March 25, 1993

Docket Nos. 50-338
and 50-339

DISTRIBUTION
See attached sheet

Mr. W. L. Stewart
Senior Vice President - Nuclear
Virginia Electric and Power Company
5000 Dominion Blvd.
Glen Allen, Virginia 23060

Dear Mr. Stewart:

SUBJECT: 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)

The Commission has issued the enclosed Amendment Nos. 170 and 149 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 December 27, 1991.

The amendments revise the pressure/temperature (P/T) operating limitations during heatup and cooldown and the low temperature/overpressure protection systems (LTOPS) setpoints for 12 and 17 effective full power years for NA-1&2, respectively.

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. 170 to NPF-4
2. Amendment No. 149 to NPF-7
3. Safety Evaluation

cc w/enclosures:
See next page

Document Name - NA83154.AMD

OFC	:LA:PDII-2	:PM:PDII-2	:D:PDII-2	:OGC	:	:
NAME	:E. Tana ETT	:L. Engle	:H. Berkow	:	:	:
DATE	: 3/5/93	: 3/8/93	: 3/8/93	: 3/18/93	:	:

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Mr. W. L. Stewart
Virginia Electric & Power Company

North Anna Power Station
Units 1 and 2

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DATED: March 25, 1993

AMENDMENT NO. 170 TO FACILITY OPERATING LICENSE NO. NPF-4-NORTH ANNA UNIT 1
AMENDMENT NO. 149 TO FACILITY OPERATING LICENSE NO. NPF-7-NORTH ANNA UNIT 2

Docket File
NRC & Local PDRs
PDII-2 Reading
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G. Lainas, 14/H/3
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OGC
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Wanda Jones, MNBB-7103
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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. 170
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 December 27, 1991, 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. 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 by August 17, 1993 or 10 effective full power years, whichever occurs first.

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: March 25, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 170

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
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 4-26	3/4 4-26
3/4 4-27	3/4 4-27
3/4 4-28	3/4 4-28
3/4 4-31	3/4 4-31
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
B 3/4 4-1	B 3/4 4-1
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
B 3/4 5-2	B 3/4 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% Δk/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 316°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 316°F or prior to cooldown below 316°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 316°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 316°F.

SHUTDOWN

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 316°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.

