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

June 12, 2019

Mr. Daniel G. Stoddard
Senior Vice President and
Chief Nuclear Officer
Innsbrook Technical Center
5000 Dominion Blvd.
Glen Allen, VA 23060-6711

SUBJECT: SURRY POWER STATION, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 295 AND 295 TO ADOPT TSTF-490, REVISION 0, AND UPDATE ALTERNATIVE SOURCE TERM ANALYSES (EPID L-2018-LLA-0068)

Dear Mr. Stoddard:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 295 to Renewed Facility Operating License No. DPR-32 and Amendment No. 295 to Renewed Facility Operating License No. DPR-37 for the Surry Power Station (Surry), Unit Nos. 1 and 2, respectively. The amendments revise the Technical Specifications (TSs) in response to your application dated March 2, 2018, as supplemented by letter dated October 25, 2018.

The amendments revise the Surry, Unit Nos. 1 and 2, TSs consistent with Revision 0 to the Technical Specifications Task Force (TSTF) Traveler, TSTF-490, "Deletion of E Bar Definition and Revision to RCS [reactor coolant system] Specific Activity Tech Spec." The amendments adopt TSTF-490, Revision 0, and make associated changes, which include replacing the current limits on primary coolant gross specific activity with limits on primary coolant noble gas specific activity. The amendments also update the Alternative Source Term analyses bases for new codes, revised atmospheric dispersion factors, new fuel handling accident fuel rod gap fractions and control room isolation operator action time, and eliminate the locked rotor accident dose consequences.

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

Sincerely,

Karen Cotton-Gross

Karen Cotton-Gross, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-280 and 50-281

Enclosures:
1. Amendment No. 295 to DPR-32
2. Amendment No. 295 to DPR-37
3. Safety Evaluation

cc: Listserv

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-280

SURRY POWER STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 295
Renewed License No. DPR-32

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

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated March 2, 2018, as supplemented by letter dated October 25, 2018, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Renewed Facility Operating License No. DPR-32 is hereby amended to read as follows:

    B. Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 295, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of the date of issuance. Implementation shall include revision of the Safety Analysis Report (Alternative Source Term analysis) approved in the Safety Evaluation and provided in the next periodic update in accordance with 10CFR 50.71(e).

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

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

Attachment:
Changes to License No. DPR-32
and the Technical Specifications

Date of Issuance: June 12, 2019

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 295
Renewed License No. DPR-37

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

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated March 2, 2018, as supplemented by letter dated October 25, 2018, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Renewed Facility Operating License No. DPR-37 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 295, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of the date of issuance. Implementation shall include revision of the Safety Analysis Report (Alternative Source Term analysis) approved in the Safety Evaluation and provided in the next periodic update in accordance with 10CFR 50.71(e).

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to License No. DPR-37
and the Technical Specifications

Date of Issuance: June 12, 2019

ATTACHMENT TO LICENSE AMENDMENT NOS. 295 AND 295

SURRY POWER STATION, UNIT NOS. 1 AND 2

RENEWED FACILITY OPERATING LICENSE NOS. DPR-32

RENEWED FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NOS. 50-280 AND 50-281

Replace the following pages of the Licenses and the Appendix A Technical Specifications (TSs) 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. DPR-32, page 3	License No. DPR-32, page 3
License No. DPR-37, page 3	License No. DPR-37, page 3
TSs	TSs
TS 1.0-5	TS 1.0-5
TS 1.0-8	TS 1.0-8
-------	TS 1.0-8a
TS 3.1-15a	TS 3.1-15a
TS 3.1-15b	TS 3.1-15b
TS 3.1-16	TS 3.1-16
TS 3.1-17	TS 3.1-17
TS 3.1-17a	TS 3.1-17a
TS 3.6-4	TS 3.6-4
TS 4.1-5a	TS 4.1-5a
-------	TS 4.1-5b
TS 4.1-10	TS 4.1-10
TS 4.1-10a	TS 4.1-10a
TS 6.6-3	TS 6.6-3

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

   A. Maximum Power Level

      The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2587 megawatts (thermal).

   B. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 295 are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

   C. Reports

      The licensee shall make certain reports in accordance with the requirements of the Technical Specifications.

   D. Records

      The licensee shall keep facility operating records in accordance with the requirements of the Technical Specifications.

   E. Deleted by Amendment 65

   F. Deleted by Amendment 71

   G. Deleted by Amendment 227

   H. Deleted by Amendment 227

   I. Fire Protection

      The licensee shall implement and maintain in effect the provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report and as approved in the SER dated September 19, 1979, (and Supplements dated May 29, 1980, October 9, 1980, December 18, 1980, February 13, 1981, December 4, 1981, April 27, 1982, November 18, 1982, January 17, 1984, February 25, 1988, and

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

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

J. Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power Levels not in excess of 2587 megawatts (thermal)

K. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 295 are hereby incorporated in this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

L. Reports

The licensee shall make certain reports in accordance with the requirements of the Technical Specifications.

M. Records

The licensee shall keep facility operating records in accordance with the Requirements of the Technical Specifications.

N. Deleted by Amendment 54

O. Deleted by Amendment 59 and Amendment 65

P. Deleted by Amendment 227

Q. Deleted by Amendment 227

K. FIRE SUPPRESSION WATER SYSTEM

A fire suppression water system shall consist of: a water source(s), gravity tank(s) or pump(s), and distribution piping with associated sectionalizing control or isolation valves. Such valves shall include yard hydrant curb valves, and the first valve ahead of the water flow alarm device on each sprinkler, hose standpipe, or spray system riser.

L. OFFSITE DOSE CALCULATION MANUAL (ODCM)

An Offsite Dose Calculation Manual (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Radiological Environmental Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.6.B.2 and 6.6.B.3.

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

N. GASEOUS RADWASTE TREATMENT SYSTEM

A gaseous radwaste treatment system is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

W. <u>STAGGERED TEST BASIS</u>

A staggerec test basis shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

X. <u>LEAKAGE</u>

LEAKAGE shall be:

a. <u>Identified LEAKAGE</u>

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 either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE, or

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

b. <u>Unidel1tified LEAKAGE</u>

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

c. <u>Pressure Boundary LEAKAGE</u>

LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

Y. 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."

D. RCS Specific Activity

Applicability

The following specifications are applicable whenever T_{avg} (average RCS temperature) exceeds 200°F.

Specification

1. The specific activity of the primary coolant shall be limited to $\leq$ 1 μCi/gm DOSE EQUIVALENT I-131.

   a. With the specific activity of the primary coolant > 1 μCi/gm DOSE EQUIVALENT I-131, verify DOSE EQUIVALENT I-131 $\leq$ 10 μCi/gm once per 4 hours.

   b. With the specific activity of the primary coolant > 1 μCi/gm DOSE EQUIVALENT I-131, but $\leq$ 10 μCi/gm, unit startup or POWER OPERATION may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT I-131 to $\leq$ 1 μCi/gm limit.

   c. With the specific activity of the primary coolant > 1 μCi/gm DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or DOSE EQUIVALENT I-131 is > 10 μCi/gm, place the reactor in HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours.

2. The specific activity of the primary coolant shall be limited to $\leq$ 234 μCi/gm DOSE EQUIVALENT XE-133.

   a. With the specific activity of the primary coolant > 234 μCi/gm DOSE EQUIVALENT XE-133, unit startup or POWER OPERATION may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT XE-133 to $\leq$ 234 μCi/gm limit.

   b. With the specific activity of the primary coolant > 234 μCi/gm DOSE EQUIVALENT XE-133 for more than 48 hours during one continuous time interval, place the reactor in HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours.

BASES

BACKGROUND - The maximum dose that an individual at the exclusion area boundary can receive for 2 hours following an accident, or at the low population zone outer boundary for the radiological release duration, is specified in 10 CFR 50.67 (Ref. 1). Doses to control room operators must be limited per GDC 19. The limits on specific activity ensure that the offsite and control room doses are appropriately limited during analyzed transients and accidents.

The Reactor Coolant System (RCS) specific activity Limiting Condition for Operation (LCO) limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the dose consequences in the event of a steam line break (SLB) or steam generator tube rupture (SGTR) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133. The allowable levels are intended to ensure that offsite and control room doses meet the appropriate acceptance criteria in the Standard Review Plan (Ref. 2).

APPLICABLE SAFETY ANALYSES - The LCO limits on the specific activity of the reactor coolant ensure that the resulting offsite and control room doses meet the appropriate Standard Review Plan acceptance criteria following a SLB or SGTR accident. The safety analyses (Refs. 3 and 4) assume the specific activity of the reactor coolant is at the LCO limits, and an existing reactor coolant steam generator (SG) tube leakage rate of 1 gpm exists. The safety analyses assume the specific activity of the secondary coolant is at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.6.H, Secondary Specific Activity.

The analyses for the SLB and SGTR accidents establish the acceptance limits for RCS specific activity. Reference to these analyses is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

The safety analyses consider two cases of reactor coolant iodine specific activity. One case assumes specific activity at 1.0 μCi/gm DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the rate of release of iodine from the fuel rods containing cladding defects to the primary coolant immediately after a SLB (by a factor of 500), or SGTR (by a factor of 335), respectively. The second case assumes the initial reactor coolant iodine activity is at 10.0 μCi/gm DOSE EQUIVALENT I-131 due to an iodine spike caused by a reactor or RCS transient prior to the accident. In both cases, the noble gas specific activity is assumed to be 234 μCi/gm DOSE EQUIVALENT XE-133.

The SGTR analysis assumes a coincident loss of offsite power. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature ΔT signal.

The loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG power operated relief valves and the main steam safety valves. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends and the Residual Heat Removal (RHR) system is placed in service.

The SLB radiological analysis assumes that offsite power is lost at the same time as the pipe break occurs outside containment. Reactor trip occurs after the generation of an SI signal on low steam line pressure. The affected SG blows down completely and steam is vented directly to the Turbine Building to maximize control room dose. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends and the RHR system is placed in service.

Operation with iodine specific activity levels greater than the LCO limit is permissible if the activity levels do not exceed 10.0 μCi/gm for more than 48 hours.

The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.

RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

<u>LCO</u> - The iodine specific activity in the reactor coolant is limited to 1.0 μCi/gm DOSE EQUIVALENT I-131, and the noble gas specific activity in the reactor coolant is limited to 234 μCi/gm DOSE EQUIVALENT XE-133. The limits on specific activity ensure that offsite and control room doses will meet the appropriate SRP acceptance criteria (Ref. 2).

The SLB and SGTR accident analyses (Refs. 3 and 4) show that the calculated doses are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of a SLB or SGTR, lead to doses that exceed the SRP acceptance criteria (Ref. 2).

<u>APPLICABILITY</u> - In REACTOR OPERATION conditions where Tavg exceeds 200°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 is necessary to limit the potential consequences of a SLB or SGTR to within the SRP acceptance criteria (Ref. 2).

In COLD SHUTDOWN and REFUELING SHUTDOWN the steam generators are not being used for decay heat removal, the RCS and steam generators are depressurized, and primary to secondary leakage is minimal. Therefore, the monitoring of RCS specific activity is not required.

ACTIONS

3.1.D.1.a

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the specific activity is $\leq$ 10.0 μCi/gm. The completion time of 4 hours is required to obtain and analyze a sample. Sampling is continued every 4 hours to provide a trend.

3.1.D.1.b

The DOSE EQUIVALENT I-131 must be restored to within limit within 48 hours. The completion time of 48 hours is acceptable since it is expected that, if there were an iodine spike, the normal coolant iodine concentration would be restored within this time period. Also, there is a low probability of a SLB or SGTR occurring during this time period.

A unit startup and/or continued plant operation is permitted relying on required actions 3.1.D.1.a and b while the DOSE EQUIVALENT I-131 LCO limit is not met. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient-specific activity excursions while the plant remains at, or proceeds to, POWER OPERATION.

3.1.D.1.c

If the required action of Condition 3.1.D.1.a or 3.1.D.1.b is not met, or if the DOSE EQUIVALENT I-131 is > 10.0 μCi/gm, the reactor must be brought to HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours. The required completion times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

3.1.D.2.a

With the DOSE EQUIVALENT XE-133 greater than the LCO limit, DOSE EQUIVALENT XE-133 must be restored to within the limit within 48 hours. The completion time of 48 hours is acceptable since it is expected that, if there were a noble gas spike, the normal coolant noble gas concentration would be restored within this time period. Also, there is a low probability of a SLB or SGTR occurring during this time period.

A unit startup and/or continued plant operation is permitted relying on required action 3.1.D.2.a while the DOSE EQUIVALENT XE-133 LCO limit is not met. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient-specific activity excursions while the plant remains at, or proceeds to, POWER OPERATION.

3.1.D.2.b

If the required action or associated Allowed Outage Time of Condition 3.1.D.2.a is not met, or if the DOSE EQUIVALENT XE-133 is > 234 μCi/gm, the reactor must be brought to HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours. The required action and completion time are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

REFERENCES

1. 10 CFR 50.67

2. Standard Review Plan (SRP) Section 15.0.1 "Radiological Consequence Analyses Using Alternative Source Terms"

3. UFSAR, Section 14.3.1 Steam Generator Tube Rupture

4. UFSAR, Section 14.3.2 Steam Line Break

3. With three auxiliary feedwater pumps inoperable, immediately initiate action to restore one inoperable pump to OPERABLE status. Specification 3.0.1 and all other required actions directing mode changes are suspended until one inoperable pump is restored to OPERABLE status.

G. The following actions shall be taken with inoperability of a component or instrumentation other than the flow instrumentation in one or both redundant auxiliary feedwater flowpaths required by Specification 3.6.C.3 on the affected unit: (See Specification 3.7 and TS Table 3.7-6 for auxiliary feedwater flow instrumentation requirements.)

1. With component or instrumentation inoperability in one redundant flowpath, restore the inoperable component or instrumentation to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 6 hours and be less than 350°F and 450 psig within the following 12 hours.

2. With component or instrumentation inoperability affecting both redundant flowpaths, immediately initiate action to restore the inoperable component or instrumentation in one flowpath to OPERABLE status. Specification 3.0.1 and all other required actions directing mode changes are suspended until the inoperable component or instrumentation in one flowpath is restored to OPERABLE status.

H. The specific activity of the secondary coolant system shall be $\leq$ 0.10 μCi/gm DOSE EQUIVALENT I-131. If the specific activity of the secondary coolant system exceeds 0.10 μCi/gm DOSE EQUIVALENT I-131, the reactor shall be placed in HOT SHUTDOWN within 6 hours after detection and in COLD SHUTDOWN within the following 30 hours.

RCS Flow

This surveillance requirement in Table 4.1-2A is modified by a note that allows entry into POWER OPERATION, without having performed the surveillance, and placement of the unit in the best condition for performing the surveillance. The note states that the surveillance requirement is not required to be performed until 7 days after reaching a THERMAL POWER of $\geq$ 90% of RATED POWER (i.e., shall be performed within 7 days after reaching 90% of RATED POWER). [Reference: NRC Safety Evaluation for License Amendments 270/269, issued October 19, 2010] The 7 day period after reaching 90% of RATED POWER is reasonable to establish stable operating conditions, install the test equipment, perform the test, and analyze the results. If reactor power is reduced below 90% of RATED POWER before completion of the RCS flow surveillance, the 7 day period shall be exited, and a separate 7 day period shall be entered when the required condition of reaching 90% of RATED POWER is subsequently achieved. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS Table 4.1-2B

Item 1 - RCS Coolant Liquid Samples

DOSE EQUIVALENT I-131 - This surveillance is performed to ensure iodine specific activity remains within the LCO limit during normal operation and following fast power changes when iodine spiking is more apt to occur. The SFCP 14 day Frequency is adequate to trend changes in the iodine activity level, considering noble gas activity is monitored every 7 days. The Frequency, between 2 and 6 hours after a power change $\geq$ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following iodine spike initiation; samples at other times would provide inaccurate results.

