DCSMS-016

Docket No. 50-338

MAR 13 1984

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Mr. W. L. Stewart
Vice President - Nuclear Operations
Virginia Electric and Power Company
Post Office Box 26666
Richmond, Virginia 23261

Dear Mr. Stewart:

The Commission has issued the enclosed Amendment No. 54 to Facility Operating License No. NPF-4 for the North Anna Power Station, Unit No. 1 (NA-1). The amendment revises the Technical Specifications in response to your letter dated December 30, 1982, as supplemented by letters dated April 25, July 6, and July 11, 1983. The amendment also revises the Technical Specifications in response to your letter dated September 29, 1983. The amendment is effective within 30 days from the date of issuance.

The amendment revises the NA-1 TS to allow operation with a Reactor Coolant System Average Temperature (RCS T_{av}) of 587.8°F. The amendment completes your Phase 1 and Phase 2 plant upgrade which increases secondary steam pressure in order to maximize the electrical output at the currently licensed reactor thermal power rating of 2775 Megawatts thermal. The amendment further revises the NA-1 TS to allow optimization of the core loading pattern by changing the fractional thermal power multiplier from 0.2 to 0.3 for a RCS T_{av} of 587.8°F.

It is noted that your currently approved RCS T_{av} of 582.8°F was previously approved with up to 7% steam generator tube plugging. However, as noted in our enclosed safety evaluation, the large break LOCA calculation submitted in support of the new RCS T_{av} of 587.8°F assumes only 5% steam generator plugging. Therefore, operation at the higher RCS T_{av} is approved for only 5% steam generator tube plugging.

Finally, as noted in the enclosed Safety Evaluation, issuance of the amendment for NA-2 is being held in abeyance until such time that feedwater regulating valve trim modifications are completed to provide operational flexibility for the NA-2 Phase II upgrade program.

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A copy of the Safety Evaluation is enclosed. The notice of issuance will be included in the Commission's next monthly Federal Register notice.

Sincerely,

Original signed by

Leon B. Engle, Project Manager
Operating Reactors Branch #3
Division of Licensing

Enclosure:
1. Amendment No. 54 to NPF-4
2. Safety Evaluation

cc: See next page

McGurren 2/2/84

ORB#3:DL	ORB#3:DL	ORB#3:DL	OELD	AD:OR:DL
PMKreutzer	LEngle/pn	JRMiller		GCLainas
2/21/84	2/28/84	2/28/84	2/ /84	3/1/84

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

VIRGINIA ELECTRIC AND POWER COMPANY

OLD DOMINION ELECTRIC COOPERATIVE

DOCKET NO. 50-338

NORTH ANNA POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 54
License No. NPF-4

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

   A. The applications for amendment by Virginia Electric and Power Company (the licensee) dated December 30, 1982 (as supplemented April 25, July 6, 1982, July 11, 1983) and September 29, 1983, 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 applications, 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. 54 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective within 30 days from the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James R. Miller

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 13, 1984

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 54 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 Amemdment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are included to maintain document completeness.

Pages

2-2
2-6
2-8
2-9
2-10
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.

