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

November 19, 2021

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, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 306 AND 306 TO UPDATE OF THE LOSS-OF-COOLANT ACCIDENT ALTERNATE SOURCE TERM DOSE ANALYSIS (EPID L-2020-LLA-0258)

Dear Mr. Stoddard:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 306 to Subsequent Renewed Facility Operating License No. DPR-32 and Amendment No. 306 to Subsequent Renewed Facility Operating License No. DPR-37 for the Surry Power Station (Surry), Units 1 and 2, respectively, in response to your application dated December 3, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20338A542), as supplemented by letter dated May 13, 2021 (ADAMS Accession No. ML21133A421).

The amendments change the facilities as described in the Updated Final Safety Analysis Report by updating the Alternate Source Term analysis for Surry, Units 1 and 2 following a loss-of-coolant accident by increasing the assumed containment depressurization profile and reducing the refueling water storage tank back leakage limit.

A copy of the related safety evaluation is also enclosed. The Commission's biweekly *Federal Register* notice will include the notice of issuance.

If you have any questions, please contact me at (301) 415-5136 or by e-mail at John.Klos@nrc.gov.

Sincerely,

/RA/

John Klos, 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. 306 to DPR-32
2. Amendment No. 306 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 SUBSEQUENT RENEWED FACILITY OPERATING LICENSE

Amendment No. 306
Subsequent 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 December 3, 2020, as supplemented by a letter dated May 13, 2021, 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, by Amendment No. 306, Subsequent Renewed Facility Operating License No. DPR-32 is hereby amended to authorize the change to the Updated Final Safety Analysis Report (UFSAR) as requested by letter dated December 3, 2020, as supplemented by a letter dated May 13, 2021, and evaluated in the NRC staff safety evaluation dated November 19, 2021. The licensee shall submit the update of the UFSAR authorized by this amendment in accordance with 10 CFR 50.71(e).

3. The license amendment is effective as of the date of issuance of the subsequent renewed license and shall be implemented within 90 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

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Date: 2021.11.19 11:37:06 -05'00'

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

Date of Issuance: November 19, 2021

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 SUBSEQUENT RENEWED FACILITY OPERATING LICENSE

Amendment No. 306
Subsequent 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 December 3, 2020, as supplemented by a letter dated May 13, 2021, 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, by Amendment No. 306, Subsequent Renewed Facility Operating License No. DPR-37 is hereby amended to authorize the change to the Updated Final Safety Analysis Report (UFSAR) as requested by letter dated December 3, 2020, as supplemented by a letter dated May 13, 2021, and evaluated in the NRC staff safety evaluation dated November 19, 2021. The licensee shall submit the update of the UFSAR authorized by this amendment in accordance with 10 CFR 50.71(e).

3. The license amendment is effective as of the date of issuance of the subsequent renewed license and shall be implemented within 90 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

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Date: 2021.11.19 11:38:09 -05'00'

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

Date of Issuance: November 19, 2021

ATTACHMENT TO

SURRY POWER STATION, UNIT NOS. 1 AND 2

LICENSE AMENDMENT NO. 306

RENEWED FACILITY OPERATING LICENSE NO. DPR-32

DOCKET NO. 50-280

AND

LICENSE AMENDMENT NO. 306

RENEWED FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NO. 50-281

Replace the following pages of the Licenses with the attached revised pages. The revised pages are identified by amendment number and contained marginal lines indicating the areas of change.

Renewed Facility Operating License No. DPR-32

REMOVE	INSERT
3	3

Renewed Facility Operating License No. DPR-37

REMOVE	INSERT
3	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. 306 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:

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. 306 are hereby incorporated in this 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 54

F. Deleted by Amendment 59 and Amendment 65

G. Deleted by Amendment 227

H. Deleted by Amendment 227

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

LICENSE AMENDMENT REQUEST TO UPDATE OF THE LOSS-OF-COOLANT ACCIDENT

ALTERNATE SOURCE TERM DOSE ANALYSIS

AMENDMENT NO. 306 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-32

AMENDMENT NO. 306 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 December 3, 2020, as supplemented by letter dated May 13, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML20338A542 and ML21133A421, respectively), Virginia Electric and Power Company (the licensee) requested to make a change, as described in the Updated Final Safety Analysis Report (USFAR), to the Alternative Source Term (AST) analysis following a loss-of-coolant accident (LOCA) for the Surry Power Station, Units 1 and 2 (Surry) by increasing the containment depressurization profile and reducing refueling water storage tank (RWST) back leakage.

