March 3, 1992

Docket No. 50-338

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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 UNIT 1 - ISSUANCE OF AMENDMENT RE: REDUCED MINIMUM REACTOR COOLANT SYSTEM FLOW RATE LIMIT (TAC NO. M82564)

The Commission has issued the enclosed Amendment No. 154 to Facility Operating License No. NPF-4 for the North Anna Power Station, Unit No. 1 (NA-1). The amendment revises the Technical Specifications (TS) in response to your letter dated January 8, 1992, as supplemented by letters dated January 31, February 10, and February 25, 1992.

This amendment revises the minimum allowable Reactor Coolant System total flow from the current value of 284,000 gpm to a value of 268,500 gpm. This revision is temporary and will remain in effect until the currently scheduled 1993 steam generator replacement. In addition, an administrative change has been made to Table 2.2-1.

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. 154 to NPF-4
2. Safety Evaluation

cc w/enclosures:
See next page

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NAME	:D. Miller	:L. Engle	:H. Berkow	: B ath	:	:
DATE	: 2/27/92	: 2/28/92	: 2/28/92	: 3/2/92	:	:

Document Name - NA1M82564.AMD

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DATED: March 3, 1992

AMENDMENT NO. 154 : TO FACILITY OPERATING LICENSE NO. NPF-4-NORTH ANNA UNIT 1

Docket File
NRC & Local PDRs
PDII-2 Reading
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G. Lainas, 14/H/3
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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. 154
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 January 8, 1992, as supplemented by letters dated January 31, February 10 and February 25, 1992, 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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 154 , are hereby incorporated in the license. VEPCO shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

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 3, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 154

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
2-1	2-1
- -	2-2a
2-6	2-6
2-9	2-9
2-10	2-10
3/4 2-15	3/4 2-15

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figures 2.1-1* for 3 loop operation and 2.1-2 and 2.1-3 for 2 loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

* For the period of operation until steam generator replacement, the combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1a.

