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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 189 AND 170 TO

FACILITY OPERATING LICENSE NOS. NPF-4 AND NPF-7

VIRGINIA ELECTRIC AND POWER COMPANY

OLD DOMINION ELECTRIC COOPERATIVE

NORTH ANNA POWER STATION, UNITS NO. 1 AND NO. 2

DOCKET NOS. 50-338 AND 50-339

1.0 INTRODUCTION

By letter dated April 15, 1994, the Virginia Electric and Power Company (the licensee) proposed changes to the Technical Specifications (TS) for the North Anna Power Station, Units No. 1 and No. 2 (NA-1&2). Specifically, the proposed TS provide pressure/temperature (PT) operating limits and low temperature overpressure protection system (LTOPS) setpoints valid to the end-of-license (EOL) for NA-1. Revised LTOPS setpoints based on existing PT limit data valid to 17 effective full power years (EFPY) are provided for NA-2. For both NA-1&2, the proposed changes incorporate analytical and operational features which (a) address the LTOPS availability and reliability concerns of Generic Letter (GL) 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 Part 50.54(f)" dated June 25, 1990, (b) provide additional PT operating and operational flexibility, and (c) reduce the potential for undesired power-operated relief valve (PORV) lifts.

The current PT operating limits and LTOPS setpoints are valid to 12 EFPY and 17 EFPY for NA-1&2 respectively. According to the most recent estimates, the burnup applicability limits will be exceeded by NA-1 in Spring of 1996. The NA-2 PT operating limits and LTOPS setpoints remain valid well into the year 2002. The proposed NA-1 TS include revised PT operating limits valid to end-of-license. Although the NA-2 TS PT operating limits are not being changed, the NA-2 LTOPS setpoints and associated reactor vessel integrity protection criteria are being changed. The reactor vessel integrity protection criteria which supports the proposed changes provide improved operational flexibility while maintaining an adequate margin of safety as demonstrated by the safety analysis.

2.0 DISCUSSION

2.1 NA-1 Surveillance Capsule Data

Credible surveillance data for NA-1 beltline materials are available from two surveillance capsules, V (14) and U (4). The NA-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).

Capsules V and U were removed from NA-1 at the end of the first cycle of operation at a cumulative burnup of 1.13 EFPY and at the end of the sixth cycle of operation at a cumulative burnup of 5.9 EFPY, respectively. The V and U capsule dosimeters were evaluated and found to have a cumulative fast (E>1.0 MeV) fluence of 2.49×10^{18} neutrons per square centimeter (n/cm^2) and 8.28×10^{18} n/cm^2, respectively. The irradiated specimens test results were compared to unirradiated specimen test results. For Capsule V, 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. For Capsule U, 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.

References (4) and (14) provide further information on Capsule V analysis results.

2.2 NA-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 TS changes is presented below. The chemistry and unirradiated RT_{NDT} data is identical to that presented in the licensee's October 22, 1992 response (26) to GL 92-01 (16) for NA-1&2.

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.

2.3 NA-1 Irradiated RT_{PTS} Values

In accordance with the methods prescribed by Regulatory Guide 1.99, Revision 2, adjusted RT_{PTS} values have been calculated for each NA-1 reactor vessel beltline material at a fluence corresponding to 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_{PTS} at the 1/4T and 3/4T locations were shown to occur in the NA-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 PT operating limits (3) are marked with an asterisk (*).

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

2.4 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 (RCP) in a reactor coolant system (RCS) loop with a 50°F difference between the steam generator (SG) 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 NA-1&2 Technical Specifications LTOPS setpoints (1), (2). As described in the licensee's 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.

2.5 Mass Addition Transient

The inadvertent startup of a single charging pump was selected as the design basis mass addition transient based on previous Updated Final Safety Analysis Report (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 NA-1&2 (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 2.8.5) 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 PT limit curve. Selection of the design PT limit curve is discussed in Section 2.8.

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

2.6 Heat Addition Transients

The heat addition transient assumes that an RCP is started with the maximum temperature difference allowed by TS (50°F) between the SGs 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.

Initial Conditions for the Heat 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

2.7 Revised Technical Specification PT Operating Limits

NA-1 PT operating limits valid to 30.7 EFPY have been developed and are presented in Reference (3). The NA-2 curve data (13) are unchanged from those currently in the TS. 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 PT 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 PT 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 PT limits do not include instrumentation uncertainties.

