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

May 2, 2001

Mr. David A. Christian
Senior Vice President - Nuclear
Virginia Electric and Power Company
5000 Dominion Blvd.
Glen Allen, Virginia 23060

SUBJECT: NORTH ANNA POWER STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS AND EXEMPTION FROM THE REQUIREMENTS OF 10 CFR PART 50, SECTION 50.60(a) RE: AMENDED PRESSURE-TEMPERATURE LIMITS (TAC NOS. MA9343, MA9344, MA9347, AND MA9348)

Dear Mr. Christian:

The Commission has issued the enclosed Amendment Nos. 226 and 207 to Facility Operating License Nos. NPF-4 and NPF-7 for the North Anna Power Station, Unit Nos. 1 and 2. The amendments change the Technical Specifications (TS) in response to your letter dated June 22, 2000, as supplemented by letters dated September 19, 2000, and January 4, February 14, March 13, March 22, and April 11, 2001. The supplements dated February 14, March 13, March 22, and April 11, 2001, contained clarifying information and did not expand the scope of the *Federal Register* notice published on February 23, 2001.

These amendments approve new pressure-temperature (P-T) limits, low-temperature overpressure protection (LTOP) system setpoints, and the LTOP system effective temperature (T_{enable}) in the TS to a maximum of 32.3 effective full-power years (EFPY) for Unit 1 and 34.3 EFPY for Unit 2. These changes specifically affect TS Figures 3.4-2, 3.4-3, and associated bases. These changes were based, in part, on the use of the American Society of Mechanical Engineers (ASME) Code Case N-641. It should be noted that the Notice of Consideration of Issuance of Amendments published in the *Federal Register* on February 23, 2001 (66 FR 11334) made reference to Code Cases N-514 and N-640. Those Code cases have been administratively combined into a single ASME Code Case, N-641, without changing the technical content. Therefore, we have reviewed your application to use Code Case N-641 instead, as requested in your letter of March 13, 2001, and we have made reference throughout the Safety Evaluation and the exemption to Code Case N-641.

In addition, your letter dated June 22, 2000, as supplemented by letters dated September 19, 2000, and January 4, February 14, March 13, March 22, and April 11, 2001, requested an exemption from the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix G to allow application of Code Case N-641. This Code case permits the use of an alternate reference fracture toughness for reactor vessel materials in determining the revised P-T curves, LTOP setpoints, and T_{enable} to maintain operator flexibility and safety during heatup and cooldown conditions. Based upon the review of the information provided, the staff has determined that application of Code Case N-641 is acceptable. Accordingly, the staff, pursuant to 10 CFR 50.12(a), has issued an exemption for the North Anna Power Station, Units 1 and 2, which is also enclosed.

NRR-058

A copy of the supporting Safety Evaluation and the exemption are enclosed. The exemption and the Notice of Issuance of Amendments have been forwarded to the Office of the Federal Register for publication.

Sincerely,

A E Edison for

Stephen R. Monarque, Project Manager, Section I
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-338 and 50-339

Enclosures:
1. Amendment No. 226 to NPF-4
2. Amendment No. 207 to NPF-7
3. Safety Evaluation
4. Exemption

cc w/encls: See next page

A copy of the supporting Safety Evaluation and the exemption are enclosed. The exemption and the Notice of Issuance of Amendments have been forwarded to the Office of the Federal Register for publication.

Sincerely,

/RA/ G. Edison for:

Stephen R. Monarque, Project Manager, Section I
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-338 and 50-339

Enclosures:
1. Amendment No. 226 to NPF-4
2. Amendment No. 207 to NPF-7
3. Safety Evaluation
4. Exemption

cc w/encls: See next page

Distribution:
PUBLIC
PDII-1 Reading
LBerry
HBerkow (RidsNrrDlpmLpdii)
REmch (hard copy)
GEdison (hard copy)
SMonarque (hard copy)
EDunnington (hard copy)
OGC (RidsOgcRp)
ACRS (RidsAcrsAcnwMailCenter)
GHill (8)
WBeckner (e-mail)
RidsRgn2MailCenter
FAkstulewicz (e-mail)

EMCB - SE Received H EEchon 4/26/01

DOCUMENT NAME: G:\PDII-1\NOANNA\ma9347-48Exemption.wpd
ADAMS ACCESSION NUMBER:

OFFICE	PDII-1/PM	PDII-1/PM	PDII-2/LA	SRXB
NAME	GEdison:mw	SMonarque	EDunnington	FAkstulewicz
DATE	4/13/2001	4/13/2001	4/13/2001	/ /2001
OFFICE	PDII-1/SC	OGC NLO with	PDII/D	D:NRR
NAME	REmch	little changes in	HBerkow	JZwolinski
DATE	4/27/2001	4/24/2001 Exempt	4/27/2001	/ /2001

OFFICIAL RECORD COPY
and SE

Mr. David A. Christian
Virginia Electric and Power Company

North Anna Power Station
Units 1 and 2

cc:
Mr. C. Lee Lintecum
County Administrator
Louisa County
P.O. Box 160
Louisa, Virginia 23093

Mr. Donald P. Irwin, Esquire
Hunton and Williams
Riverfront Plaza, East Tower
951 E. Byrd Street
Richmond, Virginia 23219

Dr. W. T. Lough
Virginia State Corporation
Commission
Division of Energy Regulation
P.O. Box 1197
Richmond, Virginia 23209

Old Dominion Electric Cooperative
4201 Dominion Blvd.
Glen Allen, Virginia 23060

Mr. Stephen P. Sarver, Director
Nuclear Licensing & Operations Support
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Blvd.
Glen Allen, Virginia 23060-6711

Office of the Attorney General
Commonwealth of Virginia
900 East Main Street
Richmond, Virginia 23219

Senior Resident Inspector
North Anna Power Station
U.S. Nuclear Regulatory Commission
1024 Haley Drive
Mineral, Virginia 23117

Mr. David A. Heacock
Site Vice President
North Anna Power Station
P.O. Box 402
Mineral, Virginia 23117-0402

Mr. Richard H. Blount, II
Site Vice President
Surry Power Station
Virginia Electric and Power Company
5570 Hog Island Road
Surry, Virginia 23883-0315

Robert B. Strobe, M.D., M.P.H.
State Health Commissioner
Office of the Commissioner
Virginia Department of Health
P. O. Box 2448
Richmond, Virginia 23218

Mr. William R. Matthews
Vice President - Nuclear Operations
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, Virginia 23060-6711

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-338

NORTH ANNA POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 226
License No. NPF-4

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

   A. The application for amendment by Virginia Electric and Power Company et al., (the licensee) dated June 22, 2000, as supplemented by letters dated September 19, 2000, and January 4, February 14, March 13, March 22, and April 11, 2001, 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 2.D.(2) of Facility Operating License No. NPF-4 is hereby amended to read as follows:

 (2) Technical Specifications

 The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 226 , are hereby incorporated in the 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

Richard L. Emch, Jr.

Richard L. Emch, Jr., Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: May 2, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 226

