June 30, 1989

Docket No. 50-338

DISTRIBUTION
See attached page

Mr. W. R. Cartwright
Vice President - Nuclear
Virginia Electric and Power Company
5000 Dominion Blvd.
Glen Allen, Virginia 23060

Dear Mr. Cartwright:

SUBJECT: NORTH ANNA UNIT 1 - ISSUANCE OF AMENDMENT RE: HEATUP AND COOLDOWN CURVES (TAC NO. 72060)

The Commission has issued the enclosed Amendment No. 117 to Facility Operating License No. NPF-4 for the North Anna Power Station, Unit No. 1 (NA-1). The amendment revises the Technical Specifications (TS) in response to your letter dated November 11, 1988, as supplemented June 19, 1989.

This amendment revises the heatup and cooldown curves and associated low temperature overpressurization setpoints to be valid for a period up to 10 effective full power years. In addition, the staff finds that the submittals supporting your amendment request comply with Generic Letter 88-11 and the methods specified in Regulatory Guide 1.99, Revision 2.

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

Sincerely,

Original signed by

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

Enclosures:
1. Amendment No. 117to NPF-4
2. Safety Evaluation

cc w/enclosures:
See next page

DF01
1/1

[NA-1 AMEND 72060]

LA:PDII-2	PM:PDII-2	D:PDII-2	OGC
DMiller	LEngle:bd	HBerkow	J Moore
06/30/89	06/30/89	06/30/89	06/30/89

8907130368 890630
PDR ADOCK 05000338
P PNU

Mr. W. R. Cartwright
Virginia Electric & Power Company

North Anna Power Station
Units 1 and 2

cc:
Mr. William C. Porter, Jr.
County Administrator
Louisa County
P.O. Box 160
Louisa, Virginia 23093

Michael W. Maupin, Esq.
Hunton and Williams
P. O. Box 1535
Richmond, Virginia 23212

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

Old Dominion Electric Cooperative
c/o Executive Vice President
Innsbrook Corporate Center
4222 Cox Road, Suite 102
Glen Allen, Virginia 23060

Mr. W. L. Stewart
Senior Vice President - Power
Virginia Electric and Power Co.
Post Office Box 26666
Richmond, Virginia 23261

Mr. Patrick A. O'Hare
Office of the Attorney General
Supreme Court Building
101 North 8th Street
Richmond, Virginia 23219

Resident Inspector/North Anna
c/o U.S. NRC
Senior Resident Inspector
Route 2, Box 78
Mineral, Virginia 23117

C. M. G. Buttery, M.D., M.P.H.
Department of Health
109 Governor Street
Richmond, Virginia 23219

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta Street N.W., Suite 2900
Atlanta, Georgia 30323

Mr. G. E. Kane
P. O. Box 402
Mineral, Virginia 23117

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

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. 117
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 November 30, 1988, as supplemented June 19, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 117 , 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 the date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, 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: June 30, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 117

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.

Page
3/4 1-9
3/4 1-12
3/4 4-3
3/4 4-27
3/4 4-28
3/4 4-31
3/4 5-3
3/4 5-6
3/4 5-6a
B3/4 1-2
B3/4 1-3
B3/4 4-1
B3/4 4-6
B3/4 4-7
B3/4 4-8
B3/4 4-9
B3/4 4-10
B3/4 4-11
B3/4 5-2

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 $\geq$ 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 324°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% $\Delta k/k$ 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 324°F or prior to cooldown below 324°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 $\geq$ 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 324°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 324°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.

*A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to 324°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 residual heat removal loop(s) shall be determined OPERABLE per Specification 4.0.5.