The supplement dated May 13, 2021, provided additional clarifying information that did not change the scope of the proposed change as described in the U.S. Nuclear Regulatory Commission (NRC) staff's original no significant hazards consideration published in the *Federal Register* on January 26, 2021 (86 FR 7112).

2.0 REGULATORY EVALUATION

2.1 Applicable Regulations

The NRC issued construction permits for Surry, Units 1 and 2, before May 21, 1971; consequently, Surry was not subject to the requirements in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria [GDC] for Nuclear Power Plants" (see SECY-92-223, "Resolution of Deviations Identified during the Systematic Evaluation Program," ADAMS Accession No. ML003763736 dated September 18, 1992). Accordingly, Surry Units 1 and 2 meet the intent of the GDC published in 1967 (draft GDC) as specified in the Surry UFSAR design criteria. The NRC staff considered the following regulations during its review of the licensee's proposed changes.

- The regulation in 10 CFR 50.54(o), which states that the "Primary reactor containments for water cooled power reactors, other than facilities for which the certifications required under §§ 50.82(a)(1) or 52.110(a)(1) of this chapter have been submitted, shall be subject to the requirements set forth in appendix J to this part."

- The regulation in 10 CFR 50.67, "Accident source term," which specifies in 10 CFR 50.67(b)(2) that the NRC may issue the amendment only if the applicant's analysis demonstrates with reasonable assurance that the dose criteria (i) An individual located at any point on the boundary of the exclusion area [EAB] for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE), (ii) An individual located at any point on the outer boundary of the low population zone [LPZ], who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) TEDE, and (iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) TEDE for the duration of the accident..

- The regulations in 10 CFR Part 50, Appendix A, "General Design Criterion for Nuclear Power Plants," Criterion 19, "Control room," which states "A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures."

  The regulations in 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," Option B, "Performance-Based Requirements," identifies the performance-based requirements and criteria for preoperational and subsequent periodic leakage-rate testing.

2.2 Regulatory Guidance

The NRC staff considered the regulatory guidance below during its review of the licensee's proposed changes. The NRC recognizes that NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," is written primarily for the review of GDC plants with Standard Technical Specifications and that Surry is a pre-GDC plant with custom TS. As such, the NRC used NUREG-0800 for guidance but based its approval on the current design and licensing basis for Surry.

- NUREG-0800, SRP, Section 2.3.4, “Short-Term Atmospheric Dispersion Estimates for Accidental Releases,” Revision 3, March 2007 (ADAMS Accession No. ML070730398).

- NUREG-0800, SRP, Section 6.2.6 “Containment Leakage Testing,” Revision 3, March 2007 (ADAMS Accession No. ML070620007).

- NUREG-0800, SRP, Section 6.5.2, “Containment Spray as a Fission Product Cleanup System,” Revision 4, March 2007 (ADAMS Accession No. ML070190178).

- NUREG-0800, SRP, Section 15.0.1, “Radiological Consequence Analyses Using Alternative Source Terms,” Revision 0, July 2000 (ADAMS Accession No. ML003734190).

- Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Rate Testing program," Revision 0, September 1995 (ADAMS Accession No. ML003740058).

- RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” Revision 0, July 2000 (ADAMS Accession No. ML003716792).

2.3 Current Licensing Bases (CLB)

The licensee stated in its submittal dated December 3, 2020, that the “current LOCA design basis radiological analysis appears in SPS [Surry’s] Updated Final Safety Analysis Report (UFSAR) [Surry Power Station, Units 1 & 2 - Revision 52 to Updated Final Safety Analysis Report, Chapter 14, Safety Analysis (EPID L-2020-LRO-0076) – Redacted] Section 14.5.5 [“Environmental Consequences of Loss-of-Coolant Accident (LOCA),” ADAMS Accession no. ML21208A367] and was submitted for approval in [letter dated March 2, 2018, ADAMS Accession No. ML18075A021] and was approved in [amendment No. 295/295 for Surry, Unit 1 and 2, by letter dated June 12, 2019, ADAMS Accession No. ML19028A384]. The analysis was performed using the RADTRAD-NAI code based on a core inventory derived with the ORIGEN-ARP code.”