TO FACILITY OPERATING LICENSE NO. NPF-4

DOCKET NO. 50-338

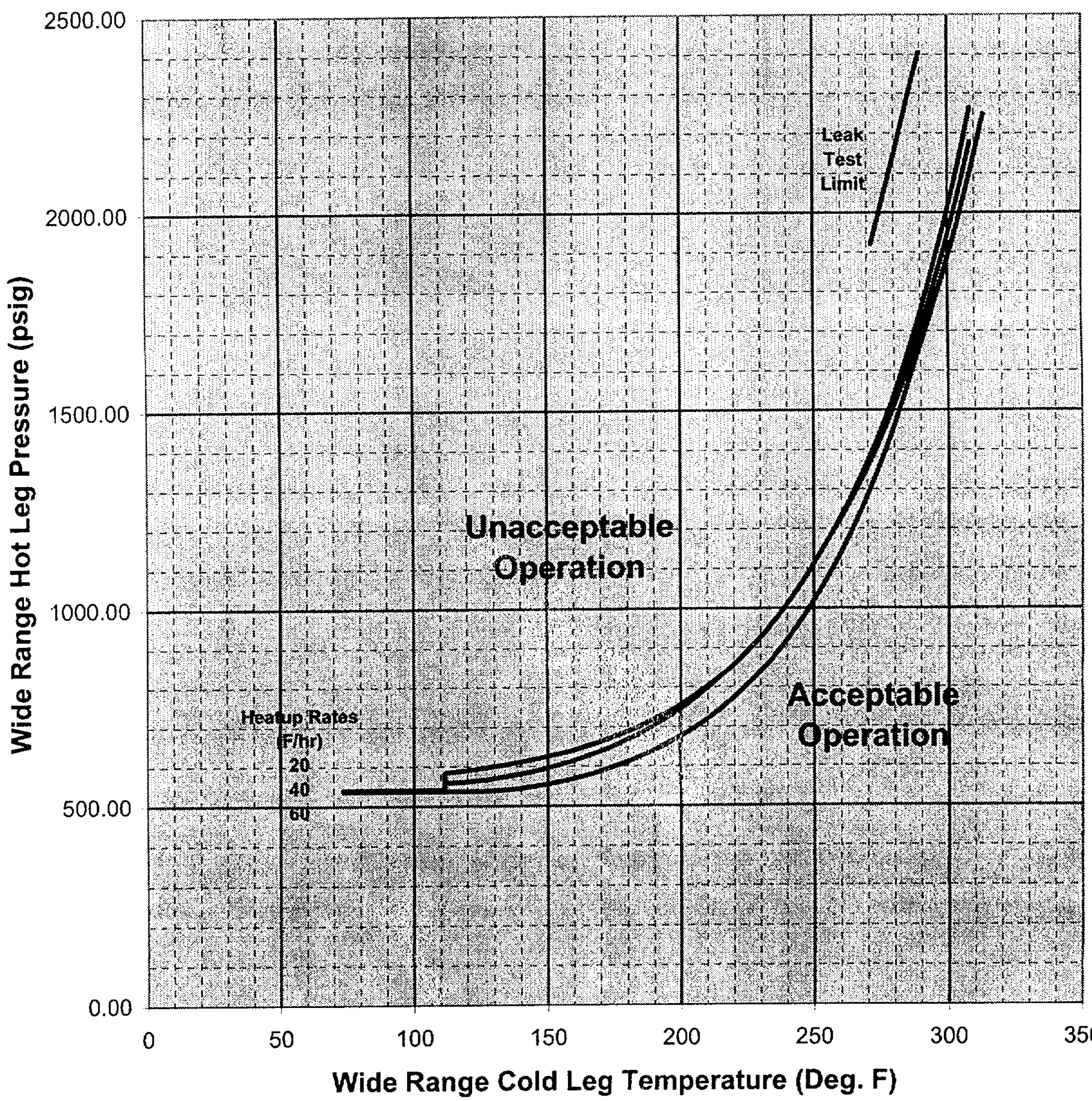
Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove Pages	Insert Pages
3/4 4-27	3/4 4-27
3/4 4-28	3/4 4-28
B 3/4 4-7	B 3/4 4-7
B 3/4 4-8	B 3/4 4-8

Figure 3.4-2

North Anna Unit 1
Reactor Coolant System Heatup Limitations

Material Property Basis
Limiting ART at 32.3 EFPY: 1/4-T, 218.5 deg. F
3/4-T, 195.6 deg. F

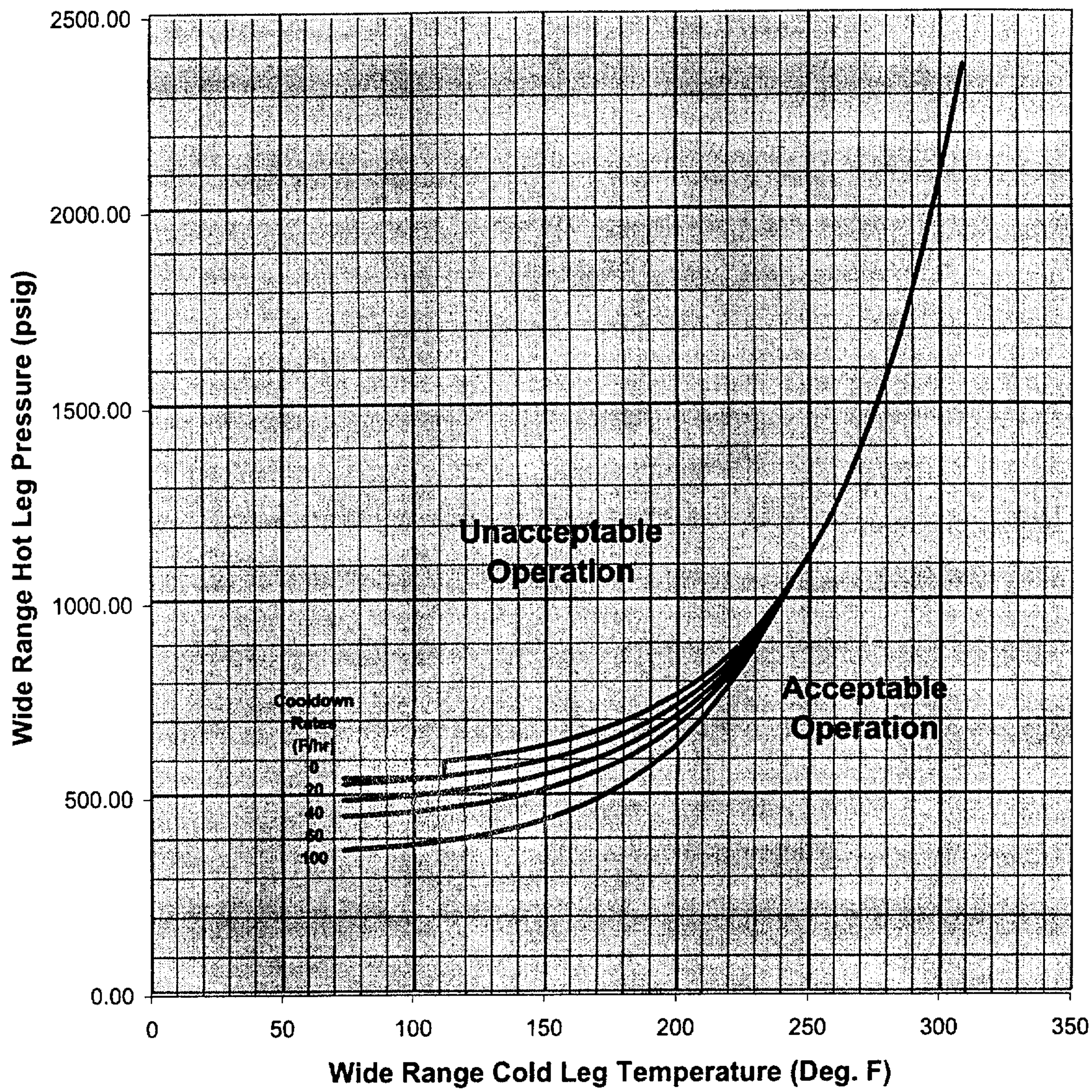


North Anna Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rates up to 60 F/hr)
Applicable for the first 32.3 EFPY (Including Margins for Instrumentation Errors)

Figure 3.4-3

North Anna Unit 1
Reactor Coolant System Cooldown Limitations

Material Property Basis
Limiting ART at 32.3 EFPY: 1/4-T, 218.5 deg. F
3/4-T, 195.6 deg. F



North Anna Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100 F/hr)
Applicable for the first 32.3 EFPY (Including Margins for Instrumentation Errors)

The heatup limit curve, Figure 3.4-2, 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 3.4-3 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 heatup and cooldown curves of Figures 3.4-2 and 3.4-3 are based upon a 1/4-T RT_{NDT} value of 218.5°F and a 3/4-T RT_{NDT} value of 195.6°F. These RT_{NDT} values conservatively bound the predicted reactor vessel beltline RT_{NDT} values for North Anna Unit 1 operation through 32.3 EFPY. The heatup and cooldown limits include margins to accommodate pressure and temperature measurement uncertainty, and the pressure difference between the point of measurement (RCS hot leg) and the point of interest (reactor vessel beltline).

The reactor vessel materials have been tested to determine their initial RT_{NDT}. Reactor operation and resultant fast neutron (E > 1 Mev) irradiation will cause an increase in the RT_{NDT}. An adjusted reference temperature, based upon the fluence and copper content of the material in question, can be predicted using US NRC Regulatory Guide 1.99, Revision 2. The heatup and cooldown limit curves (Figure 3.4-2 and Figure 3.4-3) include adjustments for this predicted shift in RT_{NDT} at the end of 32.3 EFPY.