2.8 LTOPS DESIGN

2.8.1 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.

The fraction of operating time during which significant thermal stresses are present (e.g, those associated with a >20°F/hr heatup or cooldown) is small. For example, a nuclear unit may be expected to heat up and cool down four 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 60 hours/year with the thermal stresses associated with a 20°F/hr ramp rate. This duration represents only 1.8% of plant operating time.

Industry experience and engineering evaluation support the conclusion that reliable overpressurization protection is provided by an overpressure mitigation system (OMS) designed to prevent pressure at the reactor vessel beltline from exceeding the isothermal (0°F/hr) PT 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."

2.8.2 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.

These guidelines relieve some operational restrictions yet provide adequate margins against failure for the reactor pressure vessel. By relieving these operational restrictions, the 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 TS 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. NA-1&2 will continue to operate in accordance with the heatup/cooldown rate-dependent PT 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.

2.8.3 Evaluation of ASME Section XI LTOPS Setpoint Recommendations

The licensee has performed calculations to estimate the impact of a 10% change in the rate-dependent PT limits defined by ASME Section XI Appendix G. Using the proposed NA-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 NA-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 design 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) The licensee's 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 PT limits may be evaluated in accordance with the requirements of ASME Section XI Appendix E.

2.8.4 LTOPS Enabling Temperature

Previous NA LTOPS analyses have established the LTOPS enabling temperature at RT_{NDT} + ΔT + 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 the NA-1&2 TS 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, the licensee 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, the licensee 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.

2.8.5 Proposed Design

The licensee proposes the following PORV lift setpoints and enabling temperatures:

NA-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 NA-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.

NA-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 NA-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 2.5) to the PORV lift setpoint pressure, and verifying that the resulting pressure is less than the isothermal limit curve. 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 the licensee (1), (2) and other utilities (22), (23).

2.9 COMPONENT OPERABILITY REQUIREMENTS

2.9.1 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 TS reflect the requirement that two charging pumps must be capable of automatic actuation in Modes 1,2, and 3 (as required by large break loss-of-coolant accident 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.

2.9.2 Reactor Coolant Pump Startup Criterion

To ensure that plant operating conditions are consistent with the assumptions of the heat addition accident analysis, TS require the SG secondary-to-primary temperature difference to be no greater than 50°F when an RCP 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.

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

In GL 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 LCOs for PORVs and block valves in the TS 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 GL 90-06.

To address the above requirements resulting from resolution of Generic Issue 70, the licensee proposes modification of the NA-1&2 TS 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 TS changes are modelled after those recommended in the Reference (7) GL to the extent possible for the NA-1&2 plant configuration.

2.9.4 LTOPS Availability: TS Changes to address Generic Issue 94

In GL 90-06, 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 RCP) 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, the licensee proposes that the NA-1&2 TS 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 TS changes are modelled after those recommended in GL 90-06.

3.0 SPECIFIC TS CHANGES

The TS changes described herein apply to NA-1&2. In addition to the specific changes described below, editorial changes have been made to correct grammatical errors and format inconsistencies.

TS 3.1.2.2 - REACTIVITY CONTROL SYSTEMS - FLOW PATHS - OPERATING

The existing footnote to TS 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 NA-1; 270°F for NA-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.

TS 3/4.1.2.4 - REACTIVITY CONTROL SYSTEMS - CHARGING PUMPS - OPERATING

The Action Statement, Surveillance Requirements, and footnote to TS 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 NA-1; 270°F for NA-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.

TS 3.4.1.2 (NA-2 ONLY) - REACTOR COOLANT SYSTEM - HOT STANDBY - MODE 3

A previous TS change added a footnote to NA-2 TS 3.4.1.2 to specify that an RCP 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 SG 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.

TS 3.4.1.3 - REACTOR COOLANT SYSTEM - SHUTDOWN - MODES 4 AND 5

An existing footnote to TS 3.4.1.3 has been revised to specify that an RCP 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 NA-1; 270°F for NA-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 SG when an RCP 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.

TS 3.4.3 (NA-1 ONLY) - REACTOR COOLANT SYSTEM - SAFETY VALVES - OPERATING

To be consistent with the NA-2 TS, the number of NA-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.

TS 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 GL 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 TS are consistent with the guidelines presented in GL 90-06 (7) for the NA-1&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 TS.

