



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

June 26, 2015

Mr. David A. Heacock
President and Chief Nuclear Officer
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

**SUBJECT: SURRY POWER STATION, UNIT NOS. 1 AND 2, ISSUANCE OF AMENDMENTS
REGARDING CLARIFICATION OF REACTOR COOLANT SYSTEM HEATUP
AND COOLDOWN LIMITATION TECHNICAL SPECIFICATION FIGURES
(TAC NOS. MF4701 AND MF4702)**

Dear Mr. Heacock:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 285 to Renewed Facility Operating License No. DPR-32 and Amendment No. 285 to Renewed Facility Operating License No. DPR-37 for the Surry Power Station (Surry), Unit Nos. 1 and 2, respectively. The amendments change the Technical Specifications (TSs) in response to your application dated June 3, 2014, as supplemented by letter dated February 4, 2015.

These amendments revise the Surry TS figures 3.1-1 and 3.1-2, "Surry Units 1 and 2 Reactor Coolant System Heatup Limitations" and "Surry Units 1 and 2 Reactor Coolant System Cooldown Limitations," respectively, for clarification and to be fully representative of the allowable operating conditions during Reactor Coolant System (RCS) startup and cooldown evolutions. The revisions to TS Figures 3.1-1 and 3.1-2 include: (1) the extension of the temperature axes to reflect temperatures up to RCS full power operation; (2) the extension of the pressure axes to less than 0 pounds per square inch gage (psig) to bound RCS conditions when vacuum-assist fill of the RCS loops is performed; and (3) the addition of information regarding the reactor boltup temperature.

D. Heacock

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

Sincerely,

A handwritten signature in black ink, appearing to read "Karen Cotton". The signature is fluid and cursive, with the first name "Karen" and last name "Cotton" clearly distinguishable.

Karen Cotton, 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. 285 to DPR-32
2. Amendment No. 285 to DPR-37
3. Safety Evaluation

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**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. 285
Renewed License No. DPR-32

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Virginia Electric and Power Company (the licensee) dated June 3, 2014, as supplemented by letter dated February 4, 2015, 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. 285, 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 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION



Robert J. Pascarelli, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: June 26, 2015



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. 285
Renewed License No. DPR-37

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Virginia Electric and Power Company (the licensee) dated June 3, 2014, as supplemented by letter dated February 4, 2015, 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. 285, 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 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION



Robert J. Pascarelli, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes Renewed Facility
Operating License No. DPR-37
and the Technical Specifications

Date of Issuance June 26, 2015

ATTACHMENT TO
LICENSE AMENDMENT NO. 285
RENEWED FACILITY OPERATING LICENSE NO. DPR-32
DOCKET NO. 50-280
AND
LICENSE AMENDMENT NO. 285
RENEWED FACILITY OPERATING LICENSE NO. DPR-37
DOCKET NO. 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

License

License No. DPR-32, page 3
License No. DPR-37, page 3

TSs

TS 3.1-7
TS 3.1-9
TS 3.1-12
Figure 3.1-1
Figure 3.1-2

Insert Pages

License

License No. DPR-32, page 3
License No. DPR-37, page 3

TSs

TS 3.1-7
TS 3.1-9
TS 3.1-12
Figure 3.1-1
Figure 3.1-2

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. 285, 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 byproduct 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 285 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

3. The pressurizer heatup and cooldown rates shall not exceed 100°F/hr. and 200°F/hr., respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

Basis

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G.

- 1) The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.1-1 and 3.1-2.
 - a) Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.
 - b) Figures 3.1-1 and 3.1-2 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.
 - c) Vacuum-assist fill of the Reactor Coolant System loops in COLD SHUTDOWN or REFUELING SHUTDOWN is an acceptable condition since the resulting pressure/temperature combination is located in the Acceptable Operation region of Figures 3.1-1 and 3.1-2.
- 2) These limit lines shall be calculated periodically using methods provided below.
- 3) The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT} , at the end of 48 Effective Full Power Years (EFPY) for Units 1 and 2. The heatup and cooldown limit curves were previously calculated using the most limiting value of RT_{NDT} (228.4°F) which occurred at the 1/4-T, 0° azimuthal location in the Unit 1 intermediate-to-lower shell circumferential weld. Subsequently, the reactor vessel material property basis was amended based upon new data which showed that the most limiting value of RT_{NDT} (222.5°F) at 48 EFPY occurs at the 1/4-T, 0° azimuthal location in the Unit 2 intermediate-to-lower shell circumferential weld. The revised limiting material property (i.e., Unit 2 RT_{NDT} of 222.5°F) justified continued use of the existing heatup and cooldown limit curves (based on the Unit 1 RT_{NDT} of 228.4°F) to 48 EFPY for Units 1 and 2. The limiting RT_{NDT} at the 1/4-T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. This ensures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