The actual shift in the RT_{NDT} of the vessel material is established periodically by removal and evaluation of the reactor vessel material specimens installed on the inside wall of the thermal shield. The surveillance capsule withdrawal schedule was prepared in accordance with the requirements of ASTM E-185 and is presented in the UFSAR. Regulatory Guide 1.99, Revision 2, provides guidance for calculation of the shift in RT_{NDT} using measured data. Dosimetry from the surveillance capsule is used to provide benchmarks for the calculation of the neutron fluence to which the material specimens and the reactor vessel were exposed.

The pressure-temperature limit lines shown on Figure 3.4-2 for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50. The minimum temperature for criticality specified in T.S. 3.1.1.5 assures compliance with the criticality limits of 10 CFR 50 Appendix G.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in the UFSAR to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

Pressurizer

The limitations imposed on pressurizer heatup and cooldown and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

Low-Temperature Overpressure Protection

The OPERABILITY of two PORVs or an RCS vent opening of greater than 2.07 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 235°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures or (2) the start of a charging pump and its injection into a water-solid RCS. The low temperature PORV lift setpoints were established to ensure that pressure at the reactor vessel beltline during these design basis events will not exceed 100% of the 10 CFR 50 Appendix G isothermal limit curve when the LTOP system is enabled. The LTOPS design basis pressure-temperature limit curve includes margin to accommodate pressure and temperature measurement uncertainty, and the pressure difference between the point of measurement (RCS hot leg) and the point of interest (reactor vessel beltline).

Automatic or passive low temperature overpressure protection (LTOP) is required whenever any RCS cold leg temperature is less than 235°F. This temperature conservatively bounds the water temperature corresponding to a metal temperature of the limiting RT_{NDT} + 31.9°F + instrument uncertainty. Above 235°F administrative control is adequate protection to ensure the limits of the heatup curve (Figure 3.4-2) and the cooldown curve (Figure 3.4-3) are not violated. The concept of requiring automatic LTOP at the lower end, and administrative control at the upper end, of the Appendix G curves is further discussed in NRC Generic Letter 88-11.

Surveillance limits are established for the pressure in the backup nitrogen accumulators to ensure there is adequate motive power for the PORVs to cope with an inadvertent start of a high head safety injection pump in a water solid condition, allowing adequate time for the operators to respond to terminate the event.

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-339

NORTH ANNA POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 207
License No. NPF-7

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

   A. The application for amendment by Virginia Electric and Power Company et al., (the licensee) dated June 22, 2000, as supplemented by letters dated September 19, 2000, and January 4, February 14, March 13, March 22, and April 11, 2001, 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 2.C.(2) of Facility Operating License No. NPF-7 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 207, are hereby incorporated in the 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

Richard L. Emch, Jr.

Richard L. Emch, Jr., Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: May 2, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 207

TO FACILITY OPERATING LICENSE NO. NPF-7

DOCKET NO. 50-339

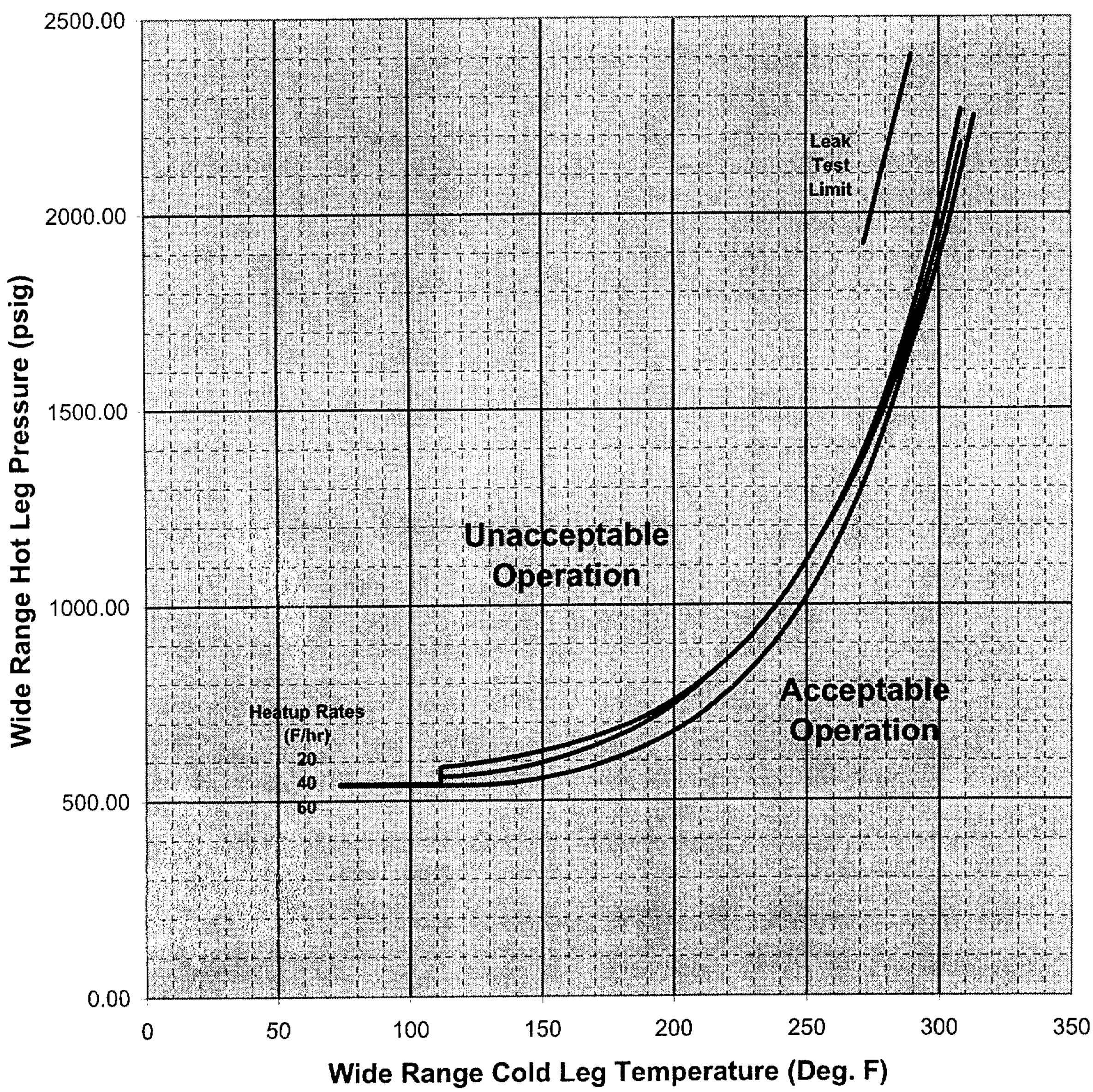
Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove Pages	Insert Pages
3/4 4-27	3/4 4-27
3/4 4-28	3/4 4-28
B 3/4 4-7	B 3/4 4-7
B 3/4 4-8	B 3/4 4-8