In addition to USFAR Section 14.5.5, “Environmental Consequences of Loss-of-Coolant Accident (LOCA),” the NRC staff identified the following Surry UFSAR sections (ADAMS Package No. ML21208A006) to be relevant to its review of the licensee’s proposed changes:

- USFAR Section 5.3.4, “Vacuum System”
- USFAR Section 5.4, “Containment Design Evaluation”
- USFAR Section 14.5.5, “Environmental Consequences of Loss-of-Coolant Accident (LOCA)”
- USFAR Section 15.5.1, “Containment Structure”

The NRC staff also identified the following Surry, Unit 1 and 2, Technical Specifications (TSs) (ADAMS Accession Nos. ML052910358 and ML052910360, respectively) as relevant to its review of the licensee’s proposed changes:

- TS 3.4, “Spray Systems”
- TS 3.8, “Containment”
- TS 4.4, “Containment Tests”

3.0 TECHNICAL EVALUATION

The licensee's LAR proposed certain changes to Surry's AST analysis following a LOCA, by increasing the containment depressurization profile and reducing RWST back leakage. The licensee did not propose changes to the primary containment structure, heat removal systems, containment integrity (peak pressure and temperature) accident analyses, containment leak testing, or the Technical Specifications (TSs).

3.1 Increased Containment Depressurization Profile

The changes in containment leak rate assumptions affect the control room, EAB, and LPZ dose consequence analyses. As stated in License Amendment Nos. 250 and 249 for Surry Power Station, Unit Nos. 1 and 2, respectively (ADAMS Accession No. ML062920499), the current design basis LOCA analysis for determining peak containment pressure and temperature is based on:

- LOCA containment peak pressure < 45 pounds per square inch gauge (psig),
- LOCA containment pressure < 1.0 psig from 1 to 4 hours and < 0.0 psig after 4 hours, and
- LOCA containment temperature < 280 degrees Fahrenheit (°F)

These peak containment pressure conditions are fully described in UFSAR Section 14.5.5 "Environmental Consequences of Loss-of-Coolant Accident (LOCA)," and states, in part, that:

> Surry has a subatmospheric containment system. During the first hour following the accident containment pressure is analyzed to remain less than 45 psig. A 0.1 volume percent per day containment leak rate was used for the first hour after the LOCA, which corresponds to a maximum containment pressure of 45 psig. The original design criterion required that within one hour, containment pressure be calculated to return to subatmospheric conditions. This original design criterion was modified in conjunction with the analyses for implementation of the alternative source term and, subsequently, Generic Letter 2004-02. The modified criteria require that, following the LOCA, the containment pressure be less than 1.0 psig within 1 hour and less than 0.0 psig within 4 hours. Therefore, from 1 hour to 4 hours after a LOCA, a 0.029 volume percent per day leak rate was assumed, which corresponds to a maximum containment pressure of 1.0 psig. Beyond 4 hours, containment pressure is assumed to be less than 0.0 psig, terminating leakage from containment.

The licensee's proposed change to the LOCA AST dose consequences analysis assumes that the containment depressurizes to 2.0 psig within 1 hour and to less than 0.0 psig within 6 hours.

The CLB for UFSAR Chapter 14 Section 14.5.5.2 "RADTRAD-NAI Model for Surry LOCA Analysis," relative to time periods and their related containment leakage rates states, in part, that:

> The transport of radionuclides to the environment is modeled by specifying flow rates between the various volumes modeled. The mixing between the sprayed and unsprayed containment volumes was modeled based on 2 unsprayed volumes per hour. The containment leakage to the environment was modeled as

> 0.1 volume percent per day for the first hour. From 1 hour until 4 hours after the LOCA, a 0.029 volume percent per day leak rate was used. Beyond 4 hours, containment pressure is assumed to be less than 0.0 psig, terminating leakage from containment. The appropriate containment leakage rates, based upon time in the accident, were applied proportionately to both the sprayed and unsprayed containment volumes during the first four hours of the LOCA.