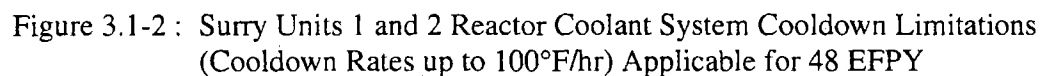
The reactor vessel materials have been tested to determine their initial RT_{NDT} ; the results are presented in UFSAR Section 4.1. Reactor operation and resultant fast neutron (E greater than 1 MEV) irradiation can cause an increase in the RT_{NDT} . Therefore, an adjusted reference temperature, based upon the copper and nickel content of the material and the fluence was calculated in accordance with the recommendations of Regulatory Guide 1.99, Revision 2 "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curves of Figures 3.1-1 and 3.1-2 include predicted adjustments for this shift in RT_{NDT} at the end of 48 EFPY for Units 1 and 2 (as well as adjustments for location of the pressure sensing instrument).

Surveillance capsules will be removed in accordance with the requirements of ASTM E185-82 and 10 CFR 50, Appendix H. The surveillance specimen withdrawal schedule is shown in the UFSAR. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure, or when the service period exceeds 48 EFPY for Units 1 and 2 prior to a scheduled refueling outage.

The reactor boltup temperature is defined in 10 CFR 50, Appendix G as "The highest reference temperature of the material in the closure flange region that is highly stressed by the bolt preload." The reactor vessel may be bolted up at a temperature greater than the initial RT_{NDT} of the material stressed by the boltup (e.g., the vessel flange). As noted on Figures 3.1-1 and 3.1-2, the limiting boltup temperature is 10°F. An administrative minimum boltup temperature limit greater than 10°F is imposed in station procedures to ensure the Reactor Coolant System temperatures are sufficiently high to prevent damage to the reactor vessel closure head/vessel flange during the removal or installation of reactor vessel head bolts. The limiting boltup temperature and the administrative minimum boltup temperature limit are in effect when the reactor vessel head bolts are under tension.

References

- (1) UFSAR, Section 4.1, Design Bases





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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 285 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-32

AND

AMENDMENT NO. 285 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 June 3, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14160A607), as supplemented by letter dated February 4, 2015 (ADAMS Accession No. ML15041A720), Virginia Electric and Power Company (Dominion, the licensee) submitted a license amendment request (LAR) to the U.S. Nuclear Regulatory Commission (NRC, the Commission), which requested changes to the Surry Power Station, Unit Nos. 1 and 2 (Surry 1 and 2), Technical Specifications (TSs). The supplemental letter dated February 4, 2015, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on September 30, 2014 (79 FR 58812).

The proposed changes would revise TS Figures 3.1-1 and 3.1-2, "Surry Units 1 and 2 Reactor Coolant System Heatup Limitations" and "Surry Units 1 and 2 Reactor Coolant System Cooldown Limitations," respectively, for clarification and to be fully representative of the allowable operating conditions during Reactor Coolant System (RCS) startup and cooldown evolutions. The proposed revisions to TS Figures 3.1-1 and 3.1-2 include: (1) the extension of the temperature axes to reflect temperatures up to RCS full power operation; (2) the extension of the pressure axes to less than 0 pounds per square inch gage (psig) to bound RCS conditions when vacuum-assist fill of the RCS loops is performed; and (3) the addition of information regarding the reactor boltup temperature.

2.0 REGULATORY EVALUATION

The NRC has established requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 to protect the integrity of the reactor coolant pressure boundary (RCPB) in nuclear power plants. The NRC staff evaluates the acceptability of a facility's pressure and temperature (P-T) limits based on the following NRC regulations and guidance: 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements"; 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements"; Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials"; Generic Letter (GL) 92-01, Revision 1, "Reactor Vessel Structural Integrity"; and GL 92-01, Revision 1, Supplement 1, "Reactor Vessel Structural Integrity." Appendix G to 10 CFR Part 50 requires that facility P-T limits for the reactor vessel (RV) be at least as conservative as those obtained by applying the linear elastic fracture mechanics methodology of Appendix G to Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). Additionally, Appendix G to 10 CFR Part 50 imposes minimum RV closure flange temperatures when system pressure is at or above 20% of the preservice hydrostatic test pressure. Appendix H to 10 CFR Part 50 establishes requirements related to facility RV material surveillance programs. RG 1.99, Revision 2, contains guidance on methodologies that the NRC staff considers acceptable for determining the increase in transition temperature and the decrease in upper shelf energy resulting from neutron irradiation.