Figure 3.4-2

North Anna Unit 2
Reactor Coolant System Heatup Limitations

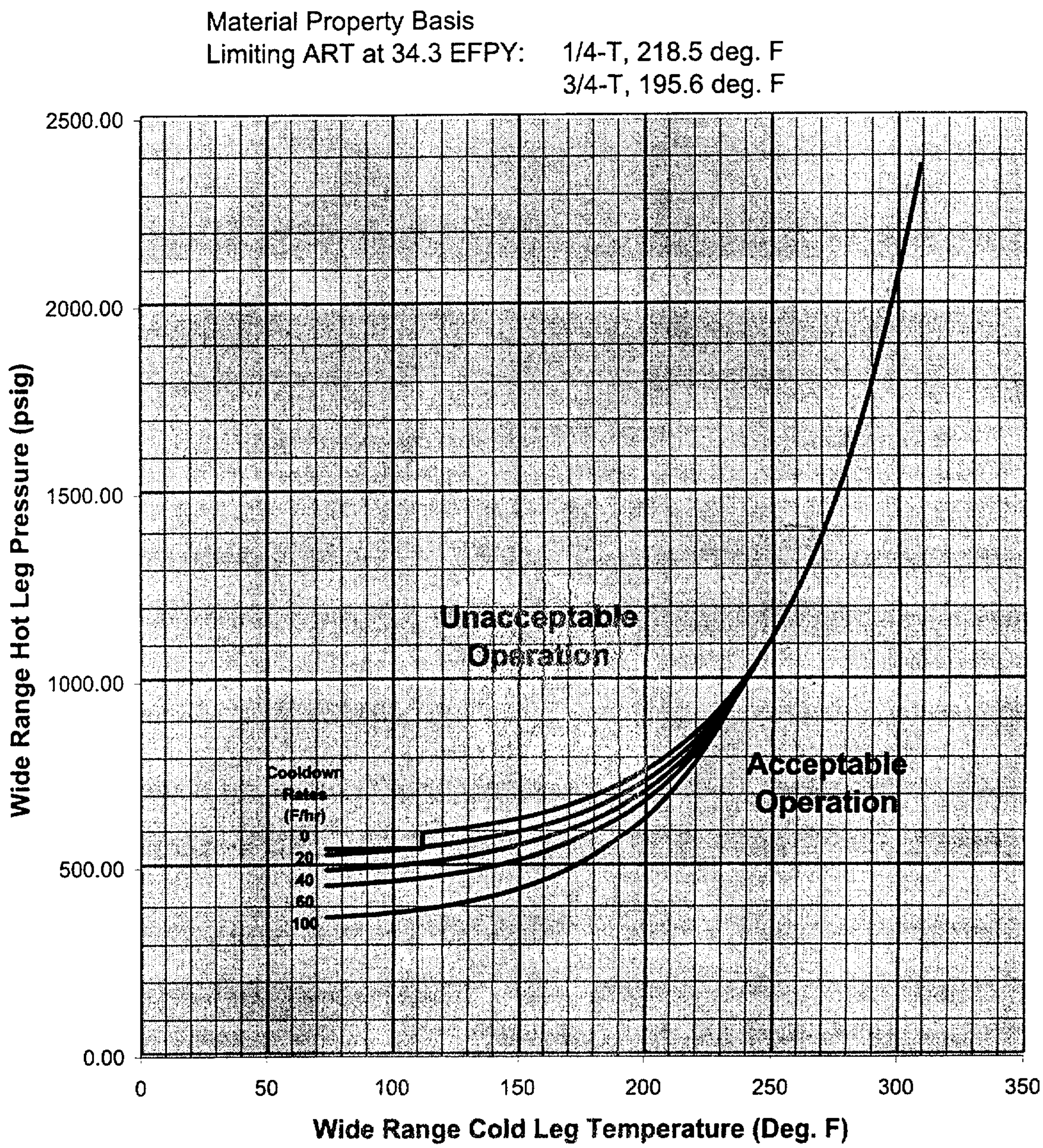
Material Property Basis
Limiting ART at 34.3 EFPY: 1/4-T, 218.5 deg. F
3/4-T, 195.6 deg. F



North Anna Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rates up to 60 F/hr)
Applicable for the first 34.3 EFPY (Including Margins for Instrumentation Errors)

Figure 3.4-3

North Anna Unit 2
Reactor Coolant System Cooldown Limitations



North Anna Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100 F/hr)
Applicable for the first 34.3 EFPY (Including Margins for Instrumentation Errors)

The heatup limit curve, Figure 3.4-2, 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 3.4-3 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 heatup and cooldown curves of Figures 3.4-2 and 3.4-3 are based upon a 1/4-T RT_{NDT} value of 218.5°F and a 3/4-T RT_{NDT} value of 195.6°F. These RT_{NDT} values conservatively bound the predicted reactor vessel beltline RT_{NDT} values for North Anna Unit 2 operation through 34.3 EFPY. The heatup and cooldown limits include margins to accommodate pressure and temperature measurement uncertainty, and the pressure difference between the point of measurement (RCS hot leg) and the point of interest (reactor vessel beltline).

The reactor vessel materials have been tested to determine their initial RT_{NDT}. Reactor operation and resultant fast neutron (E > 1 Mev) irradiation will cause an increase in the RT_{NDT}. An adjusted reference temperature, based upon the fluence and copper content of the material in question, can be predicted using US NRC Regulatory Guide 1.99, Revision 2. The heatup and cooldown limit curves (Figure 3.4-2 and Figure 3.4-3) include adjustments for this predicted shift in RT_{NDT} at the end of 34.3 EFPY.

The actual shift in the RT_{NDT} of the vessel material is established periodically by removal and evaluation of the reactor vessel material specimens installed on the inside wall of the thermal shield. The surveillance capsule withdrawal schedule was prepared in accordance with the requirements of ASTM E-185 and is presented in the UFSAR. Regulatory Guide 1.99, Revision 2, provides guidance for calculation of the shift in RT_{NDT} using measured data. Dosimetry from the surveillance capsule is used to provide benchmarks for the calculation of the neutron fluence to which the material specimens and the reactor vessel were exposed.

The pressure-temperature limit lines shown on Figure 3.4-2 for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50. The minimum temperature for criticality specified in T.S. 3.1.1.5 assures compliance with the criticality limits of 10 CFR 50 Appendix G.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in the UFSAR to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

Pressurizer

The limitations imposed on pressurizer heatup and cooldown and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

Low-Temperature Overpressure Protection

The OPERABILITY of two PORVs or an RCS vent opening of greater than 2.07 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 270°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures or (2) the start of a charging pump and its injection into a water-solid RCS. The low temperature PORV lift setpoints were established to ensure that pressure at the reactor vessel beltline during these design basis events will not exceed 100% of the 10 CFR 50 Appendix G isothermal limit curve when the LTOP system is enabled. The LTOPS design basis pressure-temperature limit curve includes margin to accommodate pressure and temperature measurement uncertainty, and the pressure difference between the point of measurement (RCS hot leg) and the point of interest (reactor vessel beltline).

Automatic or passive low temperature overpressure protection (LTOP) is required whenever any RCS cold leg temperature is less than 270°F. This temperature conservatively bounds the water temperature corresponding to a metal temperature of the limiting RT_{NDT} + 31.9°F + instrument uncertainty. Above 270°F administrative control is adequate protection to ensure the limits of the heatup curve (Figure 3.4-2) and the cooldown curve (Figure 3.4-3) are not violated. The concept of requiring automatic LTOP at the lower end, and administrative control at the upper end, of the Appendix G curves is further discussed in NRC Generic Letter 88-11.

Surveillance limits are established for the pressure in the backup nitrogen accumulators to ensure there is adequate motive power for the PORVs to cope with an inadvertent start of a high head safety injection pump in a water solid condition, allowing adequate time for the operators to respond to terminate the event.

3/4.4.10 STRUCTURAL INTEGRITY

3/4.4.10.1 ASME CODE CLASS 1, 2 and 3 COMPONENTS

The inspection programs for ASME Code Class 1, 2 and 3 Reactor Coolant System components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO THE PRESSURE-TEMPERATURE LIMITS AND LOW TEMPERATURE

OVERPRESSURE PROTECTION SYSTEM FOR

NORTH ANNA POWER STATION UNITS 1 AND 2

VIRGINIA ELECTRIC AND POWER COMPANY (VEPCO)

DOCKET NOS. 50-338 AND 50-339

1.0 INTRODUCTION

By letter dated June 22, 2000, supplemented by letters dated September 19, 2000, and January 4, February 14, March 13, March 22, and April 11, 2001, Virginia Electric and Power Company, the licensee, submitted changes related to the pressure-temperature (P-T) limit curves, the low-temperature overpressure protection (LTOP) setpoints, and the LTOP system effective temperature (T_{enable}) in the Technical Specifications (TS) for the North Anna Power Station Units 1 and 2. The licensee proposed to revise the P-T limits and T_{enable} to provide new limits that are valid to 32.3 effective full-power years (EFPY) for Unit 1 and 34.3 EFPY for Unit 2. The licensee also proposed to retain the current LTOP system setpoints. The proposed changes are based, in part, on the use of American Society of Mechanical Engineers (ASME) Code Case N-641, which was reviewed by the staff. An exemption for use of this ASME Code case for North Anna, Units 1 and 2, is enclosed. The supplements dated February 14, March 13, March 22, and April 11, 2001, contained clarifying information and did not expand the scope of the *Federal Register* notice published on February 23, 2001.

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 in nuclear power plants. The staff evaluates the P-T limit curves and T_{enable} based on the following NRC regulations and guidance: 10 CFR Part 50, Appendix G; Generic Letter (GL) 88-11; GL 92-01, Revision 1; GL 92-01, Revision 1, Supplement 1; Regulatory Guide (RG) 1.99, Revision 2 (Rev. 2); and Standard Review Plan (SRP) Sections 5.2.2 and 5.3.2. GL 88-11 advised licensees that the staff would use RG 1.99, Rev. 2, to review P-T limit curves. RG 1.99, Rev. 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, Rev. 1, requested that licensees submit their reactor pressure vessel (RPV) data for their plants to the staff for review. GL 92-01, Rev. 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations. These data are used by the staff as the basis for review of P-T limit curves and T_{enable} and as the basis for the staff's review of pressurized thermal shock (PTS) assessments (10 CFR 50.61 assessments). Appendix G to 10 CFR Part 50 requires that P-T limit curves and T_{enable} be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the ASME Boiler and Pressure Vessel 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.