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

Material Property Basis

Controlling Material:	Circumferential Weld
Copper Content:	0.086 WT%
Nickel Content:	0.11 WT%
Initial RT_{NDT}:	19°F
RT_{NDT}	1/4T, 136.3 °F 3/4T, 116.1 °F

Curves Applicable For Service Periods Up To 10 EFPY And Contain Margins Of 20 °F And 80 psi For Possible Instrument Errors

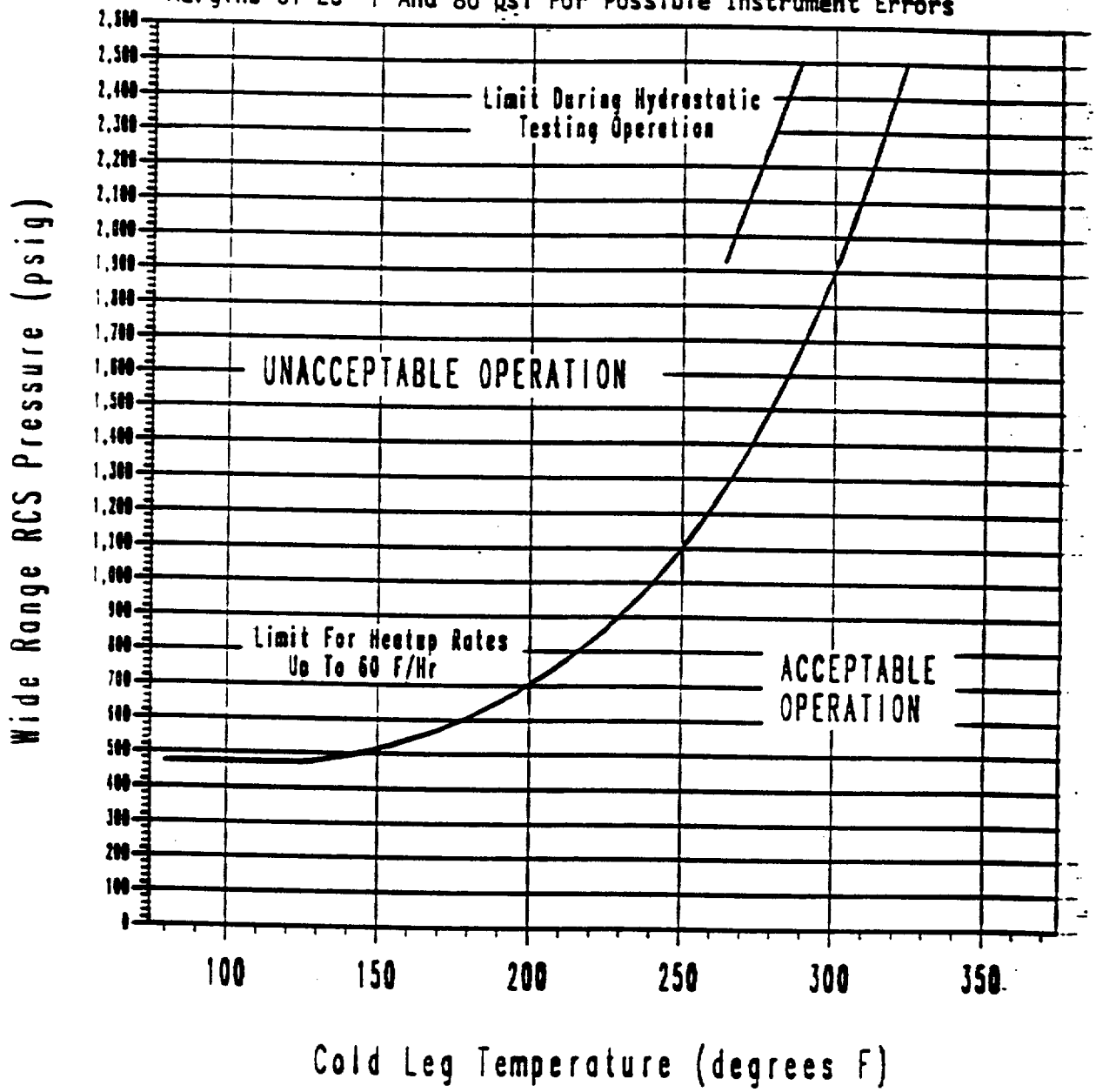

Figure 3.4.2 Reactor Coolant System Pressure-Temperature Heatup Limitations

Material Property Basis

Controlling Material:	Circumferential Weld
Copper Content:	0.086 WT%
Nickel Content:	0.11 WT%
Initial RT_{NDT}:	19 °F
RT_{NDT}	1/4T, 136.3 °F 3/4T, 116.1 °F

