October 5, 1994

Mr. J. P. O'Hanlon
Senior Vice President - Nuclear
Virginia Electric and Power Company
5000 Dominion Blvd.
Glen Allen, Virginia 23060

Distribution
See next page

SUBJECT: NORTH ANNA UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: PRESSURE/TEMPERATURE OPERATING LIMITS/LOW TEMPERATURE OVERPRESSURE PROTECTION SYSTEM PRESSURE SETPOINTS/LIMITING CONDITIONS FOR OPERATION, ACTION STATEMENTS, AND SURVEILLANCE REQUIREMENTS FOR PORVs AND BLOCK VALVES TO ADDRESS GENERIC LETTER 90-06 (TAC NOS. M77363, M77364, M77433, M77434, M89312, AND M89313)

Dear Mr. O'Hanlon:

The Commission has issued the enclosed Amendment Nos. 189 and 170 to Facility Operating License Nos. NPF-4 and NPF-7 for the North Anna Power Station, Units No. 1 and No. 2 (NA-1&2). The amendments revise the Technical Specifications (TS) in response to your letter dated April 15, 1994.

The amendments modify the pressure/temperature operating limitations during heatup and cooldown and the Low Temperature Overpressure Protection System pressure setpoints and enabling Conditions for Operation, Action Statements, and Surveillance Requirements for the power-operated relief valves and block valves to address the concerns discussed in NRC Generic Letter 90-06. The proposed changes also include several editorial/administrative changes.

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

Sincerely,

(Original Signed By)
Leon B. Engle, Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-338 and 50-339

Enclosures:
1. Amendment No. 189 to NPF-4
2. Amendment No. 170 to NPF-7
3. Safety Evaluation

cc w/enclosures: See next page

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Virginia Electric & Power Company

North Anna Power Station
Units 1 and 2

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DATED: October 5, 1994

AMENDMENT NO. 189 TO FACILITY OPERATING LICENSE NO. NPF-4-NORTH ANNA UNIT 1
AMENDMENT NO. 170 TO FACILITY OPERATING LICENSE NO. NPF-7-NORTH ANNA UNIT 2

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

VIRGINIA ELECTRIC AND POWER COMPANY

OLD DOMINION ELECTRIC COOPERATIVE

DOCKET NO. 50-338

NORTH ANNA POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 189
License No. NPF-4

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

   A. The application for amendment by Virginia Electric and Power Company et al., (the licensee) dated April 15, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.D.(2) of Facility Operating License No. NPF-4 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 189 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Victor M. McCree

Victor M. McCree, Acting Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 5, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 189

TO FACILITY OPERATING LICENSE NO. NPF-4

DOCKET NO. 50-338

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Remove Pages	Insert Pages
V	V
XIII	XIII
3/4 1-9	3/4 1-9
3/4 1-12	3/4 1-12
3/4 4-3	3/4 4-3
3/4 4-7	3/4 4-7
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3/4 4-28	3/4 4-28
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3/4 5-3	3/4 5-3
3/4 5-6	3/4 5-6
3/4 5-6a	3/4 5-6a
B 3/4 1-3	B 3/4 1-3
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--------	B 3/4 4-2a
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B 3/4 4-7	B 3/4 4-7
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3/4.4.9	PRESSURE/TEMPERATURE LIMITS	
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	ASME Code Class 1, 2 & 3 Components	3/4 4-33
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3/4.5	EMERGENCY CORE COOLING SYSTEMS (ECCS)	
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FLOW PATHS – OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 Each of the following boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System, and

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 AND 4#.

ACTION:

a. With the flow path from the boric acid tanks inoperable, restore the inoperable flow path to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1.77% $\Delta k/k$ at 200°F within the next 6 hours; restore the flow path to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the flow path from the refueling water storage tank inoperable, restore the flow path to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 Each of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the heat traced portion of the flow path from the boric acid tanks is ≥ 115°F.

Only one boron injection 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 235°F.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. At least once per 18 months during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on a safety injection test signal.

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE.

APPLICABILITY: MODES 5 and 6

ACTION:

a. With no charging pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until one charging pump is restored to OPERABLE status.

b. With no charging pump OPERABLE and the opposite unit in MODE 1, 2, 3 or 4, immediately initiate corrective action to restore at least one charging pump to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 At least the above required charging pump shall be demonstrated OPERABLE by verifying that, on recirculation flow, the pump develops a discharge pressure of $\geq$ 2410 psig when tested pursuant to Specification 4.0.5.

4.1.2.3.2 All charging pumps, except the above required OPERABLE pump, shall be demonstrated inoperable at least once per 12 hours by verifying that the switches in the Control Room have been placed in the pull to lock position.

CHARGING PUMPS – OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4 *.

ACTION:

With only one charging pump OPERABLE, restore a second charging pump to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1.77% Δk/k at 200°F within the next 6 hours; restore a second charging pump to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours. The provisions of Specification 3.0.4 are not applicable for one hour following heatup above 235°F or prior to cooldown below 235°F.

SURVEILLANCE REQUIREMENTS

4.1.2.4.1 The above required charging pumps shall be demonstrated OPERABLE by verifying, that on recirculation flow, each pump develops a discharge pressure of ≥ 2410 psig when tested pursuant to Specification 4.0.5.

4.1.2.4.2 All charging pumps, except the above required OPERABLE pump, shall be demonstrated inoperable at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than or equal to 235°F by verifying that the switches in the Control Room have been placed in the pull to lock position.

---

* 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 235°F.

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. At least two of the coolant loops listed below shall be OPERABLE:

1. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,*
2. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,*
3. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump,*
4. Residual Heat Removal Subsystem A,**
5. Residual Heat Removal Subsystem B.**

b. At least one of the above coolant loops shall be in operation.***

APPLICABILITY: MODES 4 and 5

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; be in COLD SHUTDOWN within 20 hours.

b. With no coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

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* A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to 235°F unless the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

** The offsite or emergency power source may be inoperable in MODE 5.

*** All reactor coolant pumps and residual heat removal pumps may be de-energized for up to 1 hour provided 1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required RHR subsystems shall be demonstrated OPERABLE per Specification 4.7.9.2.

4.4.1.3.2 The required reactor coolant pump(s), if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignment and indicated power availability.

4.4.1.3.3 The required steam generator(s) shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 17% at least once per 12 hours.

4.4.1.3.4 At least once per 12 hours, verify at least one coolant loop to ~~be~~ in operation and circulating reactor coolant by:

a. Verifying at least one Reactor Coolant Pump is in operation.

or

b. Verifying at least one RHR Loop is in operation and,

1. if the RCS temperature >140° F **or** the time since entry into MODE 3 is <100 hours, circulating reactor coolant at a flow rate ≥3000 gpm.

or

2. if the RCS temperature ≤140° F **and** the time since entry into MODE 3 is ≥100 hours, circulating reactor coolant at a flow rate ≥2000 gpm to remove decay heat.

SAFETY AND RELIEF VALVES – OPERATING

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.4.3.1 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2485 PSIG ± 1%.*

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal temperature and pressure.