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 (LEFM) methodology of Appendix G to Section XI of the ASME 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. 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 for hydrostatic testing curves. The methods of Appendix G postulate the existence of a sharp surface flaw in the RPV that is normal to the direction of the maximum stress. This flaw is postulated to have a depth 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 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 ASME Code Appendix G methodology requires that licensees determine the adjusted reference temperature (ART or adjusted RT_{NDT}). 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 (M) term.

The ΔRT_{NDT} is a product of a chemistry factor and a fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Rev. 2, (in accordance with Regulatory Position 1.1 of the RG) or from surveillance data (in accordance with Regulatory Position 2.1 of the RG). 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 chemistry factor (CF) was determined using the tables in RG 1.99, Rev. 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, Rev. 2, describes the methodology to be used in calculating the margin term.

In order to assure the integrity of the reactor coolant pressure boundary, an LTOP system is provided. The staff reviewed the licensee's proposed LTOP setpoints; the evaluation is provided below.

To determine the material properties for the proposed TS changes, the pressure vessel fluence values are needed. The staff reviewed the licensee's analysis of capsule fluence data to determine the acceptability of the analysis. By letters dated June 22, 2000, and September 19, 2000, VEPCO submitted the surveillance capsule W reports: BAW-2356, Revision 1, and BAW-2376, for North Anna Units 1 and 2, respectively. The staff also reviewed the acceptability of the licensee's proposed fluence values.

2.0 EVALUATION

2.1 Fluence Analysis

2.1.1 Surveillance Capsule W

The licensee provided a surveillance capsule report for capsule W for each reactor unit vessel. Capsule W was analyzed using the methods of BAW-2421P-A, Rev. 1, "Fluence and Uncertainty Methodologies," by J.R. Worsham et al., Framatome Technologies Incorporated, Lynchburg, Virginia, April 1999, which is an NRC-approved methodology for analyzing fast neutron fluence values (E > 1.0 MeV) to the pressure vessel and the surveillance capsule. The computational methods comply with the requirements of DG-1053, Draft Regulatory Guide, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," U.S. Nuclear Regulatory Commission, June 1996. As required by the approved methodology, no bias or other corrections were applied to the calculated values. The capsule dosimetry was measured using techniques complying with applicable ASTM standards.

For each capsule W, the measured dosimeter activities were compared to the calculated values for each dosimeter type at each exposure location and for the capsule exposure history. The calculated-to-measured (C/M) dosimeter ratios were found to be in excellent agreement and showed no indication of bias or abnormal distribution. The C/Ms serve as verification that the calculated values are within acceptable uncertainty limits. The staff finds the methodology and the results of surveillance capsule W analysis to be acceptable.

2.1.2 Vessel Fluence Values

The proposed pressure vessel peak fluence values for both units were calculated using an NRC-approved VEPCO methodology (letter from N. Kalyanam, US NRC, to J. P. O'Hanlon, VEPCO, North Anna Power Station, Units 1 and 2, and Surry Power Station, Units 1 and 2 - Reactor Vessel Fluence Analysis Methodology (GL 92-01, Rev. 1, Supplement 1, dated April 13, 1999)). However, the two surveillance capsule reports, BAW-2356, Rev. 1, and BAW-2376, recalculated the projected values and compared them to the measured dosimeter values. The measured values, the calculated values using the NRC-approved Framatome methodology, and the calculated values using the VEPCO methodology are all within 4 percent of each other. This is remarkable agreement indicating a high confidence level for the proposed values. Because the estimated uncertainty is much lower than the acceptable value of 20 percent, the staff considers the proposed values in the June 22, 2000, letter and those in the surveillance reports to be the same; therefore, the values in the June 22, 2000, letter are acceptable.

Both North Anna units have been operating at a load factor greater than the 0.80 assumed in the original 10 CFR 50.61 and are projected to operate at 0.90 for the remaining plant lifetime. Therefore, the projected end-of-license EFPY of operation will be 32.2 and 34.3 for Units 1 and 2, respectively. Because the RT_{PTS} values are estimated to be lower than the screening criteria in 10 CFR 50.61 at the expiration of the current license, these RT_{PTS} values are acceptable.

In summary, for the reasons stated above, the staff finds that the projected fluence values for the North Anna, Units 1 and 2, pressure vessels, at the end of their current license, are acceptable.

2.2 P-T Limits and T_{enable}

2.2.1 Licensee Evaluation

The licensee submitted P-T limit curves, LTOP setpoints and T_{enable} for operation up to 32.3 EFPY for Unit 1 and 34.3 EFPY for Unit 2. The methodology for these evaluations is described in Westinghouse Electric Company LLC report WCAP-15112, Revision 1, "North Anna Units 1 and 2 WOG Reactor Vessel 60-Year Evaluation Minigroup Heatup and

Cooldown Limit Curves for Normal Operation" (dated October 1998). Based on similarities in construction, materials and operation of the two units, the licensee considers the two units together, with the development of P-T limit curves, LTOP setpoints and T_{enable} for both units. Note that although the report addresses conditions for both the end of the (current) license (EOL) period and the end of license renewal period, the license amendment requested by the licensee is only for operation through the end of the current license period, defined by the licensee as 32.3 EFPY for Unit 1 and 34.3 EFPY for Unit 2.

From information contained in letters from the licensee to the NRC dated November 19, 1999, and September 19, 2000, and as summarized in WCAP-15112, Revision 1, the licensee determined that the limiting ART for the RPV of each unit is from the lower shell forging. The other materials considered by the licensee are the nozzle shell and intermediate shell forgings, and the nozzle shell-to-intermediate shell and intermediate shell-to-lower shell circumferential welds of each RPV. Results from the surveillance programs for North Anna Units 1 and 2 are available for the lower shell forging of Unit 1 (heat 990400/292332), the intermediate-to-lower shell circumferential weld of Unit 1 (heat 25531), the intermediate shell forging of Unit 2 (heat 990496/292424), and the intermediate-to-lower shell circumferential weld of Unit 2 (heat 716126). For each of these materials, data are available for three surveillance capsules. The surveillance data for the forgings include both the axial and tangential orientations. In addition, surveillance data for North Anna circumferential weld heats 25295 (Unit 1) and 4278 (both units) are available from the surveillance programs at Sequoyah Units 1 and 2, respectively.

Table 1 provides a summary of the licensee's evaluations of chemistry factors for each heat with available surveillance data, using the procedures of Regulatory Positions 1.1 and 2.1 of RG 1.99, Rev. 2.

For the Unit 1 lower shell forging, the licensee found that the available data did not meet the credibility requirements of RG 1.99, Rev. 2. An evaluation of the conservatism of the chemistry factor from Regulatory Position 1.1 of RG 1.99, Rev. 2, demonstrated that this chemistry factor was conservative, in that predictions of the surveillance data were within two standard deviations (2σ) of the measured values. Therefore, the licensee concluded that it was appropriate to use the Position 2.1 chemistry factor with the full margin term for evaluating the ART for this material.

The licensee's evaluation for the Unit 2 intermediate shell forging reached a similar conclusion, and likewise the licensee concluded that it was appropriate to use the Position 2.1 chemistry factor with the full margin term for use in evaluating the ART for this material.

For the Unit 1 circumferential weld heat 25531, the licensee found that the surveillance data did not meet the credibility requirements of RG 1.99, Rev. 2. An evaluation of the conservatism of the chemistry factor from Regulatory Position 1.1 of RG 1.99, Rev. 2, demonstrated that this chemistry factor was conservative, in that predictions of the surveillance data were within 2σ of the measured values. Therefore, the licensee concluded that it was appropriate to use the lower of the two values, in this case the Position 1.1 chemistry factor, with the full margin term for evaluating the ART for this material.

For the Unit 2 intermediate-to-lower shell circumferential weld (heat 716126), the licensee concluded that the available surveillance data satisfied the credibility criteria of RG 1.99, Rev. 2. Although the surveillance weld had a slightly higher copper and nickel content than the RPV weld (hence providing a higher chemistry factor than that for the RPV weld), the licensee chose to not adjust the surveillance data for the small differences in the chemical compositions. Therefore, the licensee concluded that it was appropriate to use the Position 2.1 chemistry factor with the reduced margin term for evaluating the ART of this material.

