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Docket Nos. 50-338
and 50-339

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. 46 and 29 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), respectively. The amendments consist of changes to the NA-1&2 Technical Specifications in response to your application (Serial No. 662) dated December 8, 1982 and our discussions with you regarding your application. The amendments are effective as of the date of issuance.

The amendments revise the partial power multiplier for $F^N_{\Delta H}$ from 0.2 to 0.3. Also, the NA-2 Technical Specification 4.2.2.2.g.2 is deleted to provide consistency between the NA-1&2 Technical Specifications. However, as requested in your above noted submittal, we do not find the change in the partial power multiplier for F_{xy} to be acceptable at this time. This matter has been previously discussed with the appropriate VEPCO representatives during our review of your application.

Copies of the related Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original signed by

Leon B. Engle, Project Manager
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 46 to NPF-4
2. Amendment No. 29 to NPF-7
3. Safety Evaluation
4. Notice of Issuance

cc w/enclosures:
See next page

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DATE	4/19/83	4/18/83	4/18/83	4/18/83	4/20/83		

NRC FORM 318 (10-80) NRCM 0240

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

DISTRIBUTION:
Docket File
ORB#3 Rdg
PMKreutzer

Docket No. 50-338/50-339

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: VIRGINIA EEECTRIC AND POWER COMPANY, North Anna Power Station, Units No. 1 and No. 2

Two signed originals of the Federal Register Notice identified below are enclosed for your transmittal to the Office of the Federal Register for publication. Additional conformed copies (12) of the Notice are enclosed for your use.

- ☐ Notice of Receipt of Application for Construction Permit(s) and Operating License(s).
- ☐ Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s): Time for Submission of Views on Antitrust Matters.
- ☐ Notice of Availability of Applicant's Environmental Report.
- ☐ Notice of Proposed Issuance of Amendment to Facility Operating License.
- ☐ Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.
- ☐ Notice of Availability of NRC Draft/Final Environmental Statement.
- ☐ Notice of Limited Work Authorization.
- ☐ Notice of Availability of Safety Evaluation Report.
- ☐ Notice of Issuance of Construction Permit(s).
- ☐ Notice of Issuance of Facility Operating License(s) or Amendment(s).
- ☒ Other: Amendment Nos. 46 and 29
  Referenced documents have been provided PDR.

Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

OFFICE	ORB#3:DL					
SURNAME	PMKreutzer/pn					
DATE	4/21/83					

NRC FORM 102 7—79

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Appeal Board Panel
U.S. Nuclear Regulatory Commission
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Mr. James C. Dunstance
State Corporation Commission
Commonwealth of Virginia
Blandon Building
Richmond, Virginia 23209

Mrs. June Allen
North Anna Environmental Coalition
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U.S. Environmental Protection Agency
Region III Office
ATTN: Regional Radiation Representative
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6th and Walnut Streets
Philadelphia, Pennsylvania 19106

Regional Administrator
Nuclear Regulatory Commission, Region II
Office of Executive Director for Operations
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-338

NORTH ANNA POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 46
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 (the licensee) dated December 8, 1982, 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) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 46 , 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

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 22, 1983

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 46 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. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

2-2
2-9
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2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figures 2.1-1 for 3 loop operation and 2.1-2 and 2.1-3 for 2 loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