DOSE EQUIVALENT XE-133 - This surveillance requires performing a gamma isotopic analysis as a measure of the noble gas specific activity of the reactor coolant at least once every 7 days per the SFCP. This measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken or equivalent sampling method. This surveillance provides an indication of any increase in the noble gas specific activity.

Trending the results of this surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The SFCP 7 day Frequency considers the low probability of a gross fuel failure during this time.

Due to the inherent difficulty in detecting Kr-85 in a reactor coolant sample due to masking from radioisotopes with similar decay energies, such as F-18 and I-134, it is acceptable to include the minimum detectable activity for Kr-85 in this calculation. If a specific noble gas nuclide listed in the definition of DOSE EQUIVALENT XE-133 is not detected, it should be assumed to be present at the minimum detectable activity.

TABLE 4.1-2B
MINIMUM FREQUENCIES FOR SAMPLING TESTS

	DESCRIPTION	TEST	FREQUENCY	UFSAR SECTION REFERENCE
1.	Reactor Coolant Liquid Samples	Radio-Chemical Analysis (1)	SFCP (5)	
		Tritium Activity	SFCP (5)	9.1
		* Chemistry (CL, F & O_2)	SFCP (9)	4
		* Boron Concentration	SFCP	9.1
		DOSE EQUIVALENT I-131	SFCP (4)(7)	
		DOSE EQUIVALENT XE-133	SFCP (4)	
2.	Refueling Water Storage	Chemistry (Cl & F)	SFCP	6
3.	Boric Acid Tanks	* Boron Concentration	SFCP	9.1
4.	Chemical Additive Tank	NaOH Concentration	SFCP	6
5.	Spent Fuel Pit	* Boron Concentration	SFCP	9.5
6.	Secondary Coolant	DOSE EQUIVALENT I-131	SFCP	
7.	Stack Gas Iodine and Particulate Samples	* I-131 and particulate radioactive releases	SFCP	

* See Specification 4.1.D

SFCP - Surveillance frequencies are specified in the Surveillance Frequency Control Program.

(1) A radiochemical analysis will be made to evaluate the following corrosion products: Cr-51, Fe-59, Mn-54, Co-58, and Co-60.

(2) Deleted

(3) Deleted

(4) Whenever T_{avg} (average RCS temperature) exceeds 200°F.

(5) When reactor is critical and average primary coolant temperature $\geq$ 350°F.

(6) Deleted

(7) One sample between 2 and 6 hours following a THERMAL POWER change $\geq$ 15 percent of RATED POWER within a one hour period.

(8) Deleted.

(9) Sampling for chloride and fluoride concentrations is not required when fuel is removed from the reactor vessel and the reactor coolant inventory is drained below the reactor vessel flange, whether the upper internals and/or the vessel head are in place or not. Sampling for oxygen concentration is not required when the reactor coolant temperature is below 250 degrees F.

b. Deleted

3. Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after T_{avg} exceeds 200°F following completion of an inspection performed in accordance with the Specification 6.4.Q, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each degradation mechanism,

f. The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator.

g. The results of condition monitoring, including the results of tube pulls and in-situ testing,

h. The primary to secondary LEAKAGE rate observed in each SG (if it is not practical to assign the LEAKAGE to an individual SG, the entire primary to secondary LEAKAGE should be conservatively assumed to be from one SG) during the cycle preceding the inspection which is the subject of the report,

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 295 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-32

AND

AMENDMENT NO. 295 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-280 AND 50-281

1.0 INTRODUCTION

By letter dated March 2, 2018 (Reference 1), as supplemented by letter dated October 25, 2018 (Reference 2), Virginia Electric and Power Company (the licensee) submitted a license amendment request (LAR) for the Surry Power Station (Surry), Unit Nos. 1 and 2, Technical Specifications (TSs). The amendments would revise the Surry TSs consistent with Revision 0 to Technical Specifications Task Force (TSTF) Traveler, TSTF-490, "Deletion of E Bar Definition and Revision to RCS [Reactor Coolant System] Specific Activity Tech Spec," dated September 13, 2005 (Reference 3). The amendments would adopt TSTF-490 and make associated changes, which would include replacing the current limits on primary coolant gross specific activity based on E-Bar ($\bar{E}$) average disintegration energy, with new limits on primary coolant noble gas specific activity. TSTF-490 changes inputs and assumptions in the main steam line break (MSLB) and the steam generator tube rupture (SGTR) accident analyses, which examine radiological consequences using a source term based on the release of primary coolant activity at maximum TS limits. The amendments also update the Alternative Source Term (AST) analyses bases for new codes, revised atmospheric dispersion factors, new fuel handling accident (FHA) fuel rod gap fractions and control room isolation operator action time, and elimination of the locked rotor accident dose consequences.

The Technical Specifications Task Force submitted TSTF-490 for U.S. Nuclear Regulatory Commission (NRC or the Commission) staff review on September 13, 2005 (Reference 3). The notice of availability for TSTF-490 was published in the *Federal Register* on March 19, 2007, page 12838, signifying NRC approval of TSTF-490. This TSTF involves changes to NUREG-1430 (Reference 4), NUREG-1431 (Reference 5), and NUREG-1432 (Reference 6), "Standard Technical Specifications," Section 3.4.16, "RCS Specific Activity," RCS gross specific activity limits with the addition of a new limit for noble gas specific activity. The noble gas

specific activity limit would be based on a new dose equivalent XE-133 (DEX) definition that replaces the current Ē average disintegration energy definition. In addition, the current dose equivalent I-131 (DEI) definition would be revised to allow the use of committed effective dose equivalent (CEDE) dose conversion factors (DCFs) from Table 2.1 of EPA Federal Guidance Report No. 11.

The licensee's supplement dated October 25, 2018, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on June 19, 2018 (83 FR 28465).

2.0 REGULATORY EVALUATION

The licensee's application includes a request made pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67, which provides a mechanism for licensed power reactors licensed to operate prior to January 10, 1997, to replace the traditional source term used in its radiological consequence analyses of design-basis accidents (DBAs). The current DBA radiological consequence analyses are based on the AST that was approved in Amendment Nos. 230 and 230 for Surry, Units Nos. 1 and 2 (Reference 9).

The NRC staff evaluated the licensee's analysis of the radiological consequences of the affected DBAs for implementation of the AST methodology, and the associated changes to the TS proposed by the licensee, against the radiological dose requirements specified in 10 CFR 50.67(b)(2), the dose limits specified in General Design Criterion (GDC) 19 in 10 CFR Part 50, Appendix A, the Surry Updated Final Safety Analysis Report (UFSAR; Chapter 14) (Agencywide Documents Access and Management (ADAMS) Accession Nos. ML1958A584 and ML1958A583), and the accident-specific guideline values in Regulatory Position 4.4 of Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000 (Reference 8), and Table 1 of the Standard Review Plan (SRP), Section 15.0.1. The licensee has not proposed any significant deviation or departure from the guidance provided in RG 1.183.

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

- 10 CFR 50.36(a)(1), which requires each applicant for a license to include in the application proposed TSs in accordance with the requirements of 10 CFR 50.36 and include a "summary statement of the bases or reasons for such specifications, other than those covering administrative controls." However, per 10 CFR 50.36(a)(1), these TS bases "shall not become part of the technical specifications."

- 10 CFR 50.36(b), which requires that each license authorizing reactor operation include TSs derived from the analyses and evaluation included in the safety analysis report and amendments thereto.

- 10 CFR 50.36(c), which requires that TS include certain items. Per 10 CFR 50.36(c)(2)(i), the TSs must include limiting conditions for operation (LCOs), which are the lowest functional capability or performance levels of equipment required for safe operation of the facility. That provision also requires that, when an LCO of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the condition can be met.

- 10 CFR 50.36(c)(3), which requires that TSs include surveillance requirements (SRs), which are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met.

- 10 CFR Part 50.67, which states that any licensee that was initially authorized to operate prior to January 10, 1997, and who seeks to revise its current accident source term in its design-basis radiological consequence analyses, must apply for a license amendment under 10 CFR 50.90. The regulation in 10 CFR 50.67(b)(2) states that 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 [Sievert] (25 rem) [roentgen equivalent man] total effective dose equivalent (TEDE).

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

    (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.

- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," GDC 19, "Control Room," which states that:

    Holders of operating licenses using an alternative source term under 10 CFR 50.67, shall meet the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in [10 CFR] 50.2 for the duration of the accident.

- RG 1.183, which provides guidance to licensees on performing evaluations and re-analyses in support of the implementation of an alternative source term.

- NUREG-0800, SRP Section 15.0.1, "Radiological Consequences of Analyses Using Alternative Source Terms," dated July 2000, which provides guidance for an application for the initial implementation of an alternative source term at operating power reactors and subsequent LARs from these plants.

- RG 1.23, "Onsite Meteorological Programs" (Reference 11), which provides guidance on the measurement and processing of onsite meteorological data for use as input to atmospheric dispersion models in support of plant licensing and operation.

- RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants" (Reference 12), which provides guidance on appropriate dispersion models for estimating offsite χ/Q values as a function of downwind direction and distance (i.e., at the exclusion area boundary (EAB) and outer boundary of the low population zone) for various short-term time periods (up to 30 days) after an accident.

- RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants" (Reference 13), which discusses acceptable approaches for estimating short term (i.e., 2 hours to 30 days after an accident) average χ/Q values near the buildings at control room ventilation air intakes and at other locations of significant airborne air in-leakage to the control room envelope caused by postulated design-basis accident (DBA) radiological releases.

- The NRC staff's guidance for review of TSs in Section 16, "Technical Specifications," of NUREG-0800, Revision 3, dated March 2010.

3.0 TECHNICAL EVALUATION

3.1 Radiation Protection and Consequences Review

3.1.1 TSTF-490 Background

The NRC staff evaluated the impact of the proposed TSTF-490 changes as they relate to the radiological consequences of affected DBAs that use the RCS inventory as the source term. The source term assumed in radiological analyses should be based on the activity associated with the projected fuel damage or the maximum RCS TS values, whichever maximizes the radiological consequences. The limits on RCS specific activity ensure that the offsite doses are appropriately limited for accidents that are based on releases from the RCS with no significant amount of fuel damage.

The SGTR accident and the MSLB accident typically do not result in fuel damage, and therefore, the radiological consequence analyses are generally based on the release of primary coolant activity at maximum TS limits. For accidents that result in fuel damage, the additional dose contribution from the initial activity in the RCS is not normally evaluated, and it is considered to be insignificant in relation to the dose consequence resulting from the release of fission products from the damaged fuel.

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, a separate limit is specified for the

iodine isotopes. This limit is specified in units of DE I-131, which is the normalized quantity of iodine 131 that would result in the same dose consequence as the combination of the major isotopes of iodine present in the primary coolant. The TS for DE I-131 includes both an equilibrium long-term limit, as well as a higher maximum allowable short-term limit to account for iodine spiking.

The current Surry TS 1.0.M definition of DE I-131 is based on thyroid DCFs. The numerical determination of DEI is dependent on the relative quantities of the isotopes of iodine present in the RCS and on the DCFs used in the calculation. The TS definition of DEI lists the acceptable source for the thyroid DCFs to be used in the determination of DE I-131. The DCFs used in the determination of DEI are consistent with the DCFs used in the dose consequence analyses.

For plants implementing an AST methodology pursuant to 10 CFR 50.67, thyroid and whole body doses are not reported. Instead, doses are reported as TEDE. TEDE is defined as the summation of the CEDE from inhalation and the deep dose equivalent from external exposure. RG 8.40, "Methods for Measuring Effective Dose Equivalent from External Exposure," dated July 31, 2010 (Reference 14), states that licensees are encouraged to use the effective dose equivalent (EDE) in place of the deep dose equivalent in situations in which doses are calculated rather than measured with personnel dosimetry. Therefore, in dose consequence analyses using the AST, the appropriate definition for TEDE would be the summation of the CEDE and the EDE. The EDE is equivalent to the whole body dose that is calculated for plants using Technical Information Document (TID) 14844, U.S. Atomic Energy Commission, "Calculation of Distance Factors for Power and Test Reactor Sites," dated March 23, 1962 (Reference 15), in their dose consequence analysis. RG 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors" (Reference 16), subsection 4.1, assumption 4.1.4, states that whole body doses should be calculated assuming submergence in a semi-infinite cloud with appropriate credit for attenuation by body tissue. Table III.1 of the U.S. Environmental Protection Agency (EPA) "Federal Guidance Report [FGR] No. 12. External Exposure to Radionuclides in Air, Water, and Soil" (Reference 17), provides external DCFs acceptable to the NRC staff. The factors in the column headed "effective" yield doses correspond to the whole body dose. The use of effective DCFs as a surrogate for whole body DCFs is appropriate because of the uniform body exposure associated with semi-infinite cloud dose modeling.

It is appropriate for those plants using the AST methodology to use a definition of DEI based on the CEDE DCFs instead of thyroid DCFs. Licensees converting to the AST have typically included revisions to their TS definition of DEI with reference to the inhalation DCFs from U.S. EPA "Federal Guidance Report No. 11: Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion." During Surry's conversion to the AST, the licensee did not revise its TS definition of DEI to reference the inhalation DCFs from Table 2.1 of FGR No. 11.

A second limit is used to govern the non-iodine radioisotopes in the RCS. This limit 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 $\bar{E}$. The Surry TSs define $\bar{E}$ as the average sum of the beta and gamma energies, in mega electron volts, per disintegration for isotopes other than iodines with half-lives greater than 15 minutes. The RCS non-iodine specific activity limit is then expressed as the quantity 100 divided by $\bar{E}$ in units of micro curies per cubic centimeter (μCi/cc). 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. Operating experience has

indicated that depending on the isotopes used to calculate $\bar{E}$ and the actual degree of fuel clad defects, the routinely calculated value of $\bar{E}$ may not be an effective indicator of the level of noble gas activity relative to the levels used in the DBA dose consequence analyses on which the limit is based.

3.1.2 Technical Evaluation of TSTF-490 TS Changes

3.1.2.1 Revision to Definition of DE I-131

NUREG-1431 and RG 1.183 list acceptable DCFs for use in the determination of DEI, which include the following:

- Table III of TID-14844, Atomic Energy Commission, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites."

- Table E-7 of Regulatory Guide 1.109, Revision 1, NRC, 1977 (Reference 18).

- International Commission of Radiological Protection (ICRP) 30, 1979, page 192-212, table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity" (Reference 19).

- Committed dose equivalent (CDE) or CEDE dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11 (Reference 20).

- Table 2.1 of EPA FGR No. 11, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (Reference 20).

Surry proposes to use the CEDE DCFs from Table 2.1 of EPA FGR No. 11 to determine DE I-131 such that Surry TS 1.0.M, "DOSE EQUIVALENT I-131," will state:

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

The update to the definition of DEI results in an LCO that more closely relates the iodine RCS activity limits to the dose consequence analyses that form their bases; therefore, the NRC staff finds this change is acceptable from a radiological dose perspective.

