Virginia Electric and Power Company
Richmond, Virginia 23261

April 15, 1994

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No.	94-238
NL&P/JBL	R0
Docket Nos.	50-338
	50-339
License Nos.	NPF-4
	NPF-7

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION UNITS 1 AND 2
PROPOSED TECHNICAL SPECIFICATIONS CHANGE

Pursuant to 10 CFR 50.90, the Virginia Electric and Power Company requests amendments, in the form of changes to the Technical Specifications, to Facility Operating License Numbers NPF-4 and NPF-7 for North Anna Power Station Units 1 and 2, respectively. The proposed changes modify the pressure / temperature operating limitations during heatup and cooldown and the Low Temperature Overpressure Protection System (LTOPS) pressure setpoints and enabling temperatures for Units 1 and 2. The proposed changes include revised Limiting Conditions for Operation, Action Statements, and Surveillance Requirements for the PORVs and block valves to address the concerns discussed in NRC Generic Letter 90-06. The proposed changes also include several editorial / administrative changes.

A discussion of the proposed Technical Specifications changes is provided in Attachment 1. The proposed Technical Specifications changes are provided in Attachment 2. It has been determined that the proposed Technical Specifications changes do not involve an unreviewed safety question as defined in 10 CFR 50.59 or a significant hazards consideration as defined in 10 CFR 50.92. The basis for our determination that the changes do not involve a significant hazards consideration is provided in Attachment 3. The proposed changes to the heatup and cooldown curves for North Anna Unit 1 are supported by Westinghouse Electric Corporation Report WCAP-13831, Revision 1, "Heatup and Cooldown Limit Curves for Normal Operation for North Anna Unit 1," dated November 1993. A non-proprietary version of WCAP-13831, Revision 1, is provided as Attachment 4. The proposed Technical Specifications changes have been reviewed and approved by the Station Nuclear Safety and Operating Committee and the Management Safety Review Committee.

These proposed changes will incorporate analytical and operational features into the North Anna design basis which improve system availability and reliability, provide additional pressure / temperature operating margin and operational flexibility, and reduce the potential for undesired PORV lifts. These operational improvements are

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needed for the next operating cycle for North Anna Unit 1. Therefore, NRC approval of these proposed Technical Specifications changes is requested by October 1, 1994 to support the scheduled startup on November 1, 1994.

Should you have any questions or require additional information, please contact us.

Very truly yours,

J. P. O'Hanlon

for W. L. Stewart
Senior Vice President - Nuclear

Attachments

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COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by J. P. O'Hanlon, who is Vice President - Nuclear Operations, for W. L. Stewart who is Senior Vice President - Nuclear, of Virginia Electric and Power Company. He is duly authorized to execute and file the foregoing document in behalf of that Company, and the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 15TH day of April, 1994.

My Commission Expires: May 30, 1994.

Vicki L. Hull
Notary Public

(SEAL)

ATTACHMENT 1

DISCUSSION OF CHANGES

VIRGINIA ELECTRIC AND POWER COMPANY

1.0 INTRODUCTION

Virginia Electric and Power Company proposes changes to the North Anna Units 1 and 2 Technical Specifications to provide operating limits, setpoints, and component operability requirements which ensure reactor vessel integrity during normal operation and postulated accident conditions.

The North Anna Units 1 and 2 Reactor Coolant Systems (RCS) are protected from material failure by the imposition of restrictions on allowable pressure and temperature, and on heatup and cooldown rate. The Low Temperature Overpressure Protection System (LTOPS) ensures that material integrity limits are not exceeded during the design basis overpressurization accidents. Equipment operability requirements are imposed to ensure that the assumptions of the accident analyses remain valid.

The proposed Technical Specifications changes provide revised pressure/temperature operating limits and LTOPS setpoints valid to end-of-license for Unit 1. Revised LTOPS setpoints based on existing pressure/temperature limit data valid to 17 EFPY are provided for Unit 2. For both North Anna Units, the proposed changes incorporate analytical and operational features which (a) address the LTOPS availability and reliability concerns of Generic Letter 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,' Pursuant to 10 CFR 50.54(f)," dated June 25, 1990, (b) provide additional pressure/temperature operating margin and operational flexibility, and (c) reduce the potential for undesired PORV lifts.

2.0 BACKGROUND

In 1991, Virginia Electric and Power Company performed calculations to support implementation of LTOPS setpoints and enabling temperatures based on revised pressure/temperature operating limit curves applicable to 12 EFPY and 17 EFPY for North Anna Units 1 and 2, respectively. (North Anna Unit 1 is expected to reach 12 EFPY in January of 1996; Unit 2 is expected to reach 17 EFPY in June of 2002.) Proposed Technical Specifications changes to implement the revised curves and setpoints were submitted to the NRC on December 29, 1991 (1) and were approved by the NRC on March 25, 1993 (2).

It was anticipated that North Anna Unit 1 Reactor Vessel Materials Surveillance Capsule X would be withdrawn during the Cycle 10 refueling and steam generator replacement outage in 1993. This capsule was to provide updated material properties data to support the preparation of revised pressure/temperature operating limit curves. However, despite two attempts, outage support personnel were unable to remove Capsule X from its holder on the thermal shield. Therefore, it became necessary to prepare revised Unit 1 pressure/temperature operating limits (3) and LTOPS setpoints on the basis of existing fluence estimates (4) and Regulatory Guide 1.99, Revision 2 RT_{NDT} calculations. A revised Unit 1 ASTM E-185 surveillance capsule removal schedule was prepared (5) and transmitted to the NRC on August 26, 1993 (6) in accordance with the requirements of 10 CFR 50 Appendix H.

In 1990, the NRC issued its Staff Position on Generic Issues 70 and 94 (7). Investigation of Generic Issue 70 was initiated to address concerns regarding the reliability of PORVs and block valves (8). Generic Issue 94 was initiated to address concerns regarding LTOPS availability (9),(10). As described in the December 21, 1990 response (11) to Generic Letter 90-06, Virginia Electric and Power Company intended to address the concerns raised by Generic Issue 70 through existing PORV testing requirements, and to address the concerns raised by Generic Issue 94 through implementation of the Westinghouse MERITS Technical Specification improvement program, which would have included specific Technical Specifications that established a maximum allowable outage time of 7 days for LTOPS.

By letter dated March 16, 1993 (12), the NRC provided additional guidance to Virginia Electric and Power Company for Generic Issues 70 and 94. Further, the NRC confirmed that Virginia Electric and Power Company's withdrawal from participation as lead plant in the Westinghouse MERITS program precluded the possibility of resolving Generic Issue 94 through implementation of the new Standard Technical Specifications.