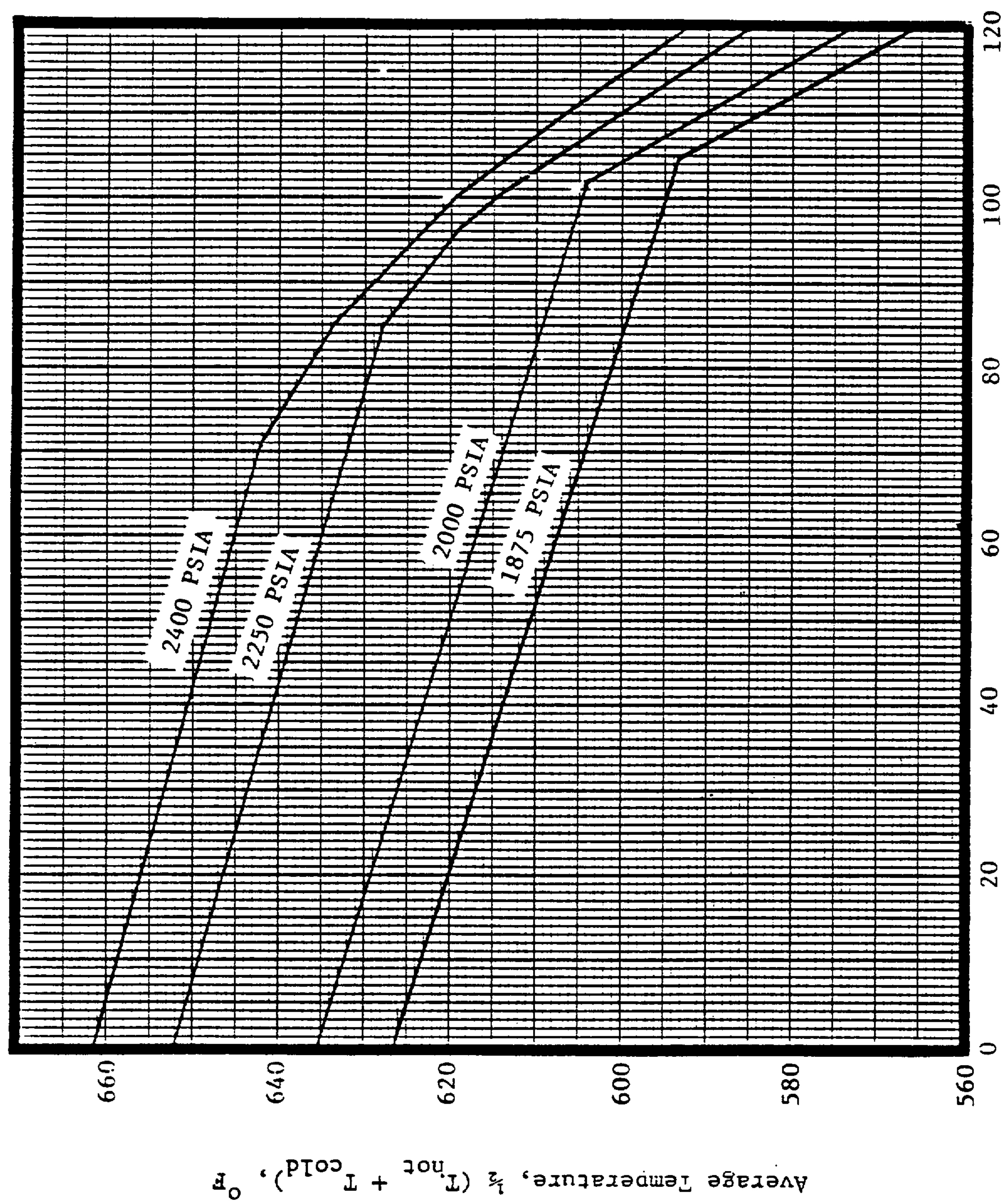

Figure 2.1-1 Reactor Core Safety Limits for Three Loop Operation,

100% Flow

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Operation with 3 Loops	Operation with 2 Loops (no loops isolated)*	Operation with 2 Loops (1 loop isolated)*
K_1 = 1.113	K_1 = ( )	K_1 = ( )
K_2 = 0.0132	K_2 = ( )	K_2 = ( )
K_3 = 0.000628	K_3 = ( )	K_3 = ( )

and f_1 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between - 35 percent and + 7 percent, $f_1(\Delta I) = 0$ (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) for each percent that the magnitude of $(q_t - q_b)$ exceeds - 35 percent, the ΔT trip setpoint shall be automatically reduced by 1.58 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of $(q_t - q_b)$ exceeds + 7 percent, the ΔT trip setpoint shall be automatically reduced by 1.24 percent of its value at RATED THERMAL POWER.

