March 6, 1996

Mr. J. P. O'Hanlon
Senior Vice President - Nuclear
Virginia Electric and Power Company
5000 Dominion Blvd.
Glen Allen, Virginia 23060

SUBJECT: NORTH ANNA UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: REVISED NEUTRON FLUX HIGH TRIP SETPOINTS WITH INOPERABLE MAIN STEAM SAFETY VALVES (SERIAL NO. 95-463) (TAC NOS. M93716 AND M93717)

Dear Mr. O'Hanlon:

The Commission has issued the enclosed Amendment Nos. 199 and 180 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 September 19, 1995.

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:

Gordon E. Edison, Sr. Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-338
and 50-339

Enclosures:
1. Amendment No. 199 to NPF-4
2. Amendment No. 180 to NPF-7
3. Safety Evaluation

cc w/enclosures: See next page
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COPY	Yes/No	Yes/No	Yes/No	Yes/No	Yes/No

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Virginia Electric & Power Company

North Anna Power Station
Units 1 and 2

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DATED: March 6, 1996

AMENDMENT NO. 199 TO FACILITY OPERATING LICENSE NO. NPF-4-NORTH ANNA UNIT 1
AMENDMENT NO. 180 TO FACILITY OPERATING LICENSE NO. NPF-7-NORTH ANNA UNIT 2

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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. 199
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 September 19, 1995, 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.D.(2) of Facility Operating License No. NPF-4 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 199, 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

David C. Trimble for

Eugene V. Imbro, Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 6, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 199

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 amendment number and contain vertical lines indicating the area of change.

Remove Pages	Insert Pages
3/4 7-1	3/4 7-1
3/4 7-2	3/4 7-2
3/4 7-3	3/4 7-3
B 3/4 7-1	B 3/4 7-1
B 3/4 7-2	B 3/4 7-2

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves associated with each steam generator of an unisolated reactor coolant loop shall be OPERABLE with lift settings as specified in Table 3.7-2.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one or more main steam line code safety valves inoperable, operation in MODES 1, 2 and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Setpoint trip is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	52
2	37
3	21

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3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensure that the secondary system pressure will be limited to within 110% of the system design pressure, during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all safety valves on all of the steam lines is 12.83 x 10^6 lbs/hr which is greater than the total secondary steam flow of 12.77 x 10^6 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are derived from the following conservative calculation such that the maximum power level allowed for operation with inoperable MSSVs is below the heat removing capability of the operable MSSVs.

In order to calculate these setpoints, the governing equation is the relationship: $q = m\,\Delta h$, where q is the heat input from the primary side, m is the steam flow rate, and Δh is the heat of the vaporization at the steam relief pressure. Therefore, the equation used in defining the revised setpoint values is:

$$\text{Hi } \Phi = \frac{100}{Q} \times \frac{(w_g \cdot h_{fg} \cdot N)}{K}$$

Where:

Hi Φ = Safety analysis power range high neutron flux setpoint, percent

Q = Nominal NSSS power rating of the plant (including reactor coolant pump heat, MWt

K = Conversion factor, 947.82 (Btu / sec) / MWt

w_g = Minimum total steam flow rate capability of the operable MSSVs on any one steam generator at the highest MSSV operating pressure including tolerance and accumulation, as appropriate, lbm / sec

h_{fg} = Heat of vaporization for steam at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, Btu / lbm

N = Number of loops in plant

The resulting values calculated from this equation are reduced by 9% power to account for instrument and channel uncertainties. With the revised values, the maximum plant operating power level would be lower than the reactor protection system setpoint by an appropriate operating margin.

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss of off-site power.

The original design basis of the AFW system provided for two motor driven AFW pumps (AFWP) each capable of delivering 340 gpm and a single turbine driven AFWP capable of delivering 700 gpm to the steam generators during accident conditions. The design basis accidents for the AFW system are the loss of normal feedwater (LONF), the loss of offsite power (LOOP), which are ANS Condition II events, and the main feedline break (MFLB), which is an ANS Condition IV event.

Current analyses of the design basis accidents for the AFW system have shown that the applicable accident analysis acceptance criteria are met, including the effects of a single active failure of any AFWP to start, if each AFWP is capable of delivering ≥ 300 gpm to its respective steam generator at the safety valve set pressure (including the effects of setpoint drift).

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. 180
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 September 19, 1995, 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. 180 , 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

David C. Trimble for

Eugene V. Imbro, Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 6, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 180

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.

Remove Pages	Insert Pages
3/4 7-1	3/4 7-1
3/4 7-2	3/4 7-2
3/4 7-3	3/4 7-3
B 3/4 7-1	B 3/4 7-1
B 3/4 7-2	B 3/4 7-2

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves associated with each steam generator of an unisolated reactor coolant loop shall be OPERABLE with lift settings as specified in Table 3.7-2.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one or more main steam line code safety valves inoperable, operation in MODES 1, 2 and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Setpoint trip is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	52
2	37
3	21

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3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensure that the secondary system pressure will be limited to within 110% of the system design pressure, during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all safety valves on all of the steam lines is 12.83×10^6 lbs/hr which is greater than the total secondary steam flow of 12.77×10^6 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1. .