Table 1 below replicates part of Table 2-1 in the licensee's submittal dated December 3, 2020:

Table 1: Comparison of Proposed Changes to CLB valves

Parameter	CLB Value		Proposed Value	
Containment Leak Rate The containment leak rate was recalculated as a result of increasing the containment depressurization profile.	Time (hour (hr))	Rate (%vol/day)	Time (hr)	Rate (%vol/day)
	0.0 - 1.0	0.1	0.0 - 1.0	0.1
	1.0 - 4.0	0.029	1.0 - 6.0	0.04
	4.0 - 720	0.0	6.0 - 720	0.0
RWST Filtered Back Leakage The RWST filtered back leakage represents twice the allowable leak rate, and it was decreased to conserve analysis margin.	Time (hr)	Rate (cc/hr)	Time (hr)	Rate (cc/hr)
	0.5 - 720	36,000	0.5 - 720	18,000
Containment Volume The containment volume was updated to reflect the value used in the revised containment analysis.	1,863,000 cubic feet (ft^3)		1,819,000 ft^3	

The licensee's "Proposed Value" in Table 1 of this SE for the containment volume of 1,819,000 ft^3 is currently part of the CLB as reflected in UFSAR Table 5.4-17 "Key Parameters In The Containment Analysis" and UFSAR Table 14.5-2 "Containment Back Pressure Analysis Input Parameters Used For Best-Estimate LBLOCA Analysis." This value for containment volume was established as part of the CLB with Amendment Nos. 250 and 249 for Surry Power Station Units 1 and 2, respectively.

TS 4.4, Specification B.2.a reads "An overall integrated leakage rate of less than or equal to La [design leak rate], 0.1 percent by weight of containment air per 24 hours, at calculated peak pressure (Pa)." The Basis for TS 4.4 states:

> The containment is designed for a maximum pressure of 45 psig. The containment is maintained at a subatmospheric air partial pressure consistent with TS Figure 3.8-1 depending upon the cooldown capability of the Engineered

> Safeguards and will not rise above 45 psig for any postulated loss-of-coolant accident.

Section 3.5.1 "Containment Leakage Model," of the licensee's submittal dated December 3, 2020, provides the parameters used and the methodology employed to derive the proposed leakage rate of 0.04% containment volume/day between 1 and 6 hours after event initiation. The analysis used corollaries of the Ideal Gas Law to derive the containment volumetric leak rate as a function of containment pressure. The analysis modeled the leakage as compressible flow through an orifice sized to allow a flow equal to the design leak rate, L_a, of 0.1% of containment volume per day at a pressure of 45 psig.

The licensee concluded that the "The containment volumetric leak rates calculated based on this approach were found to be conservative and acceptable," per Section 2.1.1, "Containment Leakage" of the NRC's amendment issuance dated March 8, 2002 [ADAMS Accession No. ML020710159]." Section 2.1.1 "Containment Leakage" of the NRC staff's Safety Evaluation (SE) for that amendment states in part that:

> … the licensee calculated the containment leak rate as a function of containment pressure by assuming that containment leakage can be modeled as an orifice in incompressible flow. For the dose calculations, the licensee assumed that the containment pressure is 0.5 psig from 1 hour to 4 hours. The leak rate corresponding to 0.5 psig is obtained using the equation for incompressible orifice flow and is approximately 20 percent of L_a.
>
> The [NRC] staff disagrees that primary containment leakage can be modeled as flow through an orifice. Leaks from containment do not behave in a regular, predictable way. Leak flow paths will not, in general, remain unchanged for the pressure and temperature ranges of a LOCA. Furthermore, the number of leaks may increase or decrease as the pressure varies. However, the [NRC] staff considers a leak rate of 20 percent of L_a to be conservative for a containment pressure of 0.5 psig since the actual leakage path would be more tortuous than the flow through an orifice, and therefore has a higher flow resistance than flow through an orifice. The licensee's calculations also assumed that the temperature of the containment atmosphere remains constant as the pressure decreases. This is conservative since the actual temperature of the containment atmosphere would be reduced by the action of passive heat sinks and the spray and a lower temperature would result in a low flow rate. Therefore, the leakage value the licensee has used for the dose analysis, 20 percent of L_a is conservative and acceptable.

