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

May 31, 2011

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 REACTOR VESSEL HEATUP AND COOLDOWN CURVES FOR 48 EFFECTIVE FULL-POWER YEARS (TAC NOS. ME3920 AND ME3921)

Dear Mr. Heacock:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 274 to Renewed Facility Operating License No. DPR-32 and Amendment No. 274 to Renewed Facility Operating License No. DPR-37 for the Surry Power Station, Unit Nos. 1 and 2 (Surry 1 and 2), respectively. The amendments change the Technical Specifications in response to your application dated May 6, 2010.

These amendments revise the pressure and temperature limit curves to provide new limits that are valid to 48 effective full-power years for Surry 1 and 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,

Karen Cotton

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. 274 to DPR-32
2. Amendment No. 274 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. 274
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 May 6, 2010, 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. 274 , 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

G K

Gloria Kulesa, 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: May 31, 2011

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. 274
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 May 6, 2010, 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. 274 , 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

Gloria Kulesa, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: May 31, 2011

ATTACHMENT

TO LICENSE AMENDMENT NO. 274

RENEWED FACILITY OPERATING LICENSE NO. DPR-32

DOCKET NO. 50-280

AND

TO LICENSE AMENDMENT NO. 274

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	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 3.1-9	TS 3.1-9
TS 3.1-11	TS 3.1-11
Fig 3.1-1	Fig 3.1-1
Fig 3.1-2	Fig 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. 274 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. 274 , 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

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 most limiting value of RT_{NDT} (222.5°F) occurs at the 1/4-T, 0° azimuthal location in the Unit 2 intermediate-to-lower shell circumferential weld. 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.

K_{It} is the stress intensity factor caused by the thermal gradients

K_{IR} is provided by the code as a function of temperature relative to the RT_{NDT} of the material.

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{It}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

The heatup limit curve, Figure 3.1-1, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour. The cooldown limit curves of Figure 3.1-2 are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The cooldown limit curves are valid for cooldown rates up to 100°F/hr. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of 48 EFPY for Units 1 and 2. The adjusted reference temperature was calculated using materials properties data from the B&W Owners Group Master Integrated Reactor Vessel Surveillance Program (MIRVSP) documented in the most recent revision to BAW-1543 and reactor vessel neutron fluence data obtained from plant-specific analyses.