*Values dependent on NRC approval of ECCS evaluation for these operating conditions.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_o[K_4 - K_5 \left(\frac{\tau_3 S}{1+\tau_3 S}\right) T - K_6 (T - T'') - f_2(\Delta I)]$

Where:

ΔT_o	=	Indicated ΔT at RATED THERMAL POWER
T	=	Average temperature, °F
T''	=	Indicated T_{avg} at RATED THERMAL POWER $\leq$ 582.8°F
K_4	=	1.088
K_5	=	0.02/°F for increasing average temperature
K_5	=	0 for decreasing average temperatures
K_6	=	0.00119 for $T > T''$; $K_6 = 0$ for $T \leq T''$
$\frac{\tau_3 S}{1+\tau_3 S}$	=	The function generated by the rate lag controller for T_{avg} dynamic compensation
τ_3	=	Time constant utilized in the rate lag controller for T_{avg} τ_3 = 10 secs.
S	=	Laplace transform operator (sec^{-1})
$f_2(\Delta I)$	=	0 for all ΔI

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 2 percent span.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the W-3 correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figures 2.1-1, 2.1-2 and 2.1-3 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than 1.30, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

The curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.55 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.55\ [1 + 0.3\ (1-P)]$$

where P is the fraction of RATED THERMAL POWER

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f(ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the setpoints to provide protection consistent with core safety limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Reactor Coolant System piping, valves and fittings, were initially designed to ANSI B 31.1 1967 Edition and ANSI B 31.7 1969 Edition (Table 5.2.1-1 of FSAR) which permits a maximum transient pressure of 120% (2985 psig) of component design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$$F^N_{\Delta H} \leq 1.55\,[1 + 0.3\,(1-P)]\,[1 - RBP\,(BU)]$$

where $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

RBP (BU) = Rod Bow Penalty as a function of region average burnup as shown in Figure 3.2-3, where a region is defined as those assemblies with the same loading date (reloads) or enrichment (first cores).

APPLICABILITY: MODE 1

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to $\leq$ 55% of RATED THERMAL POWER within the next 4 hours,

b. Demonstrate through in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a or b, above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL power and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 $F^N_{\Delta H}$ shall be determined to be within its limit by using the movable incore detectors to obtain a power distribution map:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

c. The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The measured $F^N_{\Delta H}$ of 4.2.3.1 above, shall be increased by 4% for measurement uncertainty.

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-339

NORTH ANNA POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 29
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 (the licensee) dated December 8, 1982, 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 in paragraph 2.C.(2) of Facility Operating License No. NPF-7 and pagagraph 2.C.(2) is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 29 , 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

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 22, 1983

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 29 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.

Pages

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3/4 2-9

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figures 2.1-1 for 3 loop operation and 2.1-2 and 2.1-3 for 2 loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