Curves Applicable For Service Periods Up To 10 EFPY And Contain Margins Of 20 °F And 80 psi For Possible Instrument Errors

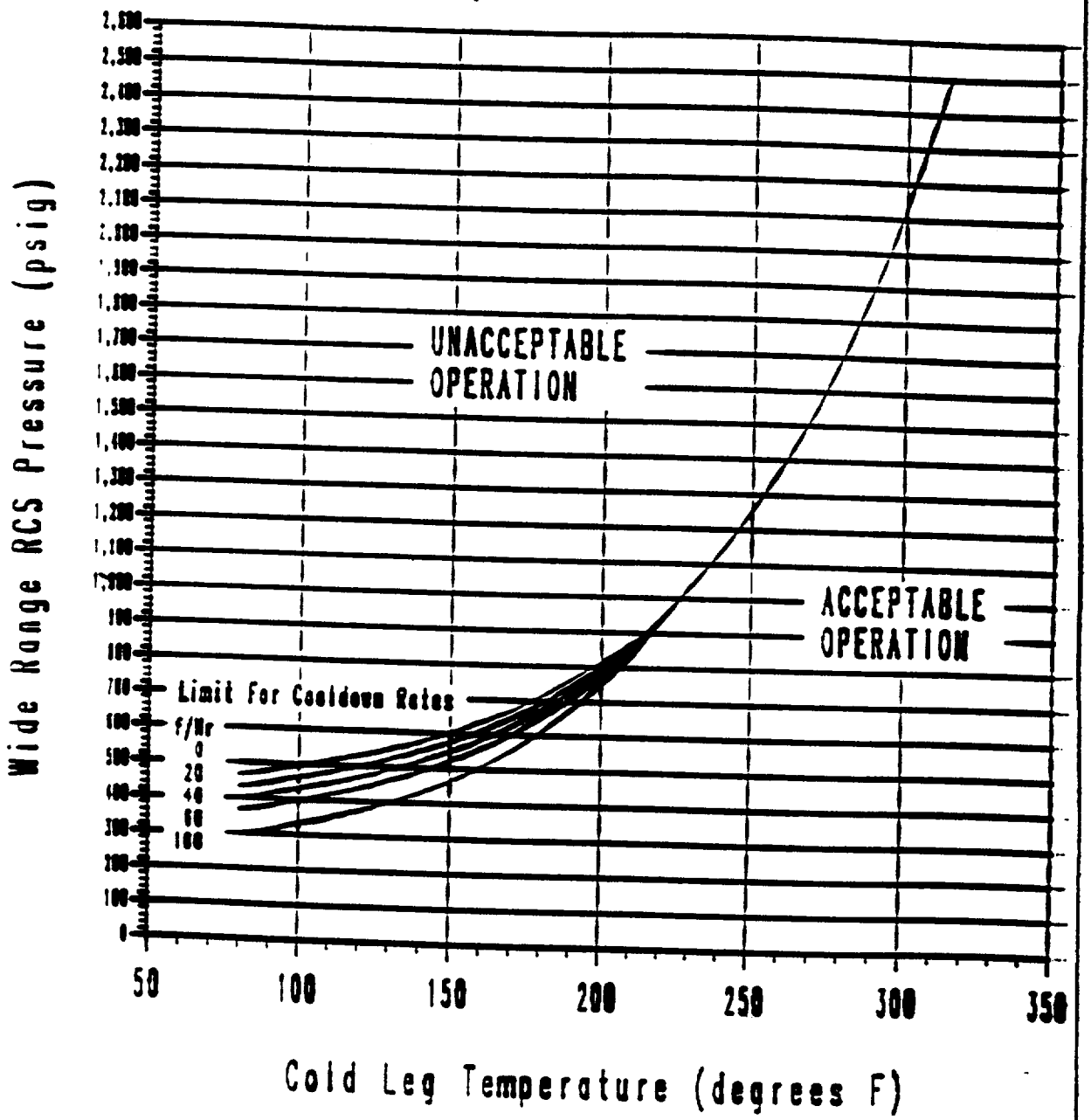

Figure 3.4.3 Reactor Coolant System Pressure-Temperature Cooldown Limitations

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following overpressure protection systems shall be OPERABLE:

a. Two power operated relief valves (PORVs) with a lift setting of: 1) less than or equal to 450 psig whenever any RCS cold leg temperature is less than or equal to 261°F, and 2) less than or equal to 390 psig whenever any RCS cold leg temperature is less than 150°F, or

b. A reactor coolant system vent of greater than or equal to 2.07 square inches.

APPLICABILITY: When the temperature of one or more of the RCS cold legs is less than or equal to 261°F, except when the reactor vessel head is removed.

ACTION:

a. With one PORV inoperable, either restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through 2.07 square inch vent(s) within the next 8 hours; maintain the RCS in a vented condition until both PORVs have been restored to OPERABLE status.

b. With both PORVs inoperable, depressurize and vent the RCS through a 2.07 square inch vent(s) within 8 hours; maintain the RCS in a vented condition until both PORVs have been restored to OPERABLE status.

c. In the event either the PORVs or the RCS vent(s) are used to mitigate a 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.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 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.

4.4.9.3.2 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

ECCS SUBSYSTEMS - $T_{avg} \geq 350°F$

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 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 accumlated 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 324°F or prior to cooldown below 324°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:

Valve Number	Valve Function	Valve Position