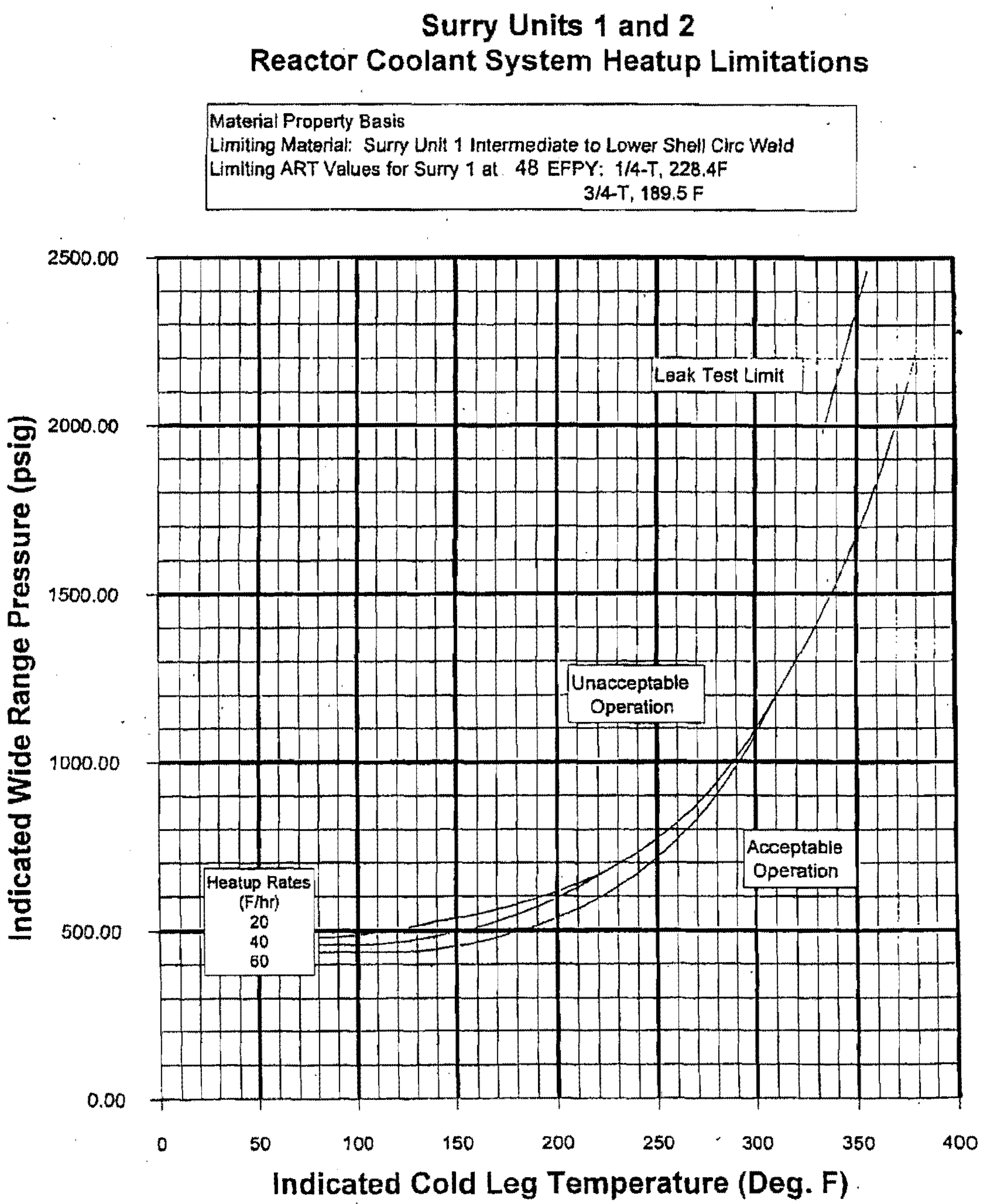

Figure 3.1-1 : Surry Units 1 and 2 Reactor Coolant System Heatup Limitations (Heatup Rates up to 60°F/hr) Applicable for 48 EFPY