3.0 TECHNICAL EVALUATION

3.1 Licensee's Evaluation

The licensee's June 3, 2014, LAR proposes the following changes to the TS figures for the Surry, 1 and 2, P-T limit curves:

- The temperature axes are extended from 400 degrees Fahrenheit (°F) to 650°F, which bounds temperatures up to RCS full power operation.
- The pressure axes are extended from 0 psig to -14.70 psig to bound RCS conditions to support vacuum-assist fill of the RCS loops.
- The phrase, "Limiting Boltup Temperature Surry 1 Initial RT_{NDT} Closure Flange Region: 10°F," is added, since the current figures do not address boltup temperature.

According to the licensee, the extension of the temperature axes reflects the maximum temperature corresponding to the reactor vessel design temperature of 650°F. The pressure axes are extended from 0 psig to 14.70 psig to bound RCS conditions to support vacuum-assist fill of the RCS loops. An engineering evaluation found that the RCS loop piping and steam generator tubing maintain their structural integrity "with considerable margin" during vacuum-assist loop fill.

The LAR also proposes the following revisions to the TS 3.1.B Basis:

- A statement indicating that vacuum-assist fill of the RCS loops in Cold Shutdown or Refueling Shutdown is an acceptable condition and is located in the Acceptable Operation region of the proposed TS Figures 3.1-1 and 3.1-2.
- A description of the reactor boltup temperature.
- A clarification regarding the limiting value of the nil-ductility reference temperature (RT_{NDT}).

The proposed revision to the TS Basis states that the reactor boltup temperature is defined in 10 CFR Part 50, Appendix G, as "the highest reference temperature of the material in the closure flange region that is highly stressed by the bolt preload."

3.2 NRC Staff's Evaluation

Regarding ferritic RCPB components that are not part of the RV beltline shell region, 10 CFR Part 50, Appendix G, Paragraph IV.A, states:

The pressure-retaining components of the reactor coolant pressure boundary that are made of ferritic materials must meet the requirements of the [ASME Code, Section III], supplemented by the additional requirements set forth in [10 CFR Part 50, Appendix G, Paragraph IV.A.2, "Pressure-Temperature Limits and Minimum Temperature Requirements"].

Therefore, 10 CFR Part 50, Appendix G, requires that P-T limits be developed for the ferritic materials in the RV beltline, as well as ferritic materials outside of the RV beltline. Further, 10 CFR Part 50, Appendix G, requires that all ferritic RCPB components meet the applicable ASME Code, Section III, requirements. The relevant ASME Code, Section III, requirements that will affect the P-T limits are the lowest service temperature requirement of NB-2332(b) for piping, pumps, and valves, and the fracture toughness requirements of NB-3211(d) for vessels.

RV nozzles, penetrations, and other discontinuities may exhibit significantly higher stresses than those for the RV beltline shell region. These higher stresses can potentially result in more restrictive P-T limits, even if the RT_{NDT} for these components is not as high as that of RV beltline shell materials that have simpler geometries.

Therefore, in a request for additional information (RAI), the NRC staff requested that the licensee describe how the P-T limit curves and the methodology used to develop these curves considered RV materials, consistent with the requirements of 10 CFR Part 50, Appendix G.

In its February 4, 2015 response (ADAMS Accession No. ML15041A720) to the RAI, the licensee stated that the fluence model used to develop the P-T limit curves was limited to the ferritic pressure boundary materials in the RV beltline region which does not include the inlet and outlet nozzles. The licensee went on to state that "the axial extent of the fluence model to encompass the reactor vessel inlet and outlet nozzles" will be completed by June 30, 2015, and "following completion of the fluence model revision, Dominion will be able to validate whether the inlet and outlet nozzles will have a neutron fluence of less than 1×10^{17} n/cm² (E > 1 MeV) at the end of the licensed operating period."

The approved P-T limit curves cover the 60-year current operating period and are applicable up to 48 effective full power years (EFPY) for Surry, 1 and 2. The NRC staff concludes that the differences in EFPY for the current operating period in comparison to the corresponding approved P-T limits provide an adequate margin so that if the revised neutron fluence model indicates that the neutron fluence exposure of the inlet and outlet nozzles is greater than 1×10^{17} n/cm² at the end of the period of extended operation, the licensee will notify the NRC accordingly and provide a schedule for revising the TS P-T limits curves for Surry, 1 and 2, if necessary.