Nominal Tavg: 587.8°F
Nominal Design Flow: 285,000 gpm

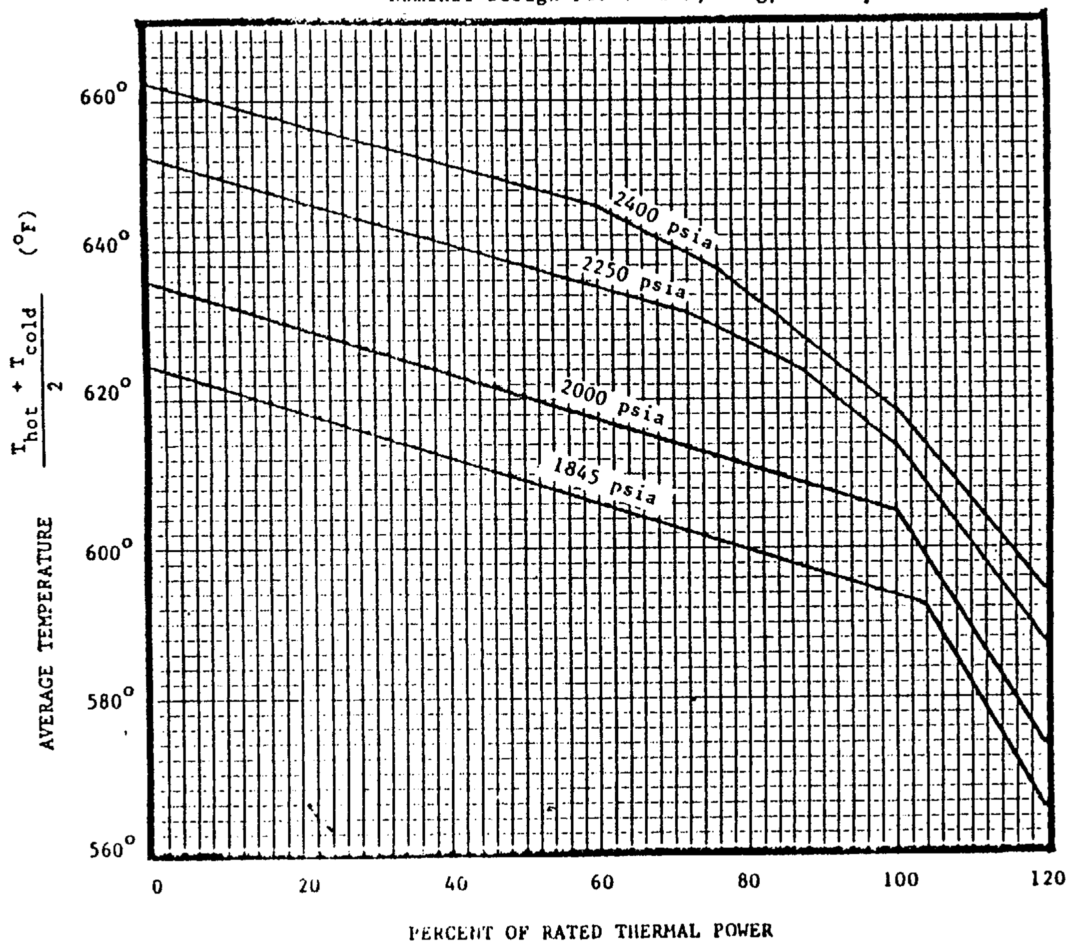

FIGURE 2.1-1 REACTOR CORE SAFETY LIMITS FOR THREE LOOP OPERATION, 100% FLOW

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 is 95,000 gpm per loop.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. Steam Generator Water Level--Low-Low	≥ 18% of narrow range instrument span-each steam generator	≥ 17% of narrow range instrument span-each steam generator
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	≤ 40% of full steam flow at RATED THERMAL POWER coincident with steam generator water level ≥ 25% of narrow range instrument span--each steam generator	≤ 42.5% of full steam flow at RATED THERMAL POWER coincident with steam generator water level ≥ 24% of narrow range instrument span--each steam generator
15. Undervoltage-Reactor Coolant Pump Busses	≥ 2905 volts-each bus	≥ 2870 volts-each bus
16. Underfrequency-Reactor Coolant Pump Busses	≥ 56.1 Hz - each bus	≥ 56.0 Hz - each bus
17. Turbine Trip		
A. Low Trip System Pressure	≥ 45 psig	≥ 40 psig
B. Turbine Stop Valve Closure	≥ 1% open	≥ 0% open
18. Safety Injection Input from ESF	Not Applicable	Not Applicable
19. Reactor Coolant Pump Breaker Position Trip	Not Applicable	Not Applicable

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: Overtemperature $\Delta T \leq \Delta T_o \left[K_1 - K_2 \left(\frac{1+\tau_1 S}{1+\tau_2 S}\right)(T-T') + K_3(P-P') - f_1(\Delta I)\right]$

where:

ΔT_o	=	Indicated ΔT at RATED THERMAL POWER
T	=	Average temperature, ^oF
T'	=	Indicated T_{avg} at RATED THERMAL POWER $\leq$ 587.8^oF
P	=	Pressurizer pressure, psig
P'	=	2235 psig (indicated RCS norminal operating pressure)
$\frac{1+\tau_1 S}{1+\tau_2 S}$	=	The function generated by the lead-lag controller for T_{avg} dynamic compensation
τ_1 & τ_2	=	Time constants utilized in the lead-lag controller for T_{avg} τ_1 = 25 secs, τ_2 = 4 secs.
S	=	Laplace transform operator (sec^{-1})

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 = 1.085	K_1 = ( )	K_1 = ( )
K_2 = 0.0150	K_2 = ( )	K_2 = ( )
K_3 = 0.000670	K_3 = ( )	K_3 = ( )

and f_1 (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 - 32 percent and + 9 percent, f_1 (ΔI) = 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 - 32 percent, the ΔT trip setpoint shall be automatically reduced by 1.92 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of ($q_t - q_b$) exceeds + 9 percent, the ΔT trip setpoint shall be automatically reduced by 1.77 percent of its value at RATED THERMAL POWER.

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

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_0 \left[K_4 - K_5 \left(\frac{\tau_3 S}{1+\tau_3 S}\right) T - K_6 (T-T') - f_2(\Delta I)\right]$

Where:

ΔT_0	=	Indicated ΔT at RATED THERMAL POWER
T	=	Average temperature, $^{\circ}F$
T'	=	Indicated T_{avg} at RATED THERMAL POWER $\leq 587.8^{\circ}F$
K_4	=	1.091
K_5	=	$0.02/^{\circ}F$ for increasing average temperature
K_5	=	0 for decreasing average temperatures
K_6	=	0.00121 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} τ_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.

TABLE 3.2-1

DNB PARAMETERS

	LIMITS		
PARAMETER	3 Loops in Operation	2 Loops in Operation** & Loop Stop Valves Open	2 Loops in Operation** & Isolated Loop Stop Valves Closed
Reactor Coolant System T_{avg}	$\leq$ 592°F		
Pressurizer Pressure	$\geq$ 2205 psig*		
Reactor Coolant System Total Flow Rate	$\geq$ 285,000 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.

AXIAL POWER DISTRIBUTION

LIMITING CONDITION FOR OPERATION

3.2.6 The axial power distribution shall be limited by the following relationship:

$$[F_j(Z)]_S = \frac{[2.20]\ [K(Z)]}{(\overline{R}_j)(P_L)(1.03)(1 + \sigma_j)(1.07)}$$

Where:

a. $F_j(Z)$ is the normalized axial power distribution from thimble j at core elevation Z.

b. P_L is the fraction of RATED THERMAL POWER.

c. K(Z) is the function obtained from Figure 3.2-2 for a given core height location.

d. $\overline{R}_j$, for thimble j, is determined from at least n=6 incore flux maps covering the full configuration of permissible rod patterns above P_m% of RATED THERMAL POWER in accordance with:

$$\overline{R}_j = \frac{1}{n} \sum_{i=1}^{n} R_{ij}$$

Where:

$$R_{ij} = \frac{F_{Qi}^{Meas}}{[F_{ij}(Z)]_{Max}}$$

and $[F_{ij}(Z)]_{Max}$ is the maximum value of the normalized axial distribution at elevation Z from thimble j in map i which had a measured peaking factor without uncertainties or densification allowance of F_Q^{Meas}.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 54 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

Introduction:

By letter dated December 30, 1982 as supplemented by letters dated April 25, July 6, and July 11, 1983, the Virginia Electric and Power Company (the licensee) requested a change to the Technical Specifications (TS) 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). Also, by letter dated September 29, 1983, the licensee requested a change to the NA-1&2 TS.

