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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING AMENDMENT NO. 41 TO
FACILITY OPERATING LICENSE NO. NPF-7
VIRGINIA ELECTRIC AND POWER COMPANY
OLD DOMINION ELECTRIC COOPERATIVE
NORTH ANNA POWER STATION, UNIT NO. 2
DOCKET NO. 50-339

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.

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 587.7°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.

On March 13, 1984 Phase II of the Plant Upgrade Program was implemented at NA-1 with the issuance of Amendment No. 54 to Facility Operating License No. NPF-4. Although our Safety Evaluation supporting Amendment No. 54 stated that we found the Phase II upgrade to be applicable to both NA-1&2, the issuance of an identical amendment for NA-2 was held in abeyance until the licensee could implement necessary feedwater valve trim at NA-2 during the Third Refueling Outage (Fall 1984).

By letter dated October , 1984 the licensee stated that the necessary feedwater valve trim had been implemented at NA-2 to support the Phase II Upgrade Program. Therefore, we are issuing the Phase II Upgrade for NA-2 at this time.

Due to the passage of time since first approved for NA-1&2 and specifically implemented for NA-1 on March 13, 1984, we are restating our safety evaluation as originally provided for NA-1&2 to support the Phase II upgrade for NA-2 at this time. Our original discussion and evaluation in addition to our comments on the NA-2 feedwater valve trim are 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

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	MASS VELOCITY	EQUIV. DIAMETER	LOCAL QUALITY	AXIAL HEIGHT	INLET TEMP.
		(psia)	(Mlb/h-f²)	(in.)		(in.)	(°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	-
[illegible]cer Factor	3,4	1490-2440	1.5-3.7		$\leq$0.15	96-168	404-524

TABLE 3

CORE CONDITION WITH TAVG INCREASE

core inlet temp. (°F)	555.5
mass velocity (mlb/h-f²)	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-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.

We stated in our safety evaluation supporting the Phase II Upgrade for NA-1 (issued March 13, 1984) that the Phase II Upgrade for NA-2 would be held in abeyance until such time that necessary feedwater valve trim could be implemented at NA-2 to compensate for a decrease in feedwater valve operational flexibility at the Upgraded Phase II conditions. By letter dated October 3, 1984, the licensee stated that necessary feedwater valve trim modifications had been completed to support the NA-2 Phase II Upgrade. We requested that Region II inspection verify the completion of these modifications. On October 4, 1984, we were so notified by Region II that the appropriate feedwater valve trim modifications were complete in support of the NA-2 Phase II Upgrade.

Therefore, based on all of the above, we find implementation of the Phase II Upgrade to be acceptable for NA-2.

Environmental Consideration:

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The 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. 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.

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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