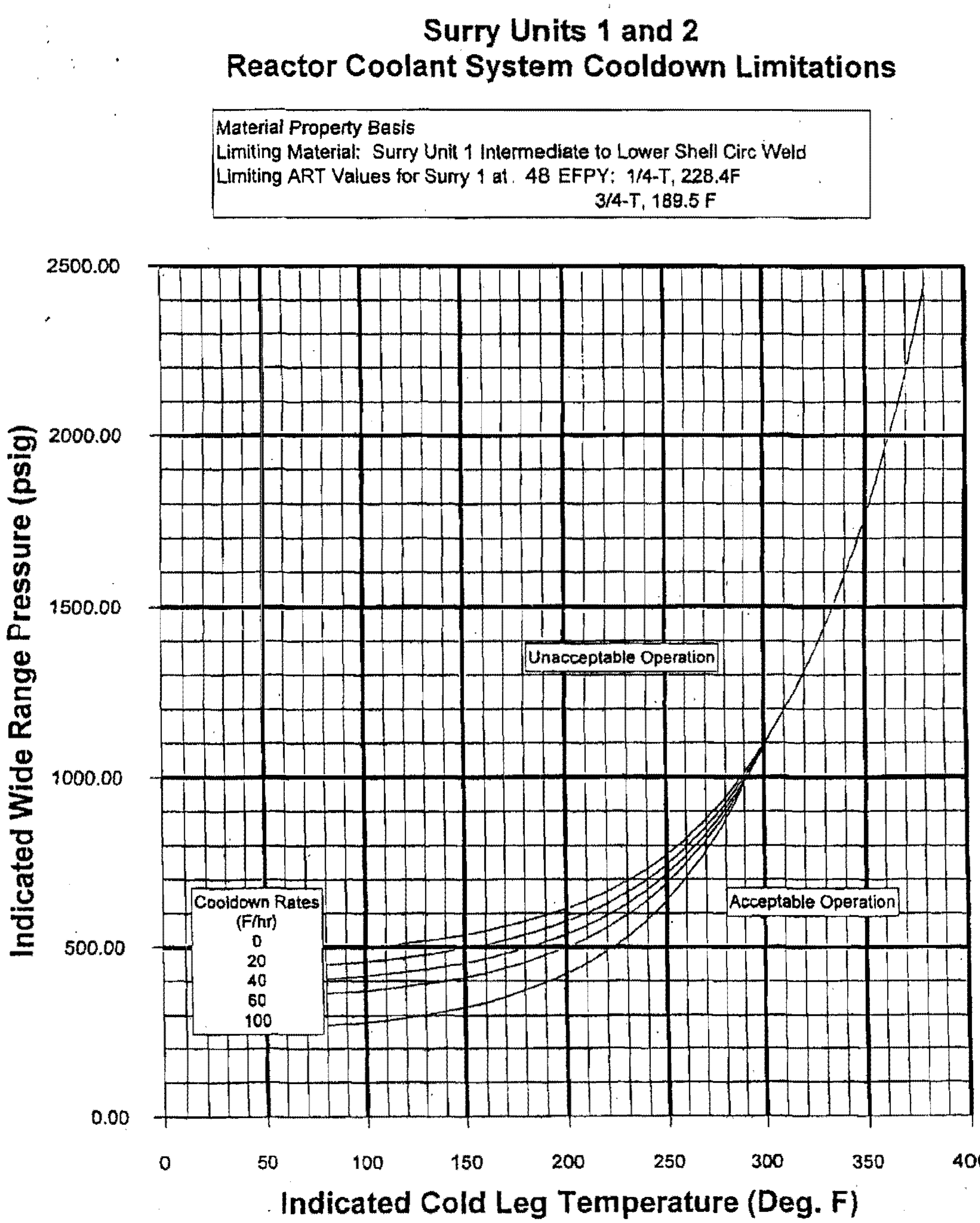

Figure 3.1-2 : Surry Units 1 and 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for 48 EFPY

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 274 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-32

AND

AMENDMENT NO. 274 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 May 6, 2010,[1] Virginia Electric and Power Company (VEPCO, the licensee) submitted a request for changes to the Surry Power Station, Unit Nos. 1 and 2 (Surry 1 and 2), Technical Specifications (TSs).

The proposed changes would revise the pressure-temperature (P-T) limit curves to provide new limits that are valid to 48 effective full power years (EFPY) for Surry 1 and 2.

2.0 REGULATORY EVALUATION

The Nuclear Regulatory Commission (NRC, the Commission) 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 in nuclear power plants. The staff evaluates the P-T limit curves based on the following NRC regulations and guidance: 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements"; Generic Letter (GL) 88-11,[2] "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations"; GL 92-01, Revision 1,[3] "Reactor Vessel Structural Integrity"; GL 92-01, Revision 1, Supplement 1,[4] Regulatory Guide (RG) 1.99, Revision 2,[5] "Radiation Embrittlement of Reactor Vessel Materials"; and Standard Review Plan (SRP), Revision 2, Section 5.3.2.[6] Appendix G to 10 CFR Part 50 requires that P-T limit curves be at least as conservative as those obtained by applying the

[1] Agencywide Documents Access and Management System (ADAMS) Accession No. ML101310604, May 6, 2010
[2] ADAMS Accession No. ML031150357, July 12, 1988
[3] ADAMS Accession No. ML031070438, March 6, 1992
[4] ADAMS Accession No. ML031070449, May 19, 1995
[5] ADAMS Accession No. ML003740284, May 1988
[6] ADAMS Accession No. ML080940723, March 2007

methodology of Appendix G to Section XI of the American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code* (ASME B&PV Code). Appendix G to 10 CFR Part 50 also provides minimum temperature requirements that must be considered in the development of the P-T limit curves. GL 88-11 advised licensees that the NRC staff would use RG 1.99, Revision 2 to review P-T limit curves. RG 1.99, Revision 2 contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy (USE) resulting from neutron radiation. GL 92-01, Revision 1, requested that licensees submit their reactor pressure vessel (RPV) materials property data for review. GL 92-01, Revision 1, Supplement 1 requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations.