LIMITING CONDITION FOR OPERATION

3.4.3.2 Both power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or both PORVs inoperable but capable of being manually cycled, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one PORV inoperable and not capable of being manually cycled, within 1 hour either restore the PORV to OPERABLE status or capable of being manually cycled, or close its associated block valve and remove power from the block valve; restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With both PORVs inoperable and not capable of being manually cycled, within 1 hour either restore at least one PORV to OPERABLE status or capable of being manually cycled, or close its associated block valve and remove power from the block valve and be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

d. With one or both block valves inoperable, within 1 hour restore the block valve(s) to OPERABLE status or place its associated PORV(s) in manual control. Restore at least one block valve to OPERABLE status within the next hour if both block valves are inoperable; restore the remaining inoperable block valve to OPERABLE status within 72 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

e. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE:

a. At least once per 31 days by performing a CHANNEL FUNCTIONAL TEST, excluding valve operation, and

b. At least once per 18 months by

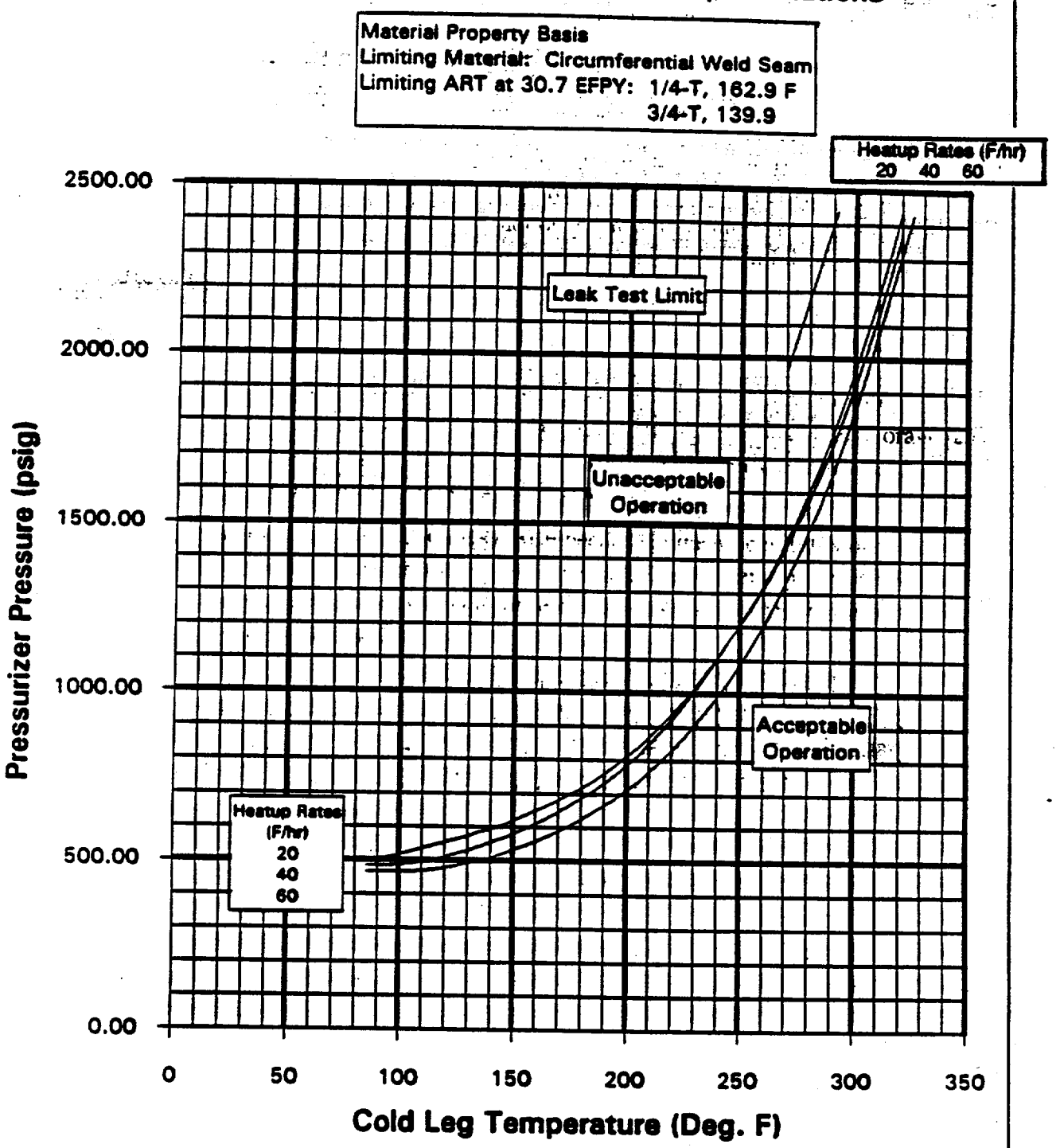
1. Operating the PORV through one complete cycle of full travel during MODES 3 or 4, and

2. Operating the solenoid air control valves and check valves on the associated accumulators in the PORV control systems through one complete cycle of full travel, and

3. Performing a CHANNEL CALIBRATION of the actuation instrumentation.

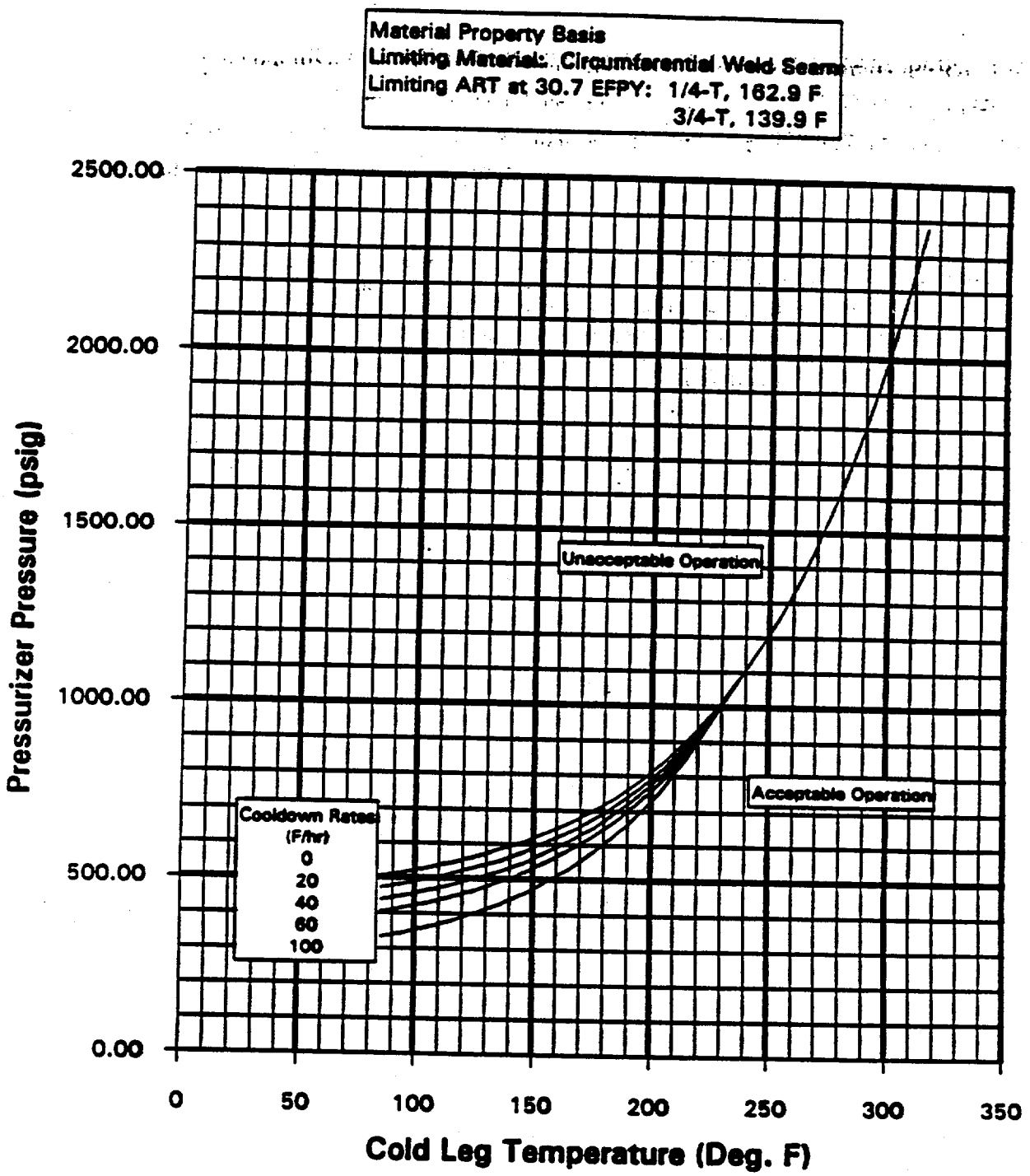
4.4.3.2.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed in order to meet the requirements of ACTION b or c in Specification 3.4.3.2.