Specifically, the licensee's requested change of December 30, 1982, as supplemented, would revise the TS to allow operation with a Reactor Coolant System (RCS) Average Temperature of 587.8 degrees Fahrenheit (°F) as opposed to the currently approved RCS T_{av} of 582.8°F. The licensee's requested change of September 29, 1983, would revise the NA-1&2 TS by changing the fractional thermal power multiplier from 0.2 to 0.3 with a RCS T_{av} of 587.8°F. Thus, the proposed change dated September 29, 1983 is germane to the requested change dated December 30, 1982, as supplemented. Therefore, these two separate request changes are being evaluated as one specific licensing action at this time.

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The requested change dated December 30, 1982 (as supplemented) would implement Phase II of a NA-1&2 plant upgrade program which would increase secondary steam pressure in order to maximize the electrical output at the currently licensed reactor thermal power rating of 2775 Megawatts thermal (MWT).

It is noted that the licensee's plant upgrade program enveloping both a Phase I and Phase II plant upgrade would increase the RCS T_{av} by a total of 7.5°F, specifically 580.3°F to 587.8°F. This total increase in T_{av} would increase secondary side steam pressure by 50 psi and result in a 5.6 MVA increase in electrical output. The licensee's Phase I plant upgrade increased the RCS T_{av} from 580.3°F to 582.8°F at the licensed reactor thermal power rating of 2775 MWT. Implementation of the NA-1&2 Phase I Upgrade Program was approved at the time the Commission issued the NA-1 Amendment No. 42 to License NPF-4 (with supporting safety analysis) on October 4, 1982 and the NA-2 Amendment No. 32 to License NPF-7 on October 19, 1983.

It is also noted that the licensee's proposed change relative to the Phase II upgrade is supported in appropriate cases by analyses covering the augmented change in the RCS T_{av} for both Phase I and Phase II representing a total change in temperature of 7.5°F even though the requested specific change for Phase II covers a T_{av} change of 5°F; specifically from the NRC approved Phase I value of 582.8°F to the requested Phase II temperature of 587.8°F.

As stated previously, the proposed change would revise the TS to allow operation with a (RCS) T_{av} of 587.8°F as opposed to the currently approved Phase I RCS T_{av} of 582.8°F. In addition to increasing the RCS T_{av} by 5°F, the net reactor coolant pump heat input has been measured to be 12 MWT instead of 10 MWT, and this 2 MWT increase would change the currently approved Nuclear Steam Supply System (NSSS) rating from 2785 MWT to 2787 MWT. TS changes have been submitted related to the RCS T_{av} safety limits, the Departure from Nucleate Boiling (DNB) parameters, and the Over Temperature Delta Temperature (OTΔT) and Over Pressure Delta Temperature (OPΔT) setpoints. The proposed change would also increase the TS value of core inlet volumetric flow rate based on actual measurements. The currently licensed reactor thermal rating of 2775 MWT remains unchanged. The proposed 5°F change in the RCS T_{av} would provide an increase in the secondary side steam pressure of approximately 32 pounds per square inch (psi) and result in a higher secondary cycle thermal efficiency and an approximate 3 MW electrical increase in output.

The licensee's safety evaluation supporting the licensee's proposed changes include the scope of the NSSS Accident Analyses and other accident analyses specified in Chapter 15 of the NA-1&2 Final Safety Analysis Report (FSAR). The safety evaluation also addressed the Balance of Plant (BOP) and NSSS/BOP Interfaces. Reanalysis of the Emergency Core Cooling System (ECCS) performance and the Loss-of-Coolant Accident (LOCA) was performed to verify that the proposed changes and the analytical techniques used by the licensee were in full compliance with 10 CFR 50, Appendix K.

Finally, the licensee's requested change of September 29, 1983 would revise the fractional thermal power multiplier from 0.2 to 0.3 with a RCS T_{av} of 587.8°F. The proposed change would allow optimization of the core loading pattern by minimizing restrictions on the fractional power limit, $F\Delta_H^N$, at low power.

Our discussion and evaluation of these changes is provided below.