Table 1: Comparison of Licensee's Evaluation of Chemistry Factors for North Anna

Material	Chemical Composition		Chemistry Factor	
	Copper (wt. %)	Nickel (wt. %)	Position 1.1	Position 2.1
Unit 1				
Lower shell forging (990400/292332)	0.156	0.817	119.8	82.9[a]
Intermediate-to-lower shell circumferential weld (25531)	0.098	0.124	56.2[a]	68.0
Nozzle-to-intermediate circumferential weld - OD 94% (25295)	0.352	0.125	163.3	144.2[a]
Nozzle-to-intermediate circumferential weld - ID 6% (4278)	0.120	0.110	63.0	92.4[a]
Unit 2				
Intermediate shell forging (990496/292424)	0.107	0.857	74.3	54.1[a]
Intermediate-to-lower shell circumferential weld (716126)	0.066	0.046	36.1	26.8[a]

[a] Value used by the licensee.

For the Unit 1 circumferential weld heat 25295, the licensee used surveillance data from Sequoyah Unit 1 for this same weld heat. For this data, the licensee used an evaluation of the data reported in NUREG/CR-6551, including updated fluence values and shift data from a computer curve fit to the data. The licensee concluded that the available surveillance data satisfied the credibility criteria of RG 1.99, Rev. 2, applied a chemistry ratio correction to account for differences between the chemical composition of the surveillance weld and the North Anna Unit 1 RPV weld, and determined a chemistry factor value in accordance with Position 2.1 of RG 1.99, Rev. 2. Therefore, the licensee concluded that it was appropriate to use the Position 2.1 chemistry factor with the reduced margin term for evaluating the ART of this material.

For circumferential weld heat 4278 identified by the licensee in the Unit 1 RPV, the licensee used surveillance data from Sequoyah Unit 2 for this same weld heat. For this data, the licensee used an evaluation of the data reported in NUREG/CR-6551, including updated fluence values and shift data from a computer curve fit to the data. The licensee found that the surveillance data did not satisfy the credibility criteria of RG 1.99, Rev. 2. An evaluation of the conservatism of the chemistry factor from Regulatory Position 1.1 of RG 1.99, Rev. 2, demonstrated that this chemistry factor was non-conservative, in that predictions of the surveillance data were not within 2σ of the measured values. The licensee then applied a chemistry ratio correction to account for differences between the chemical composition of the surveillance weld and the North Anna Unit 1 RPV weld, and determined a chemistry factor value in accordance with Position 2.1 of RG 1.99, Rev. 2. Since the Position 1.1 chemistry factor was non-conservative, the licensee concluded that it was appropriate to use the higher of the two chemistry factor values, in this case the Position 2.1 chemistry factor, with the full margin term for evaluating the ART for this material.

For determination of the ART values at the 1/4T and 3/4T locations, the licensee started with the EOL fluences for each RPV material evaluated at the wetted inner surface of the RPV. The licensee determined the neutron fluence values at the 1/4T and 3/4T locations by attenuating the neutron fluence using the combined cladding and ferritic base metal thicknesses, with the total RPV thickness, T, given by T = 0.16 + 7.705 = 7.862 inches. The resultant fluence values at the 1/4T and 3/4T locations are provided in the licensee's submittal.

Table 2 provides the licensee's evaluation of ART at the 1/4T location for each of the beltline materials in each unit's RPV. From the results summarized in Table 2, the licensee concluded that the limiting EOL ART at the 1/4T location for both units is bounded by 218.5 °F, and the limiting EOL ART at the 3/4T location for both units is bounded by 195.6 °F. With these ART values, the licensee evaluated P-T limits using the methodology provided in ASME Code Case N-641. As provided in letters dated January 4, 2001, and March 22, 2001, the curves provided by the licensee provide margins to account for temperature and pressure instrument errors (13.5 °F and 70 psi, respectively), and for the pressure difference between the point of measurement (RCS hot leg) and the RPV beltline (10 psi).

For the T_{enable}, ASME Code Case N-641 indicates that the LTOP system should be effective for coolant temperatures below the higher of (1) and (2) given below, or the higher of (1) and (3):

(1) 200 °F;

(2) RT_{NDT} + 40 °F for axial surface flaws (e.g., for axial welds and base materials) and RT_{NDT} - 85 °F for circumferential surface flaws (e.g., for circumferential welds);

(3) a plant-specific value of T_{enable} determined from an equation provided in the Code case.

For the plant-specific evaluation of T_{enable}, the licensee determined that the equation provided in the Code case results in a T_{enable} defined by RT_{NDT} + 51.9 °F, where the licensee has included an additional 20 °F to account for temperature measurement uncertainty. For Unit 1, with a peak 1/4T ART of 174.9 °F, the licensee proposes to use the present T_{enable} of 235 °F, which bounds the value permitted by the Code case (226.8 °F according to the licensee). For Unit 2 with a peak 1/4T ART of 209.4 °F, the licensee proposes to use the present T_{enable} of 270 °F, which bounds the value permitted by the Code case (261.3 °F according to the licensee).

2.2.2 Staff Evaluation

The staff performed independent calculations of the information provided in the submittal, including the ART values, the P-T limit curves, and T_{enable}. For the ART values, the staff used information in the submittal along with that in the NRC's Reactor Vessel Integrity Database (RVID) to ensure that the ART values used by the licensee in determining the P-T limit curves and T_{enable} are conservative for the EOL conditions specified by the licensee. For the P-T limit curves and T_{enable} evaluation, the staff used the bounding ART values specified by the licensee to ensure that the P-T limit curves and T_{enable} provided in the submittal are bounding for the specified values of ART.

Using the wetted surface neutron fluences provided by the licensee, the staff determined the ART values at the 1/4T and 3/4T RPV through-thickness locations in a manner consistent with the methodology in RG 1.99, Rev. 2. Specifically, the wetted-surface neutron fluence was attenuated through the full cladding thickness plus either 25 percent or 75 percent of the thickness of the ferritic base metal, as appropriate. This calculation method results in slightly reduced fluence values and hence slightly lower ART values than those submitted by the licensee.

For the North Anna RPV materials which have surveillance data available, the staff evaluation of the surveillance data is consistent with the licensee's evaluation. A summary of the staff's calculations of ART for all of the RPV materials in each unit is provided in Table 3. As described by the licensee, the staff agrees that the ART values for each RPV material in each unit are bounded by 218.5 °F for the 1/4T, and 195.6 °F for the 3/4T location.

The staff identified an omission in the licensee's evaluation of the ART values for Unit 2. Specifically, circumferential weld heat 4278 identified by the licensee in Unit 1 is also found in the Unit 2 RPV. However, in its evaluation of the ART for Unit 2 the licensee omitted consideration of the surveillance data for this weld heat from the Sequoyah Unit 2 surveillance program. The licensee has concurred with the identification of this omission. This omission does not impact the Unit 2 P-T limits because circumferential weld heat 4278 is not the limiting material.

The staff performed check calculations to verify the P-T limit curves using the licensee's EOL limiting ART values (218.5 °F for the 1/4T location and 195.6 °F for the 3/4T location) for the North Anna Units 1 and 2 RPV beltline materials. The initial submittal by the licensee provided no margins to account for pressure and temperature instrument errors. This omission is acceptable for P-T limit curves evaluated using the K_{Ia} curve to estimate the material fracture toughness, due to conservatism inherent in the use of the K_{Ia} curve. However, use of the K_{Ic} curve, as requested by the licensee through ASME Code Case N-641, requires the inclusion of margins to account for the instrument errors. The licensee resubmitted their curves to reflect these margins.

Once margins to account for instrument error were incorporated by the licensee, the staff's check calculations utilized time-dependent thermal and stress history information (specifically time-dependent RPV wall temperature and thermal stress intensity, K_{It}, for the 1/4T and 3/4T wall locations) provided by the licensee to provide a more realistic assessment of the expected conditions that the RPV would be subjected to during heat-up and cool-down operations. Initial check calculations by the staff found differences between the licensee's P-T limit values and those calculated by the staff. Discussions with the licensee indicated that, in part, the heat-up curves were evaluated utilizing a compressive thermal stress intensity at the 1/4T location. Staff questioned the use of compressive thermal stress values, with a concern that the resultant P-T limit curves could become non-conservative should a heat-up be halted for some reason. To address this concern, the licensee provided tabular evaluations of RPV wall temperature and thermal stress assuming that the heat-up was paused, with the coolant temperature held constant. Evaluation of this data by the staff indicated that the use of compressive thermal stress values does not result in a non-conservative P-T limit curve, and is therefore acceptable.