Based on the above, the NRC staff reviewed the parameters used and the methodology employed in Section 3.5.1, "Containment Leakage Model," of the licensee's submittal dated December 3, 2020. To gain a thorough understanding of any model conservatisms and non-conservatisms and to verify the accuracy of information contained in Table 3-5 "Containment Volumetric Leak Rate as Function of Containment Pressure" of the licensee's submittal dated December 3, 2020, the NRC staff performed a verification of approximately 25 percent of the containment volumetric leak rate cubic feet/minute (cfm) values contained in the table. In particular, the leakage rates values derived for the post LOCA containment pressures of 0.3, 0.5, 0.9, 2.0, 10.0 and 25.0 psig were verified for accuracy. From Table 3-5 of the licensee's submittal dated December 3, 2020, NRC notes that the containment volumetric leak rate of

0.504 cfm at a containment pressure of 2 psig corresponds to 0.399 L_a (i.e., 0.504 cfm/1.263 cfm).

The containment volumetric leak rate values displayed in Table 3-5 of the licensee's submittal dated December 3, 2020, were derived based on the maximum post-LOCA containment temperature of 280°F. The basis for using this value was established in Section 3.2 "Containment Analysis" of Surry's license amendment for Unit 1 and 2, Nos. 250 and 249, respectively, (ADAMS Accession No. ML062920499). In contrast, UFSAR Section 14.5.1 "Major Reactor Coolant System Pipe Ruptures (Large Break Loss-of-Coolant Accident)" does not clearly substantiate the basis for using 280°F. However, the use of that value is reasonable based on UFSAR Figure 5.4-2 "Containment Vapor Temperature DEHLG Peak Pressure Analysis," where DEHLG stands for "double-ended hot leg guillotine." In UFSAR Figure 5.4-2, the containment vapor temperature reaches a maximum of approximately 275°F within 20 seconds of the accident's onset before starting its descent.

The volumetric leakage rates contained in Table 3-5 of the licensee's submittal dated December 3, 2020, were derived by assuming a constant containment temperature of 280°F. A review of the equations contained in Section 3.5.1, "Containment Leakage Model" of the licensee's submittal dated December 3, 2020, shows this assumption to be conservative with respect to obtaining the most limiting containment volumetric leakage rates for each containment pressure listed in Table 3-5 of the licensee's submittal dated December 3, 2020.

Based on the NRC staff's SE for the amendment dated March 8, 2002, and based on the above evaluation, the NRC staff observes that a leakage rate of 40 percent of L_a (i.e., 0.504 cfm) for a LOCA containment pressure that decays to 2.0 psig within one hour of LOCA onset and decays to 0 psig within next 6 hours, as shown in Table 1 above, to be conservative with sufficient margin to account for any model uncertainties. Based on requirements of Surveillance Requirement TS 4.4, "Containment Tests," the NRC staff concludes that the licensee's revision of the LOCA AST dose consequences analysis based on an increased containment depressurization profile is acceptable and will continue to meet the requirements of 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B, "Performance-Based Requirements."

3.2 Reduction of the RWST Back Leakage Limit

In Section 2.2, "Reduced RWST Back Leakage Limit," of the licensee's submittal dated December 3, 2020, the licensee stated:

> The current licensing basis employs an allowable RWST back leakage rate of 18,000 cc/hr [cubic centimeters per hour] (analyzed at twice the allowable rate). The proposed change would reduce the allowable RWST back leakage rate to 9,000 cc/hr (analyzed at twice the allowable rate). This change is intended to conserve LOCA dose consequence analysis margins.

UFSAR Section 14.5.5.2 "RADTRAD-NAI Model for Surry LOCA Analysis," states that the RWST back leakage is the modeled ESF [Engineered Safety Feature] system leakage through ECCS [Emergency Core Cooling System] check valves into the RWST after switching to the recirculation cooling mode. For the ECCS and RWST leakage dose consequence analysis, the iodine activity released from the fuel is assumed to be in the sump solution until removed by radioactive decay or leakage from the ECCS and RWST. The revised dose consequence analysis is reflected in Table 1-1 "Analysis Parameters," of the licensee's submittal dated December 3, 2020. That table indicates that the Initiation of ECCS Leakage from the Outside

Recirculation Spray (ORS) System takes place at 15 minutes into the event. In agreement, UFSAR Section 14.5.5.2 “RADTRAD-NAI Model for Surry LOCA Analysis” indicates that the earliest start of the Recirculation Spray System during the LOCA analysis takes place between 0.25 - 0.50 hours into the event.