3.1.2.2 Deletion of TS 3.1.D.1, Definition of $\bar{E}$, and Addition of New TS 3.1.D.2 and Definition for DEX

The licensee proposes to delete Surry TS 3.1.D.1 and the Definition of $\bar{E}$, which is in Surry TS 3.1.D.1:

> The total specific activity of the reactor coolant due to nuclides with half-lives of more than 15 minutes shall not exceed 100/$\bar{E}$ μCi/cc whenever the reactor is

critical or the average temperature is greater than 500°F, where Ē is the average sum of the beta and gamma energies, in Mev, per disintegration. If this limit is not satisfied, the reactor shall be shut down and cooled to 500°F or less within 6 hours after detection. Should this limit be exceeded by 25%, the reactor shall be made subcritical and cooled to 500°F or less within 2 hours after detection.

The licensee proposes to add new TS 3.1.D.2 and, as TS 1.0.Y, the following definition for DEX:

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."

The new TS 3.1.D.2 states:

2. The specific activity of the primary coolant shall be limited to ≤ 234 μCi/gm DOSE EQUIVALENT XE-133.

   a. With the specific activity of the primary coolant >234 μCi/gm DOSE EQUIVALENT XE-133, unit startup or POWER OPERATION may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT XE-133 to ≤ 234 μCi/gm limit.

   b. With the specific activity of the primary coolant > 234 μCi/gm DOSE EQUIVALENT XE-133 for more than 48 hours during one continuous time interval, place the reactor in HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours.

The new definition for DEX is similar to the definition for DEI. The determination of DEX will be performed in a similar manner to that currently used in determining DEI, except that the calculation of DEX 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, 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 DEX would use the effective DCFs from Table III.1 of EPA FGR No. 12 (Reference 17). 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 DEX.

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 is 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. It also results in a TS limit that can vary during operation as different values for Ē are determined.

The licensee's proposed change implements 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. Surry TS LCO 3.1.D.1 specifies the limit for total specific activity of the RCS as 100/Ē μCi per gram. The licensee's current Ē definition includes radioisotopes that decay by the emission of both gamma and beta radiation. The action currently required in TS LCO 3.1.D.1 would rarely, if ever, be entered for exceeding 100/Ē, 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.

Because the new LCO will more closely relate the non-iodine RCS activity limits to the dose consequence analyses, which form their bases, the NRC staff concludes that the licensee's proposed deletion of the above-stated TS 3.1.D.1, including the definition for Ē and addition of a new definition for DEX in Surry's TS Section 1.0 and TS 3.1.D.2, is acceptable from a radiological dose perspective.

The DEX limit is site-specific, and the licensee's proposed change would add a limit of 234 μCi/gram (gm) DEX to TS 3.1.D.2. The site-specific limit of 234 μCi/gm DEX is based on the maximum accident analysis RCS activity derived from 1 percent fuel clad defects with sufficient margin to accommodate the exclusion of those isotopes based on low concentration, short half-life, or small DCFs. The primary purpose of TS 3.1.D.2 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 Ē definition.

The NRC staff finds that the proposed new TS 3.1.D.2.a requirement to restore DEX to within the limit within 48 hours is consistent with the completion time in current TS 3.1.D.3 for DEI. The analyses of radiological consequences for postulated SGTR and MSLB accidents demonstrate that the calculated thyroid doses are generally a greater percentage of the applicable acceptance criteria than the calculated whole body doses. Because X-133, a noble gas, contributes to the calculated whole body doses, which are a lesser percentage of the applicable acceptance criteria, the completion time for noble gas activity being out of specification in TS 3.1.D.2.a should be comparable to the completion time for iodine specific activity (which is the thyroid contributor) to be restored within specification per current TS 3.1.D.3. Therefore, the NRC staff finds that the 48-hour completion time in revised TS 3.1.D.2.a is acceptable from a radiological dose perspective.

The NRC staff finds that the proposed new TS 3.1.D.2.b requirement to place the reactor in HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours when DEX exceeds the limit for more than 48 hours is appropriate because the actions require exiting the TS mode of applicability. When RCS temperature is ≤ 200°F, the release of radioactivity in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves. In addition, the completion time of 6 hours to reach HOT SHUTDOWN and 36 hours to reach COLD SHUTDOWN is reasonable, based on operating experience, to reach HOT and COLD SHUTDOWN from full power

conditions in an orderly manner and without challenging plant systems. Therefore, the NRC staff finds new TS 3.1.D.2.b is acceptable.

The licensee proposes to include an allowance that permits unit startup or POWER OPERATION to continue, while relying on the required actions. The NRC staff finds that this allowance is acceptable due to the conservatism incorporated into the DEX specific activity limit; the low probability of an event, which 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. Therefore, the NRC staff finds the proposed change to include a DEX limit and its associated actions to be acceptable.

Proposed new TS 3.1.D.2 would also use μCi/gm instead of μCi/cc. Surveillance tests and facility instrumentation used to measure or detect the TS leakage rate limits are based on cooled liquid (liquid at standard temperature and pressure) at a density of 1.0 gm/cc, and μCi/cc is equivalent to μCi/gm for cooled liquid. The NRC staff finds that this change in units of measure does not change the TS requirements and is acceptable.

3.1.2.3 Revision of TS 3.1.D, "Maximum Reactor Coolant Activity"

The licensee proposes to retitle TS 3.1.D as "RCS Specific Activity," which is consistent with NUREG-1431 (Reference 5) and TSTF-490 (Reference 3). The word "maximum" is not needed in the title to convey that TS 3.1.D sets limits on reactor coolant activity. Because no technical requirements are altered and removal of the word maximum is an editorial change, the NRC staff finds this change acceptable.

In addition, TS 3.1.D would be reformatted. Currently, TS 3.1.D is formatted such that it includes one section labeled "Specifications." The new format would have two sections, an "Applicability" section and a "Specifications" section. This proposed change is also an editorial change because it does not change any requirements, it only restructures the TS so that it is easier to read, and therefore, provides clarity. Therefore, the NRC staff concludes that these changes are acceptable.

3.1.2.4 TS 3.1.D Applicability

The licensee proposes to modify TS 3.1.D to apply when average RCS temperatures are greater than 200 degrees Fahrenheit (°F). Currently, TS 3.1.D is applicable when the reactor is critical or the average temperature of the reactor coolant is greater than 500 °F. The NRC staff finds that to limit the potential radiological consequences of an SGTR or MSLB that may occur while temperature is greater than 200 °F, it is necessary for the TS to apply when average RCS temperature is greater than 200 °F. When average RCS temperature is less than 200 °F with the RCS loops filled, the steam generators (SGs) are operated as a backup means of decay heat removal by natural circulation and the probability of a DBA involving the release of significant quantities of RCS inventory is reduced. Therefore, monitoring of RCS specific activity is not required. With temperature less than 200 °F with the RCS loops not filled and during 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.

Based on the information above, the NRC staff finds that modifying TS 3.1.D, "Applicability," to include whenever average RCS temperature exceeds 200 °F is necessary to limit the potential

radiological consequences of an SGTR or MSLB that may occur while temperature is greater than 200 °F. Therefore, the change is acceptable from a radiological dose perspective.

3.1.2.5 Replacement of TS 3.1.D.2 and TS 3.1.D.3

The current TS 3.1.D.2 and TS 3.1.D.3 state:

2. The specific activity of the reactor coolant shall be limited to ≤ 1.0 μCi/cc DOSE EQUIVALENT I-131 whenever the reactor is critical or the average temperature is greater than 500°F.

3. The requirements of D-2 above may be modified to allow the specific activity of the reactor coolant > 1.0 μCi/cc DOSE EQUIVALENT I-131 but less than 10.0 μCi/cc DOSE EQUIVALENT I-131. Following shutdown, the unit may be restarted and/or operation may continue for up to 48 hours provided that operation under these circumstances shall not exceed 10 percent of the unit's total yearly operating time. With the specific activity of the reactor coolant > 1.0 μCi/cc DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding 10.0 μCi/cc DOSE EQUIVALENT I-131, the reactor shall be shut down and cooled to 500°F or less within 6 hours after detection.

The licensee proposes to replace TS 3.1.D.2 and TS 3.1.D.3 with new TS 3.1.D.1, which states:

1. The specific activity of the primary coolant shall be limited to ≤ 1 μCi/gm DOSE EQUIVALENT I-131.

   a. With the specific activity of the primary coolant > 1 μCi/gm DOSE EQUIVALENT I-131, verify DOSE EQUIVALENT I-131 ≤ 10 μCi/gm once per 4 hours.

   b. With the specific activity of the primary coolant > 1 μCi/gm DOSE EQUIVALENT I-131, but ≤ 10 μCi/gm, unit startup or POWER OPERATION may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT I-131 to ≤ 1 μCi/gm limit.

   c. With the specific activity of the primary coolant > 1 μCi/gm DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or DOSE EQUIVALENT I-131 is > 10 μCi/gm, place the reactor in HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours.

The proposed changes remove text that limits the applicability of TS 3.1.D.2 to "whenever the reactor is critical or the average temperature is greater than 500ºF"; changes units of measure from μCi/cc to μCi/gm in both D.2 and D.3; and reformats the information in TS 3.1.D.3, adding a 4-hour sampling requirement.

The revised applicability makes TS 3.1.D.2 more restrictive because it expands the average RCS temperature range from greater than 500 °F to greater than 200 °F. As noted in SE section 3.1.2.4, above, the this change necessary to limit the potential radiological

consequences of an SGTR or MSLB that may occur while temperature is greater than 200 °F. Therefore, the staff finds this change in the TS applicability acceptable.

TS 3.1.D.3 is also revised to require verification that DEI is less than or equal to 10 µCi/gm once per 4 hours. With DEI greater than the 1 µCi/gm limit, samples at intervals of 4 hours must be taken to demonstrate that the specific activity is less than or equal to 10 µCi/gm. The LAR indicates that a completion time of 4 hours is required to obtain and analyze a sample and that the revised TS requires sampling to continue every 4 hours to provide a trend. The NRC staff finds that a completion time of 4 hours is a reasonable amount of time to obtain and analyze a sample and that trending is an important consideration that informs the licensee's course of action for DEI. Therefore, these changes are acceptable.

The radiological dose consequence analyses for SGTR and MSLB accidents take into account the pre-accident iodine spike analyses, which assume a DEI concentration (10 µCi/gm) 10 times higher than the corresponding long-term equilibrium value, and it corresponds to the specific activity limit associated with 100 percent rated thermal power. Therefore, the NRC staff finds that it is acceptable that TS 3.1.D.1.a and TS 3.1.D.1.b are based on the short-term, site-specific DEI spiking limit, consistent with the assumptions contained in the radiological consequence analyses.

The changes in the DEI limits from 1.0 µCi/cc to 1 µCi/gm and from 10.0 µCi/cc to 10 µCi/gm do not alter substantive requirements. Therefore, the NRC staff finds these changes acceptable.

The current TS 3.1.D.3 requires reactor shutdown and cooldown to 500 °F or less within 6 hours after detection of exceeding a DEI limit. The proposed new TS 3.1.D.1.c requires, when the DEI limit is exceeded for over 48 hours, the reactor to be in HOT SHUTDOWN (i.e., reactor subcritical with average coolant temperature greater than or equal to 547 °F) within 6 hours and COLD SHUTDOWN (i.e., reactor subcritical with average coolant temperature less than or equal to 200 °F) within the following 30 hours. Thus, the reactor is required to be subcritical and cooled to less than or equal to 547 °F (instead of 500 °F) within in 6 hours and an additional 30 hours is allowed to reach 200 °F. When DEI exceeds the limits stated in the TS, the radiological consequences of the SGTR or MSLB accident analyses do not bound the resultant radiation doses at the onsite and offsite boundaries. Because these accidents are possible when the reactor coolant temperature exceeds 200 °F, and the resultant radiation doses are not bounded by the postulated design-basis SGTR and MSLB accidents when DEI exceeds the limits, the NRC staff finds that reactor cooldown to less than or equal to 200 °F is necessary to limit the potential radiological consequences of an SGTR or MSLB accident. In addition, the staff considers it reasonable to place the reactor in HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours. Therefore, NRC staff finds the changes acceptable.

Proposed new TS 3.1.D.1 also replaces the µCi/cc unit with µCi/gm. Surveillance tests and facility instrumentation used to show compliance with the leakage rates in TSs are based on cooled liquid and a density of 1.0 gm/cc. Therefore, µCi/cc is equivalent to µCi/gm for cooled liquid. The NRC staff finds that this change does not change the TS requirements and the change is acceptable.

Proposed new TS 3.1.D.1 also reformats information previously in TS 3.1.D.3. Because the reformatting is an editorial change that restructures the TS to make it easier to read and use, the NRC staff finds this change acceptable.

The NRC staff finds that new TS (1) is based on specific DEI spiking limits, which are consistent with the assumptions contained in the radiological consequence analyses for the MSLB and SGTR, and the resultant radiological doses for these accidents remain below the accident dose criteria in RG 1.183 and the dose limits in 10 CFR 50.67, (2) includes changes to DEI units and time periods to reach shutdown that are appropriate, and (3) reformats information to make it easier to comprehend and use the TS. Therefore, the staff finds that new TS 3.1.D.1 is acceptable.

3.1.2.6 TS 3.1.D.3, TS 3.1.D.4, and TS 6.6.A.2.b

The licensee proposes to delete TS 3.1.D.3, TS 3.1.D.4, and TS 6.6.A.2.b. TS 3.1.D.3 is stated above in SE Section 3.1.2.5. TS 3.1.D.4 states:

4. If the specific activity of the reactor coolant exceeds 1.0 μCi/cc DOSE EQUIVALENT I-131 or 100/$\bar{E}$ μCi/cc, a report shall be prepared and submitted to the Commission pursuant to Specification 6.6.A.2. This report shall contain the results of the specific activity analysis together with the following information:

    a. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded.

    b. Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded.

    c. History of degassing operations, if any, starting 48 hours prior to the first sample in which the limit was exceeded.

    d. The time duration when the specific activity of the primary coolant exceeded 1.0 μCi/cc DOSE EQUIVALENT I-131.

    e. Results of the last isotopic analysis for radio-iodine performed prior to exceeding the limit, results of analysis while the limit was exceeded, and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations, and

    f. Graph of the I-131 concentration and one other radio-iodine isotope concentration in μCi/cc as a function of time for the duration of the specific activity above the steady-state level.

When the primary coolant exceeds the limits stated (above) in Specification 3.1.D.4, current TS 6.6.A.2.b requires the results of the specific activity analysis along with the information itemized (above) in Specification 3.1.D.4 to be submitted to the administrator of the appropriate NRC regional office.

TS 3.1.D.3 permits startup and continued plant operation with DEI above its limit; however, TS 3.1.D.3 would be deleted since this requirement would be addressed in new TS 3.1.D.1.b. Likewise, the existing TS 3.1.D.3 requirement for unit shutdown and cooldown to 500 °F or less within 6 hours after detection when DEI >1 μCi/cc for more than 48 continuous hours or >10 μCi/cc would be included in the proposed TS 3.1.D.1.c, but expressed in different, but

equivalent, units of measure and removing ".0" after the value. The NRC staff reviewed the proposed deletion of TS 3.1.D.3 and determined that the requirements are being relocated to TS 3.1.D.1.b and TS 3.1.D.1.c. In addition, TS 3.1.D.1.c is more restrictive in that it requires a unit shutdown and cooldown to HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours. The NRC staff finds that reactor shutdown and cooldown to less than or equal to 200 °F is necessary to limit the potential radiological consequences of an SGTR or MSLB accident that may occur while temperature is greater than 200 °F, consistent with the UFSAR. In addition, it is reasonable to place the reactor in HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours. Therefore, the NRC staff finds the proposed deletion of TS 3.1.D.3 is acceptable.