Proposed Technical Specifications have been developed which address the concerns of Generic Issues 70 and 94. The proposed Technical Specifications incorporate revised Action Statements and additional Surveillance Requirements for the PORVs and PORV control system. These changes address the PORV reliability concerns of Generic Letter 90-06. Further, the Action Statements associated with the Technical Specification governing low temperature overpressure protection have been revised to address the PORV availability concerns of Generic Letter 90-06.

3.0 SPECIFIC CHANGES

The Technical Specification changes described herein apply to North Anna Units 1 and 2. In addition to the specific changes described below, editorial changes have been made to correct grammatical errors and format inconsistencies.

Technical Specification 3.1.2.2
Reactivity Control Systems - Flow Paths - Operating

The existing footnote to Technical Specification 3.1.2.2 has been revised to specify that only one boron flow path is required to be operable whenever the temperature of one or more of the RCS cold legs is less than or equal to the LTOPS enabling temperature (235°F for Unit 1; 270°F for Unit 2). This requirement is provided to ensure consistency with the requirements of TS 3.1.2.4 (charging pump operability), and to ensure that actual operating conditions are consistent with those assumed in the mass addition transient analysis. The mass addition transient analysis assumes that only one charging pump will be operable below the LTOPS enabling temperature. Below the enabling temperatures, the anticipated low temperature overpressurization accidents may be adequately mitigated by the automatic actuation of a single PORV. Above the LTOPS enabling temperature, overpressurization due to the inadvertent startup of two charging pumps is adequately mitigated by actuation of the pressurizer safety valves.

Technical Specification 3/4.1.2.4
Reactivity Control Systems - Charging Pumps - Operating

The Action Statement, Surveillance Requirement, and footnote to Technical Specification 3/4.1.2.4 have been revised to specify that a maximum of one centrifugal charging pump shall be operable whenever the temperature of one or more of the RCS cold legs is less than or equal to the LTOPS enabling temperature (235°F for Unit 1; 270°F for Unit 2). This requirement is provided to ensure that actual operating conditions are consistent with those assumed in the mass addition transient analysis. The mass addition transient analysis assumes that only one charging pump will be operable below the LTOPS enabling temperature. Below the enabling temperatures, the anticipated low temperature overpressurization accidents may be adequately mitigated by the automatic actuation of a single PORV. Above the LTOPS enabling temperature, overpressurization due to the inadvertent startup of two charging pumps is adequately mitigated by actuation of the pressurizer safety valves.

Technical Specification 3.4.1.2 (Unit 2 Only)
Reactor Coolant System - Hot Standby - Mode 3

A previous Technical Specification change added a footnote to Unit 2 TS 3.4.1.2 to specify that a reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to 358°F unless the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures. It was necessary to include this footnote in TS 3.4.1.2 (Mode 3) because the setpoint encompassed a small portion of Mode 3 (350°F < T_{avg} < 358°F). This footnote is being deleted since the proposed temperature limit is being changed from 358°F to 270°F. The proposed TS 3.4.1.3 ensures that actual operating conditions are consistent with those assumed in the heat addition transient analysis.

Technical Specification 3.4.1.3
Reactor Coolant System - Shutdown - Modes 4 and 5

An existing footnote to Technical Specification 3.4.1.3 has been revised to specify that a reactor coolant pump shall not be started with the temperature of one or more of the RCS cold legs less than or equal to the LTOPS enabling temperature (235°F for Unit 1; 270°F for Unit 2). This requirement is provided to ensure that actual operating conditions below the LTOPS enabling temperature are consistent with those assumed in the heat addition transient analysis. The heat addition transient analysis assumes that a 50°F temperature differential exists between the secondary and primary sides of the steam generator when a reactor coolant pump is started. Below the enabling temperatures, the anticipated low temperature overpressurization accidents may be adequately mitigated by the automatic actuation of a single PORV. Above the LTOPS enabling temperature, overpressurization is adequately mitigated by actuation of the pressurizer safety valves.

Technical Specification 3.4.3 (Unit 1 Only)
Reactor Coolant System - Safety Valves - Operating

To be consistent with the Unit 2 Technical Specifications, the number of Unit 1 TS 3.4.3 is being changed from "TS 3.4.3" to "TS 3.4.3.1", and the section title is being corrected. This change is editorial in nature.

Technical Specification 3/4.4.3.2
Reactor Coolant System - Relief Valves - Modes 1, 2, and 3

TS 3.4.3.2 and the associated Action Statement have been modified to address the concerns of Generic Letter 90-06 (7). The changes to TS 4.4.3.2 include revised Surveillance Requirements for PORV and control system testing. The existing PORV monthly channel functional test has been retained as TS 4.4.3.2.1.a. Surveillance Requirements for emergency (backup) power supply testing of the PORVs and block valves were not added because the valves are powered from safety grade power sources. The changes to this Specification are consistent with the guidelines presented in NRC Generic Letter 90-06 (7) for the North Anna Units 1 and 2 plant configuration.

A subtitle has been added to Unit 1 TS 3.4.3.2 to make the title of this specification consistent with the section titling convention employed in the Technical Specifications.

Technical Specification Figures 3.4-2 and 3.4-3
Reactor Coolant System - Pressure/Temperature Limits

Revised T.S. Figures 3.4-2 and 3.4-3 have been prepared to present the revised Unit 1, 30.7 EFPY and existing Unit 2, 17 EFPY pressure/temperature operating limit data. The curves have been modified to include a correction for the pressure difference between the point of measurement (i.e., the pressurizer) and the point of interest (i.e., the reactor vessel beltline), but do not include allowances for temperature and pressure measurement uncertainty. The 10 CFR 50 Appendix G criticality limit line has been excluded in favor of the more restrictive Technical Specification 3.1.1.5, Minimum Temperature for Criticality.

Technical Specification 3/4.4.9.3
Reactor Coolant System - Overpressure Protection Systems

The format and content of Technical Specification 3/4.4.9.3 have been revised to address the LTOPS availability concerns of NRC Generic Letter 90-06 (7), and to define revised LTOPS setpoints and enabling temperatures. The Applicability statement has been revised to define the Modes in which the Limiting Condition for Operation (LCO) is applicable, and to include the provision for RCS venting. Surveillance Requirement 4.4.9.3.2 and the associated footnote have been relocated to TS 3.4.9.3 Action Statement d, consistent with the guidance of Generic Letter 90-06.