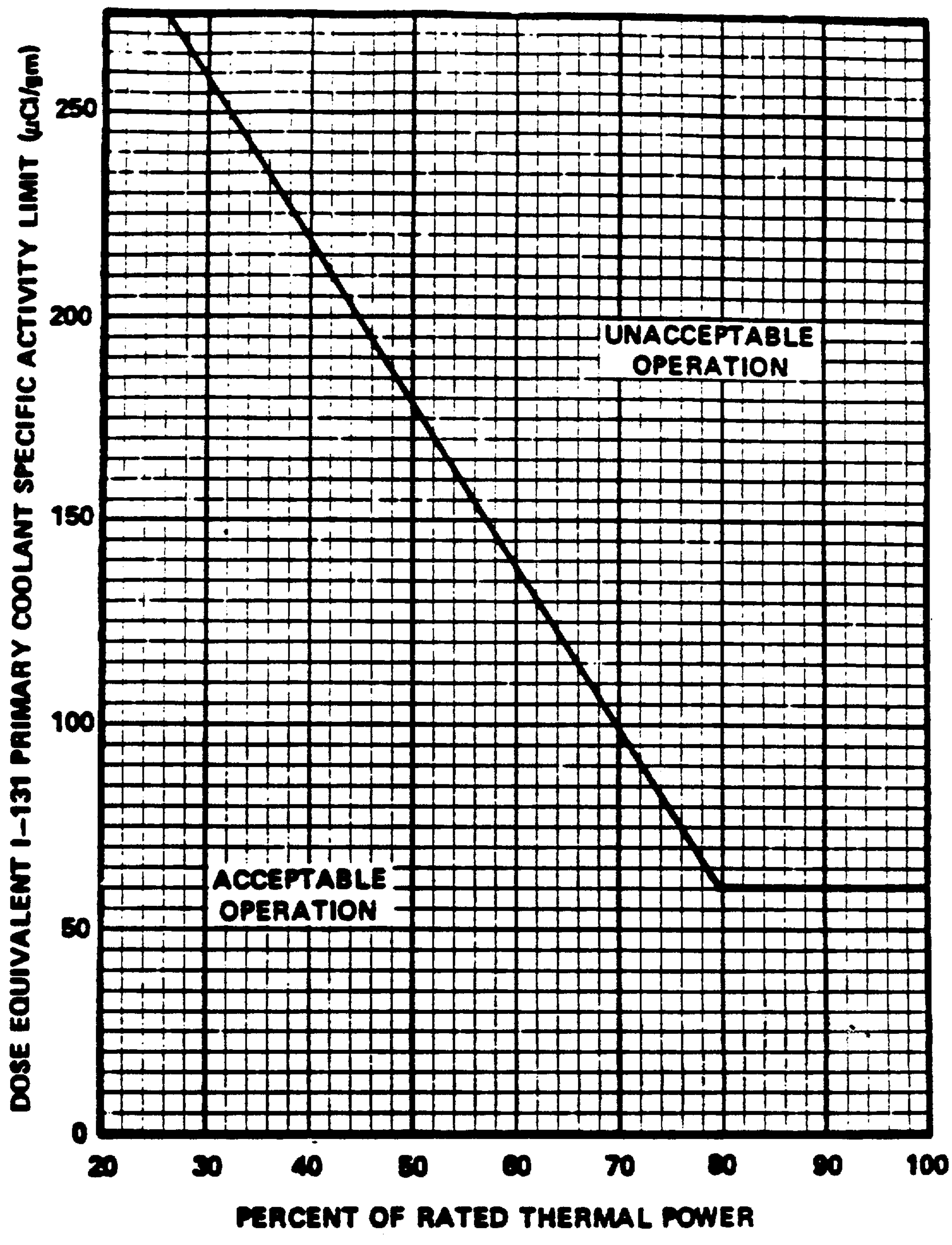


FIGURE 3.4-1

DOSE EQUIVALENT I–131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity > 1.0 μCi/gram Dose Equivalent I-131

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown in Figures 3.4-2 and 3.4-3 during heatup, cooldown, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 60°F in any one hour period.

b. A maximum cooldown of 100°F in any one hour period.

c. A maximum temperature change of less than or equal to 10°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS Tavg and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown and inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR 50, Appendix H. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3.

Figure 3.4-2 Unit 1 RCS HEATUP P/T Limits
Valid to 12 EFPY Heatup Rates: 0-60°F/Hr.
(Margins for Instrument Errors NOT Included)