The purpose of TS 3.1.D.4 and TS 6.6.A.2.b is to inform the NRC of specific activity above the TS limits. If the specific activity of the reactor coolant exceeds 1.0 µCi/cc dose equivalent I-131 or 100/Ē µCi/cc, TS 3.1.D.4 requires a report to be prepared and submitted to the NRC pursuant to TS 6.6.A.2.

If DEX exceeds 234 µCi/gm for more than 48 hours, new TS 3.1.D.2.b requires the reactor to be placed in HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours. The regulation, 10 CFR 50.72(b)(2), requires the licensee to notify the NRC as soon as practical, and in all cases, within 4 hours of the occurrence of an initiation of any nuclear plant shutdown required by the plant's TSs. In addition, 10 CFR 50.73(2)(i)(A) requires the licensee to report the completion of any nuclear plant shutdown required by the plant's TSs. Therefore, these regulations requires the licensee to notify the NRC staff if a reactor shutdown is required due to exceeding Surry's TS DEI or DEX limits.

The NRC staff concludes that the proposed deletion of TS 3.1.D.4 and TS 6.6.A.2.b is acceptable because TS 3.1.D.4 and TS 6.6.A.2.b do not contain any initial conditions, inputs, assumptions, or methodologies used in the Surry radiological consequence analyses for DBAs, and therefore, deletion of the TSs will not affect Surry's radiological consequence analyses. In addition, the NRC staff finds the proposed deletions acceptable because the regulations in 10 CFR 50.72(b)(2) already requires the licensee to inform the NRC within 4 hours of the initiation of a plant shutdown, submittal of a licensee event report per 10 CFR 50.73(a)(2)(i)(A), and the reporting to the NRC of the completion of a plant shutdown required by the actions in TS 3.1.D.

3.1.2.7 TS Table 4.1-2B Surveillance Tests

The licensee proposed the following changes to TS Table 4.1-2B, "Minimum Frequencies for Sampling Tests," Item 1, "Reactor Coolant Liquid Samples:"

- Delete the Gross Activity, Ē Determination, and Radio-Iodine Analysis tests and the associated Notes 2, 3, and 6, respectively, and add a new DEX surveillance requirement.

- Replace Note 5.

- Revise Note 7.

The proposed changes would delete the current gross activity, Ē determination, and radio-iodine analysis tests and replace them with a new surveillance to verify that the site-specific reactor coolant DEX specific activity is less than the new TS limit added to TS 3.1.D for DEX. Because

the surveillance tests at Surry seemed to differ from those evaluated in TSTF-490, the NRC staff requested that the licensee provide a comparison of the Surry SRs to those in TSTF-490, explaining any differences. The licensee responded to the NRC staff's request by letter dated October 25, 2018.

For the gross activity, the licensee stated that the SR of total specific activity and gross degassed activity at Surry is being performed for the same purpose and in only a slightly different manner than those in TSTF-490 for Westinghouse plants. The measurement of gross activity in both cases is directly tied to $\bar{E}$ determination, and the requirement for performing $\bar{E}$ determination is being removed from the TSs for the reasons noted in TSTF-490, which includes the implementation of the DEX SR. The gross activity sample test is no longer required and can be eliminated, consistent with the reasoning provided in TSTF-490. For the $\bar{E}$ determination, the licensee stated that there are no differences. For the radio-iodine analysis, the licensee stated that the Surry TSs separate DEI and radio-iodine analysis as two different line items, whereas TSTF-490 for Westinghouse plants combines the requirements under DEI. This meets the requirements for radio-iodine analysis as currently stated in the Surry TSs. The requirements currently in the Surry TSs for radio-iodine analysis are incorporated into the proposed DEI SR in the proposed Surry TS revisions which is shown by relocation of Note (7) to the DEI SR in TS table 4.1-2B. Based on the above information, the NRC staff determined that the Surry SRs for DEI, DEX, and radio-iodine analysis are essentially equivalent to those in TSTF-490 and that they are consistent with the evaluation for TSTF-490. The NRC staff finds the relocation of Note (7) from the radio-iodine analysis SR to the DEI SR to be acceptable. The proposed changes to Note (7) are discussed in this section.

The Surry site-specific reactor coolant DEX limit is 234 μCi/gm. TS 3.1.D maximum reactor coolant 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 $\bar{E}$ definition. The new surveillance requires performing a gamma isotopic analysis as a measure of the noble gas specific activity of the reactor coolant at least once every 7 days per the surveillance frequency control program. The surveillance provides an indication of any increase in the noble gas specific activity. The results of the surveillance on DEX allow proper remedial actions to be taken before reaching the TS limit under normal operating conditions. The NRC staff finds that with the elimination of the limit for RCS gross specific activity and the addition of the new TS limit for noble gas specific activity, a SR to determine $\bar{E}$ is no longer required. The NRC staff also concludes that the proposed changes are consistent with the methodology of the Surry radiological consequence analyses for DBAs, and deletion of the limit for RCS gross specific activity and the $\bar{E}$ determination SR will not affect Surry's radiological consequence analyses. Therefore, the NRC staff concludes that the proposed changes are acceptable.

As noted above, the NRC staff finds the proposed revision making TS 3.1.D applicable whenever average reactor coolant system temperature exceeds 200 °F acceptable. The licensee also proposed to revise Note (5), "When reactor is critical and average primary coolant temperature ≥ 350°F," of TS Table 4.1-2B, "Minimum Frequencies for Sampling Tests," to state, "Only required when the unit is in POWER OPERATION." However, this proposed TS change does not require the SRs to be performed whenever TS 3.1.D is applicable. The proposed change would revise the conditions for sampling and would no longer require sampling during plant conditions in which TS limits could be exceeded. The proposed change does not require performance of the SR after transient conditions. Isotopic spiking and fuel failures are more likely during transient conditions than during steady state plant operations. Because TS 3.1.D could potentially be exceeded after a plant transient or power changes, in a request for

additional information dated September 25, 2018 (ADAMS Accession No. ML18268A192), the NRC staff asked the licensee to explain why sampling is no longer needed during the reactor operations that are proposed to be eliminated and how TS 3.1.D remains consistent with the design-bases analysis from which the TS limits are derived.

Because Note (5) applies to more than the proposed DEI and DEX SRs and also applies to the radio-chemical analysis and the tritium activity tests, the NRC staff also requested in its September 25, 2018 (ADAMS Accession No. ML18268A192), request for additional information that the licensee explain why the SRs for the radio-chemical analysis and the tritium activity tests should be reduced. The licensee responded by letter dated October 25, 2018, that it was no longer requesting revision of Note (5) because the intent of the revision was to permit unit startup to POWER OPERATION prior to performing the associated sampling tests. The licensee acknowledged that isotopic spiking and fuel failures are more likely during transient condition than during steady state operations. Consequently, the licensee proposed to add a new Note (4) for the DEI and DEX sampling test frequencies, which states, "Whenever T_{avg} (average RCS temperature) exceeds 200°F." Because new Note (4) is consistent with the TS 3.1.D applicability and it addresses transient conditions, the NRC staff finds Note 4 acceptable.

The current TS Table 4.1-2B Note (7) states:

> One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 percent of RATED POWER within a one hour period provided the average primary coolant temperature ≥ 350°F.

The proposed change revises Note (7) to state:

> One sample between 2 and 6 hours following a THERMAL POWER change ≥ 15 percent of RATED POWER within a one hour period.

The NRC finds that the proposed revision to Note (7) is more restrictive requirement because it adds performance of the DEI SR for a 15 percent thermal power change, in addition to power changes that exceed 15 percent thermal power, and is required at all primary coolant temperatures. The NRC staff finds this change to be consistent with the radiological consequences of the MSLB and SGTR accidents, and, therefore, is acceptable. In addition, the licensee is replacing "&" in Note (7) with the word "and." The NRC staff finds this to be an editorial change that doesn't change any TS requirements and is acceptable.

The NRC staff reviewed the proposed changes to (1) delete the gross activity, $\bar{E}$ determination, and radio-iodine analysis tests and their associated Notes 2, 3, and 6, respectively; (2) add a new Note (4) and DEX SR and (3) revise Note 7, and has concluded that they are consistent with the methodology used to analyze the radiological consequences of the MSLB and SGTR accidents and they continue to meet 10 CFR 50.36 (c)(3) by assuring that the facility operation will be within safety limits and that the LCO would be met. Therefore, the NRC staff finds these proposed changes acceptable.

3.1.2.8 Proposed Change to TS 3.6.H

The current TS 3.6.H states:

> The specific activity of the secondary coolant system shall be ≤ 0.10 μCi/cc DOSE EQUIVALENT I-131. If the specific activity of the secondary coolant

system exceeds 0.10 µCi/cc DOSE EQUIVALENT I-131, the reactor shall be shut down and cooled to 500°F or less within 6 hours after detection and in COLD SHUTDOWN within the following 30 hours.

The licensee proposes to change TS 3.6.H to:

> The specific activity of the secondary coolant system shall be ≤ 0.10 µCi/gm DOSE EQUIVALENT 1-131. If the specific activity of the secondary coolant system exceeds 0.10 µCi/gm DOSE EQUIVALENT 1-131, the reactor shall be placed in HOT SHUTDOWN within 6 hours after detection and in COLD SHUTDOWN within the following 30 hours.

The licensee proposes to replace the µCi/cc unit with that of µCi/gm in TS 3.6.H. Surveillance tests and facility instrumentation used to show compliance with the leakage rates in TSs are based on cooled liquid and a density of 1.0 gm/cc, and µCi/cc is equivalent to µCi/gm for cooled liquid. The NRC staff finds that this change does not change the TS requirements, and the change is acceptable.

The licensee also proposes to replace the "reactor shall be shut down and cooled to 500°F or less within 6 hours after detection and in COLD SHUTDOWN within the following 30 hours," with the wording, "reactor shall be placed in HOT SHUTDOWN within 6 hours after detection and in COLD SHUTDOWN within the following 30 hours." This change does not affect the overall 36-hour time period to reach COLD SHUTDOWN. The change does impact the coolant temperature and the status of the reactor 6 hours after detection. Instead of requiring (within 6 hours after detection) that the reactor be shut down with coolant temperature 500 °F or less, the proposed change requires the reactor to be in HOT SHUTDOWN within 6 hours after detection that the limit has been exceeded. TS Section 1.0 defines HOT SHUTDOWN as the reactor is subcritical by at least 1.77 percent Δk/k (reactivity) and average RCS temperature is ≥ 547 °F. The added reactivity requirement is more restrictive than the current TS allowance, and the change in average RCS temperature to ≥ 547 °F is less restrictive than the current TS allowance. Because placing the reactor in COLD SHUTDOWN (i.e., an average coolant temperature less than or equal to 200 °F) limits the potential radiological consequences of an SGTR or MSLB accident, the NRC staff finds the change is acceptable from a radiological dose perspective.

3.1.3 Technical Evaluation of Updated AST

The licensee proposed to change its offsite and control room DBA dose consequence analysis for Surry that is currently in UFSAR Chapter 14. The proposed change updates the AST methodology for determining DBAs offsite and control room doses as described in the UFSAR. The LAR proposes to:

- Revise the methodology used to analyze design basis dose consequences to include the RADTRAD-NAI code.
- Implement revised χ/Qs for the control room and offsite receptors.
- Revise the methodology used to determine the core inventory to include the ORIGEN-ARP code.

- Revise the fuel handling accident (FHA) gap fraction methodology to use gap fractions based on PNNL-18212, Rev. 1, Table 2.9, "Maximum."
- Increase the FHA time critical operator action to manually isolate the control room from 2 minutes to 12 minutes.
- Remove the dose consequences of the locked rotor accident (LRA) from the radiological design basis because fuel damage is not predicted.

To support the changes above, the licensee reanalyzed the following accidents employing the AST, as described in RG 1.183.

- Loss-of-coolant accident (LOCA)
- FHA
- SGTR
- MSLB
- LRA

The DBA dose consequence analyses evaluated the integrated TEDE at the EAB for the worst 2-hour period following the onset of the accident. The integrated TEDE 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 TEDE to a Surry control room operator were evaluated for the duration of the accident. The licensee performed the dose consequence analyses using the RADionuclide Transport and Removal and Dose Estimation-The Numerical Applications, Inc. (RADTRAD-NAI) computer code. The development of the RADTRAD radiological consequence computer code was sponsored by the NRC, as described in NUREG/CR-6604 (Reference 21), 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 performed independent dose evaluations using the Symbolic Nuclear Analysis Package (SNAP)/RADTRAD 4.0 computer code, as described in NUREG/CR-7220 (Reference 22), to confirm the acceptability of the licensee's dose estimates. The licensee's dose estimates, as well as the applicable dose acceptance guidelines from RG 1.183, are located in Section 3.1.3.6 of this safety evaluation.

Each DBA radiological source term used in the AST analyses was developed using ORIGEN-ARP. The proposed ORIGEN-ARP based core inventory bounds cycle designs with enrichments between 3.60 and 4.75 weight percent Uranium-235 (U-235), batch average burnups up to 56 gigawatt days per metric ton of uranium (GWD/MTU), and a power level, including uncertainties of 2605 megawatts thermal (MWt). The licensee developed the equilibrium core activity inventory using the ORIGEN-ARP code. The NRC staff concludes that the use of ORIGEN-ARP is consistent with regulatory guidance in RG 1.183, and therefore, is acceptable.

The licensee used CEDE and effective dose equivalent DCFs from FGR 11 and 12 to determine the TEDE dose in accordance with AST evaluations. The use of DCFs from Table 2.1 of FGR-11 and Table III.1 of FGR-12 is in accordance with RG 1.183 guidance, and is, therefore, acceptable to the NRC staff.

3.1.3.1 LOCA

The current radiological consequences for the LOCA was approved by the NRC staff in Amendment Nos. 230 for Surry, Unit Nos 1 and 2 (ADAMS Accession No. ML020710159), and was subsequently modified in Amendment Nos. 250 and 249 for Surry, Units 1 and 2, respectively (ADAMS Accession No. ML062920499). The licensee proposes to update the radiological consequences for the LOCA by (1) revising the LOCA methodology to include the RADTRAD-NAI code, (2) implementing revised χ/Qs for the control room and offsite receptors, and (3) revising the methodology used to determine the core inventory to include the ORIGEN-ARP code. The licensee's proposed changes to the LOCA analysis in UFSAR Section 14.5.5 are listed in Attachment 2, Table 2.0-1, of the LAR.

LOCA Source Term

The licensee 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 assumed that all the fuel assemblies in the core are affected. The core inventory is stated in Attachment 2, Table 3.2-3, of the LAR. The licensee did not propose any changes to the containment sump pH analysis for the LOCA analysis. The NRC staff concludes these proposed changes are consistent with RG 1.183, and therefore, are acceptable.

Assumptions on Transport in the Primary Containment

The current LOCA radiological consequence analysis assumes the containment sprays remove elemental and particulate iodine from the containment atmosphere. The only proposed change to the containment spray assumptions is to the sprayed and unsprayed volume of the containment. The licensee is increasing the number of significant digits of these volumes. The NRC staff finds that this change increases the accuracy of the assumed volumes and is acceptable.

The current LOCA analysis assumes recirculation sprays remove elemental and particulate iodine from the containment atmosphere. The licensee proposes to update only the parameters for the recirculation sprays listed in Attachment 2, Table 2.0-1, of the LAR, but all other assumptions would remain as stated in UFSAR Section 14.5.5. The licensee determined the recirculation spray elemental iodine and aerosol removal rates using the methodology that the NRC previously found acceptable in Amendment Nos. 230. The licensee did not propose any other changes to its current assumptions on the transport in the primary containment for the LOCA analysis.