TS FIGURES 3.4-2 AND 3.4-3 REACTOR COOLANT SYSTEM - PRESSURE/TEMPERATURE LIMITS

Revised TS Figures NA-1 3.4-2 and 3.4-3 have been prepared to present the revised NA-1, 30.7 EFPY and existing NA-2, 17 EFPY PT 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 TS 3.1.1.5, Minimum Temperature for Criticality.

TS 3/4.4.9.3 REACTOR COOLANT SYSTEM - OVERPRESSURE PROTECTION SYSTEMS

The format and content of TS 3/4.4.9.3 have been revised to address the LTOPS availability concerns of GL 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 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 GL 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 2.8. Above the LTOPS enabling temperature, actuation of the pressurizer safety valves is adequate to ensure reactor vessel integrity during the LTOPS design basis transients.

TS 3.5.2 ECCS SUBSYSTEMS - T_{avg} > 350°F

A previous LTOPS TS change modified the NA-1&2 TS 3.5.2 ACTION "c" to allow the provisions of TS 3.0.4 to be not applicable to ACTIONS "a" and "b" for one hour following heatup above 316°F (358°F for NA-2) or prior to cooldown below 316°F (358°F for NA-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 NA-2 TS 3.5.2 was necessary because the NA-2 temperature limit involved Mode 3 operation between 350°F and 358°F.

NA-1&2 TS 3.5.2 are being modified to reflect the revised temperature limit (235°F for NA-1; 270°F for NA-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 NA-2 TS 3.5.2 footnote is being deleted.

A footnote to NA-1 TS 3.5.2 imposed emergency core cooling system (ECCS) component operability requirements that were to be in effect until SG replacement. Because Unit 1 SG replacement has been accomplished, this footnote is being removed.

TS 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 (NA-1; 270°F for NA-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.

TS 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 SG 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 GL 90-06 (7). The Bases for TS 3/4.4.9 have been modified to reflect current information on the development of PT operating limits and LTOPS setpoints.

TS 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 PORV AND BLOCK VALVE RELIABILITY

GL 90-06 requested that PORVs and block valves be included within the scope of an operational assurance program that is in compliance with 10 CFR 50, Appendix B. By letter dated December 21, 1990, the licensee responded to the recommendations specified in GL 90-06.

The NA-1&2 PORVs and block valves are included in a quality assurance program that meets 10 CFR 50, Appendix B requirements. These valves are on the plant operational Quality Assurance list (Q-list). The maintenance program for PORVs and block valves is based on manufacturer's recommendations and guidelines. Valve maintenance is performed by trained personnel. Spare or replacement parts are procured in accordance with the original construction codes and standards or applicable later editions of the code.

The NA-1&2 PORVs and the block valves are included in the scope of the inservice testing (Section XI) program. The block valves are included in the GL 89-10 MOV program. The solenoid-operated valves and check valves in the PORV control air system are checked indirectly when the PORVs are tested. In accordance with the licensee's approved Section XI inservice testing program, the PORVs are not tested at power, but are stroke tested on approach to each cold shutdown. In addition, the PORVs are stroke tested prior to each startup before establishing water-solid conditions. Therefore, the PORVs are tested prior to establishing conditions where the PORVs are used for low-temperature overpressure protection. The remainder of the Section XI requirements are completed in cold shutdown.

5.0 EVALUATION

The NA-1 PT operating limits required by 10 CFR 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 NA-1 reactor vessel materials surveillance capsule results. The revised Appendix G curves were prepared in accordance with approved Westinghouse methodologies and Regulatory Guide 1.99, Revision 2. New NA-2 curves are not proposed at this time.

Revised LTOPS setpoints and LTOPS enabling temperatures are proposed for NA-1&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 RCP operation, thereby minimizing the probability of undesired PORV lifts during RCS startup. Above the LTOPS enabling temperature, actuation of the pressurizer safety valves is adequate to ensure reactor vessel integrity during the design bases 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 heat/cooldown rate-dependent PT operating limits specified in the TS. 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 SG secondary side has been imposed to ensure that the assumptions of the heat addition transient are not invalidated. TS changes have been proposed to address the concerns of GL 90-06. 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 GL 90-06. Therefore, based on all of the above, the staff finds the proposed TS changes to be acceptable.

6.0 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 comment.

7.0 ENVIRONMENTAL CONSIDERATION

These 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 these amendments involve no significant hazards consideration and there has been no public comment on such finding (59 FR 27069). Accordingly, these 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.

8.0 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

9.0 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. M75386), 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.

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Date: October 5, 1994