The LTOPS setpoints and enabling temperatures were developed to provide bounding low temperature reactor vessel integrity protection during the postulated design basis mass and heat addition transients. The isothermal limit curve is used to establish the LTOPS setpoint. This approach is discussed in Section 4.6. Above the LTOPS enabling temperature, actuation of the pressurizer safety valves is adequate to ensure reactor vessel integrity during the LTOPS design basis transients.

Technical Specification 3.5.2
ECCS Subsystems - T_{avg} ≥ 350°F

A previous LTOPS Technical Specification change modified the Units 1 and 2 TS 3.5.2 ACTION "c" to allow the provisions of Specification 3.0.4 to be not applicable to ACTIONS "a" and "b" for one hour following heatup above 316°F (358°F for Unit 2) or prior to cooldown below 316°F (358°F for Unit 2). In addition, a footnote was added to Unit 1 TS 3.5.2 to indicate that a maximum of one centrifugal charging pump may be operable whenever the temperature of one or more of the RCS cold legs is less than or equal to 358°F. The footnote to Unit 2 TS 3.5.2 was necessary because the Unit 2 temperature limit involved Mode 3 operation between 350°F and 358°F.

Units 1 and 2 TS 3.5.2 are being modified to reflect the revised temperature limit (235°F for Unit 1; 270°F for Unit 2) for ensuring that actual operating conditions are consistent with those assumed in the accident analysis. Because the revised temperature limit does not involve Mode 3 operation, the Unit 2 TS 3.5.2 footnote is being deleted.

A footnote to Unit 1 TS 3.5.2 imposed ECCS component operability requirements that were to be in effect until steam generator replacement. Because Unit 1 steam generator replacement has been accomplished, this footnote is being removed.

Technical Specification 3/4.5.3
ECCS Subsystems - T_{avg} < 350°F

The existing footnote in TS 3.5.3 has been revised to specify that a maximum of one centrifugal charging pump may be operable whenever the temperature of one or more of the RCS cold legs is less than or equal to 235°F (Unit 1; 270°F for Unit 2). Surveillance Requirement 4.5.3.2 is also being changed to reflect the revised temperature limit. These requirements are provided to ensure that actual operating conditions are consistent with those assumed in the mass addition transient analysis. The mass addition transient analysis assumes that only one charging pump will be operable below the LTOPS enabling temperature. Below the enabling temperatures, the anticipated low temperature overpressurization accidents may be adequately mitigated by the automatic action of a single PORV. Above the LTOPS enabling temperature, overpressurization due to the inadvertent startup of two charging pumps is adequately mitigated by actuation of the pressurizer safety valves.

Technical Specifications 3/4 Bases

The proposed Bases for TS 3/4.1.2 (Boration Systems) and TS 3/4.5.2 and TS 3/4.5.3 (ECCS Subsystems) incorporate the revised temperature below which charging pump operability requirements must be observed. The proposed Bases for TS 3/4.4.1 (Reactor Coolant Loops) incorporate the revised temperature below which the steam generator secondary-to-primary temperature difference must be less than 50°F. The Bases for TS 3/4.4.2 and 3/4.4.3 (Safety and Relief Valves) have been modified to reflect the proposed changes made in response to Generic Letter 90-06 (7). The Bases for TS 3/4.4.9 have been modified to reflect current information on the development of pressure/temperature operating limits and LTOPS setpoints.

Technical Specification 6.9.2

The reference to the title of TS 3.4.9.3 (Item "i" of TS 6.9.2) has been modified to be consistent with the proposed title of TS 3.4.9.3.

Item "h" of TS 6.9.2 has been deleted. This item was previously deleted by License Amendments 96/83 for North Anna Units 1 and 2, but was inadvertently reinserted by a subsequent license amendment.

4.0 SAFETY SIGNIFICANCE

This section presents a safety evaluation which supports the proposed changes to the North Anna Units 1 and 2 Technical Specifications. The information presented in each section is summarized below.

Section 4.1 The reactor vessel material surveillance capsule results which support the proposed changes are identical to those which support the existing Technical Specification pressure/temperature operating limits (1),(2). Although no new surveillance capsule results are presented, a summary of the results which support the revised North Anna Unit 1 pressure/temperature operating limits is provided as background information in Section 4.1.

Section 4.2 Section 4.2 presents the reactor vessel materials data used in calculations of irradiated RT_{NDT} for North Anna Unit 1.

Section 4.3 Revised design basis end-of-license RT_{NDT} values are presented for each North Anna Unit 1 reactor vessel beltline material in Section 4.3.

Section 4.4 Section 4.4 discusses the overpressurization analysis which supports the proposed Technical Specification LTOPS setpoints. The overpressurization analysis is identical to that which supports the existing North Anna Units 1 and 2 Technical Specifications LTOPS setpoints (1),(2).

Section 4.5 The proposed Unit 1 (3) and existing Unit 2 (13) pressure/temperature operating limits are discussed in Section 4.5.

Section 4.6 A description of the development and evaluation of revised LTOPS setpoints and LTOPS enabling temperatures for each unit is presented in Section 4.6.

Section 4.7 Component operability requirements, including proposed Technical Specifications changes to address the concerns of Generic Letter 90-06, are described in Section 4.7.

4.1 North Anna Unit 1 Surveillance Capsule Results

Credible surveillance data for North Anna Unit 1 beltline materials are available from two surveillance capsules, V (14) and U (4). The North Anna Unit 1 surveillance program includes Forging 03 (SA508, Class 2) and Circumferential Weld 04 (Rotterdam Weld). Fluence estimates used in the present analysis are based on the Capsule U results (4).

4.1.1 North Anna 1 Capsule V

Capsule V was removed from North Anna Unit 1 at the end of the first cycle of operation at a cumulative core burnup of 1.13 EFPY. The capsule dosimeters were evaluated and found to have a cumulative fast neutron (E > 1.0 MeV) fluence of 2.49×10^{18} n/sq.cm. (14).

The irradiated specimens test results were compared to unirradiated specimen test results. The Charpy V-notch impact test results show the irradiation has increased the average base metal 30 ft-lb transition temperature by 21°F (axial, or transverse orientation) and 39°F (tangential, or longitudinal orientation). The weld metal 30 ft-lb transition temperature increased by 78°F.

Reference (14) should be consulted for further information on Capsule V analysis results.

