P~cember 13, 1985

Docket Nos. 50-338
and 50-339

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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 Nos. 73 and 59 to Facility Operating License Nos. NPF-4 and NPF-7 for the North Anna Power Station, Units No. 1 and No. 2 (NA-1&2). The amendments revise the Technical Specifications (TS) in response to your letter dated February 7, 1985 (Serial No. 666). The amendments are effective as of the date of issuance.

The amendments allow operation with a positive moderator temperature coefficient of plus (+) 6.0×10^{-5} $\Delta K/K°F$ for power levels below 70 percent of rated power and a zero (0) coefficient for power levels 70 percent and above. The positive moderator coefficient is acceptable for the presently rated core power level of 2775 MWT with a maximum reactor coolant system temperature of 587.8°F.

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

Sincerely,

/s/
Leon B. Engle, Project Manager
PWR Project Directorate #2
Division of PWR Licensing-A

Enclosure:
1. Amendment No. 73 to NPF-4
2. Amendment No. 59 to NPF-7
3. Safety Evaluation

cc w/enclosures:
See next page

No legal objection
McGurren

LA:PBD-8	PM:PAD-2	D:PAD-2	OELD
PMKreutzer	LEngle	LSRubenstein	
12/6/85	12/9/85	12/8/85	12/11/85

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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. 73
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 February 7, 1985, 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.

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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. 73 , 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

L.S. Rubenstein

Lester S. Rubenstein, Director
PWR Project Directorate #2
Division of PWR Licensing-A

Attachment:
Changes to the Technical Specifications

Date of Issuance: December 13, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 73

TO FACILITY OPERATING LICENSE NO. NPF-4

DOCKET NO. 50-338

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

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REACTIVITY CONTROL SYSTEMS

BORON DILUTION

VALVE POSITION

LIMITING CONDITION FOR OPERATION

3.1.1.3.2 The following valves shall be locked, sealed or otherwise secured in the closed position except during planned boron dilution or makeup activities

a. 1-CH-217 or

b. 1-CH-220, 1-CH-241, FCV-1114B and FCV-1113B.

APPLICABILITY: MODES 3, 4, 5, and 6

ACTION:

With the above valves not locked, sealed or otherwise secured in the closed position:

a. In MODES 3 and 4 be in COLD SHUTDOWN within 30 hours

b. In MODES 5 and 6 suspend all operations involving positive reactivity changes or CORE ALTERATIONS and lock, seal or otherwise secure the valves in the closed position within 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.3.2 The above listed valves shall be verified to be locked, sealed or otherwise secured in the closed position within 15 minutes after a planned boron dilution or makeup activity.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be:

a. For the all rods withdrawn, beginning of core life condition $\leq 0.6 \times 10^{-4}\ \Delta k/k/°F$ below 70 percent RATED THERMAL POWER

$\leq 0.0 \times 10^{-4}\ \Delta k/k/°F$ at or above 70 percent RATED THERMAL POWER

b. Less negative than $-4.0 \times 10^{-4}\ \Delta k/k/°F$ for the all rods withdrawn, end of core life at RATED THERMAL POWER.

APPLICABILITY: Specification 3.1.1.4.a - MODES 1 and 2* only#
Specification 3.1.1.4.b - MODES 1, 2 and 3 only#

ACTION:

a. With the MTC more positive than the limit of 3.1.1.4.a above:

1. Establish and maintain control rod withdrawal limits sufficient to restore the MTC to within its limit within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6.

2. Maintain the control rods within the withdrawal limits established above until subsequent measurement verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.

3. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

4. With the MTC more negative than the limit of 3.1.1.4b above, be in HOT SHUTDOWN within 12 hours.

*With $K_{eff} \geq 1.0$

#See Special Test Exception 3.10.3

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

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. 59
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 February 7, 1985, 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. 59 , 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

L. S. Rubenstein

Lester S. Rubenstein, Director
PWR Project Directorate #2
Division of PWR Licensing-A

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 13, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 59

TO FACILITY OPERATING LICENSE NO. NPF-4

DOCKET NO. 50-338

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

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MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be:

a. For the all rods withdrawn, beginning of core life condition $\leq 0.6 \times 10^{-4}$ $\Delta k/k/°F$ below 70 percent RATED THERMAL POWER

$\leq 0.0 \times 10^{-4}$ $\Delta k/k/°F$ at or above 70 percent RATED THERMAL POWER

b. Less negative than -4.0×10^{-4} delta k/k/°F for the all rods withdrawn, end of core life at RATED THERMAL POWER.

