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

March 3, 1992

Docket No. 50-338

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

Dear Mr. Stewart:

SUBJECT: NORTH ANNA UNIT 1 - ISSUANCE OF AMENDMENT RE: MAXIMUM REACTOR POWER (TAC NO. M82674)

The Commission has issued the enclosed Amendment No. 153 to Facility Operating License No. NPF-4 for the North Anna Power Station, Unit No. 1 (NA-1). The amendment revises the Technical Specifications (TS) in response to your letter dated January 28, 1992, as supplemented February 27, 1992.

This amendment limits maximum reactor power to 95% of rated thermal power and imposes more restrictive equipment operability requirements for the Emergency Core Cooling System. The changes will remain in effect until steam generator replacement in 1993.

Your letter of February 27, 1992, requested that the amendment be issued on March 3, 1992, prior to the end of the 30-day notice period. Your letter stated that the steam generator tube inspection and plugging processes have been performed more rapidly than expected, and NA-1 is now scheduled to restart on March 3, 1992. In addition, NA-2 was shut down on February 26, 1992, and Surry Unit 1 was shut down on February 28, 1992. If the amendment is not issued to support a timely startup of NA-1, you could be faced with a potentially adverse power supply situation with three of the four nuclear units out of service. Due to these changed circumstances, the staff has determined that the amendment can be issued prior to the end of the 30-day notice period.

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A copy of the Safety Evaluation is also enclosed. The Notice of Issuance and Final Determination of No Significant Hazards Consideration will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/s/

Leon B. Engle, Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 153 to NPF-4
2. Safety Evaluation

cc w/enclosures:
See next page

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NAME	:D. Miller	:L. Engle	:H. Berkow	:	:	:
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AMENDMENT NO. 153 : TO FACILITY OPERATING LICENSE NO. NPF-4-NORTH ANNA UNIT 1

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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. 153
License No. NPF-4

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

   A. The application for amendment by Virginia Electric and Power Company, et al., (the licensee) dated January 28, 1992, as supplemented February 27, 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, license condition 2.D.(1) to Facility Operating License NPF-4 is modified to read as follows:**

   (1) Maximum Power Level

   VEPCO is authorized to operate the North Anna Power Station, Unit No. 1, at reactor core power levels not in excess of 2893 megawatts (thermal).*

   *The maximum reactor power level shall be limited to 2748 megawatts (thermal) which is 95% of RATED THERMAL POWER in accordance with the licensee's submittal dated January 28, 1992 (Serial No. 92-042) for the period of operation until the steam generator replacement.

3. In addition, 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. 153 , are hereby incorporated in the license. VEPCO shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

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

Attachments:
1. Page 4 of License
2. Changes to the Technical Specifications

Date of Issuance:

**Page 4 is attached, for convenience, for the composite license to reflect this change.

(1) Maximum Power Level

VEPCO is authorized to operate the North Anna Power Station, Unit No. 1, at reactor core power levels not in excess of 2893 megawatts (thermal).*

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 153, are hereby incorporated in the license. VEPCO shall operate the facility in accordance with the Technical Specifications.

(3) Additional Conditions

The matters specified in the following conditions shall be completed to the satisfaction of the Commission within the stated time periods following the issuance of this amendment or within the operational restrictions indicated. The removal of these conditions shall be made by an amendment to the license supported by a favorable evaluation by the Commission:

c. Virginia Electric and Power Company shall not operate the reactor in operational modes 1 and 2 with less than three reactor coolant pumps in operation.

d. VEPCO may use two (2) fuel assemblies containing fuel rods clad with an advanced zirconium base alloy cladding material as described in the licensee's submittals dated February 20, 1987 and September 30, 1988.

e. If Virginia Electric and Power Company plans to remove or to make significant changes in the normal operation of equipment that controls the amount of radioactivity in effluents from the North Anna Station, the Commission shall be notified in writing regardless of whether the change affects the amount of radioactivity in the effluents.

*The maximum reactor power level shall be limited to 2748 megawatts (thermal) which is 95% of RATED THERMAL POWER in accordance with the licensee's submittal dated January 28, 1992 (Serial No. 92-042) for the period of operation until the steam generator replacement.