Figure 3.4-2 — **North Anna Unit 1**
Reactor Coolant System Heatup Limitations



North Anna Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rates up to 60 F/hr) Applicable for the First 30.7 EFPY (Without Margins for Instrumentation Errors)

Figure 3.4-3 — **North Anna Unit 1**
Reactor Coolant System Cooldown Limitations



North Anna Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100 F/hr) Applicable for the First 30.7 EFPY (Without Margins for Instrumentation Errors)

LIMITING CONDITION FOR OPERATION

3.4.9.3 Two power-operated relief valves (PORVs) shall be OPERABLE with lift settings of (1) less than or equal to 500 psig whenever any RCS cold leg temperature is less than or equal to 235°F, and (2) less than or equal to 395 psig whenever any RCS cold leg temperature is less than 150°F.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 235°F, MODE 5, and MODE 6 when the head is on the reactor vessel and the RCS is not vented through a 2.07 square inch or larger vent.

ACTION:

a. With one PORV inoperable in MODE 4, restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2.07 square inch vent within the next 8 hours.

b. With one PORV inoperable in MODES 5 or 6, either (1) restore the inoperable PORV to OPERABLE status within 24 hours, or (2) complete depressurization and venting of the RCS through at least a 2.07 square inch vent within a total of 32 hours.

c. With both PORVs inoperable, complete depressurization and venting of the RCS through at least a 2.07 square inch vent within 8 hours.

d. With the RCS vented per ACTIONS a, b, or c, verify the vent pathway at least once per 31 days when the pathway is provided by a valve(s) that is locked, sealed, or otherwise secured in the open position; otherwise, verify the vent pathway ever 12 hours.

e. In the event either the PORVs or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

f. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9.3 Each PORV shall be demonstrated OPERABLE by:

a. Performance of a CHANNEL FUNCTIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE.

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel, at least once per 18 months.

c. Verifying the PORV keyswitch is in the Auto position and the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

d. Testing pursuant to Specification 4.0.5.

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE low head safety injection pump,

c. An OPERABLE flow path capable of transferring fluid to the Reactor Coolant System when taking suction from the refueling water storage tank on a safety injection signal or from the containment sump when suction is transferred during the recirculation phase of operation or from the discharge of the outside recirculation spray pump.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

c. The provisions of Specification 3.0.4 are not applicable to 3.5.2.a and 3.5.2.b for one hour following heatup above 235°F or prior to cooldown below 235°F.

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE centrifugal charging pump#,

b. One OPERABLE low head safety injection pump#, and

c. An OPERABLE flow path capable of automatically transferring fluid to the reactor coolant system when taking suction from the refueling water storage tank or from the containment sump when the suction is transferred during the recirculation phase of operation or from the discharge of the outside recirculation spray pump.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of the low head safety injection pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

A maximum of one centrifugal charging pump and one low head safety injection pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 235°F.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.

4.5.3.2 All charging pumps and safety injection pumps, except the above required OPERABLE pumps, shall be demonstrated inoperable at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than or equal to 235°F by verifying that the switches in the Control Room are in the pull to lock position.

3/4.1.2 BORATION SYSTEMS (Continued)

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.77% Δk/k after xenon decay and cooldown to 200°F. This expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 6,000 gallons of 12,950 ppm borated water from the boric acid storage tanks or 54,200 gallons of 2300 ppm borated water from the refueling water storage tank.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable below 235°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1.77% Δk/k after zenon decay and cooldown from 200°F to 140°F. This condition requires either 1378 gallons of 12,950 ppm borated water from the boric acid storage tanks or 3400 gallons of 2300 ppm borated water from the refueling water storage tank.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics. The OPERABILITY of one boron injection system during REFUELING insures that this system is available for reactivity control while in MODE 6.

3/4.1.2 BORATION SYSTEMS (Continued)

The limits on contained water volume and boron concentration of the RWST ensure a pH value of between 7.7 and 9.0 for the solution recirculated within the containment after a LOCA. This pH minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

At least one charging pump must remain operable at all times when the opposite unit is in MODE 1, 2, 3, or 4. This is required to maintain the charging pump cross-connect system operational.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section (1) ensure that acceptable power distribution limits are maintained, (2) ensure that the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential effects of rod misalignment on associated accident analyses. OPERABILITY of the movable control assemblies is established by observing rod motion and determining that rods are positioned within ± 12 steps (indicated position) of the respective demand step counter position. The OPERABILITY of the individual rod position indication system is established by appropriate periodic CHANNEL CHECKS, CHANNEL FUNCTIONAL TESTS, and CHANNEL CALIBRATIONS. OPERABILITY of the individual rod position indicators is required to determine control rod position and thereby ensure compliance with the control rod alignment and insertion limits. The OPERABLE condition for the individual rod position indicators is defined as being capable of indicating rod position within ± 12 steps of the associated demand position indicator. For power levels below 50 percent of RATED THERMAL POWER, the specifications of this section permit a maximum one hour in every 24 stabilization period (thermal "soak time") to allow stabilization of known thermal drift in the individual rod position indicator channels during which time the indicated rod position may vary from demand position indication by no more than ± 24 steps. This "1 in 24" feature is an upper limit on the frequency of thermal soak allowances and is available both for a continuous one hour period or one consisting of several discrete intervals. During this stabilization period, greater reliance is placed upon the demand position indicators to determine rod position. In addition, the ± 24 step/hour limit is not applicable when the control rod position is known to be greater than 12 steps from the rod group step counter demand position indication. Above 50 percent of RATED THERMAL POWER, rod motion is not expected to induce thermal transients of sufficient magnitude to exceed the individual rod position indicator instrument accuracy of ± 12 steps. Comparison of the demand position indicators to the bank insertion limits with verification of rod position by the individual rod position indicators (after thermal soak following rod motion below 50 percent of RATED THERMAL POWER) is sufficient verification that the control rods are above the insertion limits.

The control bank FULLY WITHDRAWN position can be varied within the interval of 225 to 229 steps withdrawn, inclusive. This interval permits periodic repositioning of the parked RCCAs to minimize wear, while having minimal impact on the normal reload core physics and safety evaluations. Changes of the RCCA FULLY WITHDRAWN position within this band are administratively controlled, using the rod insertion limit operator curve.

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with all reactor coolant loops in operation and maintain the DNBR above the design limit during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODES 4 and 5, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat, but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two RHR loops to be OPERABLE.

After the reactor has shutdown and entered into MODE 3 for at least 100 hours, a minimum RHR system flow rate of 2000 gpm in MODE 5 is permitted, provided there is sufficient decay heat removal to maintain the RCS temperature less than or equal to 140°F. Since the decay heat power production rate decreases with time after reactor shutdown, the requirements for RHR system decay heat removal also decrease. Adequate decay heat removal is provided as long as the reactor has been shutdown for at least 100 hours after entry into MODE 3 and RHR flow is sufficient to maintain the RCS temperature less than or equal to 140°F. The reduced flow rate provides additional margin to vortexing at the RHR pump suction while in Mid Loop Operation. During a reduction in reactor coolant system boron concentration the Specification 3.1.1.3.1 requirement to maintain 3000 gpm flow rate provides sufficient coolant circulation to minimize the effect of a boron dilution incident and to prevent boron stratification.

The restrictions on starting a Reactor Coolant Pump with one or more RCS cold legs less than or equal to 235°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

The operation of one Reactor Coolant Pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will therefore be within the capability of operator recognition and control.