Assumptions on Engineered Safety Feature System Leakage

The licensee proposes to retain the emergency core cooling system (ECCS) leakage model in its radiological consequence LOCA analysis with the exception of the following updates. The licensee proposes to increase the amount of the ECCS filtered leakage, ECCS unfiltered leakage, and refueling water storage tank back leakage in the model with no changes to the timing.

RG 1.183, Appendix A, Regulatory Position 5.2 (Reference 8), states that the engineered safety feature system leakage should be taken as two times the sum of the simultaneous leakage from

all components in the engineered safety feature recirculation systems above, which the TSs would require declaring such systems inoperable. As reflected in the UFSAR, the current LOCA analysis uses values that are twice the allowable leak rate in accordance with RG 1.183, Appendix A, Regulatory Position 5.2. The licensee proposes to increase the values to those stated in Attachment 2, Table 2.0-1, of the LAR to add more operating margin.

The NRC staff concludes that the proposed change over-estimates the amount of leakage from the ECCS following a LOCA, and that the calculated radiological dose from this pathway is higher than that which would actually occur following a LOCA. Because the proposed change conservatively estimates the amount of ECCS leakage, and because the radiological doses remain below the acceptance criteria in RG 1.183 and SRP 15.0.1, and below the limits stated in 10 CFR 50.67 and GDC 19, the NRC staff finds that the proposed changes are acceptable.

Control Room Ventilation System

The licensee proposes to retain the majority of its current control room emergency ventilation system assumptions with the exception of the following updates. The licensee proposes to decrease the control room unfiltered in-leakage rate to 250 cubic feet per minute (cfm) based on the maximum tracer gas test results of 147 +/- 6 cfm. The NRC staff finds that the control room unfiltered in-leakage rate of 250 cfm is acceptable because it exceeds the actual amount of control room unfiltered in-leakage as determined by the maximum tracer gas test, thereby making it a conservative assumption. The licensee proposes to decrease the control room emergency ventilation filtered flow rate from 1,000 cfm to 900 cfm, which the NRC staff finds to be a conservative change because it reduces air turnover in the control room and leads to a higher dose estimate than currently estimated. The NRC staff finds the changes in filtered flow rate acceptable. In addition, the control room χ/Q values are updated based on meteorological data from 2009 to 2013, which are evaluated in Section 3.2 of this safety evaluation and found acceptable.

Control Room Operator Dose During Ingress and Egress

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

The licensee evaluated the dose received by the control room operators during routine access to the control room for the 30-day period following the LOCA. The licensee assumed the following in its calculation.

The control room occupancy is modeled as 100 percent of the time during the first 24 hours of a LOCA event. Therefore, transit to and from the control room is only expected after the first 24 hours following an accident. The licensee assumes that during the 30-day event, a total of 30 roundtrips between the site boundary and the control room occur with a one-way transit time of 30 minutes. This leads to a total of 30 hours of ingress/egress transit time per individual during the event.

The containment building returns to a sub-atmospheric condition during the first 24 hours, preventing any containment airborne release contribution to ingress/egress dose. Additionally, NUREG-0737, Item 11.B.2, "Plant Shielding" (Reference 24), states that leakage of systems outside containment need not be considered as potential sources. The only remaining

contributor to personnel dose during ingress/egress is the direct radiation exposure from the sources within the containment building.

The licensee modeled direct radiation exposure from the containment using the RADTRAD radioisotope inventories in the sprayed and unsprayed regions of the containment building at 24 hours. The resulting dose rate is determined using MicroShield. The containment building is modeled as a right circular cylinder of air with a diameter of 126 feet and a height of 65.5 feet. The height of the cylinder represents the distance from ground level to the top of the cylindrical portion of the containment building. The air cylinder is surrounded by a radial concrete shield with a thickness of 4.5 feet, and the concrete shield is surrounded by air. The default MicroShield densities of 2.35 grams per cc for concrete and 0.0122 gram per cc for air is used. The radioisotope inventories are uniformly distributed within the cylinder of air representing the containment building.

The dose rate for ingress/egress is determined at a point coplanar with the bottom surface of the cylindrical source and at a radial distance of 183.6 feet from the centerline of the containment building, representing an estimate of the closest point of approach during ingress/egress. The licensee calculated that the resulting dose rate, with buildup, at the closest point of approach to the containment building is 0.3 millirem per hour, resulting in an accumulated dose of 9 millirem during 30 hours of ingress/egress transit. The NRC staff reviewed the ingress and egress analysis and finds the resulting radiological dose is acceptable because the modeling assumptions and inputs stated above are consistent with the plant design parameters, and the radiological dose remains below the radiation dose limits stated in 10 CFR 50.67.

Offsite Boundaries

The licensee proposes to increase the EAB breathing rate to 3.5E-04 m^3/second (sec). Because the increased breathing rate is a conservative change that results in a higher estimated radiation dose received, the NRC staff finds the change acceptable. In addition, the EAB and LPZ χ/Qs values are updated based on meteorological data from 2009 to 2013, which are evaluated in Section 3.2 of this safety evaluation and found acceptable.

Conclusion

The licensee evaluated the radiological consequences resulting from the postulated LOCA and concluded that the radiological consequences at the EAB, LPZ, and control room are within the radiation dose reference values provided in 10 CFR 50.67 and the accident-specific dose criteria specified in SRP Section 15.0.1 (Reference 7). Based on its review, the NRC staff concludes that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this safety evaluation. The licensee's acceptable assumptions and inputs are presented in Attachment 2, Table 2.0-1, of the LAR, and the licensee's calculated dose results are 9.1 rem at the EAB, 1.9 rem at the LPZ, and 4.4 rem in the control room, and are summarized in Table 1 in Section 3.1.3.6 of this safety evaluation. The NRC staff compared the doses estimated by the licensee to the applicable accident dose criteria of 25 rem for the EAB and LPZ as stated in RG 1.183 and SRP 15.0.1, and to the results estimated in the NRC staff's confirmatory calculations. The NRC staff performed independent dose calculations to ensure a thorough understanding of the licensee's assumptions and methods. Specifically, the NRC staff found that (a) the inputs and assumptions used by the licensee are reasonable and (b) that the licensee's dose estimates are below the regulatory limits and comparable to those calculated by the NRC staff. The NRC staff finds that the

licensee demonstrated that there is reasonable assurance that the estimates of the TEDE due to a postulated design-basis LOCA comply with the accident dose criteria in RG 1.183, SRP 15.0.1, and the radiation dose limits 10 CFR 50.67, which are 25 rem at the EAB and LPZ, and 5 rem in the control room. Therefore, the staff finds the proposed LOCA analysis changes acceptable.

3.1.3.2 FHA

The FHA involves the drop of a fuel assembly during refueling operations. The mechanical part of the licensee's analysis remains unchanged from the current licensing basis, and it assumes that the total number of failed fuel rods is one fuel assembly out of the 157 fuel assemblies in the core. The rods are assumed to instantaneously release their fission gas contents to the water surrounding the fuel assembly. The depth of water over the damaged fuel is not less than 23 feet and is controlled by TS 3.10.

The licensee analyzed an FHA in containment and an FHA in the spent fuel pool (SFP) in the fuel building. An FHA in the SFP of the fuel building would involve a release via the ventilation vent number two. An FHA in the containment involves three release pathways. The first path is through the open equipment hatch directly to the environment, the second path is through the open containment personnel hatch to the west louver, and the third path is through the containment purge exhaust. The licensee's proposed changes to the FHA analysis of record are listed in Attachment 2, Table 2.0-1, of the LAR.

FHA Source Term

The licensee used the ORIGEN-ARP code to generate the core radionuclide inventory. The core inventory is stated in Attachment 2, Table 3.2-3, of the LAR. The licensee did not propose any changes to the amount of fuel damage assumed in the FHA or the assumed fuel decay time. The amount of fuel damage is the same whether the FHA is in the SFP or containment. For the FHA analyses, the licensee proposes to use the fraction of fission product inventory in the gap from Revision 1 of PNNL-18212 as input to RADTRAD-NAI. The NRC staff's evaluation of the fraction of fission product inventory in the gap follows below.

Fission Gas Gap Fraction

The fission gas gap fraction is the fraction of the total inventory of a radionuclide that is present in the fuel rod void volume (i.e., plenum plus fuel-to-clad gap) and is thus available for release, should the cladding fail. This quantity is used as an input to the dose calculations for accidents where clad damage is assumed to occur. Table 3 of RG 1.183 provides acceptable fission gas gap fractions for non-LOCA accidents and contains a footnote that excludes rods at relatively high burnups above a specific maximum linear heat generation rate (LHGR). Specifically, Table 3, Footnote 11 of RG 1.183 (Reference 8) states:

> The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kW/ft peak rod average power for burnups exceeding 54 GWD/MTU. As an alternative, fission gas release calculations performed using U.S. Nuclear Regulatory Commission (NRC)-approved methodologies may be considered on a case-by-case basis. To be acceptable, these

> calculations must use a projected power history that will bound the limiting projected plant-specific power history for the specific fuel load.[...]

The NRC Draft Guide (DG)-1199, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (ADAMS Accession No. ML090960464), has been released for public comment and is also known as "Proposed" RG 1.183, Revision 1 (dated July 2000). This DG provides an updated fission gas gap fraction table (also Table 3), which is applicable to fuel that is operated within bounds set by Figure 1 of DG-1199. Figure 1 is a curve of peak nodal power vs. peak nodal burnup. The NRC approved this method of evaluating the fission gas gap fraction at Diablo Canyon Power Plant, "Technical Report, WECTEC Global Project Services, Inc. – Implementation of Alternative Source Terms – Summary of Dose Analyses and Results, Revision 2, Diablo Canyon Power Plant" (Reference 30). These limits are based on an analysis documented in PNNL-18212, Revision 0, "Update of Gap Release Fractions for Non-LOCA Events Utilizing the Revised ANS 5.4 Standard," produced by the NRC and Pacific Northwest National Labs (PNNL) (Reference 31).

PNNL-18212, Revision 1 (Reference 32), includes new proposed fission gas gap fractions in Table 2.9 (the "Maximum" column) and a bounding curve of rod average power to rod average burnup for pressurized water reactors (PWRs) in Figure A.1, which are both reproduced below.

	Gap Release Fractions - 95/95 UTL			
Isotope	Calculated PWR 14x14 Design	Calculated BWR 9x9 Design	Maximum	Current RG 1.183 Table 3
Kr-85	0.357	0.372	0.38	0.10
I-131	0.077	0.052	0.08	0.08
I-132	0.087	0.058	0.09	0.05
Other Nobles	0.073	0.049	0.08	0.05
Other Halogens	0.046	0.031	0.05	0.05
Alkali Metals	0.478	0.499	0.50	0.12

a. Gap fractions for non-LOCA events with exception of RIA events

Upper Tolerance Level (UTL)

Table 2.9 (from PNNL-18212, Revision 1) PWR and BWR Fuel Rod Peak Gap Release Fractions (Based on Peak Values from Tables 2.4 and 2.8(a))

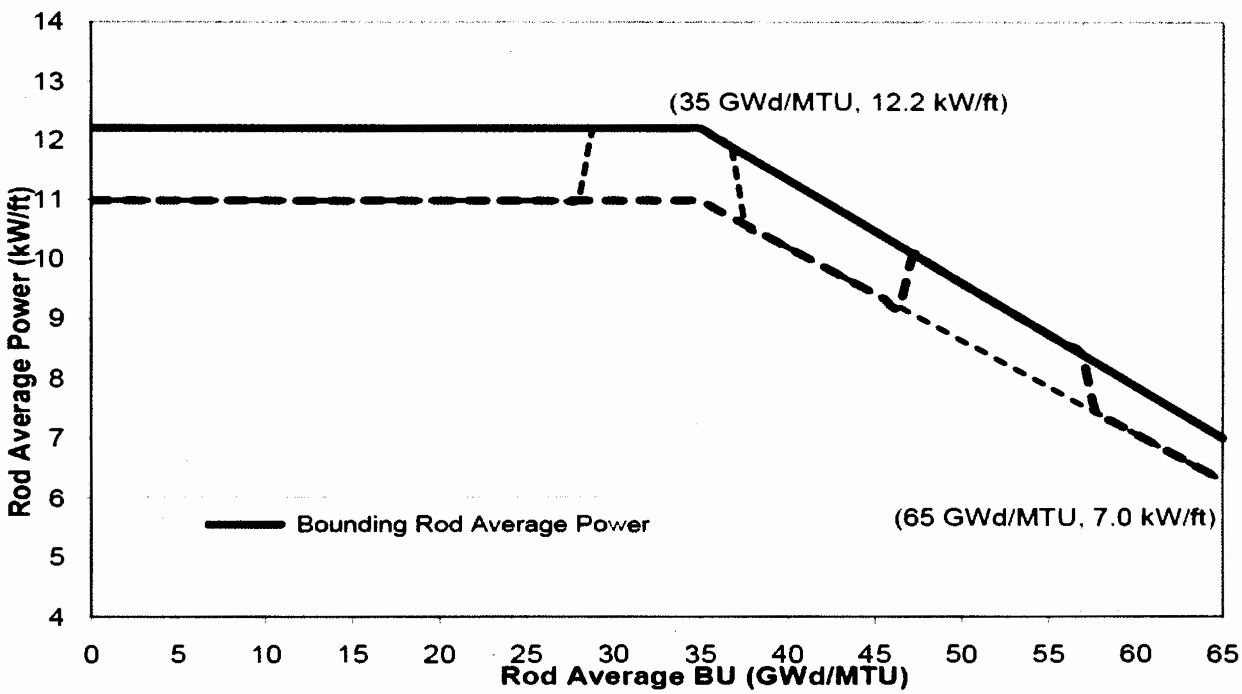

Figure A.1 (from PNNL-18212, Revision 1) Bounding PWR Linear Heat Generation Rate and Limiting Segmented Power Histories for Stable (PWR History #6) and Short-Lived Releases (PWR History #4)

Surry proposed that the bounding fission gas gap fractions in PNNL-18212, Revision 1, be used for the licensing basis FHA in lieu of the site-specific fractions currently in use by the licensee. The licensee states that the reload design will be validated against the operating curve each reload. As described in PNNL-18212, Revision 1, the "maximum" gas gap fractions are rounded up from the highest calculated fission gas release from all BWR and PWR assembly configurations currently in use or proposed for use in the United States. These values represent a 95/95 probability/confidence level (accounting for uncertainties in the fission gas release model to bound empirical data) for rods operated up to the bounding rod power envelope. All values in the "maximum" gas gap fraction column are thus calculated from fuel bundles found to have higher (for that isotope) than the Surry 15x15 fuel rod design. Because the method used in PNNL-18212, Revision 1, is based on the analysis of fission gas gap fraction and is applicable to Surry's FHA analysis, the NRC staff finds it acceptable to use the "Maximum" column fission gas gap fractions from PNNL-18212, Revision 1, Table 2.9, for Surry FHA analysis up to the rod average power versus rod average burnup curve in Figure A.1.

Water Depth

The FHA analysis assumes the resulting chemical form of the radio-iodine in the water is 95 percent cesium iodide, 4.85 percent elemental iodine, and 0.15 percent organic iodide. The cesium iodide released from the fuel is assumed to completely dissociate in the pool water. Because of the low pH of the pool water, the iodine re-evolves as elemental iodine. This is assumed to occur instantaneously, resulting in a chemical form of the radio-iodine in the water of 99.85 percent elemental iodine and 0.15 percent organic iodide.

As corrected by item 8 of Regulatory Issue Summary 2006-04 (Reference 25), RG 1.183, Appendix B, Regulatory Position 2 (Reference 8), 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 70% elemental and 30% organic species.