ATTACHMENT TO LICENSE AMENDMENT NO. 153

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.

Remove Page	Insert Page
3/4 5-3	3/4 5-3

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE low head safety injection pump,

c. An OPERABLE flow path capable of transferring fluid to the Reactor Coolant System when taking suction from the refueling water storage tank on a safety injection signal or from the containment sump when suction is transferred during the recirculation phase of operation or from the discharge of the outside recirculation spray pump.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours. *

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

c. The provisions of Specifications 3.0.4 are not applicable to 3.5.2.a and 3.5.2.b for one hour following heatup above 324°F or prior to cooldown below 324°F.

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* Adherence to ACTION "a" shall require the following equipment OPERABILITY for the period of operation until steam generator replacement:

- With one low head safety injection pump inoperable, two centrifugal charging pumps (one in each subsystem) and their associated flow paths shall be OPERABLE or be in HOT STANDBY within the next 6 hours, and be in HOT SHUTDOWN within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
a. MOV-1890A	a. LHSI to hot leg	a. closed
b. MOV-1890B	b. LHSI to hot leg	b. closed
c. MOV-1836	c. Ch pump to cold leg	c. closed
d. MOV-1869A	d. Ch pump to hot leg	d. closed
e. MOV-1869B	e. Ch pump to hot leg	e. closed

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2. Of the areas affected within containment at the completion of each containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

1. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

e. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on a safety injection test signal.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 153

FACILITY OPERATING LICENSE NO. NPF-4

VIRGINIA ELECTRIC AND POWER COMPANY

OLD DOMINION ELECTRIC COOPERATIVE

NORTH ANNA POWER STATION, UNIT NO. 1

DOCKET NO. 50-338

1.0 INTRODUCTION

By letter dated January 28, 1992, as supplemented February 27, 1992, the Virginia Electric and Power Company (the licensee) proposed changes to the Technical Specifications (TS) for the North Anna Power Station, Unit No. 1 (NA-1). Specifically, the proposed changes would increase the steam generator tube plugging (SGTP) limit value up to 35% for the most restrictive SG. The proposed changes to the operating license would limit maximum reactor power to 95% of rated thermal power for the interim period of operation until SG replacement in 1993, by adding a footnote to license condition 2.D.(1), Maximum Power Level, which states that maximum reactor power level shall be limited to 95% of rated thermal power for the period of operation until SG replacement in 1993. The proposed changes to the TS would also impose more restrictive equipment operability requirements for the Emergency Core Cooling System (ECCS) by adding a footnote to Action Statement "a" of TS 3.5.2, "ECCS Subsystems - Tavg greater than 350°F," which requires that the charging pump in each ECCS subsystem be operable to comply with the requirements of the action statement if either low head safety injection pump is inoperable. These proposals are necessary to accommodate the interim effects of increased SGTP on the large break loss of coolant accident (LOCA) analysis.

NA-1 is currently involved in a mid-cycle SG inspection outage. An extensive eddy current inspection of the NA-1 SG tubes is being performed using conservative analysis guidelines and plugging criteria. A substantially increased number of tubes are expected to be plugged.

By letter dated February 27, 1992, the licensee requested that the amendment be issued on March 3, 1992, but noted that the 30-day notice period does not end until March 6, 1992. However, the steam generator tube inspection and plugging processes have been performed more rapidly than expected, and NA-1 is now scheduled to restart on March 3, 1992. In addition, NA-2 was shut down on February 26, 1992, and Surry Unit 1 was shut down on February 28, 1992.

If the amendment is not issued to support a timely startup of NA-1, the licensee could be faced with a potentially adverse power supply situation with three of the four nuclear units out of service. Due to these changed circumstances, the staff has determined that the amendment can be issued prior to the end of the 30-day notice period.

2.0 EVALUATION

There are a number of areas of plant design which are potentially impacted by the operation with extended SGTP. Westinghouse performed reviews of components and systems within their design responsibility to confirm that operation with the proposed conditions remain in compliance with the applicable codes and standards. Westinghouse concluded that all Nuclear Steam Supply System (NSSS) and components will remain within the bounds of existing design analysis results for operation with up to 40% of the tubes plugged in any or all SGs. Stone & Webster Engineering Corporation evaluated balance of plant (BOP) systems and components to determine the effect of extended SGTP operation. They concluded that the effect on operation with extended SGTP will remain within the bounds of existing design analyses for operation with up to 37% average SGTP.