Discussion:

Reanalysis of LOCA and non-LOCA Accidents:

An increase in the RCS T_{av} will change the condition of the NSSS in several ways which can affect plant response to transients and accidents. The RCS subcooling will be reduced by 5°F, and along with it the margin to DNBR. (This effect is partially offset by the fact that the core inlet flow is higher than previously assumed.) Stored energy in the reactor fuel and in the coolant will also increase proportionally. Furthermore, the power defect in reactivity is increased. Finally secondary steam pressure is increased by about 50 psi. In light of these differences, a reanalysis of LOCA and non-LOCA accidents was submitted by the licensee for NRC staff review and approval.

Accidents Not Reanalyzed

Several transients did not require reanalysis. Transients at zero power are unchanged because the T_{av} at hot zero power remains the same. Similarly, transients which are independent of thermal-hydraulic (Fuel Handling Accidents) and transients which have been shown to be bounded by more serious accidents (Uncontrolled Boron Dilution at Power) were not reanalyzed. The spurious actuation of safety injection was not reanalyzed because the original analysis had shown that DNBR remains above the initial value throughout the transient. Finally, steam generator tube rupture was not recalculated because the principal impact of increasing T_{av} would be a slight benefit due to increased initial secondary steam pressure.

LOCA Reanalysis

The NRC has recently accepted a Large Break LOCA (LBLOCA) calculation submitted for NA-1&2. The analysis was performed with the approved "1981" Westinghouse evaluation model, assuming F_Q equal to 2.20 and 7% steam generator tube plugging. A peak clad temperature of 2194.7°F was calculated. The LBLOCA calculation submitted with the current amendment request used the same evaluation model and boundary conditions, with the following exceptions; (1) T_{av} was assumed equal to 587.8°F instead of 582.8°F, (2) a thermal design flow of 95,000 Gallons Per Minute (GPM) per loop was used rather than 92,800 GPM and (3) 5% steam generator tube plugging was assumed in place of 7%. The calculated peak clad temperature is below 2200°F, and the other acceptance criteria of 10 CFR 50.46 are satisfied.

The assumption of 5% tube plugging is acceptable, but as a consequence, operation at T_{av} equal to 587.8°F will be permissible only up to 5% tube plugging instead of the previously approved limit of 7%.

The small break LOCA (SBLOCA) has been shown in previous calculations to fall well within the acceptance criteria of 10 CFR 50.46. For instance, the worst case break (3 inch diameter) analyzed in the NA-1&2 FSAR yielded a peak clad temperature of 1852°F. Increased T_{av} could affect SBLOCA in two ways; (1) more stored energy in the primary system and (2) higher initial pressure on the secondary side. Both of these effects have minimal impact on SBLOCA, and consequently the licensee is justified in not reanalyzing the accident.

Non-LOCA Transients and Accidents

The reanalysis of non-LOCA transients and accidents was performed in conformance with the Standard Review Plan, using analytical methods which have been approved by the staff.

Because increased T_{av} would lead to higher stored energy in the primary system, the change had little effect on transients involving increased heat removal. Accidental steam generator depressurization and minor steam line breaks are bounded by the major steam line break at hot zero power, for which the calculated DNBR does not drop below 1.30. Accidents due to excessive load increase, and excessive heat removal due to feedwater malfunctions continue to meet Standard Review Plan criterion of DNBR greater than 1.30.

For events involving decreased heat removal, the increase in T_{av} results in a slightly lower calculated DNBR. Nonetheless, the criterion for DNBR greater than 1.30 is still satisfied. This category includes the loss-of load, loss-of-main feedwater and loss-of-offsite power transients. For the more serious feedline rupture event, the primary pressure and temperature transient is considerably less severe than in the original FSAR. This is primarily due to taking credit for an auxiliary feedwater system design improvement which established a one-to-one relationship between auxiliary feedwater pumps and steam generators. As in the original FSAR, heat removal by the auxiliary feedwater system is sufficient to prevent overpressurization of the Reactor Coolant System and prevent core uncovery.

The complete loss of forced coolant flow accident continues to meet the DNBR criterion, even though violation of the limit is acceptable for this class of accident. The locked RCP rotor event yields slightly higher peak pressures and clad temperatures with increased T_{av}, but the calculated results are still within acceptable limits. These results are reasonable for a 5°F increase in T_{av}.

