August 30, 1993

Docket No. 50-339

DISTRIBUTION
See attached sheet

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

Dear Mr. Stewart:

SUBJECT: NORTH ANNA UNIT 2 - ISSUANCE OF AMENDMENT RE: REDUCTION IN REACTOR COOLANT SYSTEM (RCS) FLOW RATE (TAC NO. M85868)

The Commission has issued the enclosed Amendment No. 152 to Facility Operating License No. NPF-7 for the North Anna Power Station, Unit No. 2 (NA-2). The amendment revises the Technical Specifications (TS) in response to your letter dated December 4, 1992.

This amendment allows a reduction in the measured reactor coolant system flow rate to accommodate the systems effects associated with increasing steam generator tube plugging.

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. 152 to NPF-7
2. Safety Evaluation

cc w/enclosures:
See next page

OFFICE	LA:PDII-2	PM:PDII-2	D:PDII-2	OGC	
NAME	ETana ETT	LEngle:ms	HBerkow	M Young	
DATE	08/11/93	08/12/93	08/13/93	08/17/93	/ /

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

North Anna Power Station
Units 1 and 2

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DATED: August 30, 1993

AMENDMENT NO. 152 TO FACILITY OPERATING LICENSE NO. NPF-7-NORTH ANNA UNIT 2

Docket File
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

VIRGINIA ELECTRIC AND POWER COMPANY

OLD DOMINION ELECTRIC COOPERATIVE

DOCKET NO. 50-339

NORTH ANNA POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 152
License No. NPF-7

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

   A. The application for amendment by Virginia Electric and Power Company, et al., (the licensee) dated December 4, 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.C.(2) of Facility Operating License No. NPF-7 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 152 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 30, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 152

TO FACILITY OPERATING LICENSE NO. NPF-7

DOCKET NO. 50-339

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Remove Pages	Insert Pages
2-6	2-6
3/4 2-16	3/4 2-16

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	Low Setpoint - ≤ 26% of RATED THERMAL POWER
	High Setpoint - ≤ 109% 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 x 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.

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg}

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

APPLICABILITY: MODE 1

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement at least once per 18 months.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS		
	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}	$\leq$ 591°F		
Pressurizer Pressure	$\geq$ 2205 psig*		
Reactor Coolant System Total Flow Rate	$\geq$ 275,300 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.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 152 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

1.0 INTRODUCTION

By letter dated December 4, 1992, the Virginia Electric and Power Company (the licensee) proposed a change to the Technical Specifications (TS) for the North Anna Power Station, Unit No. 2 (NA-2). The proposed changes would revise Table 2.2-1, Reactor Trip System Instrumentation Trip Setpoints, of TS 2.2-1 and Table 3.2-1, DNB Parameters, of TS 3.2.5 to allow a reduction in the minimum measured reactor coolant system (RCS) flow rate. The licensee has made this request in anticipation of increased steam generator tube plugging (SGTP) at NA-2.

The licensee has examined the Chapter 15 accident analyses and determined which transients required reanalysis and which required only reevaluation. The criteria used by the licensee to determine whether to reanalyze or reevaluate was that (1) if the event is potentially impacted by RCS flow rate and also by other effects of SGTP, the event was either evaluated or reanalyzed, (2) 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, and (3) if an event is unaffected by RCS flow rate, it is not addressed.

2.0 DISCUSSION

The primary consequences of increasing the SGTP are (1) increased reactor coolant system loop resistance, resulting in a lower RCS flow rate, (2) decreased steam generator tube heat transfer area, resulting in lower steam generator outlet steam pressure, and (3) a decreased total RCS volume. The licensee measured the NA-2 flow in April 1992 at an average SGTP level of 7.0% and found it to be 293,321 gpm, which is greater than the TS limit of 284,000 gpm. To conservatively bound the flow rates resulting from future increases in SGTP levels up to approximately 18%, the licensee proposed a minimum measured flow of 275,300 gpm, 3% lower than the current TS flow rate. The licensee selected this value based on experience gained during the NA-1 extended SGTP effort.