The licensee's June 3, 2014 LAR proposed changes to the TS figures 3.1-1 and 3.1-2 for the P-T limit curves for Surry, 1 and 2, that extend the pressure axis to -14.7 psig and the temperature axis to 650°F. The revisions clarify that pressure limits are considered to be met for pressures that are below 0 psig (i.e., up to and including full vacuum conditions). Certain operating practices, such as vacuum fill operations for the RCS, are common practice in both boiling water reactors and pressurized water reactors and can result in system pressures below 0 psig. Although the RV is not specifically evaluated for negative pressures, the magnitude of the negative pressure is limited to one atmosphere and thus is not expected to result in significant stresses. Extension of the temperature axis to 650°F bounds the operating temperatures over the range of RCS full power operation. Therefore, the NRC staff finds the revisions to the Surry, 1 and 2, TS extending the pressure and temperature axes to be acceptable.

The licensee proposed to insert a clarification of the limiting RT_{NDT} into the basis for TS 3.1. The approved P-T limits were calculated using a limiting RT_{NDT} of 228.4°F which corresponds to the $\frac{1}{4}$ -T, 0° azimuthal location in the Surry Unit 1 intermediate-to-lower shell circumferential weld. By letter dated May 31, 2011 (ADAMS Accession No. ML11110A111), the NRC staff approved the licensee's May 6, 2010, LAR (ADAMS Accession No. ML101310604) extending the cumulative core burnup applicability limit from 28.8 EFPY and 29.4 EFPY for Surry Units 1 and 2, respectively, to 48 EFPY for both units. In the request, the licensee stated "considering the 60-year fluence projections corresponding to 48 EFPY, and the Linde 80 weld material properties per Topical Report BAW-2308-2-A... the limiting $\frac{1}{4}$ -T RT_{NDT} value is 222.5°F for Surry Unit 2 Intermediate to Lower Shell Circumferential weld." This determination does not require a change in the current P-T limits due to their conservatism; therefore, the NRC staff finds this clarification to be acceptable.

As revised, Figures 3.1-1 and 3.1-2 would each contain an additional note stating, "Limiting Boltup Temperature Surry 1 Initial RT_{NDT} Closure Flange Region: 10°F." The initial RT_{NDT} values of the Surry Units 1 and 2 vessel flanges are consistent with previous submittals, including Updated Final Safety Analysis Report (UFSAR) Table 4.1-14, "Unit 1 Reactor Pressure Vessel Toughness Data (Unirradiated)" and Table 4.1-15, "Unit 2 Reactor Pressure Vessel Toughness Data (Unirradiated)." Section 3.0 of the LAR states, "An administrative minimum boltup temperature of greater than 10°F is imposed in station procedures to ensure the RCS temperatures are sufficiently high to prevent damage to the closure head/vessel flange during the removal or installation of reactor vessel head bolts." The P-T limits for Surry Units 1 and 2 were developed following WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" (ADAMS Accession No. ML050120209). WCAP-14040-A, Section 2.10, "Minimum Boltup Temperature" states, "the RT_{NDT} is calculated in accordance with the methods described in Branch Technical Position MTEB 5-2. The Westinghouse position is that the minimum boltup temperature be no lower than 60°F. Thus, the minimum boltup temperature should be 60°F or the material RT_{NDT} whichever is higher." Therefore, the administrative minimum boltup temperature, reflected in the station procedure figures for Surry, 1 and 2, of 80°F meets the requirements of Appendix G to 10 CFR Part 50.

3.2 NRC Staff's Conclusion

The NRC staff concludes that the proposed changes to the Surry, 1 and 2, TS Figures 3.1-1, and 3.1-2 and to the TS 3.1.B Basis meet the requirements of 10 CFR Part 50, Appendix G. Therefore, the changes to the TSs are acceptable. In addition, the NRC staff concludes that there is sufficient margin regarding the P-T limits to allow completion of the revised neutron fluence model.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Virginia State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding as published in the *Federal Register* on September 30, 2014 (79 FR 58812). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 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.

Principal Contributor: Austin Young NRR/DE/EVIB

Date: June 26, 2015

D. Heacock

- 2 -

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

Sincerely,

/RA/

Karen Cotton, 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. 285 to DPR-32
2. Amendment No. 285 to DPR-37
3. Safety Evaluation

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*concurring by SE

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