a. MOV-1890A	a. LHSI to hot leg	a. closed
b. MOV-1890B	b. LHSI to hot leg	b. closed
c. MOV-1836	c. Ch pump to cold leg	c. closed
d. MOV-1869A	d. Ch pump to hot leg	d. closed
e. MOV-1869B	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.

SURVEILLANCE REQUIREMENTS (Continued)

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.

f. By verifying that each of the following pumps develop the indicated discharge pressure (after subtracting suction pressure) on recirculation flow when tested pursuant to Specification 4.0.5.

   1. Centrifugal charging pump $\geq$ 2410 psig.

   2. Low head safety injection pump $\geq$ 156 psig

g. By verifying that the following manual valves requiring adjustment to prevent pump "runout" and subsequent component damage are locked and tagged in the proper position for injection:

   1. Within 4 hours following completion of any repositioning or maintenance on the valve when the ECCS subsystems are required to be OPERABLE.

   2. At least once per 18 months.

1.	1-SI-188	Loop A Cold Leg
2.	1-SI-191	Loop B Cold Leg
3.	1-SI-193	Loop C Cold Leg
4.	1-SI-203	Loop A Hot Leg
5.	1-SI-204	Loop B Hot Leg
6.	1-SI-205	Loop C Hot Leg

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying that:

   1. For high head safety injection lines, with a single pump running:

      a) The sum of the injection line flow rates, excluding the highest flow rate, is $\geq$ 384 gpm, and

      b) The total pump flow rate is $\leq$ 650 gpm.

ECCS SUBSYSTEMS - T_{avg} < 350°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 324°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 324°F by verifying that the switches in the Control Room are in the pull to lock position.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.77% $\Delta k/k$ is initially required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} <200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal. A 1.77% $\Delta k/k$ shutdown margin provides adequate protection for the boron dilution accident.