4.1.2 North Anna 1 Capsule U

Capsule U was removed from North Anna Unit 1 at the end of the sixth cycle of operation. The capsule dosimeters were evaluated and found to have a cumulative fast neutron (E > 1.0 MeV) fluence of 8.28×10^{18} n/sq.cm. (4).

The irradiated specimens test results were compared to unirradiated specimen test results. The Charpy V-notch impact test results show the irradiation has increased the average base metal 30 ft-lb transition temperature by 95°F (tangential, or longitudinal orientation) and 65°F (axial, or transverse orientation). The weld metal 30 ft-lb transition temperature increased by 75°F.

Reference (4) should be consulted for further information on Capsule U analysis results.

4.2 North Anna Unit 1 Reactor Vessel Materials Data

Reactor vessel beltline material chemistry, neutron fluence, and unirradiated RT_{NDT} data are necessary for performing Regulatory Guide 1.99, Revision 2 irradiated RT_{NDT} calculations. A summary of the data used in the calculations (3) which support the proposed Technical Specification changes is presented below. The chemistry and unirradiated RT_{NDT} data is identical to that presented in the October 22, 1992 response (26) to Generic Letter 92-01 (16) for North Anna Units 1 and 2 (15).

North Anna Unit 1

Material	Wt.% Cu	Wt.% Ni	Unirrad. RT_{NDT} (F)	Fluence * (10^{19} n/cm^2)	Fluence Reference
Forg. 03	0.15	0.80	38	3.95	(4)
Forg. 04	0.12	0.82	17	3.95	(4)
Forg. 05	0.16	0.74	6	0.277	(4)
Weld 04	0.086	0.11	19	3.95	(4)
Weld 05A	0.30	0.10	0	0.277	(4)
Weld 05B	0.11	0.10	0	0.277	(4)

* End-of-license vessel inner surface fluence values. Forging 05, Weld 05A, and Weld 05B fluences are 7% of the peak vessel inner-surface fluence.

4.3 North Anna Unit 1 Irradiated RT_{NDT} Values

In accordance with the methods prescribed by Regulatory Guide 1.99, Revision 2, adjusted RT_{NDT} values have been calculated for each North Anna Unit 1 reactor vessel beltline material at a fluence corresponding to end-of-license (EOL), or 30.7 EFPY. Surveillance data were used to calculate the adjusted RT_{NDT} for the Lower Shell Forging 03 and the Circumferential Weld 04. The limiting 30.7 EFPY values of RT_{NDT} at the 1/4T and 3/4T locations were shown to occur in the Unit 1 Circumferential Weld 04. A summary of the adjusted RT_{NDT} calculations is provided below:

Material	1/4-T ART (°F)	3/4-T ART (°F)
Lower Shell Forging 03	215.2 (146.5)	186.7 (128.3)
Inter. Shell Forg. 04	158.1	136.8
Upper Shell Forging 05	140.3	117.3
Circ. Weld 04	137.5 (162.9)*	119.1 (139.9)*
Weld 05A	143.4	111.2
Weld 05B	82.4	65.4

ART numbers within () are based on chemistry factors calculated using surveillance capsule data. Adjusted reference temperature values used to generate pressure/temperature operating limits (3) are marked with an asterisk (*).

The calculations of adjusted RT_{NDT} for North Anna Unit 1 are based on a peak vessel inner surface fluence of 3.95×10^{19} n/sq.cm. Table 6-15 of the Capsule U analysis results (4) demonstrates that a 30.7 EFPY fluence of 3.70×10^{19} n/sq.cm. is justified. Because use of the higher fluence value did not significantly impact the calculated RT_{NDT} or the pressure/temperature limit results, the higher fluence was used. This fluence margin may be used to address future analytical or operational issues.

4.4 Overpressurization Analysis

Cold overpressure protection is provided to ensure that the combined pressure and thermal stresses experienced during a design basis overpressurization accident remain well below those which could result in vessel fracture. The PORV setpoints are based on the analysis of two design basis accidents: the inadvertent startup of a charging pump and the startup of a reactor coolant pump in an RCS loop with a 50°F difference between the steam generator secondary fluid temperature and the RCS temperature. Only one PORV is assumed to operate during the transients.

The proposed LTOPS setpoints are based on the same overpressurization analysis results which were used to develop the existing North Anna Units 1 and 2 Technical Specifications LTOPS setpoints (1),(2). As described in the December 29, 1991 submittal (1), the overpressurization analysis results revealed that the mass addition transient produces the most limiting results. The following sections describe the inputs to the North Anna RETRAN (17) model and the analysis to determine the new PORV setpoints.

4.4.1 Mass Addition Transient

The inadvertent startup of a single charging pump was selected as the design basis mass addition transient based on previous UFSAR work (Reference (18), Section 5.2.2.2). Because of the valve opening characteristic associated with the air-operated relief valves used on the pressurizer at North Anna (19),(20), the inadvertent startup of a charging pump at water-solid conditions results in pressurization beyond the PORV lift setpoint. The objective of the analysis was to determine the extent to which RCS pressure exceeded the pressurizer PORV lift setpoint following inadvertent startup of a charging pump during water-solid operation.

The effects of pressure measurement location were explicitly considered in the overpressurization analysis. Specifically, pressurizer PORV actuation was based on hot leg pressure in the RETRAN model. The "PORV lift setpoint overshoot" was defined as the difference between the maximum reactor vessel beltline pressure and the PORV lift setpoint.

The mass addition analysis was performed at the initial conditions listed in the table below. The initial RCS temperature, pressure, and PORV setpoint were varied to observe the effects of changes in these parameters. A range of RCS temperatures between 100°F and 325°F were examined, as well as a range of initial pressures. The analysis revealed a gradually decreasing PORV lift setpoint overshoot with increasing initial RCS temperature and PORV setpoint. The peak RCS pressure was found to be relatively insensitive to the initial RCS pressure. The proposed PORV lift setpoints (Section 4.6.5) were validated by adding the PORV lift setpoint overshoot values to the proposed lift setpoint at each temperature, and verifying that the resulting pressures did not violate the design pressure/temperature limit curve. Selection of the design pressure/temperature limit curve is discussed in Section 4.6.