The requirement to maintain the boron concentration of an isolated loop greater than or equal to the boron concentration of the operating loops ensures that no reactivity addition to the core could occur during startup of an isolated loop. Verification of the boron concentration in an idle loop prior to opening the cold leg stop valve provides a reassurance of the adequacy of the boron concentration in the isolated loop. Operating the isolated loop on recirculating flow for at least 90 minutes prior to opening its cold leg stop valve ensures adequate mixing of the coolant in this loop and prevents any reactivity effects due to boron concentration stratification.

Startup of an idle loop will inject cool water from the loop into the core. The reactivity transient resulting from this cool water injection is minimized by delaying isolated loop startup until its temperature is

within 20°F of the operating loops. Making the reactor subcritical prior to loop startup prevents any power spike which could result from this cool water induced reactivity transient.

3/4.4.2 AND 3/4.4.3 SAFETY AND RELIEF VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 380,000 lbs per hour of saturated steam at the valve set point. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during hot shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, or the power operated relief valves (PORVs) will provide overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip setpoint is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

The power operated relief valves and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the power operated relief valves minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.

The OPERABILITY of the PORVs and block valves is determined on the basis of their being capable of performing the following functions:

a) Manual control of PORVs to control reactor coolant system pressure. This is a function that may be used to mitigate certain accidents and for plant shutdown.

b) Maintaining the integrity of the reactor coolant pressure boundary. This function is related to controlling identified leakage and ensuring the ability to detect unidentified reactor coolant pressure boundary leakage.

c) Manual control of the block valve to (1) unblock an isolated PORV to allow it to be used for manual control of reactor coolant system pressure (Item a, above), and (2) isolate a PORV with excessive seat leakage (Item b, above).

d) Automatic control of PORVs to control reactor coolant system pressure. This function reduces challenges to the code safety valves for overpressurization events.

e) Manual control of a block valve to isolate a stuck-open PORV.

Surveillance Requirements provide the assurance that the PORVs and block valves can perform their functions. Specification 4.4.3.2.1 addresses the PORVs and Specification 4.4.3.2.2 addresses the block valves. The block valves are exempt from the surveillance requirements to cycle the valves when they have been closed to comply with the ACTION requirements. This precludes the need to cycle the valves with full system differential pressure or when maintenance is being performed to restore an inoperable PORV to operable status.

Surveillance Requirement 4.4.3.2.1.b.2 provides for the testing of the mechanical and electrical aspects of control systems for the PORVs.

Testing of PORVs in HOT STANDBY or HOT SHUTDOWN is required in order to simulate the temperature and pressure environmental effects on PORVs. Testing at COLD SHUTDOWN is not considered to be a representative test for assessing PORV performance under normal plant operating conditions.

3/4.4.4 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12 hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the North Anna site such as site boundary location and meteorological conditions, were not considered in this evaluation. The NRC is finalizing site specific criteria which will be used as the basis for the reevaluation of the specific activity limits of this site.. This reevaluation may result in higher limits.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity > 1.0 ~μCi/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

Reducing T_{avg} to < 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE / TEMPERATURE LIMITS

Reactor Coolant System

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 5.2 of the UFSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure- temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Consequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

The heatup limit curve, Figure 3.4-2, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour. The cooldown limit curves of Figure 3.4-3 are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of 30.7 EFPY. The most recent capsule analysis results are documented in Westinghouse Report WCAP-11777, February 1988. The heatup and cooldown curves are documented in Westinghouse Report WCAP-13831, Rev. 1, August 1993.

The reactor vessel materials have been tested to determine their initial RT_{NDT}. Reactor operation and resultant fast neutron (E > 1 Mev) irradiation will cause an increase in the RT_{NDT}. An adjusted reference temperature, based upon the fluence and copper content of the material in question, can be predicted using US NRC Regulatory Guide 1.98, Revision 2. The heatup and cooldown limit curves (Figure 3.4-2 and Figure 3.4-3) include predicted adjustments for this shift in RT_{NDT} at the end of 30.7 EFPY. The reactor vessel beltline region material properties are listed on Figures 3.4-2 and 3.4-3.

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

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

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

Pressurizer

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

Low-Temperature Overpressure Protection

The OPERABILITY of two PORVs or an RCS vent opening of greater than 2.07 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 235°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures or (2) the start of a charging pump and its injection into a water-solid RCS.

Automatic or passive low temperature overpressure protection (LTOP) is required whenever any RCS cold leg temperature is less than 235°F. This temperature is the water temperature corresponding to a metal temperature of at least the limiting RT_{NDT} + 50°F + instrument uncertainty. Above 235°F administrative control is adequate protection to ensure the limits of the heatup curve (Figure 3.4-2) and the cooldown curve (Figure 3.4-3) are not violated. The concept of requiring automatic LTOP at the lower end, and administrative control at the upper end, of the Appendix G curves is further discussed in NRC Generic Letter 88-11.

ECCS SUBSYSTEMS (Continued)

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The limitation for a maximum of one centrifugal charging pump and one low head safety injection pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps and low head safety injection pumps except the required OPERABLE pump to be inoperable below 235°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained.

In the event of modifications to an ECCS subsystem that could alter the subsystem flow characteristics, a flow balance test shall be performed. The flow balance test criteria are established based on the system performance assumed in the safety analysis (minimum flow limit) and on HHSI pump runout protection (maximum flow limit). In performing the flow balance, the effects of flow measurement instrument uncertainties accounting for system configuration and the variability between installed pumps must be properly considered.

Numerical acceptance criteria for the flow balance test are specified in the surveillance test procedure. These criteria are established based on the following considerations:

1) The total injected flow to the core (assuming spillage of the branch line with the highest flow) must meet or exceed that assumed in the safety analysis. The limiting safety analysis is the loss of coolant accident (LOCA) analysis. This criterion may vary, particularly since the inputs to the safety analysis controlled by LCO 6.9.1.7 may vary with reload cycle. The safety analysis flow requirements are thus established by the currently applicable LOCA analysis which has demonstrated compliance with the ECCS acceptance limits of 10 CFR 50.46.

2) The total pumped flow must be less than the HHSI pump runout limit. This flow varies with the specific HHSI pump assumed to operate during the accident. Since the HHSI pumps also function as normal charging pumps, their characteristics, including runout limits, will vary over service life.

3) The requirements for reactor coolant pump seal injection must be met during normal operation, and the effects of seal injection during accidents must be considered in meeting constraints 1) and 2) above.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator, Region II, within the time period specified for each report. These reports shall be submitted pursuant to the requirement of the applicable specification:

a. Inservice Inspection Reviews, Specification 4.0.5, shall be reported within 90 days of completion.

b. MODERATOR TEMPERATURE COEFFICIENT. Specification 3.1.1.4.

c. RADIATION MONITORING INSTRUMENTATION. Specification 3.3.3.1, Table 3.3-6, Action 35.

d. SEISMIC INSTRUMENTATION. Specifications 3.3.3.3 and 4.3.3.3.2.

e. METEOROLOGICAL INSTRUMENTATION. Specification 3.3.3.4.

f. Deleted.

g. LOOSE PARTS MONITORING SYSTEMS. Specification 3.3.3.9.

h. Deleted.

i. LOW-TEMPERATURE OVERPRESSURE PROTECTION. Specification 3.4.9.3.

j. EMERGENCY CORE COOLING SYSTEMS. Specification 3.5.2 and 3.5.3.

k. SETTLEMENT OF CLASS 1 STRUCTURES. Specification 3.7.12.

l. GROUND WATER LEVEL - SERVICE WATER RESERVOIR. Specification 3.7.13.

m. Deleted.

n. RADIOACTIVE EFFLUENTS. As required by the ODCM.

o. RADIOLOGICAL ENVIRONMENTAL MONITORING. As required by the ODCM.

p. SEALED SOURCE CONTAMINATION. Specification 4.7.11.1.3.

q. REACTOR COOLANT SYSTEM STRUCTURAL INTEGRITY. Specification 4.4.10. For any abnormal degradation of the structural integrity of the reactor vessel or the Reactor Coolant System pressure boundary detected during the performance of Specification 4.4.10, an initial report shall be submitted within 10 days after detection and a detailed report submitted within 90 days after the completion of Specification 4.4.10.