SRP Section 5.3.2 provides an acceptable method of determining the P-T limit curves for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics methodology of Appendix G to Section XI of the ASME B&PV Code. The basic parameter of this methodology is the stress intensity factor K_I, which is a function of the stress state and flaw configuration. ASME B&PV Code, Section XI, Appendix G requires a safety factor of 2.0 on stress intensities resulting from reactor pressure during normal and transient operating conditions, and a safety factor of 1.5 on these stress intensities for hydrostatic testing curves. The flaw postulated in the ASME B&PV Code, Section XI, Appendix G has a depth that is equal to 1/4 of the RPV beltline thickness and a length equal to 1.5 times the RPV beltline thickness. The critical locations in the RPV beltline region for calculating heatup and cooldown P-T limit curves are the 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively. The methodology found in Appendix G to Section XI of the ASME Code requires that licensees determine the adjusted reference temperature (ART or adjusted RT_{NDT}) by evaluating material property changes due to neutron radiation. The ART is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}) and a margin term. The ΔRT_{NDT} is a product of a chemistry factor (CF) and a fluence factor. The CF is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Revision 2, or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the CF was determined using the tables in RG 1.99, Revision 2, or surveillance data. The margin term is used to account for uncertainties in the values of the initial RT_{NDT}, the copper and nickel contents, the neutron fluence and the calculational procedures. RG 1.99, Revision 2 describes the methodology to be used in calculating the margin term.

3.0 TECHNICAL EVALUATION

3.1 Licensee's Evaluation

The proposed P-T limit curves in the licensee's letter dated May 6, 2010, extend the period of applicability of the current P-T limit curves, which were approved for 28.8 EFPY and 29.4 EFPY for Surry 1 and 2, respectively.

The licensee updated material property calculations using revised initial material properties and margins of Topical Report (TR) BAW-2308, Revision 2-A (publicly available through the letter from PWR (Pressurized-Water Reactor) Owners Group, to NRC, dated April 30, 2008, Subject: PWR Owners Group, Transmittal of NRC Approved Topical Report BAW-2308-NP, Revision 2, "Initial RT_{NDT} of Linde 80 Weld Materials" (TAC No. MD4241), PA-MSC-0229).[7] NRC staff had previously approved the use of TR BAW-2308, Revision 1-A (publicly available through the letter to Mr. Jerald S. Holm, Framatome ANP, from Herbert N. Berkow, NRC, transmitting the Final Safety Evaluation for Topical Report BAW-2308, Revision 1, "Initial RT_{NDT} of Linde 80 Weld Materials (TAC No. MB6636), dated August 4, 2005)[8] for Surry 1 and 2 analyses. The revised initial RT_{NDT} values and initial margin terms for Linde 80 weld materials in BAW-2308, Revision 2-A differ slightly from the values contained in BAW-2308, Revision 1-A. The licensee also provided updated fluence values, applicable to 48 EFPY, in performing the calculations.

3.2 Staff's Evaluation

3.2.1 ART Value and P-T Limit Curves

By letter dated June 13, 2006,[9] VEPCO submitted an exemption request from the requirements of Appendix G to 10 CFR Part 50 and 10 CFR 50.61. Appendix G provides fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the system may be subjected over its service lifetime. Section 50.61 provides fracture toughness requirements for protection against pressurized thermal shock (PTS). Section 50.61(a)(5) and 10 CFR Part 50, Appendix G(II)(D)(i) require that preservice or unirradiated condition RT_{NDT} be evaluated according to the procedures in the ASME Code, Section III, Paragraph NB-2331, from Charpy V-notch impact tests and drop-weight tests. Framatome ANP TR BAW-2308, Revision 1-A provides an alternate method for determining adjusted RT_{NDT} (reference nil-ductility temperature) and margins of the Linde 80 weld materials present in the beltline region of the Surry 1 and 2 RPVs. By exemption dated June 27, 2007,[10] NRC staff approved the exemption request for the alternate material properties basis per 10 CFR 50.60(b). The exemption stated:

> The exemptions are granted to the licensee to utilize the most recent staff-approved version of BAW-2308 (currently BAW-2308, Revision 1). Future revisions of BAW-2308 could affect fracture toughness data and analyses for Surry 1 and 2. Therefore, the licensee must review any future staff-approved revisions of BAW-2308 and update the units' fracture toughness assessments, based on the information in any staff-approved revision of BAW-2308.

By letter dated March 24, 2008,[11] the NRC staff approved the Final Safety Evaluation for Pressurized Water Reactor Owners Group TR BAW-2308, Revision 2. The values of initial RT_{NDT} and initial margin terms for the Linde 80 weld materials in BAW-2308, Revision 2-A were changed slightly from the values approved in TR BAW-2308, Revision 1-A.

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[7] ADAMS Accession No. ML081270388, April 30, 2008
[8] ADAMS Accession No. ML052070408, August 4, 2005
[9] ADAMS Accession No. ML061650080, June 13, 2006
[10] ADAMS Accession No. ML071160287, June 27, 2007
[11] ADAMS Accession No. ML080770349, March 24, 2008

For Surry 1 and 2, the applicable updated values for the Topical Reports are as follows:

Linde 80 Weld Wire Heat	BAW-2308, Revision 1 Initial RT_{T0} (°F)	BAW-2308, Revision 1 Margin (°F)	BAW-2308, Revision 2 Initial RT_{T0} (°F)	BAW-2308, Revision 2 Margin (°F)
Generic	-47.6	17.2	-48.6	18.0
299L44	-81.8	11.6	-74.3	12.8
72445	-72.5	12.3	-72.5	12.0

The following Linde 80 weld materials are contained in Surry 1: 8T1554, 299L44, and 72445; and Surry 2 are: 8T1762 and 72445. Linde 80 weld wire heat numbers not listed specifically in the Table above must use the Generic values provided.

The staff evaluated the licensee's P-T limit curves for acceptability by performing independent calculations using the methodologies of Appendix G of Section XI of the ASME B&PV Code and 10 CFR Part 50, Appendix G. Based on the 48 EFPY neutron fluence projections and the updated initial values for the Linde 80 weld materials provided in TR BAW-2308, Revision 2-A, the staff's calculated ART values were in good agreement with the licensee's calculated ART values for the Surry 1 and 2 beltline materials. The limiting material for the 1/4 T RT_{NDT} was determined to be the Surry 2 Intermediate-to-Lower Shell Circumferential Weld R3008/0227, with a calculated value of 222.5 °F. The limiting material for the 3/4T RT_{NDT} was determined to be the Surry 2 Intermediate-to-Lower Shell Circumferential Weld R3008/0227, with a calculated value of 188.6 °F.

The current P-T limit curves for Surry 1 and 2 are based on limiting 1/4T and 3/4T RT_{NDT} values of 228.4 °F and 189.5 °F, respectively. The values are higher than the values calculated using the 48 EFPY neutron fluence projections and the revised Linde 80 weld material values contained in TR BAW-2308, Revision 2-A. Therefore, the NRC staff concludes that the existing P-T limit curves in the current Surry 1 and 2 TSs remain valid for cumulative core burnups up to 48 EFPY. Based upon the aforementioned limiting ART values, NRC staff verified that the licensee's proposed P-T limits are in accordance with Appendix G to Section XI of the ASME B&PV Code and satisfy the requirements in Paragraph IV.A.2 of Appendix G to 10 CFR Part 50.