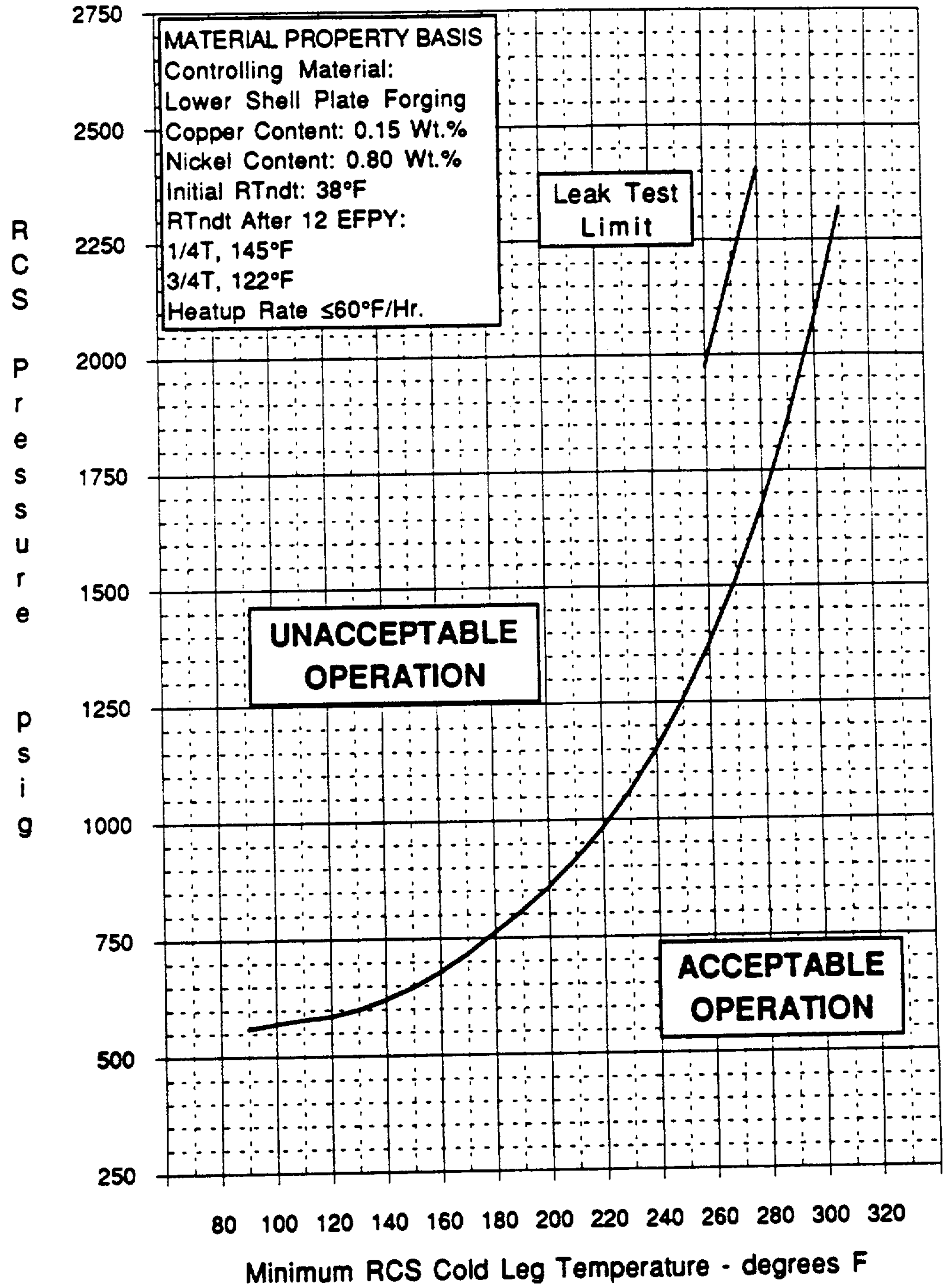
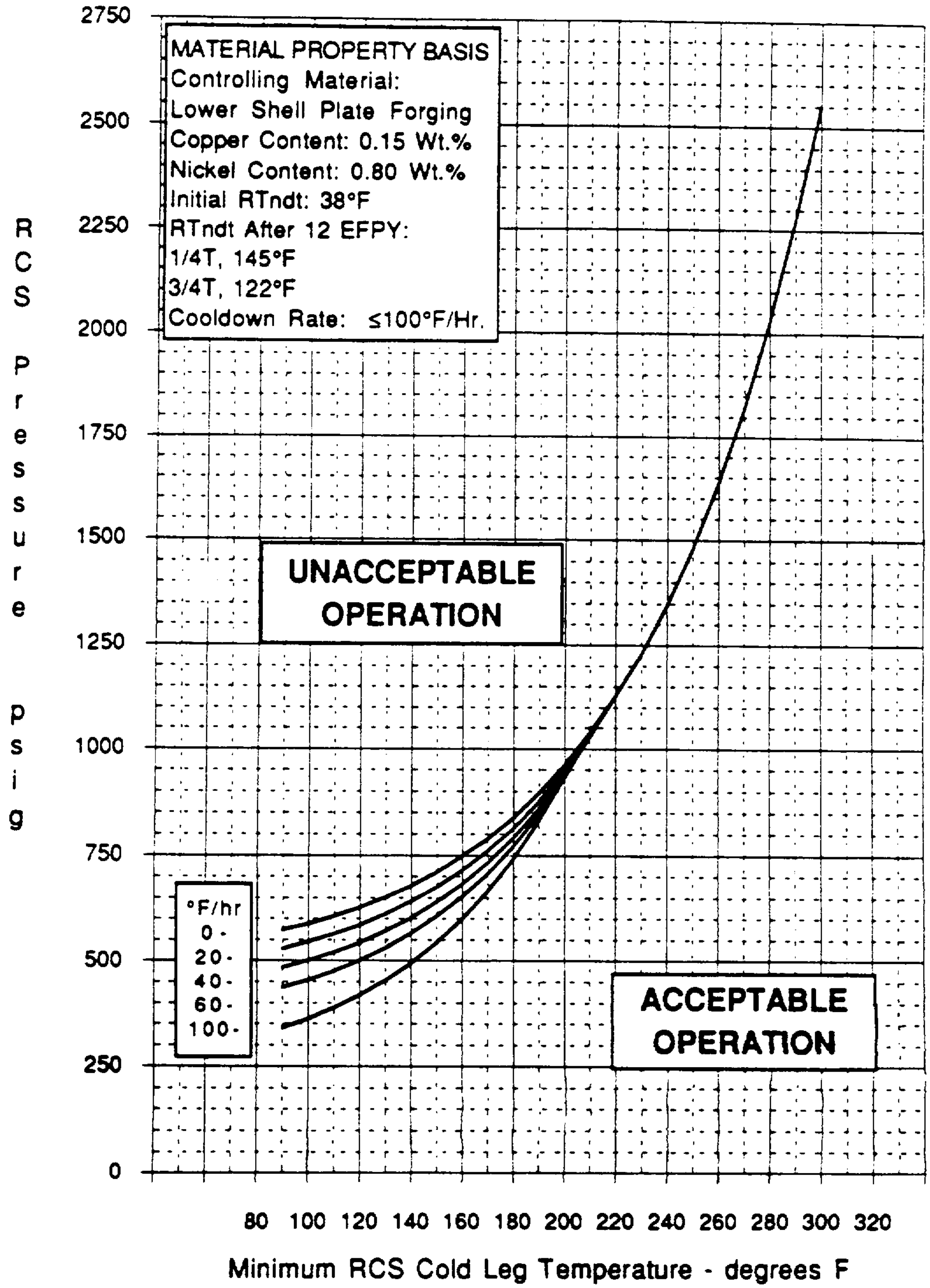


Figure 3.4-3 Unit 1 RCS COOLDOWN P/T Limits
Valid to 12 EFPY Cooldown Rates: 0-100°F/Hr.
(Margins for Instrument Errors NOT Included)



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 270°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 270°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.

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 Specifications 3.0.4 are not applicable to 3.5.2.a and 3.5.2.b for one hour following heatup above 316°F or prior to cooldown below 316°F.

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* Adherence to ACTION "a" shall require the following equipment OPERABILITY for the period of operation until steam generator replacement:

- With one low head safety injection pump inoperable, two centrifugal charging pumps (one in each subsystem) and their associated flow paths shall be OPERABLE or be in HOT STANDBY within the next 6 hours, and be in HOT SHUTDOWN within the next 6 hours.

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.

EMERGENCY CORE COOLING SYSTEMS

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 Tavg 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 316°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 316°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% $\Delta k/k$ 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 316°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% $\Delta k/k$ 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.

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 a 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 316°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 staring 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 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 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 phenomenon which may occur following changes in THERMAL POWER.