In accordance with RG 1.183, Appendix B, Regulatory Position 2, the licensee credits an overall effective iodine decontamination factor of 200 for a water cover depth of 23 feet. RG 1.183, Revision 0, states that this difference in decontamination factors for elemental and organic iodine species results in the iodine above the water being composed of 57 percent elemental and 43 percent organic species. However, the proposed Revision 1 of RG 1.183 dated July 2000 (DG-1199) states that the difference in decontamination factors for elemental and organic iodine species results in the iodine above the water being composed of 70 percent elemental and 30 percent organic species for release pressures that remain less than 1,200 pounds per square inch gauge (psig). The licensee proposes to modify the elemental and organic iodine species above the water in the FHA from 57 percent elemental and 43 percent organic to 70 percent elemental and 30 percent organic. The licensee stated that the proposed change aligns the Surry FHA analysis with DG-1199. However, the release pressure was not discussed in the LAR. Therefore, the NRC staff requested that the licensee discuss if Surry release pressure remains less than 1,200 psig for the FHA. The licensee responded by letter dated October 25, 2018, and stated:

> Refueling can begin 100 hours after shutdown (Technical Specification 3.10.A.9) with the RCS < 140°F (REFUELING SHUTDOWN Technical Specification 1.C.1). Dominion Energy Virginia's fuel vendors have verified that rod internal pressure will be <1200 psig under the conditions of the fuel handling accident (≤100 hours after shutdown, water temperature ≥140°F, and fuel rod burnup of 62,000 MWD/MTU).

Because the licensee's internal rod pressure remains less than 1,200 psig, the decontamination factors for elemental and organic iodine species result in the iodine above the water being composed of 70 percent elemental and 30 percent organic. Therefore, the NRC staff concludes that the proposed change is acceptable and consistent with draft RG 1.183, Revision 1.

Transport for FHA

The FHA in containment has three postulated release pathways. The first path is through the open equipment hatch directly to the environment, the second path is through the open containment personnel hatch to the west louver, and the third path is through the containment purge exhaust. An FHA in the SFP of the fuel building has one release pathway via the ventilation vent number two. These pathways remain open for the duration of the 2-hour release.

The current analysis of an FHA in containment in UFSAR Section 14.4.1 assumes a containment volume of 931,500 feet $(ft)^3$ and various containment release flow rates between 2,000 cfm and 80,000 cfm in order to bound all credible releases and maximize dose

consequences. The licensee proposes to change the RADTRAD-NAI model such that the containment release volume is 1 ft^3 with a 500 cfm exhaust flow rate, coupled with a 2-hour release. The proposed change removes any dilution and holdup in the containment and is consistent with the assumptions in RG 1.183; therefore, the NRC staff concludes that the proposed change is acceptable.

The current analysis of the radiological consequence of an FHA in the fuel building, as reflected in UFSAR Section 14.4.1, assumes a fuel building volume of 111,000 ft^3 and various fuel building release flow rates between 3,500 cfm and 80,000 cfm. The licensee proposes to change the RADTRAD-NAI model such that the containment release volume is 1 ft^3 with a 500 cfm exhaust flow rate, coupled with a 2-hour release. The proposed change removes any dilution and holdup in the fuel building and is consistent with the assumptions in RG 1.183; therefore, the NRC staff concludes that the proposed change is acceptable.

Control Room Habitability for the FHA

The current UFSAR analysis of the radiological consequence of an FHA assumes that the control room is manually isolated at time 0 based on identification that an FHA has occurred and a 2-minute time critical operator action to accomplish manual isolation of the control room. The licensee proposes to extend the time critical operator action to isolate the control room after initiation of the FHA from 2 minutes to 12 minutes. The proposed change also includes 20 seconds for the control room dampers to close after the time critical operator action. The licensee states that in order to accommodate the increase in time critical operator action and damper closure, the control room isolation is modeled as occurring at 10 minutes and 20 seconds or 0.1722 hours, and therefore, the control room isolation is changing from 0 hours to 0.1722 hours. Because there appears to be a disconnect between the RADTRAD model, which assumes isolation of the control room at 10 minutes and 20 seconds, and the time critical operator action, which isolates the control room at 12 minutes, the NRC staff requested that the licensee explain how the control room isolation RADTRAD modeling and the time critical operator action are consistent with each other. The licensee responded by letter dated October 25, 2018, and stated:

> The current design and licensing basis for the FHA is the control room is isolated prior to the plume reaching the control room normal ventilation intake. This is modeled as isolation at t=0 hours in the RADTRAD code. Isolation timing is currently based on a 2-minute time critical operator action and a calculation that determined that plume travel is 2 minutes from the release location to the control room normal intake. The RADTRAD model does not include the 2 minute plume travel time.
>
> The proposed design and licensing basis for the FHA models isolation at 10 minutes and 20 seconds, as described above, which includes 20 seconds for damper operation after the operator action is completed. However, the sequence of events still includes the 2 minute time for plume travel. Therefore the operator action is accomplished in 12 minutes from the drop of the fuel assembly and 10 minutes after the plume reaches the normal intake. The modeling of control room isolation and the time critical operator action are therefore consistent and correct.

The licensee assumes that the operator action is accomplished in 12 minutes from the drop of the fuel assembly and 10 minutes after the plume reaches the normal intake. The licensee

proposes to retain the majority of its current control room emergency ventilation system assumptions with the exception of the following updates. The licensee proposes to decrease the control room unfiltered in-leakage rate to 250 cfm based on the maximum tracer gas test results of 147 +/- 6 cfm. The NRC staff finds that the control room unfiltered in-leakage rate of 250 cfm is acceptable because it exceeds the actual amount of control room unfiltered in-leakage as determined by the maximum tracer gas test, thereby making it a conservative assumption. The licensee proposes to decrease the control room emergency ventilation filtered flow rate from 1,000 cfm to 900 cfm. The NRC staff finds this decrease to be a conservative change because it reduces air turnover in the control room, which leads to a higher dose estimate than currently estimated. In addition, the control room χ/Q values are updated based on meteorological data from 2009 to 2013, which are evaluated and found acceptable in Section 3.2 of this safety evaluation.

Offsite Boundaries

The licensee proposes to increase the EAB breathing rate to 3.5E-04 m^3/sec. Because the increased breathing rate is a conservative change that will result in a higher estimated radiation dose, the NRC staff finds the change acceptable. In addition, the EAB and LPZ χ/Q values are updated based on meteorological data from 2009 to 2013, which are evaluated and found acceptable in Section 3.2 of this safety evaluation.

Conclusion

The licensee evaluated the radiological consequences resulting from the postulated FHA and concluded that the radiological consequences at the EAB, LPZ, and control room 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. Based on its review, the NRC staff concludes that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this safety evaluation. The licensee used acceptable assumptions and inputs as presented in Attachment 2, Table 2.0-1, of the LAR, and the licensee's calculated dose results are 3.2 rem at the EAB, 0.2 rem at the LPZ, and 2.8 rem at the control room. They are summarized in Table 1 in Section 3.1.3.6 of this safety evaluation. The NRC staff compared the doses estimated by the licensee to the applicable accident dose criteria of 6.3 rem for the EAB and LPZ as stated in RG 1.183 and SRP 15.0.1 and to the results estimated in the NRC staff independent calculations, which were performed to ensure a thorough understanding of the licensee's assumptions and methods. Specifically, the NRC staff found that the inputs and assumptions used by the licensee are reasonable and that the licensee's dose estimates are below the regulatory limits. The NRC staff finds that the licensee demonstrated that there is reasonable assurance that the licensee's estimates of the TEDE due to a postulated design-basis FHA comply with the accident dose criteria in RG 1.183, SRP 15.0.1, and the radiation dose limits 10 CFR 50.67, which are 25 rem at the EAB and LPZ and 5 rem in the control room. Therefore, the staff finds the proposed FHA analysis changes acceptable.

3.1.3.3 SGTR

An SGTR is a break in a tube carrying primary coolant through the SG. This postulated break allows primary liquid to leak to the secondary side of one of the SGs with an assumed release to the environment through the SG pressure operated relief valves (PORVs) or the SG safety valves and the turbine driven auxiliary feed water (TDAFW) pump exhaust. The PORV on the ruptured SG is assumed to open normally at the beginning of the event due to increased pressure and then fail open. The ruptured SG discharges steam to the environment for 30

minutes until the SG is isolated by manual operator action. Flashed steam and liquid break flow from the primary into the ruptured SG is assumed to continue for the duration of the 30-minute release period.

The intact SGs discharge steam for a period of 10.5 hours. The 10.5-hour period includes 6 hours for the primary system to cool sufficiently to allow the cooldown to be completed using the residual heat removal (RHR) system plus an allowance of 4.5 hours to place the RHR system in service. No fuel damage is predicted as a result of an SGTR; therefore, the licensee performed the SGTR analysis with both a pre-accident iodine spike and a concurrent accident iodine spike. The NRC staff finds the performance of both iodine spikes is consistent with RG 1.183 and the current Surry analysis in UFSAR Section 14.3.1. In addition, the impact of a coincident loss-of-offsite power (LOOP) at the time of tube rupture was considered. The licensee's proposed changes to the SGTR analysis of record are listed in Attachment 2, Table 2.0-1, of the LAR.

SGTR Source Term

The licensee's evaluation indicates that no fuel damage would occur as a result of an SGTR accident. The licensee performed the SGTR accident analyses with two radio-iodine 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 Surry, the maximum iodine concentration allowed by TS 3.1.D.1 as a result of an iodine spike is 10 μCi/gm DE I-131.

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. The increase in primary coolant iodine concentration for the concurrent iodine spike case is estimated using a spiking model, which 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. For Surry, the RCS TS 3.1.D.1 limit for equilibrium or normal operation is 1.0 μCi/gm DE I-131. The concurrent iodine spike lasts 8 hours, which is consistent with assumptions in RG 1.183. The licensee assumes that the activity released from the concurrent iodine spiking (which is assumed to last 8 hours) mixes instantaneously and homogeneously throughout the primary coolant system. In addition, the licensee includes the radiological dose contribution from the release of iodine from the secondary system at the TS 3.6.H limit of 0.10 μCi/gm DE I-131. The NRC staff finds these source term assumptions and inputs consistent with Appendix F of RG 1.183, and therefore, acceptable.

Release Transport

The source term resulting from the radionuclides in the primary system coolant and from the iodine spiking in the primary system is transported to the ruptured SG by the break flow and to the intact SGs by the accident-induced leak rate of 1 gallon per minute, as specified by TS 6.4.Q.2.b. Releases from the ruptured SG are terminated after 30 minutes when the SG is isolated by closure of the PORV. The licensee assumes a fraction of the break flow flashes to steam in the ruptured SG and passes directly into the steam space of the ruptured SG with no credit taken for scrubbing by the SG liquid. The radionuclides entering the steam space as the result of the flashed steam pass directly to the environment through the SG PORVs. The remainder of the break flow enters the SG liquid. Radionuclides in the primary coolant leaking into the intact SGs are assumed to enter the SG liquid. Radionuclides in the SG liquid are

released as a result of secondary liquid boiling. The licensee assumes a partition factor of 100 for non-noble gas isotopes during boiling. Thus, 1 percent of the iodine and particulates are released from the SG liquid to the environment along with the steam flow. Noble gases are released from the primary system to the environment without reduction or mitigation. The licensee also assumes that the release from the intact SG continues for a period of 10.5 hours, after which the RHR system removes 100 percent of the decay heat with no need to augment the cooldown by steaming. The NRC staff concludes that the transport assumptions and inputs are consistent with Appendix F of RG 1.183, and that the transport model utilized for iodine and particulates is consistent with Appendix E of RG 1.183. Therefore, the transport assumptions, inputs and model are acceptable.

Control Room Habitability for the SGTR

The current analysis of the radiological consequence of an SGTR, as reflected in UFSAR Section 14.3.1, assumes that the control room in-leakage is either 500 cfm or 10 cfm. The licensee proposes to change the control room in-leakage to either 250 cfm or 0 cfm, and the stated reason for the change is maximum ASTM E741 tracer gas test equals 147 ± 6 cfm. However, the NRC staff noted that an assumed control room in-leakage of 0 cfm appears to be inconsistent with the tracer gas test result. Therefore, the NRC staff asked the licensee to explain how the most recent tracer gas test supports a control room in-leakage of 0 cfm, and why two different control room in-leakages are used in the SGTR analysis. By letter dated October 25, 2018, the licensee stated:

> The limiting SGTR control room dose consequences were determined for the pre-accident iodine spike with offsite power available. Under these conditions the normal ventilation system keeps operating and control room isolation is delayed until the Safety Injection (SI) signal plus 20 seconds for damper operation. As a result of the normal ventilation system high flow rate, delayed isolation, and the early availability for release of the pre-accident spike, the control room dose results increase with decreasing unfiltered inleakage ... the concurrent iodine spike cases or the pre-accident spike case with loss of offsite power ... were analyzed with up to 250 cfm of unfiltered inleakage and demonstrated an increase in dose with increasing unfiltered inleakage.

The NRC staff reviewed the licensee's response and concludes that the control room emergency ventilation system is not in operation when the unfiltered in-leakage rate is 0 cfm and that the normal ventilation system is in operation. The flow rate for the normal ventilation system is 3,300 cfm and is not filtered. Therefore, the radiological dose in the control room is highest when the normal ventilation system is in operation, and the radiological dose lowers when the control room emergency ventilation system is placed in operation, at which time the unfiltered in-leakage becomes 250 cfm. The NRC staff also concludes that the control room unfiltered in-leakage rate of 250 cfm for the concurrent iodine cases and the pre-accident iodine case with LOOP and 0 cfm for the pre-accident case with offsite power available are supported by the maximum tracer gas test results of 147 +/- 6 cfm. The licensee proposes to retain the majority of its current control room emergency ventilation system assumptions with the exception of the following updates. The licensee proposes to decrease the control room unfiltered in-leakage rate to 250 cfm for the concurrent iodine cases and the pre-accident iodine case with LOOP based on the maximum tracer gas test results of 147 +/- 6 cfm. The NRC staff finds that the control room unfiltered in-leakage rate of 250 cfm is acceptable and conservative because it exceeds the actual amount of control room unfiltered in-leakage as determined by the maximum tracer gas test.

The licensee proposes to decrease the control room emergency ventilation filtered flow rate from 1,000 cfm to 900 cfm. The NRC staff finds the use of a decreased flow rate to be acceptable because it is a conservative change that reduces air turnover in the control room, which leads to a higher dose estimate than currently estimated. The licensee also proposes to decrease the control room unfiltered in-leakage rate to 0 cfm for the pre-accident case with offsite power available. The NRC staff finds this assumption acceptable because the higher flow rate associated with the normal control room ventilation system would be used until the dampers reposition, which is a conservative assumption. In addition, the control room χ/Q values are updated based on meteorological data from 2009 to 2013, which are evaluated and found acceptable in Section 3.2 of this safety evaluation.

Offsite Boundaries

The licensee proposes to increase the EAB breathing rate to 3.5E-04 m^3/sec. Because the increased breathing rate is a conservative change that will result in a higher estimated radiation dose received, the NRC staff finds the change acceptable. In addition, the EAB and LPZ χ/Q values are updated based on meteorological data from 2009 to 2013, which are evaluated in Section 3.2 of this safety evaluation.