Reactor Coolant Temperature (°F)	100, 150, 200, 250, 300, 325
Reactor Coolant Pressure (psig)	200, 250, 300, 340, 380, 400
Maximum Charging Pump Flow Rate (Design Basis flow vs. head curve)	705 gpm
Pressurizer Steam Volume	0 ft^3
Pressurizer Water Volume	1400 ft^3
Reactor Coolant System Flow	10%
PORV OPEN Setpoint	Variable
PORV Closed Setpoint	OPEN-15 psi

Initial Conditions for the Mass Addition Transient

4.4.2 Heat Addition Transient

The heat addition transient assumes that a reactor coolant pump (RCP) is started with the maximum temperature difference allowed by Technical Specifications (50°F) between the steam generators and the RCS. This scenario has been determined to be the design basis heat addition transient for LTOPS setpoint determination (Reference (18), Section 5.2.2.2).

The heat addition transient was modelled assuming the initial conditions listed in the table below. The secondary-to-primary heat transfer modelling included a very conservative evaluation of the local secondary side convection heat transfer coefficient, and an assumed constant bulk secondary side temperature (i.e., no credit was taken for decreasing temperature due to secondary-to-primary heat transfer). The pump startup flow characteristic was also modelled in a conservative fashion. The analysis revealed that the results of the heat addition transient are easily bounded by those of the mass addition transient.

Reactor Coolant Temperature	100°F
Reactor Coolant Pressure (psig)	280, 340
RCS/SG ΔT	50°F
Pressurizer Steam Volume	0 ft^3
Pressurizer Water Volume	1400 ft^3
RCP Speeds In Affected Loop, startup In Unaffected Loop, coastdown	 10% - 100% 10% - 0%
PORV Open Setpoint	Variable
PORV Closed Setpoint	OPEN - 15 psi

Initial Conditions for the Heat Addition Transient

4.5 Revised Technical Specification Pressure/Temperature Operating Limits

North Anna Unit 1 pressure/temperature operating limits valid to 30.7 EFPY have been developed and are presented in Reference (3). The North Anna Unit 2 curve data (13) are unchanged from those currently in the Technical Specifications. References (3) and (13) should be consulted for details concerning the development of these curves. Heatup rates of 20°F/hr, 40°F/hr, and 60°F/hr, and cooldown rates of 0°F/hr (steady-state), 20°F/hr, 40°F/hr, 60°F/hr, and 100°F/hr were considered.

The criticality limit required by 10 CFR 50 Appendix G is not included with the proposed pressure/temperature operating limits, since Limiting Condition for Operation (LCO) 3.1.1.5 defines a minimum temperature for criticality that is substantially more limiting than the criticality limit required by 10 CFR 50, Appendix G. LCO 3.1.1.5 restricts the lowest operating loop average temperature to ≥541°F for Modes 1 and 2.

The proposed pressure/temperature operating limits include a correction for the effects of pressure measurement location. Specifically, the allowable pressures have been reduced to compensate for the difference between the point of measurement (i.e., the pressurizer) and the point of interest (i.e., the reactor vessel beltline). The pressure/temperature limits do not include instrumentation uncertainties, since these uncertainties are insignificant when compared to the margin terms included in the ASME Section XI Appendix G methods (i.e., 2.0 multiplier on pressure stress).

4.6 LTOPS Design

4.6.1 Industry Experience

The NRC's Value/Impact and Regulatory Analyses of Generic Issue 94, "Additional Low Temperature Overpressure Protection for Light Water Reactors" (9),(10) demonstrate that LTOPS events have historically occurred at essentially isothermal metal conditions. This service experience provides one component of a technical basis for using the isothermal ASME Section XI limit curve to establish LTOPS setpoints.

An evaluation of the overpressurization data before and after 1980 revealed that the frequency of overpressurization events (per reactor year) and the severity of overpressurization events were significantly reduced by the 1979 requirement that each plant install an overpressure mitigation system (OMS) to intercept overpressurization events and prevent them from exceeding the Technical Specifications pressure-temperature limit curves. Operating experience after 1986 indicated a further reduction in challenges to LTOP systems. Between January 1980 and December 1986, 63 PWRs logged 356 reactor years of commercial operation. During this period there were 30 OMS challenges, driven by either the addition of mass or heat energy, or a combination of the two. Evaluation of the 30 OMS challenge events revealed that none were initiated during or immediately following significant heatup or cooldown. These results follow expectations, since pressure and temperature are not changed rapidly during low temperature operation at conditions where the OMS is enabled.

The fraction of operating time during which significant thermal stresses (e.g, those associated with a >20°F/hr heatup or cooldown) are present is small. For example, a nuclear unit may be expected to heat up and cool down 4 times per year. Assuming a 20°F/hr heatup or cooldown rate, a 400°F temperature change requires 20 hours. Therefore, a plant may be conservatively estimated to spend 160 hours/year with the thermal stresses associated with a 20°F/hr ramp rate. This duration represents only 1.8% of plant operating time (e.g., 160 / (365*24)).

Industry experience and engineering evaluation support the conclusion that reliable overpressurization protection is provided by an OMS designed to prevent pressure at the reactor vessel beltline from exceeding the isothermal (0°F/hr) pressure/temperature limit curve. This conclusion was affirmed in the NRC's Safety Evaluation Report of Wisconsin Electric and Power Company's license amendment request for modification of Technical Specifications related to the Point Beach Units 1 and 2 Overpressure Mitigating System (COMS) (21): "The zero degree heatup curve is allowed since most pressure transients occur during isothermal metal conditions."

4.6.2 ASME Section XI Recommendations for LTOPS

The ASME Section XI Working Group on Operating Plant Criteria (WGOPC), which has responsibility for Appendix G to Section XI, considered the burden and safety impact imposed by regulatory requirements for LTOP, and developed Code guidelines for determining the LTOP setpoint pressure and the required LTOPS enabling temperature:

> **LTOP systems shall be effective at coolant temperatures less than 200°F or at coolant temperatures corresponding to a reactor vessel metal temperature less than RT_{NDT} + 50°F, whichever is greater.[1,2] LTOP systems shall limit the maximum pressure in the vessel to 110% of the pressure determined to satisfy Appendix G of Section XI, Article G-2215.**
>
> [1] The coolant temperature is the reactor coolant inlet temperature
>
> [2] The vessel metal temperature is the temperature at a distance one fourth of the vessel section thickness from the inside wetted surface in the vessel beltline region. RT_{NDT} is the highest adjusted reference temperature (for weld or base metal in the beltline region) at a distance one fourth of the vessel section thickness from the vessel wetted inner surface as determined by Regulatory Guide 1.99, Revision 2.

These guidelines relieve some operational restrictions yet provide adequate margins against failure for the reactor pressure vessel. Further, by relieving the operational restrictions, these guidelines result in a reduced potential for activation of pressure relieving devices, thereby improving plant safety.