Reducing Tavg 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 are prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of 12 EFPY. The adjusted reference temperature was calculated using results from a capsule removed after the sixth fuel cycle. The results are documented in Westinghouse Report WCAP-11777, February 1988 and Babcock and Wilcox Report BAW-2146, October, 1991.

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 1.98, 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 12 EFPY. The reactor vessel beltline region material properties are listed in Table B.3.4-1.

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 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}+90°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.

Table B.3.4-1

MATERIAL PROPERTY BASIS

Controlling Material:	Lower Shell Plate Forging
Copper Content:	0.15 Wt.%
Nickel Content:	0.80 Wt.%
Initial RTndt:	38°F
RTndt After 12 EFPY:	1/4T, 145°F
	3/4T, 122°F
Cooldown Rate:	≤100°F/Hr.
Heatup Rate:	≤60°F/Hr.

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 316°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. 20555-0001

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. 149
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 December 27, 1991, 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. 149 , 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 by August 1, 1993, or 10 effective full power years, whichever occurs first.

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: March 25, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 149

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.

Remove Pages	Insert Pages
3/4 1-9	3/4 1-9
3/4 1-12	3/4 1-12
3/4 4-2	3/4 4-2
3/4 4-3	3/4 4-3
3/4 4-26	3/4 4-26
3/4 4-27	3/4 4-27
3/4 4-28	3/4 4-28
3/4 4-30	3/4 4-30
3/4 5-3	3/4 5-3
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 4-6	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
B 3/4 4-9 thru B 3/4 4-16	B 3/4 4-9
B 3/4 5-2	B 3/4 5-2

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three 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 358°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 358°F or prior to cooldown below 358°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 358°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 358°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.

HOT STANDBY

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 with in 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.

* A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to 358°F unless the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

** 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).

SHUTDOWN

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 358°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.

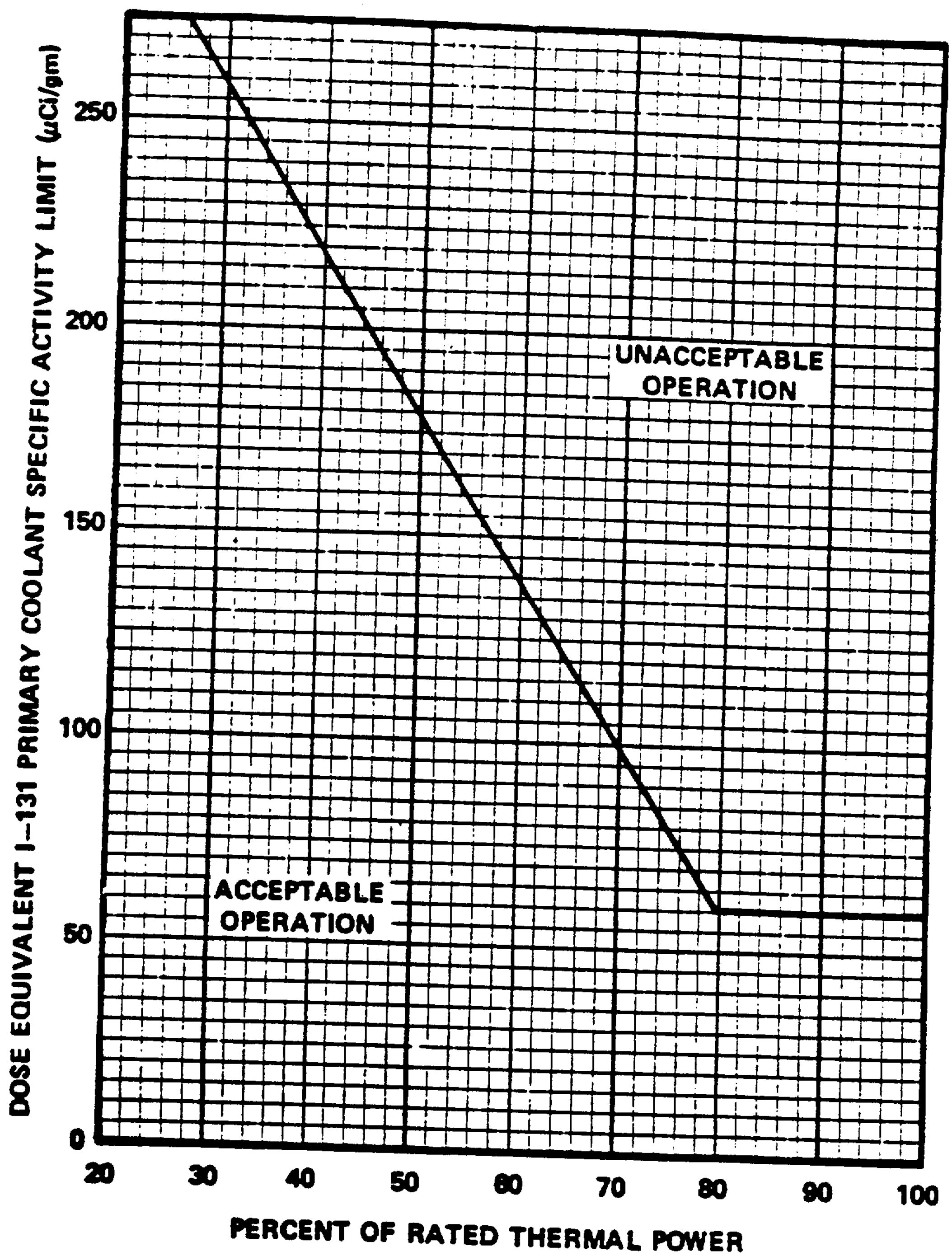


FIGURE 3.4-1

DOSE EQUIVALENT I–131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity > 1.0 μCi/gram Dose Equivalent I-131

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown in Figures 3.4-2 and 3.4-3 during heatup, cooldown, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 60°F in any one hour period.

b. A maximum cooldown of 100°F in any one hour period.

c. A maximum temperature change of less than or equal to 10°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS Tavg and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown and inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR 50, Appendix H. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3.