Accidental depressurization of the primary system with the higher T_{av} leads to a slightly lower calculated DNBR, but the DNBR criterion is still exceeded by a sizable margin.

Thermal Hydraulic Design Evaluation of Coolant System Parameters

At rated thermal load, increasing the RCS T_{av} to 587.8°F on the primary side of the steam generator tubes will increase the temperature of the steam on the secondary side by approximately 6.8°F, which corresponds to a 50 psi increase in steam pressure. Table 1 provides a comparison of the current and proposed RCS temperatures and flow rates at rated thermal power. From the table it can be seen that the reactor core thermal rating, pressure and "no load" temperature remain at current values. The core inlet volumetric flow rate has been increased based on the actual performance of the reactor coolant pumps. The total core inlet thermal flow rate is the TS minimum flow limit utilized for thermal and hydraulic analyses (e.g., DNB evaluations). Based on NA-1&2 calorimetric data, the measured core inlet volumetric flow rate is 302,100 gpm with 2.8 percent of the steam generator tubes plugged. If the steam generator tube plugging level was increased to 5 percent, the measured flow would decrease by less than 1 percent. The NA Units employ a calorimetric -ΔT method to determine the core inlet flow rate. For this flow measurement technique the maximum uncertainty in the total flow measurement is ±2.0 percent. Accounting for a 5 percent steam generator tube plugging level and the maximum flow measurement error of 2.0 percent, a total core inlet thermal flow rate of 285,000 gpm is conservatively low. Therefore, a thermal flow rate of 285,000 gpm may be utilized as a design thermal flow rate for the proposed RCS T_{av} increase and in fact was used by the licensee in their design analyses to set thermal limits. The RCS T_{av} has been increased from 582.8°F to 587.8°F. The variations in inlet temperature and temperature rises are attributable to the thermodynamic properties of compressed liquid water and the increased core inlet volumetric flow rate. The overall impact of these changes in the thermal hydraulic performance of the core has been evaluated and found to be acceptable.

Confirmation of W-3 DNB Correlation Bounds

The staff requested that the licensee confirm that the applicable range for the key parameters in the W-3 DNBR correlation bounds the conditions expected after increasing T_{av} to 587.8°F. The licensee supplied Tables 2 and 3 and associated references which demonstrate the applicability of W-3 for the proposed temperature conditions of the core. Based on this data, the staff finds that the key parameters in W-3, which have been previously reviewed and approved by the staff, acceptably bound the thermal conditions anticipated after the increase in T_{av}.

TABLE 1

COMPARISON OF REACTOR COOLANT SYSTEM PARAMETERS

Thermal and Hydraulic Design Parameters	Design Conditions	
	Current	Proposed
NSSS Power, MWt	2785	2787
Net Reactor Coolant Pump Heat Input, MWt	10	12
Reactor Core Heat Output, MWt	2775	2775
System Pressure, Nominal psia	2250	2250
System Pressure, Min., Steady State, psia	2220	2220
Total Core Inlet Thermal Flow Rate, gpm	278,400	285,000
Total Core Inlet Thermal Flow Rate, lbm/hr	105.1×10^6	106.3×10^6
Core Effective Flow Rate for Heat Transfer, lbm/hr	100.4×10^6	101.5×10^6
Reactor Coolant System Temperatures, °F		
Nominal Reactor Vessel/Core Inlet	546.9	555.5
Average Rise in Vessel	66.9	64.5
Average Rise in Core	69.7	67.2
Average in Core	583.6	591.1
Average in Vessel	580.3	587.8
No Load	547.0	547.0