3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 3000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 3000 GPM will circulate an equivalent Reactor Coolant System volume of 9957 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron reductions will therefore be within the capability for operator recognition and control.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the value of this coefficient remains within the limiting conditions assumed for this parameter in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC) (Continued)

conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC value was obtained by incrementally correcting the MTC used in the FSAR analyses to nominal operating conditions. These corrections involved adding the incremental change in the MTC associated with a core condition of Bank D inserted to an all rods withdrawn condition and an incremental change in MTC to account for measurement uncertainty at RATED THERMAL POWER conditions. These corrections result in the limiting MTC value of -5.0×10^{-4} $\Delta k/k/°F$. The MTC value of -4.0×10^{-4} $\Delta k/k/°F$ represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibruim boron concentration and is obtained by making these corrections to the limiting MTC value of -5.0×10^{-4} $\Delta k/k/°F$.

Once the equilibrium boron concentration falls below about 60 ppm, dilution operations take an extended amount of time and reliable MTC measurements become more difficult to obtain due to the potential for fluctuating core conditions over the test interval. For this reason, MTC measurements may be suspended provided the measured MTC value at an equilibrium full power boron concentration $\leq$ 60 ppm is less negative than -4.7×10^{-4} delta k/k/°F. The difference between this value and the limiting MTC value of -5.0×10^{-4} delta k/k/°F conservatively bounds the maximum credible change in MTC between the 60 ppm equilibrium boron concentration (all rods withdrawn, RATED THERMAL POWER conditions) and the licensed end-of-cycle, including the effect of boron concentration, burnup, and end-of-cycle coastdown.

The surveillance requirements for measurement of the MTC at the beginning and near the end of each fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 541°F. This limitation is required to ensure 1) the moderator temperature coefficient is within its analyzed temperature range, 2) the protective instrumentation is within its normal operating range, and 3) the P-12 interlock is above its setpoint, and 4) compliance with Appendix G to 10 CFR Part 50 (see Bases 3/4.4.9).

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.

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 capabioity 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 324°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 xenon 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 also ensure a pH value of between 8.5 and 11.0 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 ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential effects of rod misalignment on associated accident analyses. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors or a restriction in THERMAL POWER; either of these restrictions provides assurance of fuel rod integrity during continued operation. In addition those accident analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the accident analyses. Measurement with $T_{avg} \geq 500^{\circ}F$ and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above 1.30 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.

The restrictions on starting a Reactor Coolant Pump with one or more RCS cold legs less than or equal to 324°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 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 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 shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides 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 Protection System trip set point 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 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.

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

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 10 EFPY. The adjusted reference temperature was calculated using results from a capsule removed after the sixth fuel cycle. The results are documented in Westinghouse Reports WCAP-11777, February 1988 and WCAP-11791, May 1988.

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of these tests are shown in the UFSAR and WCAP-11777. Reactor operation and resultant fast neutron (E>1 Mev) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence and copper content of the material in question, can be predicted using US NRC Regulatory Guide 1.99, Revision 2. The heatup and cooldown limit curves (Figure 3.4.2 and Figure 3.4.3) include predicted adjustments for this shift in RT_{NDT} at the end of 10 EFPY, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-70, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and

vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

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 and WCAP-11777 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

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.

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 261°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 261°F. This temperature is the water temperature corresponding to a metal temperature of at least the limiting RT_{NDT} + 90°F + instrument uncertainty. Above 261°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.

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

THIS PAGE INTENTIONALLY LEFT BLANK

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 324°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.

3/4.5.4 BORON INJECTION SYSTEM

The OPERABILITY of the boron injection system as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS system cooldown. RCS cooldown can be caused by inadvertent depressurization, a loss-of-coolant accident or a steam line rupture.

The limits on injection tank minimum contained volume and boron concentration ensure that the assumptions used in the steam line break analysis are met.