The increase in tube plugging results in a reduction in RCS total flow rate through the core. This core flow reduction can challenge the DNB design limits. The statistical DNBR limit (SDL) of 1.26 was achieved by combining key DNBR analysis parameter uncertainties in a statistical manner with the WRB-1 Critical Heat Flux (CHF) correlation. The transient analyses results were assessed against a 1.46 design DNBR limit. The percentage difference between the design DNBR limit and the SDL represents the generic retained margin. This means that an additional 13.7% DNBR remains available to accommodate changing plant conditions.

2.1 REEVALUATED EVENTS

The events that were not reanalyzed, but were evaluated and found to be affected by the flow reduction, had a DNBR penalty applied. The DNBR penalty of 4.8% was calculated by approved methods (reference 2) and is based on a flow reduction of 3.0% and a bounding WRB-1 DNBR partial derivative of 1.6%. The 4.8% penalty is subtracted directly from the generic retained margin 13.7%, leaving an 8.9% margin for other plant modifications. Those accidents accommodated by this single penalty are listed below:

- 15.2.1 Rod Withdrawal from Subcritical
- 15.2.5 Partial Loss of Flow
- 15.2.10 Excessive Heat Removal
- 15.2.11 Excessive Load Increase
- 15.2.12 Accidental Depressurization of the RCS
- 15.2.13 Accidental Depressurization of the Main Steam System
- 15.2.14 Spurious Operation of the Safety Injection System
- 15.3.7 Single Rod Withdrawal at Power

The 4.8% penalty is also extracted from the available Core Thermal Limit retained DNBR margin. This is to ensure that the bounding Core Thermal Limit protection is provided by the existing Overtemperature ΔT and Overpower ΔT reactor protection system for the reduced flow rate.

The disposition of the remaining Chapter 15 transients which were reevaluated is as follows:

Main Steamline Break (15.4.2.1)

The main steamline break (MSLB) accident analysis uses the W-3 CHF correlation for DNBR calculations. The W-3 correlation has a different DNBR sensitivity to marginal changes in flow. When the conditions associated with MSLB were applied to the reduced flow rate, the flow reduction translated into a 4.3% penalty. This penalty was assessed against the generic retained margin.

Steam Generator Tube Rupture (15.4.3)

The steam generator tube rupture (SGTR) leads to increased contamination of the secondary system due to leakage of radioactive coolant from the RCS. The analysis assumes the operator terminates the primary to secondary leakage within 30 minutes. The reduction in RCS flow rate will not adversely affect the operator's ability to control an SGTR.

Small Break LOCA (15.3.1)

The Emergency Core Cooling System (ECCS) was reanalyzed by approved methods (Reference 4), and found to meet the acceptance criteria of 10 CFR 50.46, for performance during a postulated small-break loss-of-coolant accident (SBLOCA). The reanalysis included SGTP up to 35% in any single steam generator. This bounds the RCS conditions associated with the proposed flow reduction.

Large Break LOCA (15.4.1)

The licensee submitted a letter dated July 16, 1993 in compliance with the reporting requirements of 10 CFR 50.46. The licensee indicated that they have reanalyzed the LBLOCA event by approved methods and found the new LBLOCA peak cladding temperature (PCT) to be 2019°F. This PCT is within the 10 CFR 50.46 acceptance criteria of 2200°F and the analysis includes the reduced RCS flow rate of 264,400 gpm. LBLOCA remains the bounding event for NA-1&2.

2.2 Reanalyzed Events

The reanalyzed transients were initially reanalyzed and approved for the NA-1 extended SGTP evaluation (Reference 2). The licensee reanalyzed the system transient portion of the analyses using the licensee's RETRAN system transient analysis code single and double loop models. The models were modified to reflect the effects of reduced RCS flow rate associated with increased SGTP. Specifically, RCS flow rates, steam generator tube heat transfer areas (outside and inside the tubes), SG tube metal volume (heat capacity), and SG tube flow area were reduced to reflect plugging effects. The reanalyzed transients are detailed below.

Loss of External Load (Section 15.2.7)

The cases reanalyzed for loss of external load are beginning of cycle (BOC) with pressure control and BOC without pressure control. These cases represent the most limiting DNB and overpower cases, respectively. The calculated DNBR increased throughout the transient from the initial value of 2.15. Both the

RCS peak pressure and the main steam peak pressure remain below the acceptance criteria.