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

VIRGINIA ELECTRIC AND POWER COMPANY

OLD DOMINION ELECTRIC COOPERATIVE

DOCKET NO. 50-339

NORTH ANNA POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 170
License No. NPF-7

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

   A. The application for amendment by Virginia Electric and Power Company et al., (the licensee) dated April 15, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-7 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 170 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Victor M. McCree

Victor M. McCree, Acting Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 5, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 170

TO FACILITY OPERATING LICENSE NO. NPF-7

DOCKET NO. 50-339

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

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3/4 5-6	3/4 5-6
3/4 5-7	3/4 5-7
B 3/4 1-3	B 3/4 1-3
B 3/4 4-1	B 3/4 4-1
B 3/4 4-2	B 3/4 4-2
---	B 3/4 4-2a
B 3/4 4-6	B 3/4 4-6
B 3/4 4-7	B 3/4 4-7
B 3/4 4-8	B 3/4 4-8
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FLOW PATHS – OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System.

b. Two flow paths from the refueling water storage tank via charging pumps to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4#.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1.77% delta k/k at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 Each of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the heat traced portion of the flow path from the boric acid tanks is greater than or equal to 115°F when it is a required water source.

Only one boron injection 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 270°F.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. At least once per 18 months during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on a safety injection test signal.

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 One charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

a. With no charging pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until one charging pump is restored to OPERABLE status.

b. With no charging pump OPERABLE and the opposite unit in MODE 1, 2, 3 or 4, immediately initiate corrective action to restore at least one charging pump to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 The above required charging pump shall be demonstrated OPERABLE by verifying that, on recirculation flow, the pump develops a discharge pressure of greater than or equal to 2410 psig when tested pursuant to Specification 4.0.5.

4.1.2.3.2 All charging pumps, except the above required OPERABLE pump, shall be demonstrated inoperable at least once per 12 hours by verifying that the control switch is in the pull to lock position.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4#.

ACTION:

With only one charging pump OPERABLE, restore a second charging pump to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1.77% delta k/k at 200°F within the next 6 hours; restore a second charging pump to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours. The provisions of Specification 3.0.4 are not applicable for one hour following heatup above 270°F or prior to cooldown below 270°F.

SURVEILLANCE REQUIREMENTS

4.1.2.4.1 The above required charging pumps shall be demonstrated OPERABLE by verifying that on recirculation flow, each pump develops a discharge pressure of greater than or equal to 2410 psig when tested pursuant to Specification 4.0.5.

4.1.2.4.2 All charging pumps, except the above required OPERABLE pump, shall be demonstrated inoperable at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than or equal to 270°F by verifying that the control switch is in the pull to lock position.

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 270°F.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 All reactor coolant loops shall be in operation with power removed from the loop stop valve operators.

APPLICABILITY: MODES 1 and 2.*

ACTION:

With less than the above required reactor coolant loops in operation, be in at least HOT STANDBY within 1 hour.

SURVEILLANCE REQUIREMENTS

4.4.1.1 The above required reactor coolant loops shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

4.4.1.2 At least once per 31 days, with the reactor coolant loops in operation by verifying that the power is removed from the loop stop valve operators.

*See Special Test Exception 3.10.4.

LIMITING CONDITION FOR OPERATION

3.4.1.2 a. At least two of the reactor coolant loops listed below shall be OPERABLE:

1. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,
2. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,
3. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump,

b. At least one of the above coolant loops shall be in operation.*, **

APPLICABILITY: MODE 3

ACTION:

a. With less than the above required loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With no coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective actions to return the required coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 At least one cooling loop shall be verified to be in operation and circulating coolant at least once per 12 hours.

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* All reactor coolant pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

** The requirement to have one coolant loop in operation is exempted during the performance of the boron mixing tests as stipulated in License Condition 2.C(15)(f) and 2.C(20)(b).

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. At least two of the coolant loops listed below shall be OPERABLE:

1. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,*
2. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,*
3. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump,*
4. Residual Heat Removal Subsystem A,**
5. Residual Heat Removal Subsystem B.**

b. At least one of the above coolant loops shall be in operation.***

APPLICABILITY: MODES 4 and 5

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; be in COLD SHUTDOWN within 20 hours.

b. With no coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

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* A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to 270°F unless the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

** The offsite or emergency power source may be inoperable in MODE 5.

*** All reactor coolant pumps and residual heat removal pumps may be de-energized for up to 1 hour provided 1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

LIMITING CONDITION FOR OPERATION

3.4.3.2 Both power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or both PORVs inoperable but capable of being manually cycled, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one PORV inoperable and not capable of being manually cycled, within 1 hour either restore the PORV to OPERABLE status or capable of being manually cycled, or close its associated block valve and remove power from the block valve; restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With both PORVs inoperable and not capable of being manually cycled, within 1 hour either restore at least one PORV to OPERABLE status or capable of being manually cycled, or close its associated block valve and remove power from the block valve and be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

d. With one or both block valves inoperable, within 1 hour restore the block valve(s) to OPERABLE status or place its associated PORV(s) in manual control. Restore at least one block valve to OPERABLE status within the next hour if both block valves are inoperable; restore the remaining inoperable block valve to OPERABLE status within 72 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

e. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE:

a. At least once per 31 days by performing a CHANNEL FUNCTIONAL TEST, excluding valve operation, and

b. At least once per 18 months by:

1. Operating the PORV through one complete cycle of full travel during MODES 3 or 4, and

2. Operating the solenoid air control valves and check valves on the associated accumulators in the PORV control systems through one complete cycle of full travel, and

3. Performing a CHANNEL CALIBRATION of the actuation instrumentation.

4.4.3.2.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed in order to meet the requirements of ACTION b or c in Specification 3.4.3.2.

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with at least 125 kw of pressurizer heaters and a water volume of less than or equal to 1240 cubic feet.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