Nominal T_{avg} = 586.8°F
Nominal RCS flow = 289200 GPM

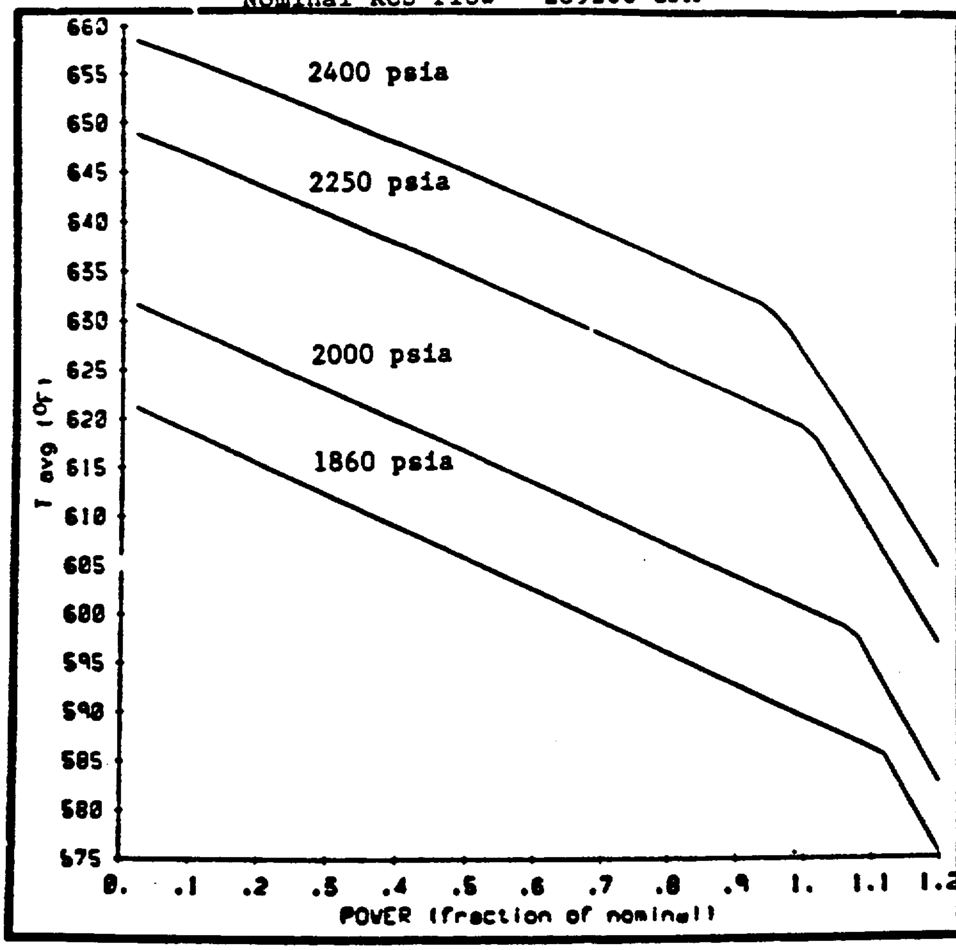

Figure 2.1-1 **REACTOR CORE SAFETY LIMITS FOR THREE LOOP OPERATION**

Nominal T_{avg} = 586.8°F

Nominal RCS flow = 268,500 GPM

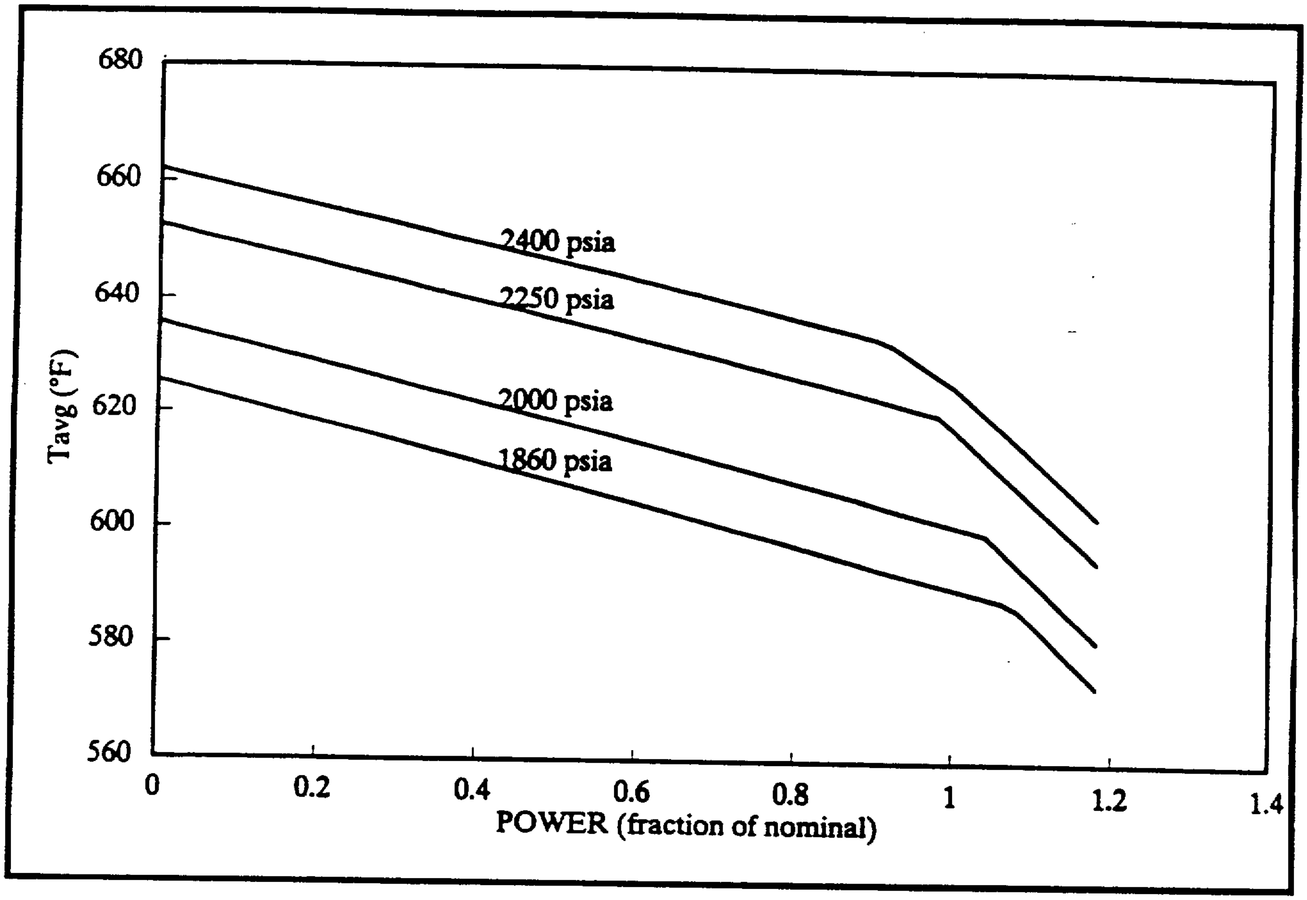

Figure 2.1-1a REACTOR CORE SAFETY LIMITS FOR THREE LOOP OPERATION FOR THE PERIOD OF OPERATION UNTIL STEAM GENERATOR REPLACEMENT

