effective Dose Equivalent per Accident in roentgen equivalent man (rem)

Accident	EAB[1]	LPZ[2]	SRP 15.0.1 and RG 1.183 Limit	Control Room	10 CFR 50.67 and GDC 19 Limit
Loss of Coolant Accident	8.4	9.6	25	4.4	5
Fuel Handling Accident in Spent Fuel Pool	1.0	0.4	6.3	3.9	5
Main Steam Line Break Pre-incident Spike Concurrent Spike	<0.1 0.2	<0.1 0.2	25 2.5	<0.1 0.2	5 5
Steam Generator Tube Rupture Pre-incident Spike Concurrent Spike	1.6 1.4	0.9 0.8	25 2.5	0.6 0.5	5 5
Control Rod Ejection Accident Containment release Secondary release	1.5 0.5	1.8 0.4	6.3 6.3	0.6 1.1	5 5

(1) Exclusion Area Boundary
(2) Low Population Zone

Table 2: Control Room Inputs and Assumptions	
Input/Assumption	**AST Value**
Control Room Volume	1.49×10^5 ft^3
Normal Operation	
Filtered Make-up Flow Rate	0 cfm
Filtered Recirculation Flow Rate	0 cfm
Unfiltered Make-up Flow Rate	2575 cfm
Unfiltered In-leakage	0 cfm
Emergency Operation – Recirculation Mode	
Filtered Make-up Flow Rate	1800 cfm
Filtered Recirculation Flow Rate	31,000 cfm
Unfiltered Make-up Flow Rate	0 cfm
Unfiltered In-leakage	190 cfm
Filter Efficiencies	
Pressurization Filters	All iodine 99%
Recirculation Filters	Elemental–99% Organic–99% Particulate–99%
Occupancy	
0-24 hours	100%
1-4 days	60%
4-30 days	40%
Breathing Rate	3.5×10^{-4} m^3/sec (0-720 hr)

Table 3: LOCA Inputs and Assumptions	
Input/Assumption	**AST Value**
Containment Purge	
Iodine Chemical Form	95% cesium iodide 4.85% elemental 0.15% organic
Containment Volume	2,930,000 ft^3
Containment Purge Filtration	0
Removal by Wall Deposition	None
Removal by Sprays	None
Containment Leakage	
Iodine Chemical Form	95% cesium iodide 4.85% elemental 0.15% organic
Containment Sump pH	>7.0
Containment Sprayed Volume	2,300,000 ft^3
Containment Unsprayed Volume	630,000 ft^3
Containment Spray Start Time	110 seconds
Containment Spray Stop Time	2 hours
Containment Spray Flow Rate	2500 gpm
Elemental Iodine Spray Removal Coefficient	13.7 hr^{-1}
Aerosol Spray Removal Coefficient	5.34 hr^{-1}
Organic Iodine Spray Removal	None
Natural Deposition	Elemental, Organic iodine–None Aerosols–0.1 hr^{-1}
Containment Leakage Rate	
0 to 24 hours	0.21%/day
24 hours to 30 days	0.105%/day
Containment Leakage Filtration	0%
ECCS Leakage to the Auxiliary Building	
Iodine Chemical Form	0% aerosol 97% elemental 3% organic
Containment Sump Volume	114,922 ft^3
ECCS Recirculation Start Time	30 minutes
ECCS Leakage Flow Rate	2 gpm
ECCS Flashing Fraction	10%

Table 4: FHA Inputs and Assumptions	
Input/Assumption	**AST Value**
Iodine Chemical Form	0% aerosol 99.85% elemental 0.15% organic
Number of Fuel Assemblies Damaged	314 Rods (1 FA + 50 rods in target assembly)
Percentage of Fuel Rods Damaged	100%
No. of rods exceeding 6.3 kw/ft above 54 GWD/MTU	314
Water Level Above Damaged Fuel	23 ft
Pool Decontamination Factors	200-Overall
Delay Before Fuel Movement	70 hours
Onsite χ/Qs FHB 0-2 hrs 2-8 hrs 8-24 hrs	 6.01×10^{-3} sec/m^3 4.44×10^{-3} sec/m^3 1.71×10^{-3} sec/m^3

Table 5: MSLB Inputs and Assumptions

Input/Assumption	AST Value
Maximum Pre-Accident Iodine Spike Concentration	60 μCi/gm Dose Equivalent I-131
Concurrent Iodine Spike Appearance Rate	500 X Equilibrium
Initial Steam Generator Iodine Source Term	0.1 μCi/g Dose Equivalent Iodine
Iodine Chemical Form	0% aerosol, 97% elemental, 3% organic
Percentage of Fuel Rods Failed	0
RCS Mass	2.258×10^8 g
Steam Generator Secondary Liquid Mass	1.78×10^8 g (all 4 SGs)
Intact Steam Generator Steam Release	
0–2 hrs	4.66×10^5 lbm
2–8 hrs	1.06×10^6 lbm
8–24 hrs	2.11×10^6 lbm
Primary-Secondary Leak Rate	0.65 gpm to two intact SGs, 0.35 gpm to faulted SG
Density Used for Leakage Volume-to-Mass Conversion	62.4 lbm/ft^3
Time to Cool RCS to RHR Cut-in	20 hrs
Intact Steam Generator Iodine partition factor	100
Intact Steam Generator Moisture Carryover Fraction	0.32% (Alkali Metal Partition Factor =312)