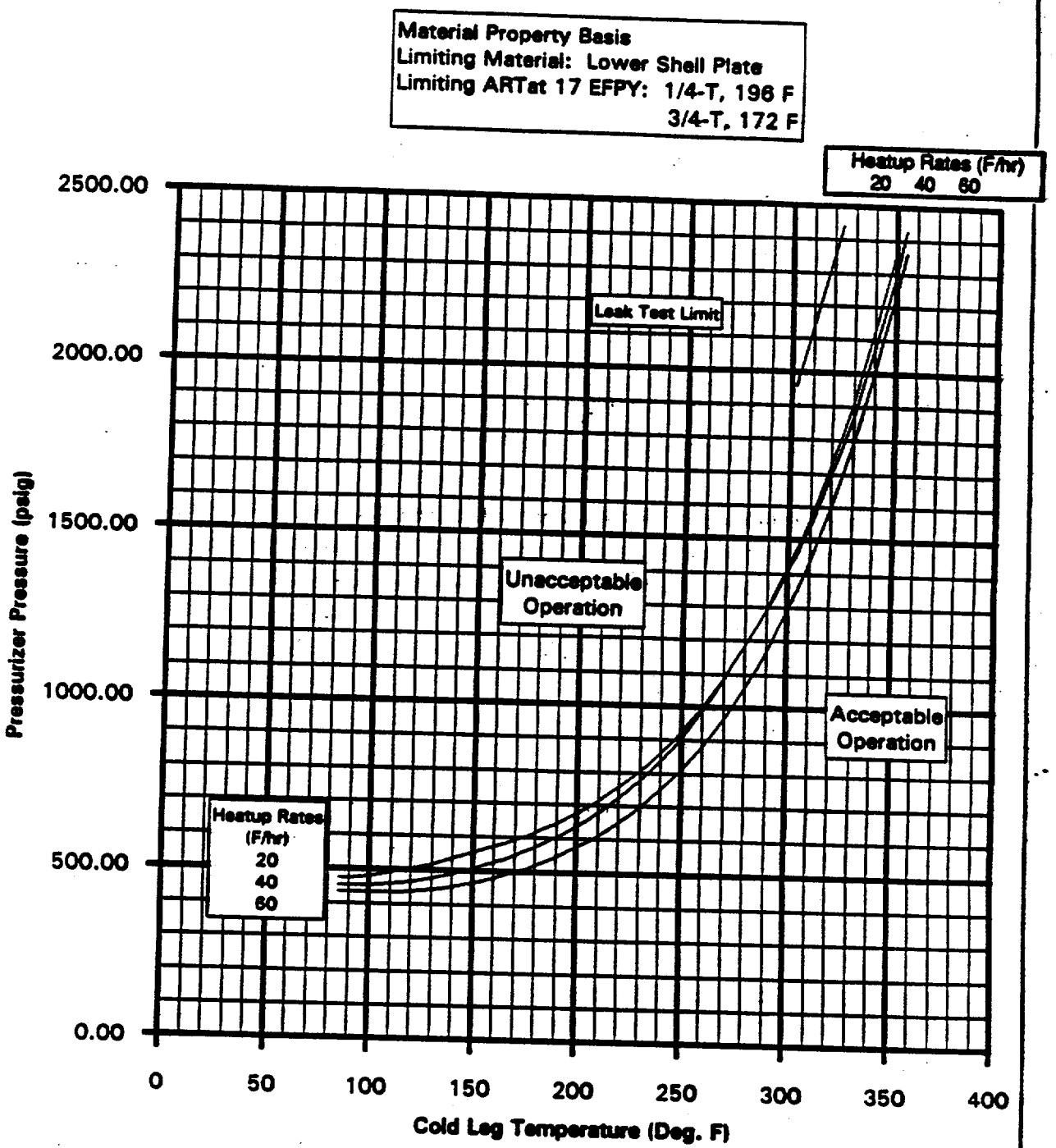
a. With the pressurizer inoperable due to an inoperable emergency power supply for the pressurizer heaters either restore the inoperable emergency power supply within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

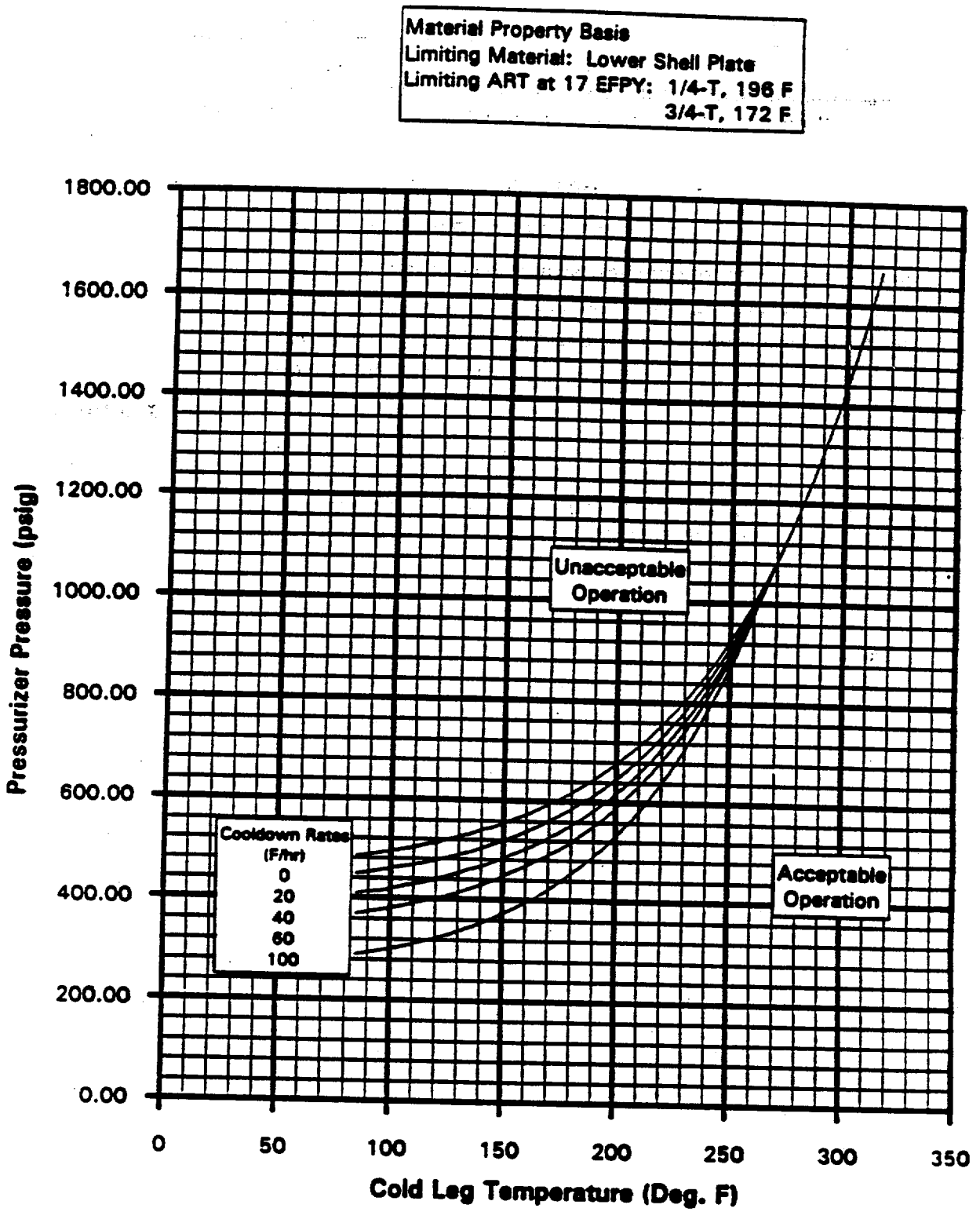
4.4.4.1 The pressurizer water volume shall be determined to be within its limit at least once per 12 hours.

Figure 3.4-2 — North Anna Unit 2 Reactor Coolant System Heatup Limitations



North Anna Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rates up to 60 F/hr) Applicable for the First 17 EFPY (Without Margins for Instrumentation Errors)

Figure 3.4-3 — **North Anna Unit 2**
Reactor Coolant System Cooldown Limitations



North Anna Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100 F/hr) Applicable for the First 17 EFPY (Without Margins for Instrumentation Errors)

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F or cooldown of 200°F, in any one hour period, and

b. A maximum spray water temperature and pressurizer temperature differential of 320°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

LIMITING CONDITION FOR OPERATION

3.4.9.3 Two power-operated relief valves (PORVs) shall be OPERABLE with lift settings of (1) less than or equal to 415 psig whenever any RCS cold leg temperature is less than or equal to 270°F, and (2) less than or equal to 375 psig whenever any RCS cold leg temperature is less than 130°F.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 270°F, MODE 5, and MODE 6 when the head is on the reactor vessel and the RCS is not vented through a 2.07 square inch or larger vent.

ACTION:

a. With one PORV inoperable in MODE 4, restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2.07 square inch vent within the next 8 hours.

b. With one PORV inoperable in MODES 5 or 6, either (1) restore the inoperable PORV to OPERABLE status within 24 hours, or (2) complete depressurization and venting of the RCS through at least a 2.07 square inch vent within a total of 32 hours.

c. With both PORVs inoperable, complete depressurization and venting of the RCS through at least a 2.07 square inch vent within 8 hours.

d. With the RCS vented per ACTIONS a, b, or c, verify the vent pathway at least once per 31 days when the pathway is provided by a valve(s) that is locked, sealed, or otherwise secured in the open position; otherwise, verify the vent pathway ever 12 hours.

e. In the event either the PORVs or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or vent(s) on the transient, and any corrective action necessary to prevent recurrence.

f. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9.3 Each PORV shall be demonstrated OPERABLE by:

a. Performance of a CHANNEL FUNCTIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE.