Table 2: Licensee Evaluation of ART Values for North Anna Units 1 and 2

Material	ID Fluence (10^{19} n/cm^2)	Chem. Factor (°F)	Init. RT_{NDT} (°F)	Margin (°F)	1/4T ART (°F)
Unit 1					
Nozzle Forging	0.136	121.5	6	69.0	121.7
Int. Forging	3.920	86.0	17	34.0	157.7
Lower Forging	3.920	82.9	38	34.0	174.9
Circ. Weld (25295)	0.136	144.2	0	48.8	104.3
Circ. Weld (4278)	0.136	92.4	0	59.8	95.3
Circ. Weld (25531)	3.920	56.2	19	56.0	144.8
Unit 2					
Nozzle Forging	0.148	51.0	9	65.2	94.6
Int. Forging	3.960	54.1	75	34.0	176.3
Lower Forging	3.960	96.0	56	34.0	209.4
Circ. Weld (4278)	0.148	63.0	0	50.9	76.2
Circ. Weld (801)	0.148	87.8	0	59.4	94.6
Circ. Weld (716126)	3.960	26.8	-48	28.0	13.4

Table 3: NRC Staff Evaluation of ART Values for North Anna Units 1 and 2

Material	Chem. Factor (°F)	Init. RT_{NDT} (°F)	1/4T Fluence (10^{19} n/cm^2)	1/4T ART (°F)	3/4T Fluence (10^{19} n/cm^2)	3/4T ART (°F)
Unit 1						
Nozzle Forging	121.5	6	0.082	121.1	0.033	100.3
Int. Forging	86.0	17	1.233	157.1	0.943	135.6
Lower Forging	82.9	38	1.233	174.2	0.943	153.5
Circ. Weld (25295)	144.2	0	0.082	103.5	0.033	82.1
Circ. Weld (4278)	92.4	0	0.082	88.2	0.033	66.7
Circ. Weld (25531)	56.2	19	1.233	144.3	0.943	129.5
Unit 2						
Nozzle Forging	51.0	9	0.036	92.5	0.036	82.6
Int. Forging	54.1	75	1.236	175.9	0.952	162.4
Lower Forging	96.0	56	1.236	208.7	0.952	184.7
Circ. Weld (4278)	63.0	0	0.036	87.4	0.036	68.2
Circ. Weld (801)	87.8	0	0.036	87.7	0.036	66.6
Circ. Weld (716126)	26.8	-48	1.236	13.1	0.952	4.9

A detailed comparison of the licensee's submitted P-T limit curves and the staff's check calculations revealed that the licensee's curves were non-conservative for heat-ups at high pressure levels. After discussions between the staff and the licensee, a coding error was identified by the licensee in their software to evaluate the P-T limit curves. This error was corrected and the P-T limit curves were subsequently recalculated and submitted by the licensee.

Using the licensee-supplied data, excellent agreement was found between the staff's calculations and licensee's curves, verifying the appropriateness of the curves submitted by the licensee. The staff also found that the minimum temperature requirements of Table 1 of Appendix G to 10 CFR Part 50 were properly implemented in the P-T limit curves.

The staff confirmed that the T_{enable} proposed by the licensee, 235 °F for Unit 1 and 270 °F for Unit 2, are more conservative than those permitted by ASME Code Case N-641, and thus are acceptable. However, the staff does not agree that ASME Code Case N-641 permits a T_{enable} of 226.8 °F for Unit 1 and 261.3 °F for Unit 2. Considering the temperature measurement uncertainty of 13.5 °F and the maximum temperature lag of 13 °F (between the RPV 1/4T location and the coolant temperature during 60 °F/h heat-up), the staff has determined that ASME Code Case N-641 permits a T_{enable} of RT_{NDT} + 58.4 °F, or 232.6 °F for Unit 1, and 267.1 °F for Unit 2.

Thus, the staff determined that the P-T limit curves and T_{enable} values satisfy the requirements in Paragraph IV.A.2 of Appendix G of 10 CFR Part 50, as modified by Code Case N-641, and hence, the requirements of 10 CFR 50.60.

The staff concludes that the proposed P-T limit curves and T_{enable} for heatup and cooldown of the reactor coolant system satisfy the requirements in Appendix G to Section XI of the ASME Code, as modified by Code Case N-641, and Appendix G of 10 CFR Part 50, for operations up to 32.3 EFPY for Unit 1 and 34.3 EFPY for Unit 2. The proposed P-T limit curves and T_{enable} also satisfy GL 88-11, because the method in RG 1.99, Rev. 2, was used to calculate the ART. Hence, the proposed P-T limit curves and T_{enable} are acceptable for incorporation in the North Anna Units 1 and 2 TS.

2.3 LTOP Setpoints

The LTOP system mitigates overpressure transients at low temperatures so that the integrity of the reactor coolant pressure boundary is not compromised by violating the 10 CFR Part 50, Appendix G, P-T limits under steady-state operating conditions. North Anna Units 1 and 2 LTOP systems use the pressurizer power-operated relief valves (PORV) or a reactor coolant system (RCS) vent with the reactor depressurized to accomplish this function. The system is manually enabled by operators and uses two lifting setpoints for the PORV during different RCS temperatures. The design basis of the North Anna, Units 1 and 2, LTOP considers both mass-addition and heat-addition transients. The limiting mass-addition analyses account for the injection from one charging pump to the water-solid RCS. The heat-addition analyses account for heat input from the secondary side of the steam generators into the RCS upon starting an idle reactor coolant pump (RCP) when the steam generator secondary water temperature is less than 50°F above the RCS cold-leg temperature.

In North Anna Unit 1, the current Limiting Condition for Operation (LCO) in TS 3.4.9.3 requires that an LTOP shall be operable with two operable PORVs with a lifting setting of: 1) less than or equal to 500 psig whenever any RCS cold leg temperature is less than or equal to 235 °F, and 2) less than or equal to 395 psig whenever any RCS temperature is less than or equal to 150 °F. This LCO is applicable when any RCS cold-leg temperature is less than or equal to 235 °F when the head is on the reactor vessel and the RCS is not vented through a 2.07 square inch or larger vent. In Unit 2, the current LCO 3.4.9.3 requires that an LTOP shall be operable with two operable PORVs with a lifting setting of 1) less than or equal to 415 psig whenever any RCS cold-leg temperature is less than or equal to 270 °F, and 2) less than or equal to 375 psig whenever any RCS cold-leg temperature is less than or equal to130 °F. This LCO is applicable when any RCS cold-leg temperature is less than or equal to 270 °F when the head is on the reactor vessel and the RCS is not vented through a 2.07 square inch or larger vent.

Since the use of the ASME Code Case N-641 compensates for the more restrictive 10 CFR Part 50, Appendix G P-T limits in the proposed TS affecting LTOP setpoints, the licensee has proposed to maintain the current LTOP setpoints unchanged. The staff's evaluation of the licensee's justification regarding the LTOP setpoints is discussed below.

The LTOP system is required to mitigate overpressure transients at low temperature operations to prevent violating 10 CFR Part 50, Appendix G P-T limits. However, the licensee is requesting exemption to this rule and adopting the use of ASME Code Case N-641. The use of ASME Code Case N-641 will result in predictions of P-T limiting values less restrictive than the current TS values. In addition, the staff has accepted the use of the P-T limits that are developed using the provisions in ASME Code Case 641 for the design of the LTOP system. The LTOP system actuation setpoints are the pressures at which the PORVs will lift, when the LTOP system is enabled, to limit the peak RCS pressure within the acceptable limits during a pressurization transient.

The licensee has proposed that the current PORV actuation setpoints for North Anna Units 1 and 2 remain unchanged to protect the proposed P-T limits in TS Figures 3.4-2 and 3.4-3. Since the P-T limits at Units 1 and 2 would be only slightly changed, and the method for developing the PORV setting to protect these P-T limits has been accepted by the staff, we find that the current PORV setpoints will provide adequate protection to the P-T limits proposed in TS Figures 3.4-2 and 3.4-3 during a design-basis overpressure transient (mass-addition or heat-addition) as described above. Based on the above discussion, we find the proposed PORV setpoints acceptable.

The staff has reviewed the licensee's justification for the unchanged LTOP system enable temperature and PORV actuation setpoint as discussed above. We find that the proposed P-T limits up to 32.3 EFPY at Unit 1 and 34.3 EFPY at Unit 2 will be adequately protected with these current LTOP setpoints; therefore, the licensee's proposal is acceptable.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement 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 (66 FR 11334). 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.