Conclusion

The licensee evaluated the radiological consequences resulting from the postulated SGTR and concluded that the radiological consequences at the EAB, LPZ, and control room 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 concludes that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this safety evaluation. The assumptions acceptable to the NRC staff are presented in Attachment 2, Table 2.0-1, of the LAR, and the licensee's calculated dose results are 1.1 rem at the EAB, 0.1 rem at the LPZ, and 0.3 rem at the control room for the concurrent iodine spike analysis and 0.8 rem at the EAB, 0.1 rem at the LPZ, and 0.7 rem at the control room for the pre-accident iodine spike analysis and are summarized in listed in Table 1 in Section 3.1.3.6 of this safety evaluation. The NRC staff compared the doses estimated by the licensee to the accident dose criteria of 25 rem at the EAB and LPZ for the pre-accident iodine spike analysis and 2.5 rem at the EAB and LPZ for the concurrent iodine spike analysis as stated in RG 1.183 and SRP 15.0.1, and to the results estimated in the NRC staff independent calculations, which were performed to ensure a thorough understanding of the licensee's assumptions and methods. Specifically, the NRC staff found that the inputs and assumptions used by the licensee are reasonable and that the licensee's dose estimates, which are comparable to those calculated by the NRC staff, are below the regulatory limits . The NRC staff finds that the licensee demonstrated that there is reasonable assurance that the estimates of the total effective dose equivalent due to a postulated design-basis FHA comply with the accident dose criteria in RG 1.183, SRP 15.0.1, and the radiation dose limits 10 CFR 50.67, which are 25 rem at the EAB and LPZ and 5 rem in the control room. Therefore, the staff finds the proposed SGTR analysis changes acceptable.

3.1.3.4 MSLB

The MSLB accident is a postulated break in one of the main steam lines leading from an SG to the turbine. The break occurs in the turbine building, which also contains the control room emergency ventilation intakes. The affected SG blows down into the turbine building for 30 minutes, after which it is isolated. Two models are analyzed for the MSLB, one with a

coincident LOOP and the other that has power available for the duration of the accident. Each model utilizes steam release rates that are specific to the scenario. In addition, both scenarios model turbine building exhaust differently. The LOOP scenario models the turbine building exhaust fans as not having the power required to operate. Therefore, natural circulation (0.2 volumes per hour) is modeled. The power available scenario models the turbine building exhaust flow rate at the maximum capability of the turbine building exhaust fans (12 volumes per hour). The total modeled flow rate from the turbine building includes the affected SG blow down flow rates. In order to maximize dose consequences, neither scenario credits the condenser as being available. The licensee's proposed changes to the MSLB analysis of record are listed in Attachment 2, Table 2.0-1, of the LAR.

MSLB Source Term

Appendix E of RG 1.183 identifies acceptable radiological analysis assumptions for an MSLB accident. RG 1.183, Appendix E, Regulatory Position 2, states that if no or minimal fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by TSs and that two cases of iodine spiking should be assumed. The licensee's evaluation indicates that no fuel damage would occur as a result of an MSLB accident. The licensee stated it performed the MSLB accident analyses with two radio-iodine spiking cases, as recommended by 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 TSs for a spiking condition. For Surry, the maximum iodine concentration allowed by TS 3.1.D.1 as a result of an iodine spike is 10 μCi/gm DE I-131.

The second case assumes that the primary system transient associated with the MSLB causes an iodine spike in the primary system. This case is referred to as a concurrent iodine spike. The increase in primary coolant iodine concentration for the concurrent 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 500 times greater than the iodine equilibrium release rate. For Surry, the RCS TS 3.1.D.1 limit for equilibrium or normal operation is 1.0 μCi/gm DE I-131. The concurrent iodine spike lasts 8 hours, which is consistent with RG 1.183. The licensee assumes that the activity released from the iodine spiking mixes instantaneously and homogeneously throughout the primary coolant system. In addition, the licensee included the radiological dose contribution from an initial secondary side liquid concentration at the TS limit of 0.1 μCi/gm DE I-131. Noble gases are assumed to not be present in the bulk SG liquid. The NRC staff finds these source term assumptions and inputs are consistent with RG 1.183, and therefore, are acceptable.

Release Transport

For the affected SG, the licensee assumes the primary and secondary activity is released directly into the turbine building with no credit taken for partitioning or scrubbing of the SG liquid. The primary-to-secondary leak rate in the affected SG is 500 gallons per day (gpd). The MSLB is terminated at 30 minutes when it is isolated by closure of a main steam trip valve. From the turbine building, the activity is assumed to pass into the control room via emergency intakes and unfiltered inleakage, as well as pass into the environment through vents, louvers, and other openings located around the turbine building. The affected SG releases activity into the turbine building until isolated at 30 minutes. The release from the turbine building is determined by each scenario. In the LOOP scenario, the turbine building ventilation system is not operating due to offsite power being unavailable. With power available, the system is determined to be

energized and is modeled as operating at its maximum capacity. The models use 0.2 and 12 volumes per hour for each scenario, respectively.

Cooldown of the primary system is through the release of steam from the two intact SGs. Steam releases from the intact SGs are modeled to occur through the PORVs and TDAFW turbine exhaust. The intact SGs are modeled to have a primary to secondary leakage rate equal to 940 gpd, which is the remaining primary-to-secondary leakage (1440 gpd - 500 gpd). Intact SG steaming will continue until sufficient cooldown allows the RHR system to be placed into service. The time for the RHR system to be placed into service is modeled as 38 and 12.5 hours following the initiation of the event for the LOOP and power available scenarios, respectively. Due to the effects of partitioning and moisture carryover, the total radionuclides released to the environment are reduced by a factor of 100. The effect of partitioning and moisture carryover is modeled by reducing the steam release rate by a factor of 100 to conserve radionuclides in the intact SG liquid. Releases of noble gases are assumed to occur directly to the environment without any mitigation or holdup. The NRC staff finds that the licensee's inputs and assumptions are consistent with Appendix E of RG 1.183.

Control Room Habitability for the MSLB

The licensee proposes to retain the majority of its current control room emergency ventilation system assumptions with the exception of the following updates. The licensee proposes to decrease the control room unfiltered in-leakage rate to 250 cfm based on the maximum tracer gas test results of 147 +/- 6 cfm. The NRC staff finds that the control room unfiltered in-leakage rate of 250 cfm is acceptable and conservative because it exceeds the actual amount of control room unfiltered in-leakage as determined by the maximum tracer gas test. The licensee proposes to decrease the control room emergency ventilation filtered flow rate will decrease from 1,000 cfm to 900 cfm, which the NRC staff finds to be a conservative change because it reduces air turnover in the control room, leading to a higher dose estimate than currently estimated. The normal control room intake will be isolated within 39 seconds of accident initiation for LOOP and within 36 seconds with power available. The timings are derived from the safety injection signal actuation plus signal processing and damper response times. Prior to isolation of the control room, unfiltered in-leakage is treated as entering the control room envelope via the normal intakes. After isolation, unfiltered inleakage is treated as entering the control room envelope via the emergency intakes. Emergency ventilation is modeled as providing a filtered air supply within 1 hour of control room isolation until termination of the event. The NRC staff finds these changes acceptable. In addition, the control room χ/Q values are updated based on meteorological data from 2009 to 2013, which are evaluated and found acceptable in Section 3.2 of this safety evaluation.

Offsite Boundaries

The licensee proposes to increase the EAB breathing rate to 3.5E-04 m^3/sec. Because the increased breathing rate is a conservative change that will result in a higher estimated radiation dose, the NRC staff finds the change is acceptable. In addition, the EAB and LPZ χ/Q values are updated based on meteorological data from 2009 to 2013, which are evaluated and found acceptable in Section 3.2 of this evaluation.

Conclusion

The licensee evaluated the radiological consequences resulting from the postulated MSLB and concluded that the radiological consequences at the EAB, LPZ, and control room 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 reviewed the analysis and verified that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this safety evaluation. The assumptions acceptable to the NRC staff are presented in Attachment 2, Table 2.0-1, of the LAR, and the licensee's calculated dose results are 0.6 rem at the EAB, 0.1 rem at the LPZ, and 1.6 rem at the control room for the concurrent iodine spike analysis and 0.4 rem at the EAB, 0.1 rem at the LPZ, and 1.4 rem at the control room for the pre-accident iodine spike analysis and are summarized in Table 1 in Section 3.1.3.6 of this safety evaluation. The NRC staff compared the doses estimated by the licensee to the accident dose criteria of 25 rem at the EAB and LPZ for the pre-accident iodine spike analysis and 2.5 rem at the EAB and LPZ for the concurrent iodine spike analysis as stated in RG 1.183 and SRP 15.0.1 and to the results estimated in the NRC staff independent calculations, which were performed to ensure a thorough understanding of the licensee's assumptions and methods. Specifically, the NRC staff found that the inputs and assumptions used by the licensee are reasonable and that the licensee's dose estimates, which are comparable to those calculated by the NRC staff, are below the regulatory limits. The NRC staff finds that the licensee demonstrated that there is reasonable assurance that the licensee's estimates of the total effective dose equivalent due to a postulated design-basis MSLB comply with the accident dose criteria in RG 1.183, SRP 15.0.1, and the radiation dose limits 10 CFR 50.67, which are 25 rem at the EAB and LPZ and 5 rem in the control room. Therefore, the NRC staff finds the proposed MSLB analysis changes acceptable.

3.1.3.5 Locked Rotor Accident (LRA)

The licensee proposes to remove the existing analysis of the radiological consequences of an LRA from the design basis in UFSAR, Section 14.2.9.2, because fuel damage is not predicted. Fuel damage, defined as fuel rods that experience departure from nucleate boiling, is not predicted during the design-basis LRA transient analysis. RG 1.183 Appendix G, Regulatory Position 2, states that if no fuel damage is postulated for the limiting event, a radiological analysis is not required, as the consequences of the locked rotor events are bounded by the consequences projected for the MSLB outside containment.

The NRC staff concludes that the LRA is bounded by the MSLB because fuel damage does not occur during the LRA. Therefore, the NRC staff finds that the proposed change to remove the radiological consequence analysis for the LRA from the UFSAR is consistent with RG 1.183 and is acceptable.

3.1.3.6 Table 1

This table provides a summary of the licensee's estimates for each analyzed DBA.

Table 1
Total Effective Dose Equivalent in rem

DBA Analyses	EAB	LPZ	RG 1.183 and SRP 15.0.1 Accident Dose Criteria at EAB and LPZ	10 CFR 50.67 EAB and LPZ Limit	Control Room	10 CFR 50.67 Limit
LOCA	9.1	1.9	25	25	4.4	5
FHA	3.2	0.2	6.3	25	2.8	5
SGTR Concurrent Iodine Spike	1.1	0.1	2.5	25	0.3	5
SGTR Pre-accident Iodine Spike	0.8	0.1	25	25	0.7	5
MSLB Concurrent Iodine Spike	0.6	0.1	2.5	25	1.6	5
MSLB Pre-accident Iodine Spike	0.4	0.1	25	25	1.4	5

These estimates are comparable to those calculated by the NRC staff.

3.2 Meteorological Review

3.2.1 Atmospheric Dispersion Estimates

The licensee's LAR involves a request to revise the DBA atmospheric dispersion factor values using meteorological data collected on site for the period from 2009-2013. Atmospheric dispersion factors are significant inputs in assessments performed to demonstrate compliance with 10 CFR Part 50. The determination of control room χ/Q values were made using the ARCON96 atmospheric dispersion model NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes" (Reference 26), pursuant to the guidance of RG 1.194 (Reference 13), which includes replacing values determined based on the Murphy and Campe methodology[1] for SG safety valve and PORV releases. Offsite χ/Q values were developed using the PAVAN computer code NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design-Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations" (Reference 27), in accordance with the guidance provided in RG 1.145 (Reference 12). The NRC staff reviewed the licensee's new atmospheric dispersion analyses, as described below.

[1] 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 Atomic Energy Air Cleaning Conference, San Francisco, CONF 740807, U.S. Atomic Energy Commission (now U.S. NRC), dated August 1974.

3.2.1.1 Meteorological Data

The licensee used hourly meteorological data from calendar years 2009-2013 in its new analysis. The licensee stated that it compiled and archived hourly meteorological data as described in RG 1.23 (Reference 11). The licensee reviewed the data for missing or anomalous observations, instrumentation problems, and trends indicative of local effects such as building wakes and excessive vegetation effects. The licensee did not perform data substitutions as a result of its review. The licensee provided the 2009-2013 data in hourly format, as specified in RG 1.23 and as required for input into the ACRON96 computer code, and joint frequency distributions (JFDs) of wind speed and direction as a function of atmospheric stability for input into the PAVAN computer code.

The NRC staff performed a review of the 2009-2013 hourly meteorological database using the methodology described in NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data" (Reference 28). The NRC staff reviewed the overall quality of the data and finds it to be consistent with the guidance outlined in RG 1.23. The NRC staff finds the data revealed generally stable, and neutral atmospheric conditions at night and unstable conditions during the day are generally consistent with expected meteorological conditions. The NRC staff also finds that wind speed and wind direction data showed similar results from year to year.

Based on its review and the consistency of the data with the guidance, the NRC staff considers the onsite meteorological dataset from calendar years 2009-2013 acceptable for use in calculations for the atmospheric dispersion analyses used to support this LAR.

3.2.1.2 Control Room Atmospheric Dispersion Factors

In support of the LAR, the licensee used the computer code ARCON96 to estimate χ/Q values for the control room for potential accidental releases of radioactive material. RG 1.194 endorses the ARCON96 model for determining χ/Q values to be used in the design-basis evaluations of control room radiological habitability.

The ARCON96 code estimates χ/Q values for various time-averaged periods ranging from 2 hours to 30 days. The meteorological input to ARCON96 consists of hourly values of wind speed, wind direction, and atmospheric stability class. The χ/Q values calculated through ARCON96 are based on the theoretical assumption that material released to the atmosphere will be normally distributed (Gaussian) about the plume centerline. A straight-line trajectory is assumed between the release points and receptors. The diffusion coefficients account for enhanced dispersion under low wind speed conditions and in building wakes.

The hourly meteorological data are used to calculate hourly relative concentrations (χ/Q values). The hourly relative concentrations are then combined to estimate concentrations ranging in duration from 2 hours to 30 days. Cumulative frequency distributions prepared from the average relative concentrations and the relative concentrations that are exceeded no more than 5 percent of the time for each averaging period are determined.

The dispersion coefficients used in ARCON96 have three components. The first component is the diffusion coefficient used in other NRC models such as PAVAN. The other two components are corrections to account for enhanced dispersion under low wind speed conditions and in building wakes. These components are based on analysis of diffusion data collected in various building wake diffusion experiments under a wide range of meteorological conditions. Because

the dispersion occurs at short distances within the plant's building complex, the ARCON96 dispersion parameters are not affected by nearby topographic features such as bodies of water.

Attachment 2 to the LAR describes the analyses done to estimate the χ/Q values for the control room. Control room estimates were made for the following accidents:

- LOCA
- FHA
- SGTR
- MSLB

The licensee stated that prior to control room isolation or LOOP, the control room is provided with air from the normal ventilation intake. The normal ventilation intake was not previously modeled during an FHA or LOCA due to automatic or manual control room isolation prior to uptake. This remains the case for a LOCA as a safety injection signal initiates automatic control room isolation prior to uptake. However, changes to the FHA manual isolation assumption would require χ/Q values for the normal control room intake. After control room isolation, since the four emergency ventilation intakes draw air from the interior of the turbine building, the control room χ/Q receptor locations considered for the LOCA and FHA were the turbine building fresh air louvers 1-6, roll-up doors 1 and 2, and the turbine building fresh air intakes. The licensee states that composite χ/Q values were developed for the emergency control room intakes weighted based upon the cross-sectional areas of the various openings in the turbine building, as allowed by RG 1.194.