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor trip system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a reactor trip system instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - ≤ 25% of RATED THERMAL POWER High Setpoint - ≤ 109%** of RATED THERMAL POWER	Low Setpoint - ≤ 26% of RATED THERMAL POWER High Setpoint - ≤ 110%*** of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 seconds	≤ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 seconds	≤ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 seconds
5. Intermediate Range, Neutron Flux	≤ 25% of RATED THERMAL POWER	≤ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	≤ 10^5 counts per second	≤ 1.3×10^5 counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 3
9. Pressurizer Pressure--Low	≥ 1870 psig	≥ 1860 psig
10. Pressurizer Pressure--High	≤ 2385 psig	≤ 2395 psig
11. Pressurizer Water Level--High	≤ 92% of instrument span	≤ 93% of instrument span
12. Loss of Flow	≥ 90% of design flow per loop *	≥ 89% of design flow per loop *

* Design flow per loop is one-third of the minimum allowable Reactor Coolant System Total Flow Rate as specified in Table 3.2-1.

** The high trip setpoint for Power Range, Neutron Flux, shall be ≤ 103% RATED THERMAL POWER for the period of operation until steam generator replacement.

*** The allowable value for the high trip setpoint for Power Range, Neutron Flux, is required to be ≤ 104% RATED THERMAL POWER for the period of operation until steam generator replacement.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Operation with 3 Loops	Operation with 2 Loops (no loops isolated) *	Operation with 2 Loops (1 loop isolated) *
K_1 = 1.264**	K_1 = ( )	K_1 = ( )
K_2 = 0.0220	K_2 = ( )	K_2 = ( )
K_3 = 0.001152	K_3 = ( )	K_3 = ( )

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between -44 percent and +3 percent, $f_1(\Delta I) = 0$ (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) for each percent that the magnitude of $(q_t - q_b)$ exceeds -44 percent, the ΔT trip setpoint shall be automatically reduced by 1.67 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of $(q_t - q_b)$ exceeds +3 percent, the ΔT trip setpoint shall be automatically reduced by 2.00 percent of its value at RATED THERMAL POWER.

* Values dependent on NRC approval of ECCS evaluation for these operating conditions.

** The value for K_1 shall be equal to 1.132 for the period of operation until steam generator replacement.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_o \left[K_4 - K_5 \left(\frac{\tau_3 S}{1+\tau_3 S} \right) T - K_6 (T - T') - f_2(\Delta I) \right]$

Where:	ΔT_o	=	Indicated ΔT at RATED THERMAL POWER
	T	=	Average temperature, °F
	T'	=	Indicated T_{avg} at RATED THERMAL POWER $\leq$ 586.8°F
	K_4	=	1.079*
	K_5	=	0.02/°F for increasing average temperature
	K_5	=	0 for decreasing average temperatures
	K_6	=	0.00164 for $T > T'$; $K_6 = 0$ for $T \leq T'$
	$\frac{\tau_3 S}{1+\tau_3 S}$	=	The function generated by the rate lag controller for T_{avg} dynamic compensation
	τ_3	=	Time constant utilized in the rate lag controller for T_{avg} $\tau_3 = 10$ secs.
	S	=	Laplace transform operator (sec^{-1})
	$f_2(\Delta I)$	=	0 for all ΔI

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 2 percent span.

* The value for K_4 shall be equal to 1.016 for the period of operation until steam generator replacement.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS: 3 Loops in Operation	LIMITS: 2 Loops in Operation ** & Loop Stop Valves Open	LIMITS: 2 Loops in Operation ** & Isolated Loop Stop Valves Closed
Reactor Coolant System T_{avg}	$\leq$ 591°F		
Pressurizer Pressure	$\geq$ 2205 psig *		
Reactor Coolant System Total Flow Rate	$\geq$ 284,000 gpm ***		

* Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of 10% RATED THERMAL POWER.