The philosophy adopted by the WGOPC in considering guidelines for LTOP limits was that administrative controls should be imposed to ensure that the Technical Specification pressure/temperature limits were not exceeded, and that the physical protection system must provide adequate protection against failure of the reactor pressure vessel below the enabling temperature where experience indicates the events occur. North Anna Units 1 and 2 will continue to operate in accordance with the heatup/cooldown rate-dependent pressure/temperature limits. An administrative maximum heatup/cooldown rate limit of 50°F/hr will continue to be observed. This administrative limitation on heatup and cooldown rate ensures that the instantaneous heatup or cooldown rate does not inadvertently exceed the range of analyzed rates as a result of equipment malfunction or operator error.

4.6.3 Evaluation of ASME Section XI LTOPS Setpoint Recommendations

Virginia Electric and Power Company has performed calculations to estimate the impact of a 10% change in the rate-dependent pressure/temperature limits defined by ASME Section XI Appendix G. Using the proposed North Anna Unit 1 cooldown curve data at 100°F, it was determined that a 10% reduction in allowable pressure is approximately equivalent to a 29°F/hr increase in cooldown rate. Similar calculations with North Anna Unit 1 heatup data suggest that a 10% reduction in allowable pressure is approximately equivalent to a 40°F/hr increase in heatup rate. Therefore, use of the isothermal limit curve to establish LTOPS setpoints provides margins to fracture equivalent to those provided by the ASME Section XI Appendix G LTOPS recommendations for heatup or cooldown rates up to 29°F/hr.

Reliable overpressure protection is provided by PORV lift setpoints designed to prevent reactor vessel beltline pressure from exceeding the isothermal (0°F/hr) limit curve, since:

(a) Industry experience and engineering evaluation demonstrate that events which challenge the LTOPS setpoint may be expected to occur at essentially isothermal conditions.

(b) Virginia Electric and Power Company calculations demonstrate a margin of safety equivalent to that provided by the ASME Section XI LTOPS recommendations for heatup and cooldown rates up to 29°F/hr when 100% of the isothermal curve is used to establish LTOPS setpoints.

(c) Physically achievable cooldown rates decrease with decreasing temperature. (For example, the maximum achievable cooldown rate approaches zero at the lowest allowable RCS operating temperature.)

(d) The NRC has approved use of the isothermal curve for establishing LTOPS setpoints in other utility submittals (21).

Operational occurrences which violate the rate-dependent Appendix G pressure/temperature limits may be evaluated in accordance with the requirements of ASME Section XI Appendix E.

4.6.4 LTOPS Enabling Temperature

Previous North Anna Low Temperature Overpressure Protection System (LTOPS) analyses have established the LTOPS enabling temperature at $RT_{NDT} + \Delta T + 90°F$ + temperature measurement uncertainty (1),(2). (ΔT is the maximum temperature difference between the water and metal at the 1/4-T and 3/4-T locations during heatup or cooldown at the maximum allowable rate.) The ASME Section XI recommendations provide for the establishment of the LTOPS enabling temperature at $RT_{NDT} + 50°F$ + temperature measurement uncertainty. This value ensures LTOPS protection in the temperature range where service experience has demonstrated the events may occur. Above this temperature, ASME Section XI Appendix G margins are sufficient to ensure that pressures up to the RCS design pressure will not result in propagation of the design flaw. Overpressure protection in this temperature range is provided by a combination of (a) administrative and procedural controls, (b) actuation of the PORVs (high setpoint), and (c) actuation of the pressurizer safety valves.

As stated in the Basis for Technical Specifications 3.4.2 and 3.4.3, the steam relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. Therefore, a single pressurizer safety valve provides adequate overpressurization protection against the startup of two charging pumps when a bubble has been drawn in the pressurizer.

Adequate water-solid overpressurization protection above the LTOPS enabling temperature is provided by only the passive actuation of the pressurizer safety valves. Specifically, Virginia Electric and Power Company calculations demonstrate that two pressurizer safety valves (PSV) can accommodate sufficient flow to compensate for the inadvertent and simultaneous startup of two charging pumps. Each PSV is capable of relieving 380,000 lbm/hr of saturated steam at 2500 psia (24). The water relief capacity of the PSVs was assumed to be 40% of their steam relief capacity (25). The calculations considered the pressure difference between the reactor vessel beltline and the pressurizer, and a 3% PSV lift setpoint tolerance.

On the basis of the evaluations described above, Virginia Electric and Power Company proposes establishment of the LTOPS enabling temperature at the temperature corresponding to $RT_{NDT} + 50°F$ + temperature measurement uncertainty. Margin is not added to compensate for the maximum calculated temperature difference between the downcomer fluid and the 1/4-T and 3/4-T reactor vessel locations, since use of the isothermal limit curve as the LTOPS design limit implies a uniform temperature distribution.

4.6.5 Proposed Design

Virginia Electric and Power Company proposes the following PORV lift setpoints and enabling temperatures:

North Anna Unit 1 PORV Setpoints (Valid to 30.7 EFPY):

Current:
≤450 psig for Cold Leg T≤270°F (i.e., 270°F Enabling Temp.) ≤390 psig for Cold Leg T≤150°F
Proposed:
≤500 psig for Cold Leg T≤235°F (i.e., 235°F Enabling Temp.) ≤395 psig for Cold Leg T≤150°F

The enabling temperature is calculated as RT_{NDT} + 50°F + instrument uncertainty. As previously described, the RT_{NDT} for the limiting North Anna Unit 1 material at 30.7 EFPY is 162.9°F. A bounding temperature measurement and instrumentation uncertainty of 20°F is utilized. The calculated enabling temperature of 232.9°F is rounded up to 235°F.

North Anna Unit 2 PORV Setpoints (Valid to 17 EFPY):

Current:
≤510 psig for Cold Leg T≤321°F (i.e., 321°F Enabling Temp.) ≤360 psig for Cold Leg T≤210°F
Proposed:
≤415 psig for Cold Leg T≤270°F (i.e., 270°F Enabling Temp.) ≤375 psig for Cold Leg T≤130°F

The enabling temperature is calculated as RT_{NDT} + 50°F + instrument uncertainty. The RT_{NDT} for the limiting North Anna Unit 2 material at 17 EFPY is 196°F (13). A bounding temperature measurement and instrumentation uncertainty of 20°F is utilized. The calculated enabling temperature of 266°F is rounded up to 270°F.