Figure 3.4-2 Unit 2 RCS HEATUP P/T Limits
Valid to 17 EFPY Heatup Rates: 0-60°F/Hr.
(Margins for Instrument Errors NOT Included)

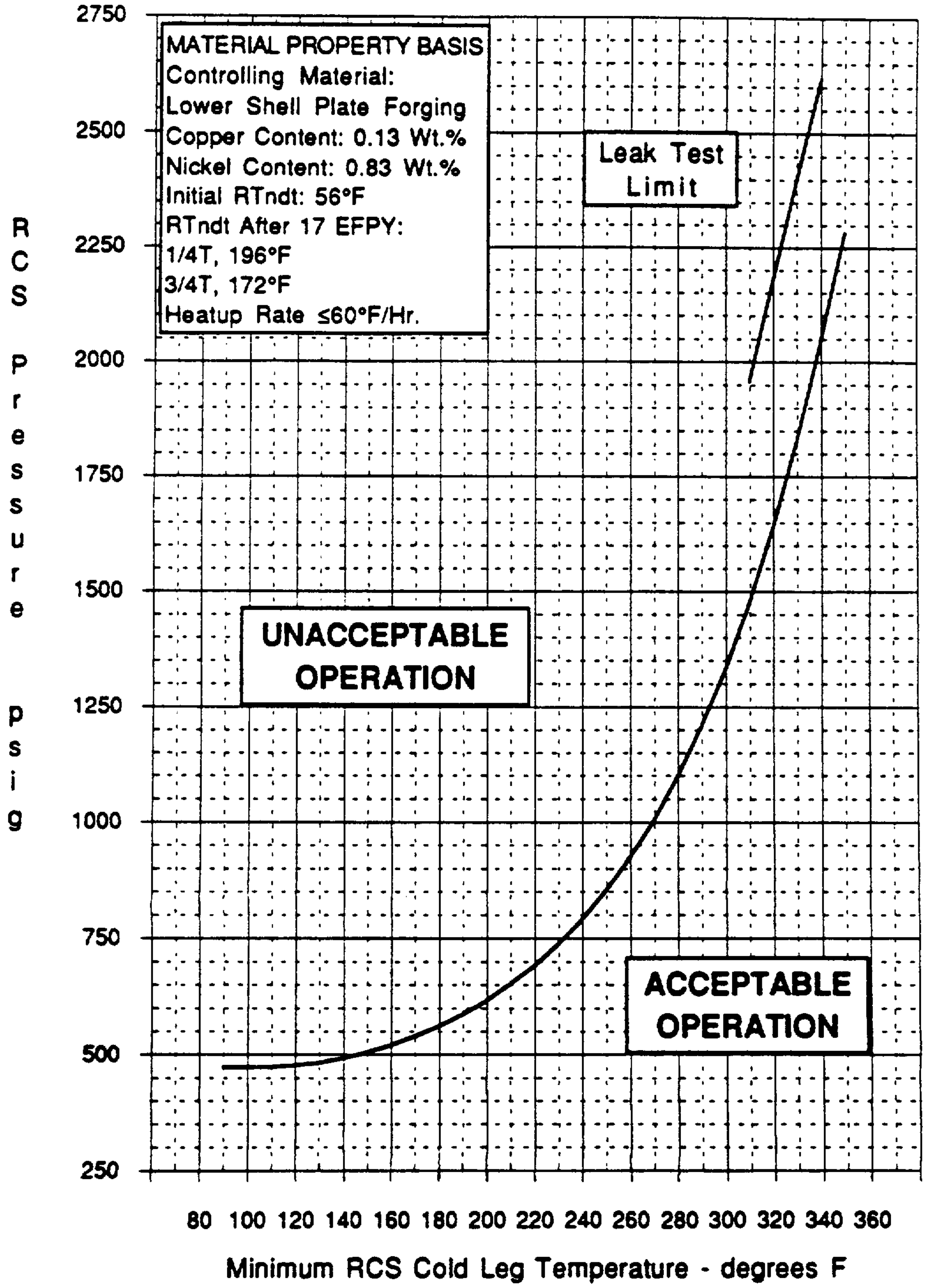
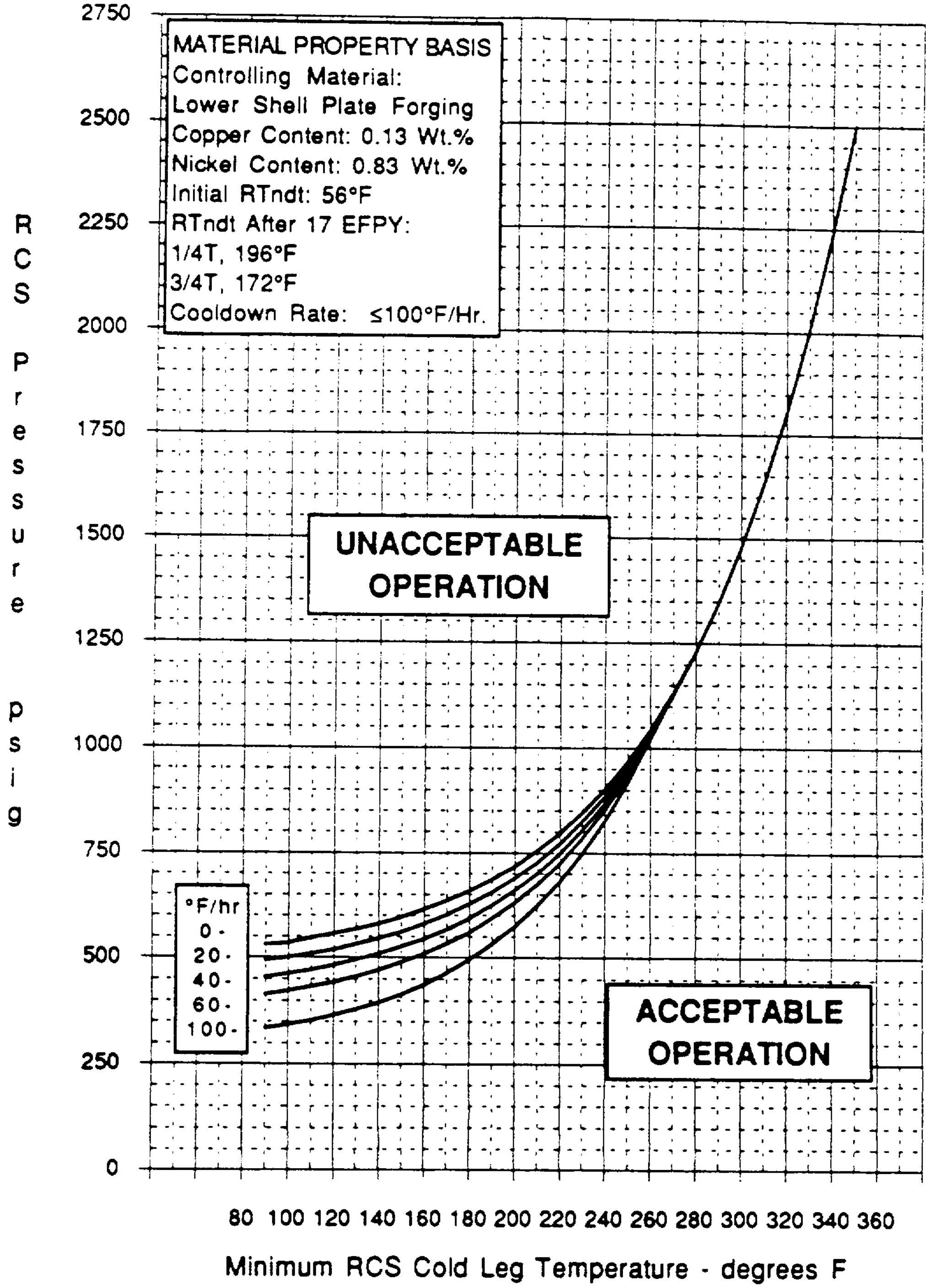