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are derived from the following conservative calculation such that the maximum power level allowed for operation with inoperable MSSVs is below the heat removing capability of the operable MSSVs.

In order to calculate these setpoints, the governing equation is the relationship: $q = m\,\Delta h$, where q is the heat input from the primary side, m is the steam flow rate, and Δh is the heat of the vaporization at the steam relief pressure. Therefore, the equation used in defining the revised setpoint values is:

$$\text{Hi } \Phi = \frac{100}{Q} \times \frac{(w_g \cdot h_{fg} \cdot N)}{K}$$

Where:

Hi Φ = Safety analysis power range high neutron flux setpoint, percent

Q = Nominal NSSS power rating of the plant (including reactor coolant pump heat, MWt

K = Conversion factor, 947.82 (Btu / sec) / MWt

w_g = Minimum total steam flow rate capability of the operable MSSVs on any one steam generator at the highest MSSV operating pressure including tolerance and accumulation, as appropriate, lbm / sec

h_{fg} = Heat of vaporization for steam at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, Btu / lbm

N = Number of loops in plant

The resulting values calculated from this equation are reduced by 9% power to account for instrument and channel uncertainties. With the revised values, the maximum plant operating power level would be lower than the reactor protection system setpoint by an appropriate operating margin.

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss of off-site power.

The original design basis of the AFW system provided for two motor driven AFW pumps (AFWP) each capable of delivering 340 gpm and a single turbine driven AFWP capable of delivering 700 gpm to the steam generators during accident conditions. The design basis accidents for the AFW system are the loss of normal feedwater (LONF), the loss of offsite power (LOOP), which are ANS Condition II events, and the main feedline break (MFLB), which is an ANS Condition IV event.

Current analyses of the design basis accidents for the AFW system have shown that the applicable accident analysis acceptance criteria are met, including the effects of a single active failure of any AFWP to start, if each AFWP is capable of delivering ≥ 300 gpm to its respective steam generator at the safety valve set pressure (including the effects of setpoint drift).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 199 AND 180 TO

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

VIRGINIA ELECTRIC AND POWER COMPANY

OLD DOMINION ELECTRIC COOPERATIVE

1.0 INTRODUCTION

By letter dated September 19, 1995, Virginia Electric and Power Company (VEPCO), the licensee, requested changes to the Technical Specifications (TS) for North Anna Nuclear Power Plant, Units 1 and 2. The proposed changes would revise the maximum allowable power range neutron flux high setpoints for operation with inoperable Main Steam Safety Valves (MSSV). This modification of the TS has been proposed by the licensee, specifically, to address the concern expressed by the vendor, the Westinghouse Electric Corporation (Westinghouse), in its Nuclear Safety Advisory Letter, NSAL-94-001, "Operation at Reduced Power Levels with Inoperable Main Steam Safety Valves," issued on January 20, 1994.

2.0 EVALUATION

Westinghouse identified a potential concern in its NSAL-94-001 regarding plant operation within the limits established in the TS. Table 3.7-1 of the TS allows the plant to operate at reduced power levels with a reduced number of operable MSSVs. Westinghouse identified that the current method used for reducing the neutron flux high trip setpoints was potentially inadequate for protecting the main steam system from an overpressure condition following certain design transients. In particular, the assumption that the maximum allowable initial power level is a linear function of the available MSSV relief capacity was determined invalid by Westinghouse. The vendor further noted that, although it has not been shown directly that the reduced neutron flux high trip setpoints generated by this assumption would result in an overpressure condition of the main steam system, the potential exists due to the non-conservative nature of the assumption.

Five MSSVs are provided for each steam generator in North Anna (3 loop plant), and are designed to protect the integrity of the main steam piping from overpressurization. In the event one or more MSSVs are inoperable, the Reactor Protection System power range neutron flux high trip setpoints are reduced to ensure that the Main Steam System is not overpressurized as a result of various transients. A detailed analysis performed for a Loss of Load/Turbine Trip (LOL/TT), the bounding transient, as part of the North Anna Updated Final Safety Analysis Report (UFSAR) Chapter 15 for a full power condition with all MSSVs operable, determined that no overpressure condition would occur. The LOL/TT event was analyzed in the UFSAR to show that core protection margins (e.g., DNBR) are maintained, the Reactor Coolant System (RCS) will not overpressurize, and the main steam system will not overpressurize.