PORV lift setpoints were validated by adding the mass addition transient "setpoint overshoot" (described in Section 4.4.1) to the PORV lift setpoint pressure, and verifying that the resulting pressure is less than the isothermal limit curve.

Pressure and temperature measurement uncertainties have been excluded from consideration in the development of LTOPS setpoints on the basis that these uncertainties are insignificant when compared to the margin terms included in the ASME Section XI Appendix G methods (i.e., 2.0 multiplier on pressure stress). Instrumentation uncertainties have been excluded from consideration in previous submittals made by Virginia Electric and Power Company (1),(2) and other utilities (22),(23).

4.7 Component Operability Requirements

4.7.1 Charging Pump Operability Requirements

To ensure that plant operating conditions are consistent with the assumptions of the inadvertent charging pump startup accident analysis, it is necessary to require that only one charging pump be capable of automatic injection at temperatures below the LTOPS enabling temperature. Above the LTOPS enabling temperature, two pressurizer safety valves are capable of relieving the flow from two charging pumps. Therefore, no additional restrictions on charging pump operability need to be implemented at temperatures above the LTOPS enabling temperature. The proposed Technical Specifications reflect the requirement that two charging pumps must be capable of automatic actuation in Modes 1, 2, and 3 (as required by large break LOCA analyses), but that only one charging pump may be capable of automatic actuation below the LTOPS enabling temperature in Modes 4 and 5.

The LTOP system is enabled on the basis of the cold leg temperature. Under conditions of natural circulation cooldown, the average RCS temperature may differ from the cold leg temperature by as much as 25°F. To ensure adequate LTOPS protection during a natural circulation cooldown, operating procedures will implement the charging pump operability requirements described above at a temperature 25°F above the LTOPS enabling temperature.

4.7.2 Reactor Coolant Pump Startup Criterion

To ensure that plant operating conditions are consistent with the assumptions of the heat addition accident analysis, Technical Specifications require the steam generator secondary-to-primary temperature difference to be no greater than 50°F when a reactor coolant pump is started. This requirement is in effect when the cold leg temperature is less than or equal to the LTOPS enabling temperature. Above the LTOPS enabling temperature, overpressurization is adequately mitigated by actuation of two pressurizer safety valves.

4.7.3 PORV, Block Valve, and Control System Reliability (TS Changes to Address Generic Issue 70)

In Generic Letter 90-06 (7), the NRC documented its conclusions concerning the actions which needed to be taken to improve the reliability of PORVs and block valves. It was determined that the limiting conditions of operation for PORVs and block valves in the Technical Specifications for Modes 1, 2, and 3 needed to be modified to incorporate the position adopted by the NRC. Guidance for the modifications was provided in Attachments A-1 through A-3 of the Generic Letter (7).

To address the above requirements resulting from resolution of Generic Issue 70, Virginia Electric and Power Company proposes modification of the North Anna Units 1 and 2 Technical

Specifications 3/4.4.3.2, and associated Bases, to revise the PORV and control system testing requirements. Surveillance Requirements for emergency (backup) power supply testing of the PORVs and block valves were not added because the valves are powered from safety grade power sources. The proposed Technical Specification changes are modelled after those recommended in the Reference (7) Generic Letter to the extent possible for the North Anna Units 1 and 2 plant configuration.

4.7.4 LTOPS Availability (Technical Specification Changes to Address Generic Issue 94)

In the Reference (7) Generic Letter, the NRC staff determined that LTOP protection system unavailability is the dominant contributor to risk from low-temperature transients. The staff further concluded that a substantial improvement in availability when the potential for an overpressure event is highest, and especially during water-solid operations, can be achieved through improved administrative restrictions on the LTOP system.

The staff concluded that the LTOP system performs a safety-related function, and inoperable LTOP equipment should be restored to an operable status in a short period of time. The current 7-day allowed outage time for a single channel is considered to be too long under certain conditions. The staff concluded that the allowed outage time for a single channel should be reduced to 24 hours when operating in Mode 5 or 6, when the potential for an overpressure transient is highest. The operating reactor experiences indicate that these events occur during planned heatup (restart of an idle reactor coolant pump) or as a result of maintenance and testing errors while in Mode 5. The reduced allowed outage time for a single channel in Modes 5 and 6 will help emphasize the importance of the LTOP system in mitigating overpressure transients, and provide additional assurance that plant operation is consistent with the design basis transient analyses.

To address the above requirements resulting from resolution of Generic Issue 94, Virginia Electric and Power Company proposes that the North Anna Units 1 and 2 Technical Specifications be modified to specify a maximum allowed outage time of 24 hours for LTOPS when the plant is operating in Modes 5 or 6. The Mode 4 allowed outage time is specified to be 7 days. The proposed Technical Specification changes are modelled after those recommended in the Reference (7) Generic Letter.

5.0 SUMMARY AND CONCLUSIONS

The North Anna Unit 1 pressure/temperature operating limits required by 10 CFR 50 Appendix G have been revised to be valid to 30.7 EFPY (end-of-license) by including the effects of the incremental radiation exposure on the reactor vessel beltline region. The curves are based on analyses of North Anna Unit 1 reactor vessel materials surveillance capsule results. The revised Appendix G curves were prepared in accordance with standard Westinghouse methodologies including Regulatory Guide 1.99, Revision 2. New North Anna Unit 2 curves are not proposed at this time.

Revised LTOPS setpoints and LTOPS enabling temperatures are proposed for North Anna Units 1 and 2. The setpoints and enabling temperatures were developed to provide bounding low temperature reactor vessel integrity protection during the design basis mass and heat addition transients. The isothermal limit curve is used to establish the LTOPS setpoint. The validity of this approach is demonstrated by consideration of the conditions at which overpressurization events have been demonstrated to occur, by an analysis which demonstrates margins for this design equivalent to those provided by ASME Section XI Appendix G recommendations for anticipated LTOPS events, and by licensing precedent. This design maximizes the operating margin above the minimum RCS pressure for reactor coolant pump (RCP) operation, thereby minimizing the probability of undesired PORV lifts during RCP startup. Above the LTOPS enabling temperature, actuation of the pressurizer safety valves is adequate to ensure reactor vessel integrity during the design basis LTOPS transients.