The licensee assessed the impact of extended SGTP operation upon the NSSS accident analyses. With the exception of the large break LOCA, the existing analyses are valid for operation of NA-1 at rated thermal power of 2893 MWt with up to 35% SGTP in any or all SGs. The licensee performed a reanalysis of the ECCS performance for the postulated large break LOCA in compliance with the Appendix K of 10 CFR 50.46. This analysis was performed with the NRC-approved version of the Westinghouse ECCS-LOCA evaluation model, BASH, WCAP-10266-P-A, Rev. 2 "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," March 1987. The analytical techniques are in full compliance with 10 CFR 50.46, Appendix K. Based on sensitivity studies in WCAP-8356, "Westinghouse ECCS Plant Sensitivity Studies," July 1974, the licensee postulated a double-ended cold leg guillotine pipe break as the most limiting case. The analysis assumed that 35% of the tubes in each SG are plugged which resulted in a reduced RCS total flowrate of 264,400 gpm. This value bounds the expected RCS flow associated with 35% SGTP. In addition, Westinghouse sensitivity studies set forth in WCAP-8471-P-A, "The Westinghouse ECCS Evaluation Model: Supplementary Information," April 1975, have demonstrated that the limiting single failure is the assumption that one low head safety injection pump fails. This assumption, combined with Appendix K requirements, leaves flow available from two high head and one low head safety injection pumps and flow from both containment spray systems.

Using these assumptions in the BASH ECCS evaluation model, it was determined that operation at maximum power of 2748 MWt (i.e., 95% of rated thermal power) with SGTP of up to 35% in any or all SGs will comply with the 10 CFR 50.46, Appendix K criteria. The LOCA reanalysis results show that a peak cladding temperature of 2140.8°F, a maximum local cladding oxidation level of 7.22% and a total core metal-water reaction of less than 1% will satisfy Appendix K criteria.

3.0 SUMMARY

Based on the licensee evaluation of NSSS/components, BOP/components and a reanalysis of LOCA, the NRC staff concludes that the proposed TS changes are acceptable.

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards considerations if operation of the facility in accordance with the amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The Commission has determined that the amendment involves no significant hazards consideration per 10 CFR 50.92, based on the licensee's analysis provided in their January 28, 1992 letter and presented below:

1. [The proposed change] does not involve a significant increase in the probability or consequences of an accident previously evaluated. The impact of the increased level of [SG] tube plugging (up to 35% peak) with a maximum reactor power of 95% on the large break LOCA was analyzed. The analysis demonstrated that operation with increased [SG] tube plugging will not result in more severe consequences than those of the currently applicable analyses. The probability of occurrence of these accidents is not increased, because an increased level of [SG] tube plugging as an initial condition for the accident has no bearing on the probability of occurrence of these accidents.

2. [The proposed change] does not create the possibility of a new or different kind of accident from any accident previously evaluated. The implementation of the increased [SG] tube plugging large break LOCA analysis into the [NA-1] design basis will not create the possibility of an accident of a different type than was previously evaluated in the [Updated Final Safety Analysis Report (UFSAR)]. No changes to plant configuration or modes of operation are implemented by the revised accident analysis. Therefore, no new mechanisms for the initiation of accidents are created by the implementation of the analysis.

3. [The proposed change] does not involve a significant reduction in a margin of safety. The [NA-1] operating characteristics, and accident analyses which support [NA-1] operation, have been fully assessed. The results of the revised large break LOCA analysis [demonstrate] that the consequences of this accident are not

increased as a result of the increased [SG] tube plugging up to 35% with a maximum reactor power of 95%. The results of the accident analysis remain below the limits established by the currently applicable [UFSAR] analyses. Therefore, there is no significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, concludes that the analysis demonstrates that the applicable criteria are met. Accordingly, the Commission has made a final determination that the amendment involves no significant hazards consideration.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding (57 FR 4503). Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

7.0 CONCLUSION

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

Principal Contributor: K. Desai

Date: March 3, 1992