Table 3.1-1 of Attachment 2 to the LAR shows the ARCON96 inputs specific to the LOCA and FHA. Table 3.1-2 of Attachment 2 to the LAR shows ARCON96 inputs common to both the LOCA and FHA. The licensee modeled the releases as ground level releases and as point sources with the exception of the containments U1 and U2, which were modeled as diffuse area sources.

Table 3.1-3 of Attachment 2 to the LAR contains the cross-sectional areas used to weight the individual LOCA and FHA emergency control room intakes and derive the composite χ/Q values. Table 3.1-4 of Attachment 2 to the LAR contains the individual receptor, as well as the composite χ/Q values for the LOCA and FHA emergency control room intakes. The licensee derived the composite χ/Q values by summing the individual receptor χ/Q values that were multiplied by the receptor area/total area. Table 3.1-5 of Attachment 2 to the LAR contains the normal control room intake χ/Q values for the LOCA and FHA. The licensee used the limiting control room χ/Q values in each accident scenario, including both normal and composite emergency intakes, which are listed in Table 2.0-1 of Attachment 2 to the LAR.

The licensee stated that new control room χ/Q values for the MSLB and SGTR were derived using ARCON96 for the source/receptor pairs addressing the most viable locations and limiting accident cases. Prior to control room isolation or LOOP, the control room is provided with air from the normal ventilation intake. The licensee stated that control room isolation occurs based upon a safety injection signal and that, after control room isolation, the control room χ/Q receptor locations considered for the MSLB and SGTR were the control room emergency intakes.

The licensee determined the ARCON96 input source-to-receptor horizontal distances using the straight-line distance between the source and the receptor without consideration for intervening buildings. Tables 3.1-6 and 3.1-2 of Attachment 2 to the LAR provide the ARCON96 inputs for

limiting source-to-receptor combinations. However, for the MSLB and SGTR χ/Q values, the licensee used the RG 1.194 (Reference 13) default building area of 2,000 m^2 instead of the calculated value of 1,453 m^2 shown in LAR Table 3.1-2. The licensee stated that sensitivities have shown the ARCON96 results to be relatively insensitive to changes in the building area input. The licensee modeled the releases as ground level releases and as point sources.

Table 3.1-6 of Attachment 2 to the LAR shows the specific ARCON96 inputs for the MSLB and SGTR accident analysis. Table 3.1-7 of Attachment 2 to the LAR contains the results of the ARCON96 runs completed for the MSLB and SGTR normal and emergency control room χ/Q values. The licensee used limiting control room χ/Q values in each accident scenario, including both normal and emergency intakes, which are listed in LAR Table 2.0-1.

The MSLB and SGTR χ/Q values in LAR Table 2.0-1 also reflect a reduction by a factor of 5 as allowed RG 1.194 based upon buoyant plume rise. The licensee provided a justification for this reduction in Section 3.1.2.3 of Attachment 2 to the LAR. The licensee stated that, in accordance with the guidance of RG 1.194, the buoyant plume rise associated with energetic releases from steam relief values or atmospheric steam dumps can be credited if (1) the release is uncapped and vertical, and (2) the time-dependent vertical velocity exceeds the 95th percentile wind speed, at the release point height, by a factor of 5. The licensee stated that the SG safety valves (safeties), PORVs, and TDAFW pump exhausts represented by sources P1 and T1 are uncapped and orient their exhaust vertically.

Table 3.1-8 of Attachment 2 to the LAR shows the results of the 95th percentile wind speed calculations for the MSLB and SGTR release locations, the 5 times 95th percentile wind speed converted to an equivalent mass flow rate, and the minimum steam release exhaust flow rate. The licensee stated that based on the results presented in LAR Table 3.1-8, reduction of the MSLB and SGTR control room χ/Q values by a factor of 5 is justified for the duration of the releases from the SG PORVs and the TDAFW pump exhaust.

The NRC staff confirmed licensee control room atmospheric dispersion estimates by running the ARCON96 computer model and obtaining similar results. Both the staff and licensee used the same release assumption for each of the release pathway-receptor combinations, as well as the previously discussed source-to-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. Because the results of the NRC staff's confirmatory analysis are similar, and the licensee followed applicable NRC guidance, the NRC staff concludes that the licensee's control room χ/Q values are acceptable for use in the radiological consequence assessments evaluated above.

3.2.1.3 Offsite Atmospheric Dispersion Factors

In support of the LAR, the licensee used the PAVAND computer code, which is the Dominion compiled version of the PAVAN computer code, to estimate χ/Q values for the EAB and outer boundary of the LPZ for accident analyses for potential accidental releases of radioactive material.

PAVAN is used to estimate relative ground level air concentrations for potential accidental releases of radioactive material from nuclear facilities. The program implements the guidance provided in RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants" (Reference 12).

Using JFDs of wind direction and wind speed by atmospheric stability, the program provides relative air concentration values as functions of direction for various time periods at the EAB and the outer boundary of the LPZ. Calculations of χ/Q values can be made for assumed ground level releases (e.g., through building penetrations and vents) or elevated releases from freestanding stacks.

The χ/Q calculations are based on the theory that material released to the atmosphere will be normally distributed (Gaussian) about the plume centerline. A straight-line trajectory is assumed between the point of release and all distances for which χ/Q values are calculated.

Table 3.1-9 of Attachment 2 to the LAR includes EAB minimum source-to-receptor distances by sector for the sources described in Figure 3.1-1 plus the four corners of the turbine building (TB1 through TB4), which are used in the MSLB analysis to bound releases from the potential source release points of the turbine building.

The licensee developed JFDs of wind speed and direction using the 2009-2013 meteorological data for input into PAVAND. The licensee stated that the wind speed groups were chosen to obtain a larger number of wind speed groups at lower wind speeds, while maintaining a relatively even distribution of data amongst the groups. The first wind speed group is designated for calms, defined as hourly average wind speeds below the vane or anemometer starting speed, which is 0.5 miles per hour (mph). The wind speed groups chosen for the JFD tables are defined by maximum values of 0.4 mph (calms), 1.3 mph, 1.9 mph, 2.6 mph, 3.3 mph, 3.8 mph, 4.3 mph, 4.8 mph, 5.4 mph, 6.0 mph, 6.7 mph, 7.8 mph, 9.5 mph, and 30 mph. The option was selected to allow the code to automatically distribute calms into the first wind speed category.

Table 3.1-11 of Attachment 2 to the LAR includes a summary of PAVAN input parameters related to instrument elevations, release heights, and containment parameters. The licensee stated that the building wake effects were conservatively ignored and the releases were modeled as ground level.

The licensee stated that Table 3.1-12 of Attachment 2 to the LAR provides the most limiting offsite χ/Q values calculated for each of the source locations. In accordance with RG 1.145, the licensee reported the larger of the two values calculated for maximum sector χ/Q and overall site χ/Q for each time period. Limiting EAB and LPZ χ/Q values for each accident scenario are listed in LAR Table 2.0-1.

The NRC staff confirmed licensee offsite atmospheric dispersion estimates by using the licensee's assumptions and input values to run the PAVAN computer model and obtain similar results. The NRC staff also confirmed that the analysis and assumptions were consistent with the guidance of RG 1.145. Because the licensee followed NRC guidance and the results of the staff's confirmatory analysis were similar, the NRC staff concludes that the licensee's offsite χ/Q values are acceptable for use in the radiological consequence assessments.

3.2.2 Conclusion

The NRC staff reviewed the guidance, assumptions, and methodology used by the licensee to assess the χ/Q values associated with postulated releases from the potential release pathways. The staff found that the licensee used methods consistent with regulatory guidance identified in Section 2.0 of this safety evaluation. Specifically, the licensee used onsite meteorological data that followed the guidance of RG 1.23. The licensee used inputs and assumptions to calculate

the control room χ/Q values that are consistent with the guidance of RG 1.194. In addition, the licensee used the inputs and assumptions to calculate the offsite χ/Q values that are consistent with the guidance of RG 1.145. Therefore, on the basis of this review of the atmospheric dispersion analysis and the similarity of the results of the NRC staff's confirmatory analysis, the staff concludes that the licensee's proposed χ/Q values are acceptable for use in calculating the radiological consequences assessments associated with this LAR.

4.0 TECHNICAL CONCLUSION

The NRC staff reviewed the licensee's proposed changes to revise the Surry, Unit Nos. 1 and 2, TSs to be consistent with TSTF-490, and make associated changes, which include replacing the current limits on primary coolant gross specific activity with limits on primary coolant noble gas specific activity. The NRC staff reviewed the proposed changes to (1) delete the gross activity, $\bar{E}$ determination, and radio-iodine analysis tests and their associated Notes 2, 3, and 6, respectively, (2) add a new DEX SR, (3) revise Note 7, (4) revise the definition of DEI, and (5) reformat TS 3.1.D and has concluded that the changes are consistent with the methodology used to analyze the radiological consequences of the MSLB and SGTR accidents. Therefore, the NRC staff finds that these proposed changes are acceptable.

The NRC staff also reviewed the updated AST implementation proposed by the licensee for Surry as described in section 3.1.3 of this SE. In performing this review, the NRC staff relied upon information placed on the docket by the licensee, NRC staff experience in performing similar reviews, and on the NRC staff's confirmatory calculations. As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological impacts of the proposed updated AST. The NRC staff concludes that the licensee used analysis methods and assumptions consistent with the guidance in RG 1.183, 1.194, and SRP 15.0.1. The NRC staff compared the radiation doses estimated by the licensee to the acceptance criteria stated in RG 1.183, SRP 15.0.1, and the limits stated in 10 CFR 50.67, and to the results estimated by the NRC staff in its confirmatory calculations, and found that the licensee's dose estimates were below the regulatory limits and comparable to those estimated by the NRC staff. The NRC staff concludes, with reasonable assurance, that the licensee's estimates of the TEDE from postulated DBAs comply with the acceptance criteria in RG 1.183 and SRP 15.0.1 and the radiation dose limits in 10 CFR 50.67. The NRC staff concludes there is reasonable assurance that Surry 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. Therefore, the NRC staff finds that the proposed updated AST implementation acceptable.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Virginia State official was notified of the proposed issuance of the amendments on November 28, 2018. The State official confirmed that the Commonwealth of Virginia had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change SRs. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The

Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (83 FR 28465). The amendments also include format, editorial or other minor revisions. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (c)(10)(v). 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.

8.0 REFERENCES

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2. Virginia Electric and Power Company, letter to U.S. Nuclear Regulatory Commission, "Surry Power Station Units 1 and 2 Proposed License Amendment Request TSTF-490 and Update of Alternative Source Term Analyses Response to Request for Additional Information and Revised Proposed Technical Specification Pages," dated October 25, 2018 (ADAMS Accession No. ML18304A330).

3. Crowthers, Michael, Technical Specification Task Force, letter to U.S. Nuclear Regulatory Commission, "TSTF-490, Revision 0, "Deletion of E Bar Definition and Revision to RCS Specific Activity Tech Spec," dated September 13, 2005 (ADAMS Accession No. ML052630462).

4. U.S. Nuclear Regulatory Commission, "Standard Technical Specification – Babcock and Wilcox Plants, NUREG-1430, Revision 4.0, Volume 1, April 2012 (ADAMS Accession No. ML12100A177).

5. U.S. Nuclear Regulatory Commission, "Standard Technical Specification – Westinghouse Plants, NUREG-1431, Revision 4.0, Volume 1, April 2012 (ADAMS Accession No. ML12100A222).

6. U.S. Nuclear Regulatory Commission, "Standard Technical Specification – Combustion Engineering Plants, NUREG-1432, Revision 4.0, Volume 1, April 2012 (ADAMS Accession No. ML12102A165).

7. U.S. Nuclear Regulatory Commission, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition, Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, July 2000 (ADAMS Accession No. 003734190).

8. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.183 "Alternative Radiological Source Terms For Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000 (ADAMS Accession No. ML003716792).

9. Edison, Gordon E., Sr., NRC letter to David A. Christian, Virginia Electric and Power Company, "Surry Power Station, Units 1 and 2 - Issuance of Amendments Regarding Alternative Source Term," Amendments 230 and 230, dated March 8, 2002 (ADAMS Accession No. ML020710159).

10. U.S. Nuclear Regulatory Commission, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition, Section 2.3.4, "Short-Term Atmospheric Dispersion Estimates for Accident Releases," March 2007 (ADAMS Accession No. 070730398).

11. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.23 "Onsite Meteorological Programs," dated February 17, 1972 (ADAMS Accession No. ML020360030).

12. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.145 "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," February 1983 (ADAMS Accession No. ML003740205).

13. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.194 "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," April 30, 1976 (ADAMS Accession No. ML003740305).

14. U.S. Nuclear Regulatory Commission, Regulatory Guide 8.40 "Methods for Measuring Effective Dose Equivalent from External Exposure," July 2010 (ADAMS Accession No. ML100610534).

15. U.S. Atomic Energy Commission, Technical Information Document 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," dated March 23, 1962 (ADAMS Accession No. ML021720780).

16. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," May 2003 (ADAMS Accession No. ML031490640).

17. Environmental Protection Agency, Federal Guidance Report No. 12., "External Exposure to Radionuclides in Air, Water, and Soil."

18. Environmental Protection Agency, Federal Guidance Report No. 11.,"Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

19. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," October 1977 (ADAMS Accession No. ML003740384).

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21. Sandia National Laboratories prepared for U.S. Nuclear Regulatory Commission, "RADSTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation," NUREG/CR-6604, April 1998 (ADAMS Accession No. ML15092A284).

22. Information System Laboratories, Inc. for U.S. Nuclear Regulatory Commission, "SNAP/RADTRAD 4.0: Description of Models and Methods," NUREG/CR-7220, June 2016 (ADAMS Accession No. ML16160A019).

23. Lingam, Siva P., NRC letter to David A. Christian, Virginia Electric and Power Company, "Surry Power Station, Units 1 and 2 - Issuance of Amendments Regarding Implementation of Generic Safety Issue 191," Amendments 250 and 249, dated October 12, 2006 (ADAMS Accession No. ML062920499).

24. U.S. Nuclear Regulatory Commission, NUREG-0737, "Clarification of TMI Action Plan Requirements – Requirements for Emergency Response Capabillity," February 1989 (ADAMS Accession No. ML102560009).

25. U.S. Nuclear Regulatory Commission "NRC Regulatory Issue Summary 2006-04, Experience With Implementation of Alternative Source Terms," dated March 7, 2006 (ADAMS Accession No. ML053460347).

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

28. U.S. Nuclear Regulatory Commission, NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data" July 1982 (ADAMS Accession No. ML12061A136).

29. Singal, Balwant, NRC Letter to Edward D. Halpin, Pacific Gas and Electric Company, Diablo Canyon Power Plant, "Issuance of Amendments Re: Revise Licensing Bases to Adopt Alternative Source Term [DG-1199, Revision 0]," dated April 27, 2017 (ADAMS Accession No. ML17012A246).

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31. Pacific Northwest National Labs prepared for the U.S. Department of Energy, PNNL-18212, Revision 0, "Update of Gap Release Fractions for Non-LOCA Events Utilizing the Revised ANS 5.4 Standard," February 2009 (ADAMS Accession No. ML090400853).

32. Pacific Northwest National Labs prepared for the U.S. Nuclear Regulatory Commission, PNNL-18212, Revision 1, "Update of Gap Release Fractions for Non-LOCA Events Utilizing the Revised ANS 5.4 Standard" June 2011 (ADAMS Accession No. ML112070118).

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Date: June 12, 2019