Restrictions on allowable operating conditions and equipment operability requirements have been established to ensure that operating conditions are consistent with the assumptions of the accident analyses. Specifically, RCS pressure and temperature must be maintained within the heatup/cooldown rate-dependent pressure/temperature operating limits specified in the Technical Specifications. An administrative upper limit on heatup and cooldown rate of 50°F/hr will continue to be observed. Restrictions on the number of charging pumps capable of inadvertent startup have been imposed to ensure that the assumptions of the mass addition transient analysis are not invalidated. A restriction on the allowable temperature difference between the RCS and steam generator secondary side has been imposed to ensure that the assumptions of the heat addition transient are not invalidated.
Technical Specification changes are proposed to address the concerns of Generic Letter 90-06 (7). The proposed changes include revised PORV and block valve allowed outage time requirements, and revised PORV, block valve, and control system testing requirements to ensure the availability and reliability of these pressure relieving devices. The proposed changes are consistent with the guidance of the Reference (7) Generic Letter.

6.0 REFERENCES

(1) Letter from W. L. Stewart to USNRC, "Proposed Technical Specifications Changes (North Anna 1 and 2 Heatup and Cooldown Curves and Revised LTOPS Setpoints," Serial No. 91-707, dated December 29, 1991.

(2) Letter from L. B. Engle (USNRC) to W. L. Stewart, "North Anna Units 1 and 2 - Issuance of Amendments Re: Pressure/Temperature Operating Limits and Low Temperature Overpressure Protection System Setpoints, (TAC Nos. M83154 and M83155," dated March 25, 1993.

(3) J. M. Chicots and M. J. Malone: "Heatup and Cooldown Curves for North Anna Unit 1," WCAP-13831, Revision 1, dated August 1993.

(4) S. E. Yanichko, et al.: "Analysis of Capsule U from the Virginia Electric and Power Company North Anna Unit 1 Reactor Vessel Radiation Surveillance Program," WCAP-11777, dated February, 1988.

(5) Westinghouse Letter Report, "North Anna 1 Surveillance Capsule Withdrawal Schedule dated July 1993, Virginia Power Contract ER-MI2002, Westinghouse G.O. RM30416, Attachment to VRA-93-107."

(6) Letter from W. L. Stewart to USNRC, "Virginia Electric and Power Company; North Anna Power Station Unit 1; Revised Surveillance Capsule Withdrawal Schedule," Serial No. 93-526, dated August 26, 1993.

(7) Letter from USNRC to All Pressurized Water Reactor Licensees and Construction Permit Holders, "Resolution of Generic Issue 70, 'Power Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low Temperature Overpressure Protection for Light Water Reactors,' Pursuant to 10 CFR 50.54(f) (Generic Letter 90-06)," dated June 25, 1990.

(8) "Technical Findings and Regulatory Analysis Related to Generic Issue 70 - Evaluation of Power Operated Relief Valve and Block Valve Reliability in PWR Nuclear Power Plants," NUREG-1316.

(9) B. F. Gore, et al.: "Value/Impact Analysis of Generic Issue 94, 'Additional Low Temperature Overpressure Protection for Light Water Reactors,'" NUREG/CR-5186, dated November, 1988.

(10) E. D. Throm: "Regulatory Analysis for the Resolution of Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light Water Reactors,'" NUREG-1326, dated December, 1989.

(11) Letter from W. L. Stewart to USNRC, "Virginia Electric and Power Company; Surry Power Station Units 1 and 2; North Anna Power Station Units 1 and 2; PORV and Block Valve Reliability; Response to Generic Letter 90-06," Serial No. 90-446, dated December 21, 1990.

(12) Letter from L. B. Engle to W. L. Stewart, "North Anna Units 1 and 2 - Staff Review of Generic Letter 90-06, Resolution of Generic Issue 70, 'Power Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low Temperature Overpressure Protection for Light Water Reactors,' Pursuant to 10 CFR 50.54(f)," dated March 16, 1993.

(13) N. K. Ray, et. al.: "North Anna Unit 2 Reactor Vessel Heatup and Cooldown Limit Curves for Normal Operation (Capsule U)," WCAP-12503, dated March, 1990.

(14) A. L. Lowe, Jr., et al.: "Analysis of Capsule V; Virginia Electric and Power Company North Anna Unit 1 Reactor Vessel Materials Surveillance Program," BAW-1638, dated May, 1981.

(15) M. J. DeVan and A. L. Lowe, Jr.: "Response to Generic Letter 92-01 for Virginia Electric and Power Company North Anna Unit 1 and North Anna Unit 2," BAW-2168, Rev. 1, dated September, 1992.

(16) NRC Generic Letter 92-01, "Reactor Vessel Structural Integrity," dated March 6, 1992.

(17) "Reactor System Transient Analyses Using the RETRAN Computer Code," VEP-FRD-41, March, 1981, as supplemented by letter from W. L. Stewart to USNRC, "Virginia Electric and Power Company, Surry and North Anna Power Stations, Reactor System Transient Analysis," Letter Serial No. 85-753, dated November 19, 1985.

(18) Updated Final Safety Analysis Report, North Anna Power Station Units 1 and 2, Virginia Electric and Power Company.

(19) "EPRI PWR Safety and Relief Valve Test Program, Safety and Relief Valve Test Report," EPRI, NP-2628-SR, December, 1982.

(20) "Safety and Relief Valves in Light Water Reactors," EPRI, NP-4306-SR, December, 1985.

(21) Letter from R. A. Clark (USNRC) to Sol Burstein, Amendment 45 to Operating License DPR-24, and Amendment 50 to Operating License DPR-27 (NRC Approval of Point Beach Units 1 and 2 LTOPS Submittal), dated May 20, 1980.

(22) Letter from J. H. Goldberg (FP&L) to USNRC, St. Lucie Unit 1, Docket No 50-335, Proposed License Amendment, P-T Limits and LTOP Analysis, dated December 5, 1989.

(23) Letter from USNRC to J. H. Goldberg (FP&L), St. Lucie Unit 1 -Issuance of Amendment Re: Pressure/Temperature (P/T) Limits and Low Temperature Overpressure Protection (LTOP) Analysis (TAC No. 75386), Docket No. 50-335, dated June 11, 1990.

(24) North Anna Units 1 and 2 Technical Specifications, Basis for TS 3.4.2 and 3.4.3.

(25) G. O. Barrett, et al.: "Pressurizer Safety Valve Set Pressure Shift; Westinghouse Owners Group Project MUHP2351," WCAP-12910, dated March, 1991.

(26) Letter from W. L. Stewart to USNRC, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Revised Response to Generic Letter 92-01, Reactor Vessel Structural Integrity," Serial No. 92-211C, dated October 22, 1992.