UFSAR Section 14.5.5, “Environmental Consequences of Loss-of-Coolant Accident (LOCA),” states, in part, that:

> The ESF leakage was assumed to be 2 times the total allowable ECCS leakage of 15,000 cc/hour (Section 14.5.5.3, [“Results of Dose Calculations for LOCA”]) and the total allowable back-leakage into the RWST of 18,000 cc/hour. The RWST release to the atmosphere is modeled at 1000 cfm through the safeguards building and out ventilation vent No. 2. Filtration by the auxiliary building ventilation system was credited for the portion of the ECCS leakage that occurs in the Safeguards Building, primarily the 3,000 cc/hour of Outside Recirculation Spray System leakage identified in Section 14.5.5.3, and the RWST backleakage. The release of radionuclides to the environment was determined both for containment leakage and Engineered Safety Feature components leakage.

UFSAR Section 14.5.5.3 “Results of Dose Calculations for LOCA” reads in part:

> Filtration by the auxiliary building ventilation system was credited for the portion of the ECCS leakage that occurs in the Safeguards Building, primarily the Outside Recirculation Spray System leakage and RWST back-leakage. Filtration was not credited for the portion of the ECCS leakage that occurs in the Auxiliary Building, primarily the SI [Safety Injection] and Charging System leakage. However, only certain areas (the charging pump cubicles and safeguards) are provided with dedicated exhaust paths to the filters. This has the potential to lead to releases to the environment that may bypass the auxiliary building filters. However, since no filtration credit is taken (in areas without dedicated exhaust paths to filters) and all atmospheric dispersion factors were modeled as ground releases (Section 14.5.5.1, [“Methodology to Determine Atmospheric Dispersion Factors, Control Room Occupancy, and Breathing Rates”]) this analysis remains conservative. The total allowable RWST back-leakage is 18,000 cc/hour. The maximum allowable unfiltered leakage is limited to the SI and Charging Systems leakage of 12,000 cc/hour. The total allowable ECCS leakage of 15,000 cc/hour includes SI and Charging Systems leakage of 12,000 cc/hour and Outside Recirculation Spray System leakage of 3,000 cc/hour.

Table 1, in Section 3.1 of this safety evaluation, lists the comparison values of the CLB and the proposed change to the RWST filtered back-leakage.

The NRC staff reviewed the licensee’s proposed change that would reduce the allowable RWST back-leakage rate to 9,000 cc/hr. Since the current licensing basis LOCA dose calculation results (i.e., UFSAR Section 14.5.5.3) indicate “The total allowable RWST back-leakage is 18,000 cc/hour,” the NRC staff concludes that the current licensing basis LOCA dose calculation results conservatively bound the licensee’s proposed change in RWST back leakage rate. Based on the above and requirements of Surveillance Requirement TS 4.4, “Containment Tests,” the NRC concludes the proposed changes would continue to meet the requirements of

10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, “Performance-Based Requirements,” and are, therefore, acceptable.

3.3 Revised Loss of Coolant Dose Consequence Analysis

The NRC staff assessed the proposed changes associated with the licensee’s revised dose consequence LOCA analysis. The current licensing basis LOCA design-basis radiological analysis is described in the Surry UFSAR Section 14.5.5, ”Environmental Consequences of Loss-of-Coolant Accident (LOCA),” and was approved by the NRC by letter dated June 12, 2019, "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)," (ADAMS Accession No. ML19028A384). The revised analysis includes an increase in the containment depressurization profile and a reduction in the allowable back leakage to the RWST. The licensee stated in the original submittal that there are no changes to any CLB atmospheric dispersion factors and no changes to the CLB core inventory were proposed as part of this amendment.