The OPERABILITY of the redundant heat tracing channels associated with the boron injection system ensure that the solubility of the boron solution will be maintained above the solubility limit of 111°F at 15,750 ppm boron.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 117

FACILITY OPERATING LICENSE NO. NPF-4

VIRGINIA ELECTRIC AND POWER COMPANY

OLD DOMINION ELECTRIC COOPERATIVE

NORTH ANNA POWER STATION, UNIT NO. 1

DOCKET NO. 50-338

1.0 INTRODUCTION

In response to Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," the Virginia Electric and Power Company (the licensee) requested changes to the pressure/temperature (P/T) limits for the North Anna Power Station, Unit No. 1 (NA-1) Technical Specifications (TS). The request was documented in letters from the licensee dated November 30, 1988 and June 19, 1989. The purpose of the changes is to revise the P/T limits, which would be valid up to 10 effective full power years (EFPY). The proposed P/T limits were developed and based on the data from actual surveillance capsules. The proposed revision provides up-to-date P/T limits for operation of the reactor coolant system during heatup, cooldown, criticality, and hydrotest.

The licensee's letter dated June 19, 1989 provided supplemental information concerning the temperature difference between the water at the vessel surface and the metal at the ¼T and 3/4T locations. This information did not change the staff's initial determination that the proposed amendment does not involve significant hazards considerations.

To evaluate the P/T limits, the staff used the following NRC regulations and guidance: Appendices G and H to 10 CFR Part 50; the ASTM Standards and ASME Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); Regulatory Guide 1.99, Revision 2; Standard Review Plan (SRP), Section 5.3.2 and Generic Letter 88-11.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide TS for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the TS. The P/T limits are among the limiting conditions of operation in the TS for all commercial nuclear plants in the U.S. Appendices G and H to 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance which must be considered in setting P/T limits. An acceptable method in constructing the P/T limits is described in SRP Section 5.3.2.

Appendix G to 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, to test the beltline materials in the surveillance capsules in accordance with Appendix H to 10 CFR Part 50. Appendix H, in turn, refers to the ASTM Standards. These tests define the condition of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE). Generic Letter 88-11 requested that licensees and permittees use the methods in Regulatory Guide (RG) 1.99, Revision 2 to predict the effect of neutron irradiation on reactor vessel materials. This RG defines the ART as the sum of the unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and margin to account for uncertainties in the prediction method.

Appendix H to 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and that they contain test specimens that are made from plate, weld, and heat-affected-zone materials of the reactor beltline.

Based on the above changes, the licensee also proposed changes to several other TS to assure a proper low temperature overpressure protection (LTOP) at NA-1.

2.0 DISCUSSION

2.1 Pressure/Temperature Limits

The staff has evaluated the effect of neutron irradiation embrittlement on each beltline material in the NA-1 reactor vessel. The amount of neutron irradiation embrittlement was calculated in accordance with RG 1.99, Revision 2. The staff has determined that the material with the highest ART (most embrittled) at 10 EFPY was the circumferential weld between the intermediate and lower shell forgings.

The licensee has removed two surveillance capsules from the NA-1 reactor pressure vessel. The results from capsule V were published in a Babcock and Wilcox report BAW-1683. The results from capsule U were published in a Westinghouse report WCAP-11777. All surveillance capsules contained Charpy impact specimens and tensile specimens which were made from base metal and weld metal, and Charpy impact specimens from the heat-affected zone material.

For the limiting beltline material, the weld between the intermediate and lower shell forgings, the staff has calculated the amount of embrittlement per RG 1.99, Revision 2. The ART at 10 EFPY at ¼T was calculated to be 136°F. The ART was calculated by applying the least squares extrapolation method in paragraph 2.1 of RG 1.99, Revision 2 to NA-1 surveillance data.

The licensee used the method in RG 1.99, Revision 2, to calculate an ART of 136°F for the weld between the intermediate and lower forgings. The staff performed a similar calculation and verified the licensee's ART value to be acceptable

(see Table 1). Substituting the ART of 136°F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, criticality, and hydrotest meet the beltline material requirements in Appendix G to 10 CFR Part 50.

TABLE 1

The NRC Staff Calculated Adjusted Reference Temperature For The Limiting Reactor Beltline Material at North Anna, Unit 1.