Figure 3.4-3 Unit 2 RCS COOLDOWN P/T Limits
Valid to 17 EFPY Cooldown Rates: 0-100°F/Hr.
(Margins for Instrument Errors NOT Included)



REACTOR COOLANT SYSTEM

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.

REACTOR COOLANT SYSTEM

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 510 psig whenever any RCS cold leg temperature is less than or equal to 321°F, and 2) less than or equal to 360 psig whenever any RCS cold leg temperature is less than 210°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 321°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.

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 Specifications 3.5.2.a and 3.5.2.b for one hour following heatup above 358°F or prior to cooldown below 358°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:

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 358°F.

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.

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 greater than or equal to 2410 psig.

   2. Low head safety injection pump greater than or equal to 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.	2-SI-89	Loop A Cold Leg
2.	2-SI-97	Loop B Cold Leg
3.	2-SI-103	Loop C Cold Leg
4.	2-SI-116	Loop A Hot Leg
5.	2-SI-111	Loop B Hot Leg
6.	2-SI-123	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.

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS - Tavg LESS THAN 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.

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 Tavg 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 358°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 358°F by verifying that the control switch is in the pull to lock position.

EMERGENCY CORE COOLING SYSTEMS

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 358°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 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 a 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 358°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 staring 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 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 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 (PORVs) 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.

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 phenomenon which may occur following changes in THERMAL POWER.

Reducing Tavg 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

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.

BASES

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 are prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of 17 EFPY. The adjusted reference temperature was calculated using results from a capsule removed after the sixth fuel cycle. The results are documented in Westinghouse Report WCAP-12497, January 1990 and WCAP-12503, March, 1990.

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-12497. 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 1.98, 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 in Table B.3.4-1.