** Values dependent on NRC approval of ECCS evaluation for these conditions.

*** The value for the minimum allowable Reactor Coolant System Total Flow Rate is reduced to 268,500 gpm until steam generator replacement.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 154

TO FACILITY OPERATING LICENSE NO. NPF-4

VIRGINIA ELECTRIC AND POWER COMPANY

OLD DOMINION ELECTRIC COOPERATIVE

NORTH ANNA POWER STATION, UNIT NO. 1

DOCKET NO. 50-338

1.0 INTRODUCTION

By letter dated January 8, 1992, as supplemented by letters dated January 31, February 10 and February 25, 1992, the Virginia Electric and Power Company (the licensee) proposed changes to the Technical Specifications (TS) for the North Anna Power Station, Unit No. 1 (NA-1). The proposed revisions would reduce the minimum allowable Reactor Coolant System (RCS) total flow rate specified in Table 3.2-1 of TS 3.2.5 from the current value of 284,000 gpm to a value of 268,500 gpm. This revision is temporary and would remain in effect until the currently scheduled 1993 steam generator replacement. Additionally, an administrative change has been proposed to Table 2.2-1 of TS 2.2.1, which would permanently revise the footnote to specify that the "Design flow per loop is one-third the minimum allowable Reactor Coolant System Total Flow Rate as specified in Table 3.2-1."

The January 31 and February 25, 1992 letters provided additional information requested by the staff regarding the requested changes. The February 10, 1992 letter revised the minimum allowable RCS flow rate based on a revised increase in SG tube plugging (SGTP) projections. These submittals are further discussed under Section 3.0 of this SE. The additional information requested by the staff, as well as the revised minimum allowable RCS flow, did not alter the proposed action or affect the staff's initial determination of no significant hazards consideration as noticed in the Federal Register on February 5, 1992 (57 FR 4563).

The proposed reduction in RCS minimum flow rate has been prompted by inspection data obtained during a mid-cycle NA-1 SG inspection outage. These results, based on extensive eddy current inspection of the NA-1 SG tubes, together with the use of conservative analysis guidelines and plugging criteria, indicate that a substantially increased number of tubes will require plugging. The attendant reduction in RCS flow rate through the tubes will increase the likelihood that the current TS 3.2.5 requirement on minimum

RCS flow rate may be violated during continued operation with the existing SGs. The proposed decrease of approximately 3% in specified minimum RCS flow rate is intended to bound any future measurements of RCS flow (required by the TS once per fuel cycle) and any anticipated tube plugging up until the 1993 SG replacement. The 3% reduction in RCS flow rate correlates with an approximate 32% level of tube plugging.

RCS total flow rate is a critical input parameter to the analyses presented in Chapter 15 of the Updated Final Safety Analysis Report (UFSAR). Accordingly, to support operation of NA-1 with extended SGTP, the impact of the proposed reduction in flow rate on Chapter 15 analyses must be evaluated. The licensee's January 8, 1992 submittal provides a summary of these reevaluations and, in addition, provides assessments of the following: (1) whether the current engineered safety features (ESF) and reactor protection system (RPS) setpoints set forth in the TS continue to provide adequate plant protection under the reduced flow conditions due to extended SGTP, (2) whether the current core thermal limits remain bounding under the reduced flow conditions due to extended SGTP, and (3) whether the nuclear steam supply system (NSSS) and balance of plant (BOP) systems and components continue to meet the applicable acceptance criteria under the reduced flow conditions due to extended SGTP.

The staff has completed the review of the licensee's proposed TS revisions and the technical evaluations submitted to support these revisions. Our evaluation follows.

2.0 EVALUATION

The licensee has examined all of the transients addressed in Chapter 15 and has determined which ones require reanalysis and which require only reevaluation. If an event is potentially impacted by RCS flow rate and also by other effects of SGTP (e.g., increased hydraulic resistance, reduced heat transfer area, reduced RCS volume), then the event is either reanalyzed or evaluated using available sensitivity data for that specific event.

If a departure from nucleate boiling ratio (DNBR)-limited event is impacted by RCS flow rate but not by other effects of SGTP, a DNBR penalty is assessed, as described below. Finally, events which are unaffected by RCS flow rate but are impacted by other effects of SGTP, and those events which are impacted by neither flow nor other effects, have not been addressed further. These events have no bearing on the requested TS revisions.