Limiting Beltline Material:	Intermediate-to-lower shell weld
Code No.:	25531/1211/SMIT 89
Copper Content:	0.086%
Nickel Content:	0.11%
Initial Reference Temperature:	19°F
Neutron Fluence n/cm^2 at inside surface at 10 EFPY:	1.39E19
ART at ¼T at 10 EFPY:	136°F (Licensee Calculated 136°F)

In addition to beltline materials, Appendix G to 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.2 of Appendix G states that when pressure exceeds 20 percent of the preservice system hydrostatic test pressure, the temperature of the closure flange regions that are highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Based on the flange reference temperature of -22°F, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

Section IV.B of Appendix G requires the predicted Charpy upper shelf energy (USE) at end-of-life to be above 50 ft-lb. At end-of-life, the limiting material (the materials with the highest adjusted RT_{NDT} and lowest Charpy USE) is the lower shell forging. At 5.9 EFPY, its measured Charpy USE is 93 ft-lb. Using the prediction method in RG 1.99, Revision 2, the Charpy USE at end-of-life will still be above 50 ft-lb.

2.2 Low Temperature Overpressure Protection

LTOP is provided by the power operated relief valves (PORVs) on the pressurizer. These PORVs are set at pressures low enough to prevent violation of the Appendix G heatup and cooldown curves should a reactor coolant system (RCS) pressure transient occur during low temperature operations. The licensee, in References 1 and 2, provides the results of analyses of the most limiting overpressure transients in determining the PORV setpoints for LTOP.

A most limiting mass addition transient was analyzed assuming an inadvertent actuation of a charging pump. TS 3.1.2.4 and 3.5.3 allow only one charging pump to be operable when the RCS temperature is less than 324°F, that is the

maximum RCS temperature during which the LTOP is required. In response to the staff's request, the licensee provided in Reference 2 the results of a sensitivity analysis. It is indicated that an inadvertent initiation of an additional low head safety injection pump would not significantly increase the peak pressure during the mass addition transient. This is because the shutoff head of the low head safety injection pump will be reached in a short time and the contribution to mass addition of the low head pump will be negligible. In this analysis, the licensee assumes initial RCS temperature of 100°F and 200°F each in combination with the PORV setpoints of 365 psia and 435 psia. The results indicate that the initial RCS temperature has a minor effect on the transient peak pressure. The heatup and cooldown P/T curves increase sharply at temperatures above 200°F. The slightly increased peak pressure due to the higher initial temperatures will be offset by the increased space between the PORV setpoint, determined to prevent reaching the Appendix G curve at a lower temperature and the Appendix G curve. Thus, the Appendix G curve allows higher pressure overshoot at temperatures above 200°F. Considering the above factors, the staff concludes that the assumptions applied to the licensee's analysis are reasonably conservative.

The licensee uses RETRAN 02/MOD02 to perform its analysis in supporting the proposed TS changes. Both RETRAN 01/MOD03 and RETRAN 02/MOD02 have been generically approved by the staff on September 4, 1984 (Reference 3). Also, the licensee's topical report on its plant-specific application of RETRAN 01/MOD03 was reviewed and approved by the staff on April 11, 1985 (Reference 4).

In addition, the licensee submitted comparisons between RETRAN 01/MOD03 and RETRAN 02/MOD02 for a series of plant transients (Reference 5). This information demonstrated that the RETRAN 01/MOD03 and RETRAN 02/MOD02 code results are nearly identical for the NA-1&2 plant-specific models except for the changes caused by the nonequilibrium pressurizer model in RETRAN 02/MOD02. However, the LTOP transients deal with RCS in water solid conditions, thus, it is not affected by the nonequilibrium pressurizer model in RETRAN 02/MOD02. While the licensee's application on the use of RETRAN 02/MOD02 has not been reviewed and approved by the staff, the staff finds reasonable assurance exists that the results of the licensee's analysis using RETRAN 02/MOD02 supports the proposed TS changes on the LTOP. However, if any concerns should be raised during the staff's review of the licensee's RETRAN 02/MOD02 application, the staff will reevaluate the licensee's evaluation on the subject TS changes.