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-12497 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 321°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 321°F. This temperature is the water temperature corresponding to a metal temperature of at least the limiting RT_{NDT}+90°F+ instrument uncertainty. Above 321°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.

Table B.3.4-1

MATERIAL PROPERTY BASIS

Controlling Material:	Lower Shell Plate Forging
Copper Content:	0.13 Wt.%
Nickel Content:	0.83 Wt.%
Initial RTndt:	56°F
RTndt After 17 EFPY:	1/4T, 196°F
	3/4T, 172°F
Cooldown Rate:	≤100°F/Hr.
Heatup Rate:	≤60°F/Hr.

3/4.4.10 STRUCTURAL INTEGRITY

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

The inspection program 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.

Pages B 3/4 4-9 thru B 3/4 4-16 have been deleted.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

BASES

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 358°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. Surveillance requirements for throttle valve position stops and flow balance testing provide assurance that proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS - LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS - LOCA analyses.

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 contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

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. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 170 AND 149 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 December 27, 1991, the Virginia Electric and Power Company (the licensee) requested permission to revise the pressure/temperature (P/T) limits in the North Anna Power Station, Unit 1 and Unit 2 (NA-1&2) Technical Specifications (TS), Section 3/4. The proposed P/T limits were requested for 12 effective full power years (EFPY) for NA-1, and 17 EFPY for NA-2. The proposed P/T limits were developed using Regulatory Guide (RG) 1.99, Revision 2. Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Effect on Plant Operations," recommends RG 1.99, Rev. 2, be used in calculating P/T limits, unless the use of different methods can be justified. The P/T limits provide for the operation of the reactor coolant system during heatup, cooldown, criticality, and hydrotest.

To evaluate the P/T limits, the staff uses the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; the ASTM Standards and the ASME Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); RG 1.99, Rev. 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 of 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits. An acceptable method for constructing the P/T limits is described in SRP Section 5.3.2.

Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the extent 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 RG 1.99, Rev. 2, to predict the effects of the neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H of 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 made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline.

The proposed changes incorporate operability requirements for the low temperature overpressure protection system (LTOP) system. TS 3.1.2.2, 3.1.2.4, 3.4.1.2, 3.4.1.3, 3.5.2 and 3.5.3 were modified to reflect the new limits on the number of operable centrifugal charging pumps in Mode 4, minimizing the potential for and severity of low-temperature overpressure transients. Specification 3.4.9.3 was modified to reflect the revised LTOP setpoints. All TS changes apply to both units with the exception of TS 3.4.1.2, which is only applicable to NA-2.

The NA-1&2 reactor coolant systems (RCS), particularly the reactor pressure vessels (RPV), are protected from material failure during low temperature operations by imposing restrictions on RCS pressure. The licensee's analysis indicates that one power-operated relief valve (PORV) is sufficient to mitigate the pressure excursions; however, the use of two PORVs provides redundant protection against such accidents.

The heatup and cooldown curves as well as the LTOP setpoints provide restrictions to bound the area of operation. Currently, the LTOP setpoints are valid through 10 effective full power years (EFPY), which NA-1 is expected to reach on August 17, 1993 and NA-2 on August 1, 1993.

The staff's discussion and evaluation of the proposed P/T limits and LTOP setpoints is provided below.

2.0 DISCUSSION

2.1 P/T LIMITS

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the NA-1&2 reactor vessels. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2.

For NA-1, the staff has determined that the material with the highest ART at 12 EFPY were the welds with 0.09% copper (Cu), 0.11% nickel (Ni), and an intial RT_{ndt} of 19°F. For the limiting beltline material and the welds, the staff calculated the ART to be 140.6°F at 1/4T (T = reactor vessel beltline thickness) and 116.2°F for 3/4T at 12 EFPY. The staff used a neutron fluence of 1.02E19 n/cm^2 at 1/4T and 3.97E18 n/cm^2 at 3/4T. The ART was determined by the least squares extrapolation method using the NA-1 surveillance data. The least squares method is described in Section 2.1 of RG 1.99, Rev. 2. The licensee calculated an ART of 145°F at 1/4T and an ART of 122°F at 3/4T for the lower shell forging. The staff judges that the licensee's ART of 145°F is more conservative than the staff's ART of 140.6°F, and it is acceptable. Substituting the ART of 140.6°F, into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

For NA-2, the staff has determined that the material with the highest ART at 17 EFPY was the lower shell forging with 0.13% copper (Cu), 0.83% nickel (Ni), and an initial RT_{ndt} of 56°F. For the limiting beltline material and the lower shell forging, the staff calculated the ART to be 196.4°F at 1/4T and 171.8°F for 3/4T at 17 EFPY. The staff used a neutron fluence of 1.48E19 n/cm^2 at 1/4T and 5.9E18 n/cm^2 at 3/4T. The ART was determined by the least squares extrapolation method using the NA-2 surveillance data. The licensee calculated an ART of 196°F at 1/4T and an ART of 172°F at 3/4T for the lower shell forging. The staff judges that a difference of 0.4° between the licensee's ART of 196°F and the staff's ART of 196.4°F is acceptable. Substituting the ART of 196.4°F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.A.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions 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.A.2 of Appendix G.