The secondary side overpressure protection is provided by actuation of the MSSVs, which are designed to relieve at least full power nominal steam flow. The UFSAR analysis verifies that the MSSVs capacity is sufficient to prevent secondary side pressure from exceeding 110 percent of the design pressure. It should be noted, however, that the UFSAR analyzes the LOL/TT transient only from the full power initial condition, with cases examining the effects of assuming primary side pressure control and different reactivity feedback conditions. With fully operational MSSVs, it can be demonstrated that overpressure protection is provided for all initial power levels. TS Table 3.7-1 allows operation with a reduced number of operable MSSVs at a reduced power level as determined by resetting the power range neutron flux high trip setpoints. This TS requirement was not based upon a detailed analysis, but as mentioned earlier, was based upon the assumption that the maximum allowable initial power level is a linear function of the available MSSV relief capacity. Thus, a subsequent detailed analysis for the LOL/TT event for individual cases of inoperable MSSVs was not performed at that time since the original full power analysis was determined to bound such an event. Westinghouse has determined that the assumption that a linearly reduced neutron flux high trip setpoint would limit the heat addition rate below the removal capacity of the remaining operable MSSVs is not valid, and is non-conservative. Therefore, the licensee performed new calculations, as recommended by NSAL-94-001, to support the proposed conservative setpoints. In lieu of a detailed analysis using the TS Table 3.7-1 power range neutron flux setpoint trip values to determine whether a true overpressure condition would result with one or more MSSVs inoperable, calculations were performed by the licensee such that the maximum power level allowed for operation with inoperable MSSVs is below the heat removing capability of the operable MSSVs. This calculation is based upon Westinghouse's recommendation, and represents the conservative methodology by setting the power range neutron flux high setpoint to this level, thus ensuring that the actual power level cannot exceed this value.

Furthermore, the information in Table 3.7-1 and the Limiting Condition for Operation Action Statement associated with the two loop operation have been deleted from the TS since the licensee is prohibited by the license from operating in this configuration.

In order to calculate the new setpoints, the governing equation is the relationship: $q = M \times Hfg$, where q is the heat input from the primary side, M is the steam flow rate, and Hfg is the heat of the vaporization at the steam relief pressure. Therefore, the equation used in defining the revised TS Table 3.7-1 setpoint values is:

$$Hi = \frac{(Wg \times Hfg \times N)}{K} \times \frac{100}{Q}$$

Where:

Hi = Safety analysis power range high neutron flux setpoint (percent)

Q = Nominal NSSS power rating of the plant (including reactor coolant pump heat, MWt)
K = Conversion factor, 947.82 (Btu/sec)/MWt
Wg = Minimum total steam flow rate capability of the operable MSSVs on any one steam generator at the highest MSSV operating pressure including tolerance and accumulation, as appropriate (lbm/sec)
Hfg= Heat of vaporization for steam at the highest MSSV opening pressure including tolerance and accumulation, as appropriate (Btu/lbm)
N = Number of loops in plant

The resulting values calculated from this equation were reduced by 9% power to account for instrument and channel uncertainties. With the revised values, the maximum plant operating power level would be lower than the reactor protection system setpoint by an appropriate operating margin. These revised values, by the use of the above equation, resolves the issue identified by the NSAL-94-001 by enabling the licensee to re-calculate and establish more restrictive power range neutron flux high setpoints as listed in the proposed changes in TS Table 3.7-1.

3.0 SPECIFIC CHANGES TO THE TS

The following specific TS changes apply to both Units 1 and 2:

Change Section 3.7.1.1, "Limiting Condition for Operation" Action Item "a" by deleting "With 3 reactor coolant loops and associated steam generators in operation and" and begin the sentence by capitalizing "With".

Delete the following Section 3.7.1.1, "Limiting Condition for Operation" Action Item "b" :
"With 2 reactor coolant loops and associated steam generators in operation and with one or more main steam line code safety valves associated with an operating loop inoperable, operation in MODES 1,2, and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Setpoint trip is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours."

Change Section 3.7.1.1, "Limiting Condition for Operation" Action Item "c" to Action Item "b".

The following changes were made to Table 3.7-1:
- Delete "During 3 Loop Operation" from title of table

- Change the Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER) for one inoperable safety valve in Table 3.7-1 from "87" to "52"

- Change the Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER) for two inoperable safety valves in Table 3.7-1 from "65" to "37"

- Change the Maximum Allowable Power Range Neutron Flux High Setpoint

(Percent of RATED THERMAL POWER) for three inoperable safety valves in Table 3.7-1 from "44" to "21"

- Delete all information which discusses two Loop Operation

In addition to the above changes, the licensee proposed to change the relevant TS Bases in order to be consistent with the proposed TS changes.

Based on the above evaluation, the staff has concluded that the proposed reactor trip setpoint reductions were derived from conservative calculations such that the maximum power level allowed for operation with inoperable MSSVs is below the heat removing capability of the operable MSSVs. The conservatism of the proposed methodology was further verified by noting that the resulting setpoint values are lower than the current setpoints in the TS (see the proposed changes in the TS Table 3.7-1). Therefore, the proposed changes in the TS, and the associated Bases are 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

These amendments change 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 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 issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding (60 FR 54724). 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 the amendments.

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

Principal Contributor: M. Razzaque

Date: March 6, 1996