Loss of Normal Feedwater (Section 15.2.8 and 15.2.9)

The loss of normal feedwater was reanalyzed with consideration given to the availability of offsite power. In both cases (with and without offsite power) the analysis demonstrated that increased steam generator tube plugging levels did not adversely impact the ability of the auxiliary feedwater system to adequately perform its function.

Rod Bank Withdrawal at Power (Section 15.2.2)

A wide range of initial plant conditions were reanalyzed to identify the most limiting rod withdrawal at power event cases. The results indicated that for all cases the minimum DNBR remained above the design limit value. Also, the reanalysis confirmed that the current TS setpoints for overtemperature and overpower ΔT trip continue to provide core thermal limit protection under extended SGTP conditions.

Complete Loss of Flow (Section 15.3.4)

The complete loss of flow event was analyzed for two cases, the complete loss of voltage at the RCP breakers, undervoltage (UV), and 5.0 Hz/sec decay rate of the supply frequency, underfrequency (UF). The transient DNBRs remained above the statistical DNBR design limit throughout the transient for both the UF and UV events.

Locked Reactor Coolant Pump Rotor Event (Section 15.4.4)

The locked rotor event was analyzed in two parts: (1) a peak pressure calculation was performed assuming that no fuel rods experience DNB; and (2) the calculation was repeated assuming a rod experiences DNB with cladding failure. For the former case, the results indicate that peak RCS pressure remains 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 Chapter 15 events that were not impacted by the extended SGTP and were not reanalyzed or reevaluated by the licensee are Inactive Loop Start-up (Section 15.2.6) and Misloaded Fuel Assembly (Section 15.3.3).

3.0 EVALUATION

The staff reviewed the licensee's submittal proposing the reduction of RCS flow rate in anticipation of increased SGTP. The licensee's proposal evaluated the Chapter 15 events for potential impact of reduced RCS flow rate and extended SGTP on accident analyses. Of the Chapter 15 events evaluated, five were specifically reanalyzed; (1) Complete Loss of Flow, (2) Loss of Normal Feedwater, (3) Rod Bank Withdrawal at Power, (4) Locked Reactor Coolant Pump Rotor Event, and (5) Loss of External Load.

Those events that were evaluated, but not reanalyzed, accommodated the decrease in RCS flow by imposing a single 4.8% penalty on the retained DNBR margin. Finally, those Chapter 15 events that were not impacted by the reduced flow or increased SGTP were not addressed.

The events were reanalyzed by approved methods and the DNBRs were found to remain within the statistical DNBR design limits throughout the transients. The single penalty was derived by approved methods (Reference 3) and the DNBR for each of these events remained within the generic retained margin.

The licensee has shown that with the decreased RCS flow NA-2 continues to meet the acceptance criteria for the retained DNBR margin and also under those circumstances where the DNBR penalty was applied, the licensee continues to satisfy the limits of the generic retained margin. By doing so, the licensee ensures that a 95% confidence level exists against DNB occurring on at least 95% of the limiting fuel rods. The staff, therefore, finds the proposed NA-2 TS RCS flow reduction to be acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Virginia State official was notified of the proposed issuance of the amendment. The State official had no comment.

5.0 ENVIRONMENTAL CONSIDERATION

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 (58 FR 7008). 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.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Letter from W. L. Stewart, Virginia Electric Power Company, to NRC, "Proposed Technical Specifications Changes Reduction in Minimum Measured RCS Flow Rate," dated December 4, 1993.

2. Letter from G. S. Lainas, NRC, to W.R. Cartwright, Virginia Electric Power Company, Surry Units 1 and 2 and North Anna Units 1 and 2, "Qualification of WRB-1 CHF Correlation in the Virginia Power COBRA Code," dated July 25, 1989.

3. Letter from W. L. Stewart to NRC, "North Anna Power Station Unit 1," Supplemental Information Regarding Proposed Technical Specification Change for Reduced Minimum RCS Flow Rate Limit," dated January 31, 1992.

4. "North Anna Power Station Units 1 and 2 - Implementation of Extended SGTP Small Break LOCA Analysis," dated January 21, 1992.

5. Letter from W. L. Stewart to NRC, "Report of ECCS Evaluation Model Changes and 30 Day Report Per Requirements of 10 CFR 50.49 North Anna Power Station Units 1 and 2" dated July 16, 1993.

Principal Contributor: S. Brewer

Date: August 30, 1993