APPLICABILITY: Specification 3.1.1.4.a - MODES 1 and 2* only#.
Specification 3.1.1.4.b - MODES 1, 2 and 3 only#.

ACTION:

a. With the MTC more positive than the limit of 3.1.1.4.a above, operations in MODES 1 and 2 may proveed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than 0 delta k/k/°F within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6.

2. The control rods are maintained within the withdrawal limits established above until subsequent measurement verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.

3. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the limit of 3.1.1.4.b above, be in HOT SHUTDOWN within 12 hours.

*With K_{eff} greater than or equal to 1.0

#See Special Test Exception 3.10.3

MODERATOR TEMPERATURE COEFFICIENT

SURVEILLANCE REQUIREMENTS

4.1.1.4 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOL Limit of Specification 3.1.1.4.a. above, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. The MTC shall be measured at any THERMAL POWER and compared to -3.1 x 10^{-4} delta k/k/^{o}F (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicated the MTC is more negative than -3.1 x 10^{-4} delta k/k/^{o}F, the MTC shall be remeasured, and compared to the EOL MTC limit of specification 3.1.1.4.b., at least once per 14 EFPD during the remainder of the fuel cycle.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 73 AND 59 TO

FACILITY OPERATING LICENSE NOS. NPF-4 AND NPF-7

VIRGINIA ELECTRIC AND POWER COMPANY

OLD DOMINION ELECTRIC COOPERATIVE

NORTH ANNA POWER STATION, UNITS NO. 1 AND NO. 2

DOCKET NOS. 50-338 AND 50-339

Introduction:

By letter dated February 7, 1985 (Serial No. 666), the Virginia Electric and Power Company (the licensee) requested a change to the Technical Specifications (TS) for the North Anna Power Station, Units No. 1 and No. 2 (NA-1&2). Specifically, the proposed change would allow operation with a positive Moderator Temperature Coefficient (MTC) at reduced power levels. The proposed change would allow greater flexibility in core designs at NA-1&2 for future cycles.

Discussion:

The proposed change would also minimize the necessity of having the control rods significantly inserted in the core during initial startup and the potential for operating restrictions due to the delta flux limits associated with constant axial offset control. The licensee proposed a TS which allows a MTC of $+6 \times 10^{-5}$ ΔK/K/°F for power levels below 70 percent of rated power and a zero coefficient for power levels 70 percent and above. The power dependent MTC was proposed to minimize the effect on accidents initiated from high power levels.

The present NA-1&2 TS does not allow the reactor to be critical unless the MTC is negative, except during physics tests. Design calculations for recent NA cycles indicate that a positive MTC may occur at the beginning of cycle for hot zero power conditions with all rods removed from the core. While control rod insertion may be used to make the coefficient negative, this makes startup more complicated and takes longer.

As power level increases, the allowed average coolant temperature becomes higher and the MTC becomes more negative. Also the boron concentration is reduced as xenon builds into the core. Thus a positive MTC is not needed as full power is approached. As fuel burnup is achieved, boron is further reduced and the MTC becomes more negative over the entire operating power range. It is expected that the MTC would be positive only for low powers at beginning of cycle.

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Evaluation:

The licensee reanalyzed those NA-1&2 Updated Final Safety Analysis Report (UFSAR) Chapter 15 incidents which were sensitive to minimum or near zero MTC. All the reanalysis was done with a MTC of 6×10^{-5} ΔK/K/°F. No credit was taken for change in MTC due to increases in temperature or power. In general, the reanalysis was based on the assumptions and methods used for the UFSAR Accidents. The accidents not reanalyzed include those resulting in excessive heat removal from the reactor coolant system (for which a large negative MTC is limiting) and those which experience heatup following a reactor trip (which are not sensitive to the MTC).

The following transients were found to be not affected by a positive moderator coefficient.

- Rod Cluster Control Assembly Misalignment
- Startup of an Inactive Reactor Coolant Loop
- Excessive Heat Removal Due to Feedwater System Malfunctions
- Excessive Load Increase
- Loss of Normal Feedwater, Loss of Offsite Power to Station Auxiliaries
- Accidental Depressurization of the Reactor Coolant System
- Rupture of a Main Steam Pipe/Accidental Depressurization of the Main Steam System
- Spurious Operation of Safety Injection
- Rupture of a Main Feedwater Pipe
- Loss of Coolant Accident

Transients Sensitive to a Positive Moderator Coefficient

Uncontrolled Boron Dilution

Boron dilution at power causes an increase in power and reactor coolant system temperature if the reactor is in manual control. With a positive MTC, the temperature increase would result in adding additional reactivity and increasing the severity of the transient. However, this incident is less severe than the rod withdrawal at power and therefore is bounded by the results of that analysis.