3.2.2 Pressurized Thermal Shock

The NRC staff evaluated the RPV beltline materials to ensure adequate resistance to failure during PTS events. The staff performed PTS reference temperature (RT_{PTS}) calculations. Projected values of RT_{PTS} for PWR reactor vessel beltline materials are determined in accordance with 10 CFR 50.61. The limiting material with respect to PTS for Surry 1 is Lower Shell Longitudinal Weld SA-1526/299L44. The value of RT_{PTS} for this material was determined to be 210.3 °F, which is below the PTS screening criterion of 270 °F for plates, forgings, and axial welds. The limiting material with respect to PTS for Surry 2 is Intermediate-to-Lower Shell Circumferential Weld R3008/0227. The value of RT_{PTS} for this material was determined to be 236.4 °F, which is below the PTS screening criterion of 300 °F for circumferential welds.

The projected values of RT_{PTS} for Surry 1 and 2 RPV beltline materials at 48 EFPY are below the 10 CFR 50.61 PTS screening criteria through 48 EFPY. Therefore, the NRC staff concludes that the submittal is in accordance with the requirements of 10 CFR 50.61.

3.2.3 Upper-Shelf Energy

Charpy USE calculations for Surry 1 and 2 remain consistent with the USE calculations previously submitted by letter dated June 13, 2006.[12] The NRC staff reviewed the USE calculations and determined that the calculations were acceptable and in accordance with the USE requirements of Appendix G to 10 CFR Part 50. The exemption dated June 27, 2007, approved the USE calculations. The 48 EFPY neutron fluence projections and updated Linde 80 weld materials values contained in TR BAW-2308, Revision 2-A do not alter the neutron fluence or copper composition values previously submitted and approved. Therefore, the USE percentage drops, calculated using the RG 1.99, Revision 2, Position 1.2 methodology, and equivalent margins analyses (EMAs) for the Surry 1 and 2 RPV beltline materials remain unchanged from the calculations and EMAs previously approved. The NRC staff concludes that the USE analyses are consistent with the previously approved USE analyses, demonstrating that the Surry 1 and 2 RPV beltline materials meet the requirements of Appendix G to 10 CFR Part 50 by satisfying the 50 ft-lb USE, or demonstrating through EMAs that acceptable results relative to Appendix K to Section XI of the ASME Code at 48 EFPY.

3.3 Conclusions

The staff concludes that the proposed P-T limit curves for Surry 1 and 2 satisfy the requirements in Appendix G to 10 CFR Part 50 and Appendix G to Section XI of the ASME Code. Hence, the proposed P-T limit curves may be incorporated into the Surry 1 and 2 TSs and are valid through 48 EFPY. The staff also concludes that the PTS and USE evaluations for the Surry 1 and 2 RPVs continue to meet the requirements of 10 CFR 50.61 and 10 CFR Part 50, Appendix G, respectively.

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 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 (75 FR 54396) published on September 7, 2010. 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.

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[12] ADAMS Accession No. ML071010469, June 13, 2006

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: there is reasonable assurance that (1) the health and safety of the public will not be endangered by operation in the proposed manner, (2) 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: Carolyn Fairbanks

Date: May 31, 2011

May 31, 2011

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 REACTOR VESSEL HEATUP AND COOLDOWN CURVES FOR 48 EFFECTIVE FULL-POWER YEARS (TAC NOS. ME3920 AND ME3921)

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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,

/RA/

Karen Cotton, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
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Docket Nos. 50-280 and 50-281

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LPL2-1 R/F
RidsNrrLAMO'Brien Resource
RidAcrsAcnw_MailCTR Resource
RidsNrrDciCvib Resource
RidsNrrDorlLpl2-1Resource
RidsNrrDirsItsb Resource
RidsOgcRp Resource
C.Fairbanks, NRR
RidsNrrPMSurry Resource
RidsRgn2MailCenter Resource
RidsNrrDorlDpr Resource

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OFFICE	NRR/LPL2-1/PM	NRR/LPL2-1/LA	NRR/DCI/CVIB/BC	OGC	NRR/LPL2-1/BC	NRR/LPL2-1/PM
NAME	KCotton	MO'Brien (BClayton for)	MMitchell	AJones "NLO"	GKulesa	KCotton
DATE	5/26/11	5/24/11	04/15/11*	04/26/11	5/31/11	5/31/11

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