The DNBR design limit and the statistical DNBR limit (SDL) for NA-1 are 1.46 and 1.26, respectively. The SDL is based on the WRB-1 CHF correlation with DNBR parameter and correlation uncertainties combined in a statistical manner. The minimum DNBR value computed as part of a transient analysis is assessed against the design limit. The percentage difference between the computed DNBR value and the design limit is termed the "DNBR analysis margin." The "generic retained margin" is defined as the percentage difference between the design limit and the SDL (i.e., 13.7%). As noted above, a DNBR penalty is assessed for certain Chapter 15 events to compensate for adverse effects on DNBR due to the proposed reduction in RCS flow rate. The penalty, which is the same for all these events, is assessed against the generic retained margin without taking credit for the analysis margin. The magnitude of the penalty (4.8%) is computed as the product of the bounding DNBR partial derivative with respect to RCS flow rate (1.6), and the proposed percent reduction in flow (3%). The value of this partial derivative was based on the WRB-1 CHF correlation for DNBR and determined by considering a wide range of statepoint conditions which bounded both normal operation and accident conditions. The penalty is directly subtracted from the 13.7% generic retained margin, leaving a margin balance of 8.9% against which other penalties such as, for example, the effects of fuel rod bowing, can be assessed. The Chapter 15 transients for which the 4.8% penalty has been assessed are the following:

- Accidental depressurization of the RCS
- Accidental depressurization of the Main Steam System
- Excessive load increase
- Excessive heat removal (feedwater malfunction)
- Partial loss of RCS flow
- Rod withdrawal from subcritical
- Spurious operation of Safety Injection System
- Single rod withdrawal at power

The disposition of the remaining Chapter 15 transients which were reevaluated but not reanalyzed are as follows:

1) For the main steamline break (MSLB) event, a different penalty (4.3%) was assessed in an analogous manner against a different retained margin (10%) because the MSLB analysis employs another correlation (W-3 CHF rather than WRB-1 CHF) for DNBR calculations. The correlation has a different DNBR sensitivity to change in RCS flow rate and therefore yields a different value (1.4) for the bounding partial derivative.

2) For the control rod drop/misalignment event, revised DNBR limit lines applicable to reduced RCS flow rates have been developed for this transient for application to reload cores. As such, no penalty need be assessed against generic retained margin. Although the negative flux rate trip has been retained at NA, the dropped rod analysis methodology employed does not take credit for this trip.

3) The Chemical Volume Control System (CVCS) malfunction (boron dilution) event is not affected by a reduction in RCS flow rate but is, however, sensitive to the reduction in RCS volume resulting from SGTP. Therefore, this event has no impact on the proposed TS revisions regarding RCS flow rate.

4) The small break loss of coolant accident (LOCA) transient has been demonstrated to be insensitive to marginal changes in RCS flow rate. The critical consideration in conservatively treating the reactor coolant pumps (RCP) for this event is the time during the event when these pumps are tripped rather than their steady-state flow rate. The analysis of record, therefore, remains valid for operation at the proposed reduced flow rate.

5) The DNBR impact of reduced RCS flow rate on the minor secondary steam pipe break transient is bounded by the MSLB event discussed above.

6) The rupture of a main feedwater pipe (main feedline break) event can result in either RCS cooldown or heatup, depending on break size and operating conditions at the time the break occurs. The bounding cooldown scenario for a secondary system pipe rupture is the MSLB transient addressed above. Regarding RCS heatup, the consequences of a feedline break upstream of the feedline check valve are bounded by the consequences of the loss of normal feedwater event. This is one of the events that has been reanalyzed by the licensee and will be discussed below. A break downstream of the check valve, however, may also result in a loss of SG secondary side inventory and can prevent auxiliary feedwater addition to the affected generator. This scenario is not DNB-limited but, rather, is limited by RCS subcooling margin. For the analysis of record, this margin is 36°F. Existing sensitivity data for changes in RCS flow rate and SGTP level has indicated that a 5% reduction in RCS flow rate for this event results in a reduction in subcooling margin of less than 10°F. The impact of the proposed 3% reduction in RCS flow rate, therefore, should be readily accommodated by the existing subcooling analysis margin.

7) For the control rod ejection event, existing sensitivity data indicates that a 5% reduction in RCS flow rate results in a 36°F increase in peak cladding temperature (PCT). When this increase is applied to the most limiting value of 2575°F (for the zero power, end-of-life case), an 89°F margin to the clad embrittlement temperature of 2700°F still remains. Therefore, the impact of the proposed 3% reduction in flow rate on PCT should be accommodated by this margin.

8) In the analysis of the SG tube rupture event, operator action is assumed to terminate the primary-to-secondary side mass transfer within 30 minutes and bounding values of key parameters are assumed in the calculation of consequences. These assumptions are not impacted by a reduction in RCS flow rate and, therefore, the analysis of record remains valid.

9) The requirements imposed by 10 CFR Part 50 Appendix K with regard to RCP for purposes of analysis of the large break LOCA event result in a relative insensitivity of the event to marginal changes in RCS flow rate. Reduced flow rates cause a decrease in core inlet temperatures with an attendant decrease in the RCS Tavg. Existing sensitivity data shows that, for the proposed decrease in RCS flow rate, the corresponding reduction in inlet temperatures results in an increase in PCT of approximately 2° F. Considering the conservatisms dictated by Appendix K, this increase is deemed insignificant. Therefore, with regard to the proposed TS revisions, the analysis of record remains valid.