Section IV.A.1 of Appendix G requires that the predicted Charpy USE at end of life be above 50 ft-lb. The material with the highest Cu content and the lowest initial transverse USE for NA-1 is the lower shell forging with 85 ft-lb. Using RG 1.99, Rev. 2, the staff calculated that the end of life

(EOL) USE will be 52.2 ft-lb. The material with the highest Cu content and the lowest initial transverse USE for NA-2 is the intermediate shell forging with 74 ft-lb. Using RG 1.99, Rev. 2, the staff calculated that the EOL USE will be 50.8 ft-lb. These values are greater than 50 ft-lb and, therefore, are acceptable.

The licensee has removed two surveillance capsules each from NA-1&2. The results from NA-1 capsule V were published in Babcock and Wilcox Report BAW-1638, and the results from capsule U were published in Westinghouse Report WCAP-11777. The results from NA-2 capsule V were published in Babcock and Wilcox Report BAW-1794, and the results from capsule U were published in Westinghouse Report WCAP-12497. All surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

2.2 LTOP LIMITS

The PORVs on the pressurizer are set at a pressure low enough to prevent violation of the composite modified heatup and cooldown curve should a pressure transient occur. The limits have been set by two design basis accidents: (1) the inadvertent start of a charging pump and (2) the startup of a reactor coolant pump in an RCS loop with a 50°F (10°C) temperature difference between the steam generator secondary fluid and the RCS. These transients represent the limiting mass addition and heat input transients with the RCS solid and were selected based on the previous NA-1&2 Updated Final Safety Analysis Report (UFSAR).

The LTOP setpoints were determined so that a pressure overshoot allowance exists to prevent exceeding the composite modified heatup and cooldown curve during a mass addition transient. The licensee indicated this allowance was necessary because of the valve opening characteristic associated with the air operated relief valves on the NA-1&2 pressurizer.

The inadvertent operation of a single charging pump was modeled by varying initial reactor coolant temperature, pressure and PORV setpoint to observe the effects of changes in these parameters. A range of RCS temperatures between 100 and 325°F (38 and 163°C) were examined, as well as a range of initial pressures. The results indicated a gradual decrease in the PORV setpoint pressure overshoot as the initial RCS temperature and PORV setpoint increase. Also, the peak RCS pressure was found to be relatively insensitive to the initial RCS pressure.

The licensee analyzed the heat addition transient by modelling the secondary to primary heat transfer with a conservative local secondary side convection heat transfer coefficient and an assumed constant bulk secondary side temperature. The pump startup flow characteristic was also modelled in a conservative manner. The analysis indicated that the results of the heat addition transient are bounded by those of the mass addition transient.

2.3 UNCERTAINTY

The licensee developed the new setpoints using a plant-specific RETRAN model. It was reasoned that measurement uncertainties be excluded based on the uncertainties being insignificant when compared to the margin terms included in the ASME Section III Appendix G methods.

The licensee excluded instrumentation uncertainties on the basis of their insignificance relative to the conservatism of stress intensity factors.

The temperature measurement uncertainty was considered in the development of the minimum LTOP enabling temperatures. Automatic low temperature overpressurization protection is required whenever any RCS cold leg temperature is less than 270°F (132°C) [321°F (161°C) for NA-2]. The enabling temperature consists of $RT_{NDT}+\Delta T+90°F+$instrument uncertainty. For NA-1, the 12 EFPY RT_{NDT} temperature is 145°F (63°C) for ¼T and 122°F (50°C) for ¾T. For NA-2, the 17 EFPY RT_{NDT} temperature is 196°F (91°C) for ¼T and 172°F (78°C) for ¾T. The ΔT is the maximum temperature difference between the water and metal. An instrument uncertainty of 20°F (-6.7°C) was added. The licensee concluded that a 90°F (32°F) addition is a reasonable range to require the automatic low temperature overpressure protection and sufficient for automatic protection during startup and shutdown.

The proposed NA-1 setpoints are $\leq$ 450 psig (3.1 MPa) when the RCS temperature is $\leq$ 270°F (132°C), and $\leq$ 390 (2.7 MPa) psig when the RCS temperature is $\leq$ 150°F (66°C). The proposed NA-2 setpoints are $\leq$ 510 psig (3.5 MPa) when the RCS temperature is $\leq$ 321°F (161°C), and $\leq$ 360 psig (2.5 MPa) when the RCS temperature is $\leq$ 210°F (99°C).

3.0 EVALUATION

The staff concludes that the proposed P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 12 EFPY for NA-1 and through 17 EFPY for NA-2 because the proposed limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The proposed P/T limits also satisfy Generic Letter 88-11 since the method specified in RG 1.99, Rev. 2 was used to calculate the ART.

Finally, the licensee has proposed to revise the Appendix G curves using standard B&W and Westinghouse methodologies including Regulatory Guide 1.99 Rev. 2. The PORV setpoints were developed to provide bounding heatup and cooldown curve protection for the worst case mass and heat addition low temperature overpressure transients and, therefore, are acceptable. Therefore, based on all of the above, the staff finds the proposed NA-1&2 TS changes for P/T and LTOP limits to be acceptable.

4.0 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.

5.0 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 pubic comment on such finding (57 FR 20519). 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.

6.0 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.

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