The modified Appendix G curve temperature corresponding to the pressurizer safety valve setpoint of 2485 psig is 324°F. This point is used to bound all of the low temperature transient analyses. Below 324°F, the anticipated low temperature overpressurization transients may be adequately mitigated by the automatic action of the pressurizer PORVs or by allowing sufficient time for operator response. Automatic LTOP is required whenever any RCS cold leg temperature is less than 261°F. This LTOP temperature is determined in accordance with the requirements set forth in Section B.2 of the Branch Technical Position RSB 5-2. Above 261°F administrative control will provide adequate protection because of the Appendix G fracture criterion. The analysis has an increased margin at higher temperatures. In addition, operation of the RCS above 261°F decreases the effects of the two design basis transients. Based on the results of the most limiting LTOP transient, the licensee's proposed TS PORV setpoints are less

than or equal to 450 psig when the RCS temperature is less than or equal to 261°F and less than or equal to 390 psig when the RCS temperature is less than or equal to 150°F.

The licensee proposed changes to TS 3.1.2.2, 3.1.2.4, 3.4.1.3, 3.4.9.3, 3.5.2, and 3.5.3 and their associated bases sections reflect the above discussed LTOP alignment temperatures and the heatup and cooldown rates identified by the updated TS Figures 3.4-2 and 3.4-3. The staff finds that they are reasonably conservative and acceptable.

3.0 EVALUATION

Based on the discussion above, the staff concludes that the proposed NA-1 P/T limits on the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 10 EFPY, because the limits conform to requirements of Appendices G and H to 10 CFR Part 50. The licensee's submittal also satisfies Generic Letter 88-11, because it used the method in RG 1.99, Revision 2 to calculate the ART. Hence, the proposed P/T limits may be incorporated into the NA-TS. In addition, the staff concludes that the licensee's proposed changes to TS 3.1.2.2, 3.1.2.4, 3.4.1.3, 3.4.9.3, 3.5.2, and 3.5.3 and their associated bases sections are acceptable to support the updated P/T limits applicable for a period up to 10 EFPY. Therefore, the staff finds the proposed changes to be acceptable.

4.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to surveillance requirements. The 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 published a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. 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 these amendments.

5.0 CONCLUSION

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: June 30, 1989

Principal Contributors:

C. Liang
B. Elliot

6.0 REFERENCES

1. Letter from W. Cartwright of Virginia Electric and Power Company to USNRC, "Proposed Technical Specifications Change," dated November 30, 1988.

2. Letter from W. Stewart of Virginia Electric and Power Company to USNRC, "Proposed Technical Specifications Change - Supplement," dated June 19, 1989.

3. Letter from C. Thomas of USNRC to T. Schnatz of UGRA, "Acceptance for Referencing of Licensing Topical Reports EPRI CCM-5 and EPRI NP-1850-CCM," dated September 1984.

4. Letter from C. Thomas of USNRC to W. Stewart of Virginia Electric and Power Company, "Acceptance for Referring of Licensing Topical Report VEP-FRD-41, VEPCO Reactor System Transient Analysis Using the RETRAN Computer Code," dated April 11, 1985.

5. Letter from W. Stewart of VEPCO to USNRC, "Surry and North Anna Power Stations Reactor System Transient Analyses," dated November 1985.

June 30, 1989

DATED: June 30, 1989

AMENDMENT NO. 117 TO FACILITY OPERATING LICENSE NO. NPF-4-NORTH ANNA UNIT 1

Docket File
NRC & Local PDRs
PDII-2 Reading
S. Varga, 14/E/4
G. Lainas, 14/H/3
H. Berkow
D. Miller
L. Engle
OGC
D. Hagan, 3302 MNBB
E. Jordan, 3302 MNBB
B. Grimes, 9/A/2
T. Meek (4)
Wanda Jones, P-130A
J. Calvo, 11/F/23
ACRS (10)
GPA/PA
OC/LFMB
B. Sinkule, R-II

DF01
1/1