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel, at least once per 18 months.

c. Verifying the PORV keyswitch is in the AUTO position and the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

d. Testing pursuant to Specification 4.0.5.

3/4.4.10 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 & 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

3/4.10.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.1.

APPLICABILITY: ALL MODES

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limits or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.10.1.1 In addition to the requirements of Specification 4.0.5, the Reactor Coolant pump flywheels shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

4.4.10.1.2 In addition to the requirements of Specification 4.0.5, at least one third of the main member to main member welds, joining A572 material, in the steam generator supports, shall be visually examined during each 40 month inspection interval.

ECCS SUBSYSTEMS – Tavg GREATER THAN 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE low head safety injection pump,

c. An OPERABLE flow path capable of transferring fluid to the Reactor Coolant System when taking suction from the refueling water storage tank on a safety injection signal or from the containment sump when suction is transferred during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

c. The provisions of Specification 3.0.4 are not applicable to 3.5.2.a and 3.5.2.b for one hour following heatup above 270°F or prior to cooldown below 270°F.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

SURVEILLANCE REQUIREMENTS (Continued)

Valve Number	Valve Function	Valve Position
a. MOV-2890A	a. LHSI to hot leg	a. closed
b. MOV-2890B	b. LHSI to hot leg	b. closed
c. MOV-2836	c. Ch pump to cold leg	c. closed
d. MOV-2869A	d. Ch pump to hot leg	d. closed
e. MOV-2869B	e. Ch pump to hot leg	e. closed

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2. Of the areas affected within containment at the completion of each containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

1. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

e. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on a safety injection test signal.

2. Verifying that each of the following pumps start automatically upon receipt of a safety injection test signal:

a) Centrifugal charging pump, and

b) Low head safety injection pump.

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE centrifugal charging pump#,

b. One OPERABLE low head safety injection pump#, and

c. An OPERABLE flow path capable of automatically transferring fluid to the reactor coolant system when taking suction from the refueling water storage tank or from the containment sump when the suction is transferred during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of the low head safety injection pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

A maximum of one centrifugal charging pump and one low head safety injection pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 270°F.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.

4.5.3.2 All charging pumps and safety injection pumps, except the above required OPERABLE pumps, shall be demonstrated inoperable at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than or equal to 270°F by verifying that the control switch is in the pull to lock position.

3/4.5.4 BORON INJECTION SYSTEM

BORON INJECTION TANK

LIMITING CONDITION FOR OPERATION

3.5.4.1 The boron injection tank shall be OPERABLE with:

a. A contained borated water volume of at least 900 gallons,

b. Between 12,950 and 15,750 ppm of boron, and

c. A minimum solution temperature of 115°F.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the boron injection tank inoperable, restore the tank to OPERABLE status within 1 hour or be in HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to 1.77% $\Delta k/k$ at 200°F within the next 6 hours; restore the tank to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.4.1 The boron injection tank shall be demonstrated OPERABLE by:

a. Verifying the contained borated water volume at least once per 7 days,

b. Verifying the boron concentration of the water in the tank at least once per 7 days, and

c. Verifying the water temperature at least once per 24 hours.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid transfer pumps, 5) associated heat tracing systems, and 6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operation conditions of 1.77% delta k/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 6000 gallons of 12,950 ppm borated water from the boric acid storage tanks or 54,200 gallons of 2300 ppm borated water from the refueling water storage tank.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable below 270°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1.77% delta k/k after zenon decay and cooldown from 200°F to 140°F. This condition requires either 1378 gallons of 12,950 ppm borated water from the boric acid storage tanks or 3400 gallons of 2300 ppm borated water from the refueling water storage tank.

3/4.1.2 BORATION SYSTEMS (Continued)

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics. The OPERABILITY of one boron injection system during REFUELING insures that this system is available for reactivity control while in MODE 6.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.7 and 9.0 for the solution recirculated within the containment after a LOCA. This pH minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

At least one charging pump must remain operable at all times when the opposite unit is in MODE 1, 2, 3, or 4. This is required to maintain the charging pump cross-connect system operational.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section (1) ensure that acceptable power distribution limits are maintained, (2) ensure that the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential effects of rod misalignment on associated accident analyses. OPERABILITY of the movable control assemblies is established by observing rod motion and determining that rods are positioned within ± 12 steps (indicated position) of the respective demand step counter position. The OPERABILITY of the individual rod position indication system is established by appropriate periodic CHANNEL CHECKS, CHANNEL FUNCTIONAL TESTS, and CHANNEL CALIBRATIONS. OPERABILITY of the individual rod position indicators is required to determine control rod position and thereby ensure compliance with the control rod alignment and insertion limits. The OPERABLE condition for the individual rod position indicators is defined as being capable of indicating rod position within ± 12 steps of the associated demand position indicator. For power levels below 50 percent of RATED THERMAL POWER, the specifications of this section permit a maximum one hour stabilization in every 24 period (thermal "soak time") to allow stabilization of known thermal drift in the individual rod position indicator channels during which time the indicated rod position may vary from demand position indication by no more than ± 24 steps. This "1 in 24" feature is an upper limit on the frequency of thermal soak allowances and is available for both a continuous one hour period or one consisting of several discrete intervals. During this stabilization period, greater reliance is placed upon the demand position indicators to determine rod position. In addition, the ± 24 step/hour limit is not applicable when the control rod position is known to be greater than 12 steps from the rod group step counter demand position indication. Above 50 percent of RATED THERMAL POWER, rod motion is not expected to induce thermal transients of sufficient magnitude to exceed the individual rod position indicator instrument accuracy of ± 12 steps. Comparison of the demand position indicators to the bank insertion limits with verification of rod position by the individual rod position indicators (after thermal soak following rod motion below 50 percent of RATED THERMAL POWER) is sufficient verification that the control rods are above the insertion limits.

The control bank FULLY WITHDRAWN position can be varied within the interval of 225 to 229 steps withdrawn, inclusive. This interval permits periodic repositioning of the parked RCCAs to minimize wear, while having minimal impact on the normal reload core physics and safety evaluations. Changes of the RCCA FULLY WITHDRAWN position within this band are administratively controlled, using the rod insertion limit operator curve.

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with all reactor coolant loops in operation and maintain the DNBR above design limit during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODES 4 and 5, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat, but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two RHR loops to be OPERABLE.

After the reactor has shutdown and entered into MODE 3 for at least 100 hours, a minimum RHR system flow rate of 2000 gpm in MODE 5 is permitted, provided there is sufficient decay heat removal to maintain the RCS temperature less than or equal to 140°F. Since the decay heat power production rate decreases with time after reactor shutdown, the requirements for RHR system decay heat removal also decrease. Adequate decay heat removal is provided as long as the reactor has been shutdown for at least 100 hours after entry into MODE 3 and RHR flow is sufficient to maintain the RCS temperature less than or equal to 140°F. The reduced flow rate provides additional margin to vortexing at the RHR pump suction while in Mid Loop Operation. During a reduction in reactor coolant system boron concentration the Specification 3.1.1.3.1 requirement to maintain 3000 gpm flow rate provides sufficient coolant circulation to minimize the effect of a boron dilution incident and to prevent boron stratification.

The restrictions on starting a Reactor Coolant Pump with one or more RCS cold legs less than or equal to 270°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

The requirement to maintain the boron concentration of an isolated loop greater than or equal to the boron concentration of the operating loops ensures that no reactivity addition to the core could occur during startup of an isolated loop. Verification of the boron concentration in an idle loop prior to opening the cold leg stop valve provides a reassurance of the adequacy of the boron concentration in the isolated loop. Operating the isolated loop on recirculating flow for at least 90 minutes prior to opening its cold leg stop valve ensures adequate mixing of the coolant in this loop and prevents any reactivity effects due to boron concentration stratifications.