10) Those Chapter 15 events which are unaffected by reduced RCS flow rate or other effects of SGTP have been identified as:

- Inactive loop startup from reduced power
- Misloaded fuel assembly
- Volume control tank rupture
- Waste gas decay tank rupture
- Fuel handling accident outside containment
- Fuel handling accident inside containment

As noted earlier, those Chapter 15 transients which are potentially impacted by both RCS flow rate reductions and other effects of SGTP have either been reanalyzed or evaluated using existing event-specific sensitivity data. The latter category of events has already been addressed above. The disposition of the reanalyzed events follows. All reanalyses were performed using the RETRAN single- and double-loop models and all assumptions were consistent with or conservative with respect to the assumption employed in the analyses of record. Modifications were made to the models to reflect the various effects of the extended SGTP.

1) For the loss of external load event, the BOC "With Pressure Control" case and the BOC "Without Pressure Control" case represent the limiting cases for DNB and overpressurization, respectively. These cases were reanalyzed. Results indicated that the DNBR increased throughout the transient from an initial value of 2.15 (for the DNB-limited case). In addition, the peak RCS and secondary pressures remained below their respective acceptance criteria values for the overpressure-limited case.

2) For the loss of normal feedwater event, both the case of "Loss of Offsite Power" and the case of "Offsite Power Available" were reanalyzed (the latter being more limiting). In both cases, results indicated that extended SGTP does not adversely impact the ability of the auxiliary feedwater system to adequately perform its safety function.

3) For the rod withdrawal at power event, a wide range of initial plant conditions were reanalyzed to identify the most limiting cases. Results indicated that, in all cases, the minimum DNBR remained above the design limit value. Additionally, the reanalysis confirmed that the current TS setpoints for overtemperature and overpower delta-T (OT delta-T and OP delta-T) trip continue to provide bounding core thermal limit protection under extended SGTP conditions.

4) For the complete loss of flow event, two cases were reanalyzed: complete loss of voltage at the RCP breakers (the "undervoltage case"), and the more limiting 5.0 Hz/sec decay in supply frequency (the "underfrequency case"). Results indicate that, in both cases, the transient DNBR remained above the SDL DNBR at all times.

5) For the locked rotor/sheared shaft event, only the locked rotor case has been reanalyzed, since previous analyses have shown this case to be bounding. The analysis consists of two parts: (1) calculation of peak

RCS pressure (assuming no fuel rods experience DNB), and (2) a conservative determination of the fraction of the core experiencing DNB. For the former case, results indicate that peak RCS pressure remains well within the acceptance limit of 2750 psia. For the latter case, the criterion of less than 13% of fuel rods experiencing DNB at the limiting time in core life (for the current operating cycle) continues to be met.

The core thermal limit lines of TS Figure 2.1-1 define the range of acceptable operating conditions which satisfy two important limits: DNBR and vessel exit boiling. The DNBR-limited segment of each line bounds the design DNBR limit of 1.46 and is based on a minimum measured flow of 289,200 gpm. As discussed earlier, the generic retained margin is large enough to absorb the DNBR penalty assessed against it and to compensate for the proposed reduction in RCS flow without taking credit for the analysis margin. Therefore, the existing DNBR portions of the limit lines continue to remain bounding. The vessel exit boiling-limited segments of the limit lines shown in Figure 2.1-1 are based on an RCS flow rate of 278,400 gpm. The proposed reduction in flow rate results in less than a 1° F reduction in subcooling margin. The licensee states that the existing vessel exit boiling limits contain sufficient margin to offset this impact and, therefore, the existing limit lines continue to remain bounding. These conclusions are based on the current TS values for OT (delta-T) and OP (delta-T).

The licensee has performed evaluations of key NSSS and BOP systems and components to confirm whether their operation under the proposed reduced flow and extended SGTP conditions remain in compliance with the applicable codes and standards. Results of these evaluations indicate continued acceptable performance for all NSSS and BOP systems/components considered.