The revised analysis supports an increase in the containment depressurization profile. The CLB assumption is that the containment will depressurize to 1.0 psig within 1 hour and less than 0.0 psig within 4 hours. The proposed change will require the containment to depressurize to 2.0 psig within 1 hour and less than 0.0 psig within 6 hours. This change will be reflected in the LOCA dose consequence analysis by increasing the allowable containment leak rate. To compensate for the increase in calculated dose resulting from the increase in the allowable containment leakage, the licensee proposed a decrease in the allowable RWST back-leakage. These changes to the CLB dose assessment are shown in Table 2-1, “Comparison of Proposed Changes,” of the licensee’s submittal dated December 3, 2020.

The licensee included all the pertinent radiological data used in the LOCA dose consequence analysis in Table 2-1. In its review, the NRC staff limited its focus on the significant changes to the CLB parameters; namely the changes to the containment pressure response and the changes to the RWST back leakage assumptions. The NRC staff assessed these changes to the CLB dose assessment and concludes that the changes in RWST back leakage do not fully compensate for the increase in containment leakage. The results indicate a small increase in the calculated LOCA dose consequences; however, the calculated doses remain below the regulatory acceptance criteria. The NRC reviewed Section 3.6, “Results,” and Table 3-6, LOCA Dose Acceptance Criteria,” of the licensee’s submittal dated December 3, 2020. As shown in Table 2 of this safety evaluation, the changes in the current licensing basis LOCA analysis assumptions result in relatively minor changes in the calculated dose consequences.

Table 2: Comparison of Revised LOCA Dose to CLB Values and Acceptance Criteria

Location	CLB TEDE (rem)	Revised TEDE (rem)	Acceptance Criteria (rem)
EAB	9.1	10.6	25.0
LPZ	1.9	2.1	25.0
Control Room	4.4	4.7	5.0

The NRC staff notes that the changes in the control room dose consequences meet the guidelines for requesting NRC staff approval as described in the NRC-endorsed NEI 96-07, Revision 1, "Guidelines for 10 CFR 50.59 Implementation" (ADAMS Accession No. ML003771157). The CLB control room dose margin to the acceptance criterion of 5 rem is 600 millirem (mrem). Since the increase in calculated dose of 300 mrem exceeds one tenth of the CLB dose margin, the guidance for meeting 10 CFR 50.59 instructs the licensee to submit a license amendment NRC staff approval.

The NRC staff found that the licensee's analysis was performed using approved methodologies, reasonable assumptions, and inputs consistent with RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Revision 0. The NRC staff reviewed the licensee's evaluation of the radiological consequences resulting from the postulated LOCA, and determined that the evaluation was acceptable; therefore, the staff concludes that the radiological consequences at the EAB, LPZ, and control room continue to meet the acceptance criteria provided in 10 CFR 50.67(b)(2), and 10 CFR Part 50, Appendix A, Criterion 19, since the proposed calculated doses remain below the regulatory acceptance criteria of 25 rem at the EAB and LPZ, and 5 rem in the control room, respectively.

4.0 TECHNICAL CONCLUSION

The NRC staff reviewed the proposed submittal dated December 3, 2020, and its supplement dated May 13, 2021, that would modify Surry's AST analysis following a LOCA by increasing the containment depressurization profile and reducing RWST back leakage. Based on the above, the NRC staff concludes that the proposed change is acceptable and meets the requirements of 10 CFR 50.54(o), and 10 CFR Part 50, Appendix J. The NRC staff also concludes that the radiological consequences at the EAB, LPZ, and control room would continue to meet the acceptance criteria provided in 10 CFR 50.67, and 10 CFR Part 50, Appendix A, Criterion 19.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the NRC staff notified an official from the Virginia Division of Radiological Health of the proposed issuance of the amendment. On October 8, 2021, the State official confirmed that the Commonwealth of Virginia had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission previously issued a proposed finding that the amendments involve no significant hazards consideration, published in the *Federal Register* on January 26, 2021 (86 FR 7112), and has received no public comments on this finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Under 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

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

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Date: November 19, 2021

SUBJECT: SURRY POWER STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 306 AND 306 TO UPDATE OF THE LOSS OF COOLANT ACCIDENT ALTERNATE SOURCE TERM DOSE ANALYSIS (EPID L-2020-LLA-0258) DATED NOVEMBER 19, 2021

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