5.0 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) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: A. Hiser
C. Liang
L. Lois

Date: May 2, 2001

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

VIRGINIA ELECTRIC AND POWER COMPANY

NORTH ANNA POWER STATION UNITS 1 AND 2

DOCKET NOS. 50-338 AND 50-339

EXEMPTION

1.0 BACKGROUND

The Virginia Electric and Power Company (the licensee) is the holder of Facility Operating Licenses NPF-4 and NPF-7, which authorize operation of the North Anna Power Station, Units 1 and 2. The licenses provide, among other things, that the facility is subject to all rules, regulations, and orders of the U.S. Nuclear Regulatory Commission (NRC, the Commission) now or hereafter in effect.

The facility consists of two pressurized-water reactors located in Louisa County in the Commonwealth of Virginia.

2.0 PURPOSE

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix G requires that pressure-temperature (P-T) limits be established for reactor pressure vessels (RPVs) during normal operating, and hydrostatic pressure or leak testing conditions. Specifically, 10 CFR Part 50, Appendix G states that "[t]he appropriate requirements on...the pressure-temperature limits and minimum permissible temperature must be met for all conditions." Appendix G of 10 CFR Part 50 specifies that the requirements for these limits are the American Society of Mechanical Engineers (ASME) Code, Section XI, Appendix G Limits. RG 1.99, Rev. 2, provides guidance for implementing 10 CFR Part 50, Appendix G. In GL 88-11, the NRC staff advised licensees that the staff would use RG 1.99, Rev. 2, to review P-T limit curves. RG 1.99, Rev. 2, contains conservative methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy resulting from neutron radiation.

To address provisions of amendments to the technical specifications (TS) regarding the P-T limits, low temperature overpressure protection (LTOP) system setpoints, and LTOP system effective temperature (T_{enable}), the licensee requested in its submittal dated June 22, 2000, as supplemented by letters dated September 19, 2000, and January 4, February 14, March 13, March 22, and April 11, 2001, that the staff exempt North Anna Units 1 and 2 from application of specific requirements of 10 CFR Part 50, Appendix G, and substitute use of ASME Code Case N-641. Code Case N-641 permits the use of an alternate reference fracture toughness (K_{IC} fracture toughness curve instead of K_{Ia} fracture toughness curve) for reactor vessel materials in determining the P-T limits, LTOP system setpoints and T_{enable}, and provides for plant-specific evaluation of T_{enable}. Since the K_{IC} fracture toughness curve shown in ASME Section XI, Appendix A, Figure A-2200-1 (the K_{IC} fracture toughness curve) provides greater allowable fracture toughness than the corresponding K_{Ia} fracture toughness curve of ASME Section XI, Appendix G, Figure G-2210-1 (the K_{Ia} fracture toughness curve) and a plant-specific evaluation of T_{enable} would give lower values of T_{enable} than use of a generic bounding evaluation for T_{enable}, use of Code Case N-641 for establishing the P-T limits, LTOP system setpoints and T_{enable} would be less conservative than the methodology currently endorsed by 10 CFR Part 50, Appendix G and, therefore, an exemption to apply the Code Case would be required by 10 CFR 50.60. Although the use of the K_{IC} fracture toughness curve in ASME Code Case N-641 was recently incorporated into Appendix G to Section XI of the ASME Code, an exemption is still needed because 10 CFR Part 50, Appendix G requires the licensee's analysis to use an edition and addenda of Section XI of the ASME Code incorporated by reference into 10 CFR 50.55a, i.e., the editions through 1995 and addenda through the 1996 addenda (which do not include the provisions of Code Case N-641).

The proposed amendments submitted by the licensee will revise the P-T limits of TS 3/4.4.9 related to the heatup and cooldown of the reactor coolant system (RCS), the LTOP system setpoints and T_{enable} for the LTOP system, for operation to 32.3 effective full power years (EFPY) for Unit 1 and 34.3 EFPY for Unit 2.

ASME Code Case N-641

The licensee has proposed an exemption to allow use of ASME Code Case N-641 in conjunction with ASME Section XI, 10 CFR 50.60(a) and 10 CFR Part 50, Appendix G, to determine the P-T limits, LTOP system setpoints and T_{enable}.

The proposed amendments to revise the P-T limits, LTOP system setpoints and T_{enable} for North Anna Units 1 and 2 rely in part on the requested exemption. The revised P-T limits, LTOP system setpoints and T_{enable} have been developed using the K_{Ic} fracture toughness curve, in lieu of the K_{Ia} fracture toughness curve, as the lower bound for fracture toughness of the RPV materials.

Use of the K_{Ic} curve in determining the lower bound fracture toughness of RPV steels is more technically correct than use of the K_{Ia} curve since the rate of loading during a heatup or cooldown is slow and is more representative of a static condition than a dynamic condition. The K_{Ic} curve appropriately implements the use of static initiation fracture toughness behavior to evaluate the controlled heatup and cooldown process of a reactor vessel. The staff has required use of the conservatism of the K_{Ia} curve since 1974, when the curve was adopted by the ASME Code. This conservatism was initially necessary due to the limited knowledge of the fracture toughness of RPV materials at that time. Since 1974, additional knowledge has been gained about RPV materials, which demonstrates that the lower bound on fracture toughness provided by the K_{Ia} curve greatly exceeds the margin of safety required to protect the public health and safety from potential RPV failure. In addition, P-T curves, LTOP setpoints, and T_{enable} based on the K_{Ic} curve will enhance overall plant safety by opening the P-T operating window, with the greatest safety benefit in the region of low temperature operations.

Since an unnecessarily reduced P-T operating window can reduce operator flexibility without just basis, implementation of the proposed P-T curves, LTOP setpoints, and T_{enable} as allowed by ASME Code Case N-641 may result in enhanced safety during critical plant operational periods, specifically heatup and cooldown conditions. Thus, pursuant to 10 CFR 50.12(a)(2)(ii), the underlying purpose of 10 CFR 50.60 and Appendix G to 10 CFR Part 50 will continue to be served.

In summary, the ASME Section XI, Appendix G, procedure was conservatively developed based on the level of knowledge existing in 1974 concerning RPV materials and the estimated effects of operation. Since 1974, the level of knowledge about these topics has been greatly expanded. The staff concurs that this increased knowledge permits relaxation of the ASME Section XI, Appendix G requirements by application of ASME Code Case N-641, while maintaining, pursuant to 10 CFR 50.12(a)(2)(ii), the underlying purpose of the NRC regulations to ensure an acceptable margin of safety.

3.0 DISCUSSION

Pursuant to 10 CFR 50.12, the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of 10 CFR Part 50, when (1) the exemptions are authorized by law, will not present an undue risk to public health or safety, and are consistent with the common defense and security; and (2) when special circumstances are present. The staff accepts the licensee's determination that an exemption would be required to approve the use of Code Case N-641. The staff examined the licensee's rationale to support the exemption request and concurs that the use of the Code case would meet the underlying intent of these regulations. Based upon a consideration of the conservatism that is explicitly incorporated into the methodologies of 10 CFR Part 50, Appendix G; Appendix G of the Code; and Regulatory Guide 1.99, Revision 2, as discussed above, the staff concludes that application of the Code case as described would provide an adequate margin of safety against brittle failure of the RPV. This conclusion is also consistent with the determinations that the staff has reached for other licensees under similar conditions based on the same considerations.

Therefore, the staff concludes that granting an exemption under the special circumstances of 10 CFR 50.12(a)(2)(ii) is appropriate and that the methodologies of Code Case N-641 may be used to revise the P-T limits, LTOP setpoints, and T_{enable} for North Anna Power Station, Units 1 and 2.

4.0 CONCLUSION

Accordingly, the Commission has determined that, pursuant to 10 CFR 50.12(a), the exemption is authorized by law, will not endanger life or property or common defense and security, and is, otherwise, in the public interest. Therefore, the Commission hereby grants the licensee an exemption from the requirements of 10 CFR Part 50, Appendix G, for North Anna Power Station, Units 1 and 2.

Pursuant to 10 CFR 51.32, the Commission has determined that the granting of this exemption will not have a significant effect on the quality of the human environment (66 FR 22018).

This exemption is effective upon issuance.

Dated at Rockville, Maryland, this 2nd day of May 2001.

FOR THE NUCLEAR REGULATORY COMMISSION

John A. Zwolinski, Director
Division of Licensing Project Management
Office of Nuclear Reactor Regulation