3.0 SUPPLEMENTAL EVALUATION

As SG tube inspections progressed during the current outage, it was found that the original tube plugging projections (on which the above TS change to reduce minimum total RCS flow to 275,300 gpm was based), needed to be modified. In particular, updated projections indicated that the plugging level for SG "C" may exceed 30%. To compensate for the effects of this anticipated increase in tube plugging, the licensee proposed to limit the maximum reactor power level to 95% of rated thermal power until the planned 1993 SG replacement. This revision, requested by letter dated January 28, 1992, was necessary because the analysis of record for the large break LOCA event would not support 100% power operation with more than 30% SGTP. (The proposed change and revised large break LOCA analysis are being reviewed under separate cover). To provide the required margin, the event was reanalyzed assuming a reduced power level of 95%, a 35% level of SGTP, and an RCS flow rate of 264,400 gpm. Note, however, that no additional reduction in RCS flow rate (from 275,300 gpm) was requested in the January 28, 1992 submittal. Following that request, the licensee proposed an additional 2 1/2% reduction in minimum total RCS flow rate to 268,500 gpm to account for the uncertainties inherent in RCS flow measurement and in plugging estimates. This further revision to TS Table 3.2-1 was requested by letter dated February 10, 1992. Following a meeting with the licensee on February 10, 1992 and several subsequent telephone conferences, additional information of a clarifying nature was submitted by letter dated February 25, 1992.

To support the additional reduction in RCS flow rate, an accompanying revision in the reactor core safety limits was requested. The revision would add a footnote to TS 2.1.1 referencing new Figure 2.1-1a in lieu of existing Figure 2.1-1. The newly generated figure presents revised thermal limit lines which are based on the proposed flow rate of 268,500 gpm and which bound the existing design DNBR limit of 1.46. These limits would be in effect for the period until the SGs are replaced.

Based on the revised core thermal limits, new setpoint values for the OT (delta-T) and OP (delta-T) trips and the power range neutron flux (PRNF) high trip, as specified in TS Table 2.2-1, were generated. These values would also apply until SG replacement. Specifically, the PRNF high trip setpoint is decreased from 109% to 103% thermal power, while the allowable value is reduced from 110% to 104%. For the OT (delta-T) setpoint, the value of K_1 is reduced from 1.264 to 1.132 and for the OP (delta-T) setpoint, the value of K_4 is reduced from 1.079 to 1.016. The following describes the staff's evaluation of the additional TS revisions requested by the licensee's February 10, 1992 letter and the technical bases submitted in support of these revisions.

As noted in Section 2.0 above, for the group of Chapter 15 transients affected by RCS flow only, the adverse DNBR impact of a reduction in RCS flow to 275,300 gpm was accounted for by assessing a 4.8% DNBR penalty against the generic retained margin of 13.7%. With the proposed further reduction in RCS flow to 268,500 gpm, it was determined that for certain events, this additional reduction could not be compensated for by simply increasing the DNBR penalty. Because other factors (e.g., rod bowing) require additional penalties to be assessed against generic retained margin, the margin balance was insufficient to accommodate a greater DNBR penalty. However, the licensee's proposed TS revision to limit maximum power to 95% of rated thermal power has a favorable impact on DNBR. Sensitivity studies were performed to determine the overall DNBR impact of the 5% reduction in maximum power level, the additional reduction in RCS flow rate to 268,500 gpm, and the 1.5% increase in F (delta-H) associated with the 5% power reduction. To accomplish this, selected thermal hydraulic statepoints representing normal operating conditions and limiting accident conditions were perturbed on these three variables. The studies indicated that the favorable DNBR impact of a 5% decrease in power more than compensates for the adverse impacts of the additional RCS flow reduction and increase in F (delta-H). The net result was a DNBR benefit of 1.2% to 2.5%. Therefore, the proposed reduction in flow to 268,500 gpm, in combination with the 5% reduction in power, is acceptable with respect to this group of transients.

Concerning the group of events for which reanalyses were performed and documented in the January 8, 1992 submittal, these events were reevaluated to determine the overall impact of the additional reduction in flow from 275,300 gpm to 268,500 gpm combined with the power reduction to 95% thermal rated power. The following summarizes the results of these reevaluations.

1) For the loss of external load event, the effect of the proposed reductions is bounded by the DNBR and overpressurization reanalyses documented in the January 8, 1992 submittal.

2) For the loss of normal feedwater event, the additional reduction in RCS flow has an insignificant adverse impact on heat removal capability. This is more than compensated for by the proposed power reduction which reduces the initial stored energy and post trip decay heat. Therefore, the reanalysis of this event, as documented in the January 8, 1992 submittal, remains bounding.