Control Rod Withdrawal from a Subcritical Condition

A control rod assembly withdrawal incident when the reactor is subcritical results in an uncontrolled addition of reactivity leading to a power excursion. The analysis showed that the peak heat flux, peak coolant temperature and thermal power did not exceed normal full power values. Since the heat flux does not exceed the normal full power value and remains bounded by the UFSAR results, the conclusions presented in the UFSAR are still valid.

Uncontrolled Control Rod Assembly Withdrawal at Power

An uncontrolled control rod assembly withdrawal at power produces a mismatch in steam flow and core power, resulting in an increase in reactor coolant temperature. A positive MTC increases the power mismatch and reduces margin to DNB. The event was analyzed with a $+6 \times 10^{-5}$ Δ/K/K°F MTC, even though a positive MTC would be allowed only below 70 percent of power. The minimum DNBR was found to be greater than the limit of 1.3 for the earlier range of reactivity insertion values. Thus it was demonstrated that the positive MTC did not lower the DNBR associated with control rod assembly withdrawal at power to below the design limit.

Loss of Reactor Coolant Flow

The most severe loss of flow transient is caused by simultaneous loss of electric power to all three reactor coolant pumps. For the case reanalyzed, the reactor coolant average temperature increase was less than 3°F above the initial value. Analysis with the positive MTC showed that the minimum DNBR was greater than 1.30. Thus it was demonstrated that the results of the complete loss of flow transient remain above the 1.30 limit for DNBR.

Locked Rotor

The UFSAR shows that the most severe locked rotor incident is an instantaneous seizure of a reactor coolant pump rotor at 100 percent power with three loops operating. The transient was reanalyzed because of the potential effect on the peak reactor coolant system pressure and fuel temperature. The analysis used $+6 \times 10^{-5}$ ΔK/K/°F MTC. The results of the analysis showed that less than 2 percent of the fuel rods experienced departure from nucleate boiling (DNB) and that the peak clad temperature reached was 2250°F. This assures that the fuel damage will be minimal, the offsite radioactive release will be a small fraction (less than 10 percent) of the 10 CFR 100 guidelines, and that no loss of core cooling capability will result. The analyses showed that the maximum pressure within the reactor coolant and main steam system did not exceed 110 percent of the design pressures. Thus is was demonstrated that the analysis results are acceptable.

Loss of External Electrical Load

The UFSAR cases analyzed for both beginning and end of life conditions are:

1) Reactor in manual rod control with operation of the pressurizer spray and the pressurizer power operated relief valves and

2) Reactor in manual rod control with no credit for pressurizer spray or pressurizer power operated relief valves.

Since the MTC will be negative at end of life, only the beginning of life cases were reanalyzed. The positive MTC will cause increases in both peak nuclear power and pressurizer pressure. For the first case the reactor trips on high pressurizer pressure. The maximum pressurizer pressure reaches 2520 psia. The minimum DNBR is reached shortly after reactor trip and is greater than 1.30. For the second case peak pressurizer pressure reaches 2546 psia and the minimum DNBR increases from its initial value throughout the transient. Since in both cases the DNB ratio remains well above the 1.3 level and the peak reactor coolant pressure is less than 110 percent of design, the conclusions presented in the UFSAR are still applicable.

Rupture of a Control Rod Drive Mechanism Housing, Control Rod Ejection

The rod ejection transient is analyzed at full power and hot standby. The reactivity addition increases nuclear power and hot spot fuel temperatures. The limiting peak hot spot clad temperature, 2493°F, and the minimum fuel temperature were reached in the hot full power transient. The peak fuel and clad temperatures do not exceed the fuel and clad limits as outlined in the licensee's rod ejection topical (VEP-NFE-2). Since these criteria are more conservative than the requirement of General Design Criterion 28, the results are acceptable.

To evaluate the effect on operation of NA-1&2 with a slightly positive moderator temperature coefficient, a safety analysis of transients sensitive to a zero or positive MTC was performed. This study indicated that the small moderator temperature coefficient does not result in the violation of safety limits for the transients analyzed. Thus it was concluded that the change to a positive MTC will not cause safety limits to be exceeded. We have reviewed the licensee's submittal and agree with this conclusion. Therefore, we find the proposed NA-1&2 TS change to be acceptable for a full power level of 2775 MWt core power with a maximum reactor coolant system average temperature of 587.8°F.

Environmental Consideration

These amendments involve a change in the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously published a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR §51.22(c)(9). Pursuant to 10 CFR §51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

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

Principal Contributor: M. Chatterton

Dated: December 13, 1985