Table 6: SGTR Inputs and Assumptions	
Input/Assumption	**AST Value**
Maximum Pre-Accident Iodine Spike Concentration	60 μCi/gm Dose Equivalent I-131
Concurrent Iodine Spike Appearance Rate	335 X Equilibrium
Initial Steam Generator Iodine Source Term	0.1 μCi/gm Dose Equivalent I-131
Iodine Chemical Form	0% aerosol 97% elemental 3% organic
Percentage of Fuel Rods Failed	0
RCS Mass	2.258×10^8 g
Steam Generator Secondary Liquid Mass	4.44×10^7 g (each)
Intact Steam Generator Steam Release 0–2 hours 2–20 hr	 696,500 lbm 2,495,600 lbm
Ruptured Steam Generator Steam Release 0–2 hours 2–20 hr	 214,000 lbm 37,100 lbm
Time of Reactor Trip	49.7 s
Primary-Secondary Leak Rate	1 gpm
Density Used for Leakage Volume-to-Mass Conversion	62.4 lbm/ft^3
Ruptured Tube Break Flow	0–5502 s – 204,900 lbm
Break Flow Flashing Fraction	0.15 Constant
Time to Cool RCS to RHR cut-in	20 hours
Intact Steam Generator Iodine Partition Coefficient	100
Intact Steam Generator Moisture Carryover Fraction	0.32% (Partition factor for Alkali Metals = 312)

Table 7: CRE Inputs and Assumptions	
Input/Assumption	**AST Value**
Fuel Rod Gap Fractions	Iodine/noble gases–0.10 Other halogens–0.05 Alkali metals–0.12
Fuel Rod Peaking Factor	1.7
Percentage of Fuel Rods Damaged	0.1
Percentage of Fuel That Experiences Melting	0.0025
Number of rods exceeding 6.3 kw/ft above 54 GWD/MTU	0

Table 7: CRE Inputs and Assumptions	
Input/Assumption	**AST Value**
Initial RCS Iodine Source Term	1.0 µCi/gm DE I-131
Initial RCS non-Iodine Activity	1% defective fuel (alkali metals and other halogens) 280 µCi/gm DE Xe-133 (noble gases)
Initial Steam Generator Iodine Source Term	0.1 µCi/gm DE I-131
Iodine Chemical Form–Secondary Release	97% elemental 3% organic elemental
Iodine Chemical Form–Containment Release	95% aerosol 4.85% elemental 0.15% organic
Containment Volume	2.93×10\3 ft^3
Containment Leakage Rate	
0 to 24 hours	0.201%
24 hours to 30 days	0.105%
Natural Deposition in Containment	Elemental iodine – None Aerosols – 3.005 x 10^{-2} hr-1
Iodine/Particulate Removal by Containment Sprays	Not credited
RCS Mass	2.258 x 10^8 grams
Steam Generator Secondary Liquid Mass	1.778 x 10^8 grams
Primary-Secondary Leak Rate	1 gpm total
Duration of Primary-to-Secondary Leakage	20 hours
Density Used for Leakage Volume-to-Mass Conversion	62.4 lbm/ft^3
Secondary Steam Release	
0-2 hrs	5.5 x 10^5 lbm
2-8 hrs	1.365 x 10^6 lbm
8-20 hr	2.73 x 10^6 lbm
Duration of SG Tube Uncovery Following Reactor Trip	0 minutes
Steam Generator Iodine Partition Coefficient	100
Steam Generator Moisture Carryover Fraction	0.32% Partition factor for Alkali Metals = 312)

Table 8: LRA Inputs and Assumptions	
Input/Assumption	**AST Value**
Fuel Rod Gap Fractions	I-131–0.08 Kr-85–0.10 Other Halogens and Noble Gases–0.05 Alkali Metals–0.12 5% Failed Fuel
Fuel Rod Peaking Factor	1.7
Number of rods exceeding 6.3 kw/ft above 54 GWD/MTU	0
Initial Steam Generator Iodine Source Term	0.1 µCi/gm
Iodine Chemical Form	95% particulate 4.85% elemental 0.15% organic
RCS Mass	2.258 x 10^8 lbm
Steam Generator Secondary Liquid Mass	1.78 x 10^8 lbm
Primary-Secondary Leak Rate	1 gpm total
Density Used for Leakage Volume-to-	62.4 lbm/ft^3
Secondary Steam Release Mass Conversion	
0–2 hr	610,500 lbm
2–8 hr	1,501,500 lbm
8–20 hr	3,003,000 lbm
Duration of SG Tube Uncovery Following Reactor Trip	0 minutes
Steam Generator Iodine Partition Coefficient	100
Intact Steam Generator Moisture Carryover Fraction	0.32% (partition coefficient of 312 for Alkali Metals)

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Georgia State official was notified of the proposed issuance of the amendments on June 7, 2023. On June 27, 2023, the State official confirmed that the State of Georgia had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change the requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration published in the *Federal Register* on September 6, 2022 (87 FR 54553), and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

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

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Date: July 31, 2023

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2, ISSUANCE OF AMENDMENTS NOS. 219 AND 202, REGARDING ALTERNATE SOURCE TERM, TSTF-51, TSTF-471, AND TSTF-490 (EPID L-2022-LLA-0096) DATED JULY 31, 2023

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