3) The DNBR sensitivity studies described above indicated a net DNBR benefit resulting from the proposed reductions in power and RCS flow for a wide range of thermal-hydraulic statepoints covering normal operation and limiting accident conditions. For the complete loss of flow event, the impact of the proposed reductions was evaluated by examining their effect on RCS flow coastdown vs. time behavior. The thermal hydraulic statepoint for the revised coastdown was shown to be bounded by these sensitivities. Therefore, the reanalysis of this event, as documented in the January 8 1992 submittal, remains bounding.

4) For the locked rotor event, the proposed reductions result in a smaller number of rods experiencing DNBR than was the case for the reanalysis documented in the January 8, 1992 submittal. This is attributed to the net DNBR benefit shown to exist through the above described sensitivity analyses. Therefore, the acceptance criterion of less than 13% of fuel rods experiencing DNBR (based on the current offsite dose calculation) continues to be met. For the peak RCS pressure portion of the analysis, the locked rotor still represents the dominant source of flow resistance in the loop. Therefore, the increased resistance resulting from additional SGTP has only a minor effect on the flow coastdown vs. time behavior. This is more than offset by the favorable impact of the 5% power reduction on peak RCS pressure during the event.

5) The rod withdrawal at power event was reanalyzed at the reduced flow rate of 268,500 gpm and the revised OT (delta-T) and PRNF high trip setpoints mentioned above. In the reanalysis described in the January 8, 1992 submittal, it was noted that the limiting case corresponded to initiation of the transient from hot full power. Thus, for the revised reanalysis of this event, a range of reactivity insertion rates from an initial condition of 95% rated thermal power was examined. A comparison of the results of the revised and original (January 8, 1992) reanalysis indicates the latter is bounding. Furthermore, the results obtained for the revised case confirm that the updated thermal limits (i.e., TS Figure 2.1-1a) are not exceeded for the complete range of possible system conditions with the revised OT (delta-T) and PRNF setpoint values employed. Concerning the OP (delta-T) reactor trip setpoint, the revised value of the constant K_4 is consistent with the proposed 5% reduction in the PRNF high trip setpoint. The OP (delta-T) trip serves as a backup to the PRNF; no credit was taken for it in the Chapter 15 analyses.

For the MSLB event, as noted in Section 2.0, the adverse DNBR impact of the original proposed reduction in RCS flow rate was compensated for by assessing a DNBR penalty of 4.3% against the 10% generic retained margin. With the further reduction in RCS flow to 268,500 gpm, it was necessary to impose an additional penalty of 3.5%. It should be noted that the RCS flow rate reduction associated with SGTP and the reduction in heat transfer area resulting from SGTP can be treated as separate effects with opposite impacts. The latter has a favorable impact on the MSLB transient because it results in reduced SG heat removal capability which, in turn, leads to a decrease in RCS cooldown rate and associated power excursion. This benefit was not taken credit for in the licensee's evaluation.

The licensee has reevaluated the operation of key NSSS and BOP systems and components under the further reduced RCS flow rate and reduced thermal power limit. Using, as a baseline, the Westinghouse and Stone & Webster evaluations performed for the originally proposed flow rate reduction, as documented in the January 8, 1992 submittal, the licensee has concluded that these evaluations bound operation under the new conditions. Therefore, the systems and components reexamined remain in compliance with the applicable codes and standards.

4.0 SUMMARY

On the basis of the above evaluation (Section 2.0) and supplementary evaluation (Section 3.0) the staff finds that, with regard to the proposed TS revisions, the licensee has provided adequate supporting analyses and evaluations to demonstrate the following:

1) For all UFSAR Chapter 15 events, the proposed reductions in RCS flow rate (to 268,500 gpm) and thermal power limit (to 95% of rated value) are accommodated by current thermal margins or by the assessment of a DNBR penalty against generic retained margin. All acceptance criteria continue to be met.

2) The revised values of the OT (delta-T), OP (delta-T), and PRNF high trip setpoints, as well as the current values of the remaining ESF and RPS setpoints, provide adequate plant protection at the reduced RCS flow rate and power.

3) The revised core thermal limits (Figure 2.1-1a) bound NA-1 Cycle 9 operation at the reduced RCS flow rate and power.

4) NSSS and BOP systems and components will continue to remain in compliance with the applicable codes and standards for operation at the reduced RCS flow rate and power.

Therefore, the staff finds the proposed TS revisions to be acceptable. With the exception of the revision to TS Table 2.2-1, which is a permanent administrative change, all other proposed TS will remain in effect for the balance of the operating period until the NA-1 SGs are replaced in 1993.

5.0 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.

6.0 ENVIRONMENTAL CONSIDERATION

This amendment changes 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 amendment involves 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding (57 FR 2291). Accordingly, this amendment meets 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 this amendment.

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

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