TABLE 2

W-3 CORRELATION LIMITS

CORRELATION	REF. NO.	PRESSURE RANGE (psia)	MASS VELOCITY ($Mlb/h-f^2$)	EQUIV. DIAMETER (in.)	LOCAL QUALITY	AXIAL HEIGHT (in.)	INLET TEMP. (°F)
W-3	1,2	1000-2400	1.0-5.0	0.2-0.7	$\leq$0.18	10-144	>400
F-factor	1,2	1000-2400	1.0-3.0	0.2-0.7	$\leq$0.15	10-144	
Coldwall Factor	1,2 3,4	1000-2400	1.0-5.0		$\leq$0.15	>10	
Spacer Factor	3,4	1490-2440	1.5-3.7		$\leq$0.15	96-168	404-624

TABLE 3

CORE CONDITION WITH TAVG INCREASE

core inlet temp. (°F)	555.5
mass velocity ($mlb/h-f^2$)	2.442
pressure (psia)	2250

Containment Safety Margin

The following acceptance criteria for subatmospheric containment functional design form the basis for the licensee's evaluation of containment safety margin for the uprated RCS T_{av} conditions of the NSSS:

(1) The calculated peak containment pressure shall not exceed the design pressure of 45 psig;

(2) The containment shall be depressurized to below one atmosphere absolute pressure in less than 60 minutes;

(3) Once depressurized, the containment shall be maintained at a pressure less than one atmosphere absolute for the duration of the accident.

The licensee has re-analyzed the postulated loss of coolant accident (LOCA) for the uprated NSSS conditions assuming a pump suction double ended rupture (PSDER), and evaluated the effect on the Net Positive Suction Head Available (NPSHA) for the Recirculation Spray (RS) and Low Head Safety Injection (LHSI) pumps. The analysis results were compared with the appropriate design criteria. We conclude, based on these results, that the proposed uprated NSSS conditions will have a negligible impact on the containment functional design.

Subcompartment analyses for the reactor cavity and steam generator and pressurizer compartments were not redone. The licensee's calculations confirm that, for a subcooled reactor coolant system, mass and energy releases would decrease with increased reactor coolant temperature. Therefore, the analyses documented in the NA-1&2 FSAR are bounding for the uprated conditions. We concur with this finding.

The licensee did not reanalyze the main steam line break (MSLB) accident for the uprated conditions. The current design basis MSLB is a full guillotine break at the no-load (hot shutdown) condition and this analysis remains unchanged for the uprated NSSS conditions. Although there would be some additional energy release for a MSLB at power because of the uprated NSSS conditions, the no-load condition would remain the limiting case. We concur with this finding since the steam generator inventory at no-load conditions would continue to dominate any additional energy release that would occur for a MSLB at power.

Main Steam System

Consideration of the change in the RCS T_{av} for the main steam system involved main steam safety valve capacity and main steam isolation capability. The main steam safety valves have a total relieving capacity of 12,826,269 pounds per hour (lb/hr) which is more than the total uprated main steam flow of 12,251,367 lb/hr. The main steam trip and non-return valves were evaluated for rapid closure impact loads applied subsequent to main steam system pipe rupture at uprated conditions (increased steam pressure) by the licensee. The results of the computer runs that modeled the transients effect on the valves showed that these valves would close as required without jeopardizing the integrity of the pressure boundary.

Auxiliary Feedwater System

Consideration of the change in the RCS T_{av} for the auxiliary feedwater (AFW) system involved AFW ability to provide adequate flow for decay heat removal. The AFW pumps are designed to deliver rated flow to the steam generators at a static head equivalent to the set pressure of the lowest main steam safety valve. Because this setpoint pressure will not change, the resistance parameters associated with the AFW system will remain the same, and this AFW flow requirement (based on 2910 MWT core power plus 2%) for NA-1&2 remains unchanged. Therefore, the existing AFW system will be adequate at the uprated conditions.

Condensate and Feedwater System

Consideration of the change in the RCS T_{av} for the condensate and feedwater system involved its isolation capability following transients and accidents. The small decrease in feedwater pressure (by approximately 2 psi) does not affect the closure capability of the feedwater isolation valves.