Startup of an idle loop will inject cool water from the loop into the core. The reactivity transient resulting from this cool water injection is minimized by delaying isolated loop startup until its temperature is within 20°F of the operating loops. Making the reactor subcritical prior to loop startup prevents any power spike which could result from this cool water induced reactivity transient.

3/4.4.2 AND 3/4.4.3 SAFETY AND RELIEF VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 380,000 lbs per hour of saturated steam at the valve set point. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during hot shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, or the power operated relief valves (PORVs) will provide overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip setpoint is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

The power operated relief valves and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.

The OPERABILITY of the PORVs and block valves is determined on the basis of their being capable of performing the following functions:

a) Manual control of PORVs to control reactor coolant system pressure. This is a function that may be used to mitigate certain accidents and for plant shutdown.

b) Maintaining the integrity of the reactor coolant pressure boundary. This function is related to controlling identified leakage and ensuring the ability to detect unidentified reactor coolant pressure boundary leakage.

c) Manual control of the block valve to (1) unblock an isolated PORV to allow it to be used for manual control of reactor coolant system pressure (Item a, above), and (2) isolate a PORV with excessive seat leakage (Item b, above).

d) Automatic control of PORVs to control reactor coolant system pressure. This function reduces challenges to the code safety valves for overpressurization events.

e) Manual control of a block valve to isolate a stuck-open PORV.

Surveillance Requirements provide the assurance that the PORVs and block valves can perform their functions. Specification 4.4.3.2.1 addresses the PORVs and Specification 4.4.3.2.2 addresses the block valves. The block valves are exempt from the surveillance requirements to cycle the valves when they have been closed to comply with the ACTION requirements. This precludes the need to cycle the valves with full system differential pressure or when maintenance is being performed to restore an inoperable PORV to operable status.

Surveillance Requirement 4.4.3.2.1.b.2 provides for the testing of the mechanical and electrical aspects of control systems for the PORVs.

Testing of PORVs in HOT STANDBY or HOT SHUTDOWN is required in order to simulate the temperature and pressure environmental effects on PORVs. Testing at COLD SHUTDOWN is not considered to be a representative test for assessing PORV performance under normal plant operating conditions.

3/4.4.4 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12-hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE ensures that the plant will be able to establish natural circulation.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the North Anna site such as site boundary location and meteorological conditions, were not considered in this evaluation. The NRC is finalizing site specific criteria which will be used as the basis for the reevaluation of the specific activity limits of this site. This reevaluation may result in higher limits.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 ~μCi/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE / TEMPERATURE LIMITS

Reactor Coolant System

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 5.2 of the UFSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Consequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

The heatup limit curve, Figure 3.4-2, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour. The cooldown limit curves of Figure 3.4-3 are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of 17 EFPY. The most recent capsule analysis results are documented in Westinghouse Reports WCAP-12497, January 1990. The heatup and cooldown curves are documented in Westinghouse Report WCAP-12503, March, 1990.

The reactor vessel materials have been tested to determine their initial RT_{NDT}. Reactor operation and resultant fast neutron (E > 1 Mev) irradiation will cause an increase in the RT_{NDT}. An adjusted reference temperature, based upon the fluence and copper content of the material in question, can be predicted using US NRC Regulatory Guide 1.98, Revision 2. The heatup and cooldown limit curves (Figure 3.4-2 and Figure 3.4-3) include predicted adjustments for this shift in RT_{NDT} at the end of 17 EFPY. The reactor vessel beltline region material properties are listed on Figures 3.4-2 and 3.4-3.

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

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

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

Pressurizer

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

Low-Temperature Overpressure Protection

The OPERABILITY of two PORVs or an RCS vent opening of greater than 2.07 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 270°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures or (2) the start of a charging pump and its injection into a water-solid RCS.

Automatic or passive low temperature overpressure protection (LTOP) is required whenever any RCS cold leg temperature is less than 270°F. This temperature is the water temperature corresponding to a metal temperature of at least the limiting RT_{NDT} + 50°F + instrument uncertainty. Above 270°F administrative control is adequate protection to ensure the limits of the heatup curve (Figure 3.4-2) and the cooldown curve (Figure 3.4-3) are not violated. The concept of requiring automatic LTOP at the lower end, and administrative control at the upper end, of the Appendix G curves is further discussed in NRC Generic Letter 88-11.

3/4.4.10 STRUCTURAL INTEGRITY

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

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

3/4.5.1 ACCUMULATORS

The OPERABILITY of each RCS accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

ECCS SUBSYSTEMS (Continued)

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The limitation for a maximum of one centrifugal charging pump and one low head safety injection pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps and low head safety injection pumps except the required OPERABLE pump to be inoperable below 270°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained.

In the event of modifications to an ECCS subsystem that could alter the subsystem flow characteristics, a flow balance test shall be performed. The flow balance test criteria are established based on the system performance assumed in the safety analysis (minimum flow limit) and on HHSI pump runout protection (maximum flow limit). In performing the flow balance, the effects of flow measurement instrument uncertainties accounting for system configuration and the variability between installed pumps must be properly considered.

Numerical acceptance criteria for the flow balance test are specified in the surveillance test procedure. These criteria are established based on the following considerations:

1) The total injected flow to the core (assuming spillage of the branch line with the highest flow) must meet or exceed that assumed in the safety analysis. The limiting safety analysis is the loss of coolant accident (LOCA) analysis. This criterion may vary, particularly since the inputs to the safety analysis controlled by LCO 6.9.1.7 may vary with reload cycle. The safety analysis flow requirements are thus established by the currently applicable LOCA analysis which has demonstrated compliance with the ECCS acceptance limits of 10 CFR 50.46.

2) The total pumped flow must be less than the HHSI pump runout limit. This flow varies with the specific HHSI pump assumed to operate during the accident. Since the HHSI pumps also function as normal charging pumps, their characteristics, including runout limits, will vary over service life.

3) The requirements for reactor coolant pump seal injection must be met during normal operation, and the effects of seal injection during accidents must be considered in meeting constraints 1) and 2) above.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator, Region II, within the time period specified for each report. These reports shall be submitted pursuant to the requirement of the applicable specification:

a. Inservice Inspection Reviews, Specification 4.0.5, shall be reported within 90 days of completion.

b. MODERATOR TEMPERATURE COEFFICIENT. Specification 3.1.1.4.

c. Deleted.

d. RADIATION MONITORING INSTRUMENTATION. Specification 3.3.3.1, Table 3.3-6, Action 35.

e. Deleted.

f. LOW-TEMPERATURE OVERPRESSURE PROTECTION. Specification 3.4.9.3.

g. EMERGENCY CORE COOLING SYSTEMS. Specification 3.5.2 and 3.5.3.

h. SETTLEMENT OF CLASS 1 STRUCTURES. Specification 3.7.12.

i. GROUND WATER LEVEL - SERVICE WATER RESERVOIR. Specification 3.7.13.

j. Deleted.

k. Deleted.

l. RADIOACTIVE EFFLUENTS. As required by the ODCM.

m. RADIOLOGICAL ENVIRONMENTAL MONITORING. As required by the ODCM.

n. SEALED SOURCE CONTAMINATION. Specification 4.7.11.1.3.

o. REACTOR COOLANT SYSTEM STRUCTURAL INTEGRITY. Specification 4.4.10. For any abnormal degradation of the structural integrity of the reactor vessel or the Reactor Coolant System pressure boundary detected during the performance of Specification 4.4.10, an initial report shall be submitted within 10 days after detection and a detailed report submitted within 90 days after the completion of Specification 4.4.10.

p. CONTAINMENT STRUCTURAL INTEGRITY. Specification 4.6.1.6. For any abnormal degradation of the containment structure detected during the performance of Specification 4.6.1.6, an initial report shall be submitted within 10 days after completion of Specification 4.6.1.6. A final report, which includes (1) a description of the condition of the liner plate and concrete, (2) inspection procedure, (3) the tolerance on cracking, and (4) the corrective actions taken, shall be submitted within 90 days after the completion of Specification 4.6.1.6.

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.

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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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