Component Cooling and Service Water Systems

Consideration of the change in the RCS T_{av} for the component cooling system and service water system involved their ability to remove heat from safety related equipment. The increased RCS cold leg temperature increases the heat loadings on the component cooling water (CCW) system during normal operating conditions due to the slightly increased heat load from the chemical and volume control system heat exchangers. The affected heat exchangers are the non-regeneration, excess letdown and seal water return heat exchangers. The cumulative heat loadings to the CCW system at the uprated operating conditions remain less than the design value used for the original plant design. Heat removal capability for safety related equipment cooled by the CCW system is not affected by this change. Consequently, the service water system is also not impacted by the uprating.

Spent Fuel Pool Cooling System

There is no impact on the spent fuel pit heat loads as a result of the uprating since core thermal power and the associated decay heat levels for spent fuel remain unchanged.

Fractional Thermal Power Multiplier

The licensee has proposed to revise the TS by changing the fractional thermal power multiplier from 0.2 to 0.3 with a RCS T_{av} equal to 587.8°F. The proposed change would allow optimization of the core loading pattern by minimizing restrictions on the fractional power limit, $F\Delta_H^N$, at low power. At full power, the $F\Delta_H^N$ limit will remain unchanged. In the expression for $F\Delta_H^N$, as specified in the NA-1& 2 TS, $F\Delta_H^N = 1.55\ [1+0.3(1-P)]$. The proposed change would increase the partial power multiplier from 0.2 to 0.3 in the expression above; however, at full power, P becomes 1.0 and the multiplicative effect of the 0.3 partial multiplier is zero (0). The increase in the fraction power $F\Delta_H^N$ will be compensated for by more restrictive fractional power core thermal limits. These more restrictive core thermal limit lines will maintain the current design bases DNB criteria. Analyses supporting the proposed change used analytical techniques consistent with North Anna design bases and previously NRC-approved Westinghouse fractional power multiplier analyses which are appropriately applied to NA-1&2. Therefore, we find the proposed change to be acceptable.

Evaluation:

Based on the above, we have determined that the licensee has satisfactorily reexamined the impact of increasing the RCS T_{av} to 587.8°F for a full range of transients and accidents. We have further determined that the licensee's proposed change encompasses the analysis of all transients and accidents specified in the Standard Review Plan. Although there is some loss of margin in many of the events, the relative acceptance criteria are met. In addition, all acceptance criteria of 10 CFR 50.46 are satisfied and the analytical techniques as used by the licensee are in full compliance with 10 CFR 50, Appendix K.

We have also reviewed and evaluated the thermal-hydraulic aspects of the licensee's proposed change and conclude the proposed increase in RCS T_{av} and associated increase in core design flow rate are acceptable. The licensee has provided acceptable documentation regarding containment functional design. We have determined that the increase in the RCS T_{av} does not result in any containment safety concern.

We have further reviewed the potential effects of the proposed change regarding BOP/NSSS interfaces and find that predicted changes are small and are within the envelope of the approved NA-1&2 system design.

Finally, we have determined that increasing the partial power multiplier from 0.2 to 0.3 for a RCS T_{av} of 587.8°F will be compensated for by more restrictive core thermal limits. These limits will maintain the current DNB criteria. In addition, the proposed change used analytical techniques previously approved by the NRC which are appropriately applied to NA-1&2 and therefore we find the proposed change to be acceptable.

Based on all of the above, we find the proposed change to be acceptable. We further find that the proposed changes to the NA-1&2 TS regarding these matters are acceptable.

As noted above, the licensee's submittal of the large break LOCA calculation submitted in support of the proposed RCS T_{av} of 587.8°F assumed only 5% steam generator tube plugging. Therefore, operation at a RCS T_{av} of 587.8°F is approved for only up to 5% steam generator tube plugging.

Finally, it is noted that the above safety evalution is for both NA-1&2. However, at this time, the proposed change is applicable to NA-1 only. The licensee has noted that the BOP review for the Phase II upgrade conditions at NA-2 identified a decrease in feedwater valve operational flexibility at the uprated conditions. Necessary modification in feedwater valve trim will be completed during the NA-2 Third Refueling Outage (Fall 1984). Therefore, issuance of the Phase II upgrade program for NA-2 will be held in abeyance until such modifications are completed and verified by the NRC.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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