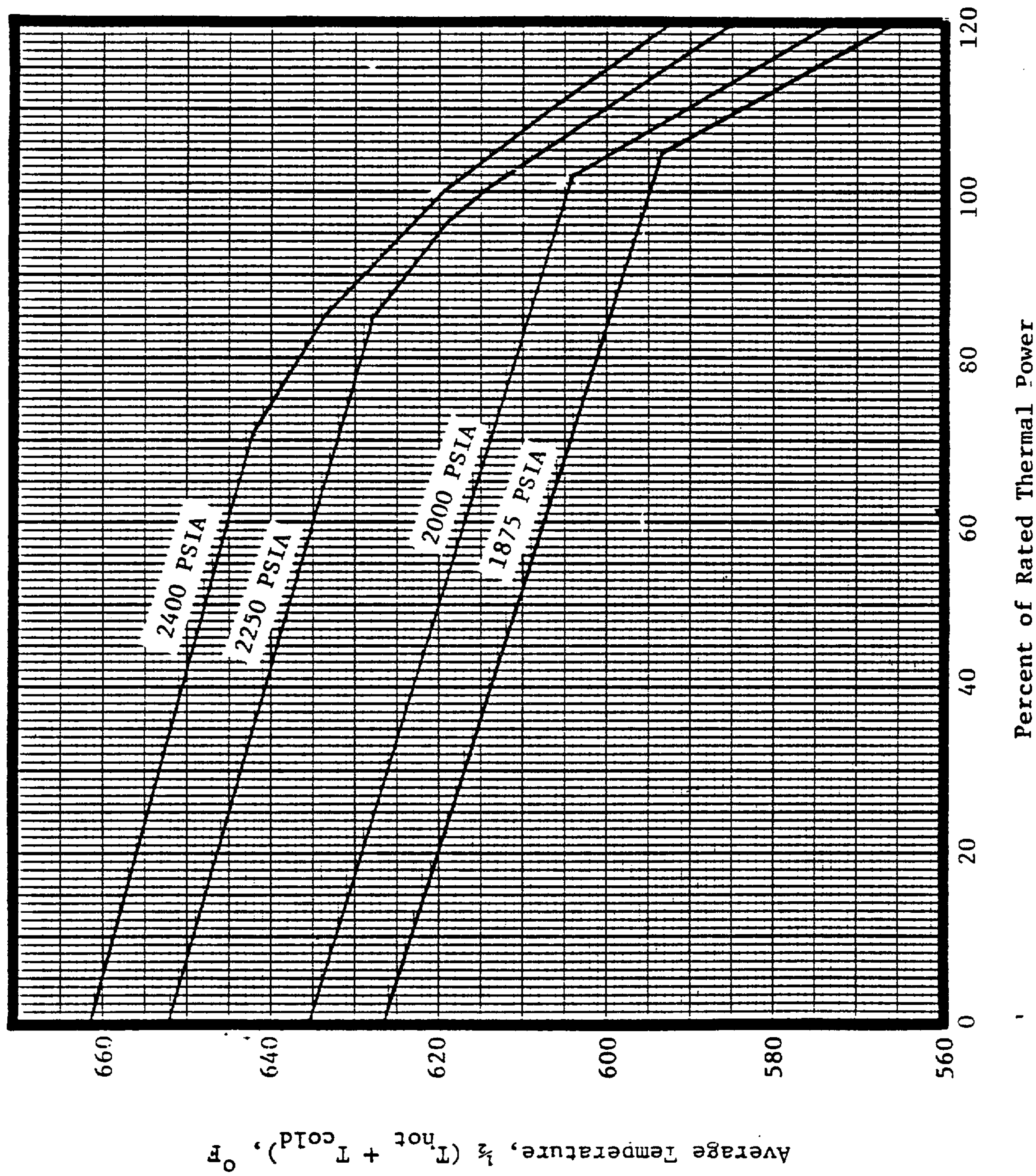

Figure 2.1-1 Reactor Core Safety Limits for Three Loop Operation,

100% Flow

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Operation with 3 Loops	Operation with 2 Loops (no loops isolated)*	Operation with 2 Loops (1 loop isolated)*
K_1 = 1.141	K_1 = ( )	K_1 = ( )
K_2 = 0.0128	K_2 = ( )	K_2 = ( )
K_3 = 0.000608	K_3 = ( )	K_3 = ( )

and f_1 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between - 35 percent and + 7 percent, f_1 (ΔI) = 0 (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) for each percent that the magnitude of ($q_t - q_b$) exceeds - 35 percent, the ΔT trip setpoint shall be automatically reduced by 1.56 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of ($q_t - q_b$) exceeds + 7 percent, the ΔT trip setpoint shall be automatically reduced by 1.21 percent of its value at RATED THERMAL POWER.

*Values dependent on NRC approval of ECCS evaluation for these operating conditions.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_o[K_4 - K_5 \left[\frac{\tau_3 S}{1+\tau_3 S}\right] T - K_6 \; (T-T") - f_2(\Delta I)]$

where: ΔT_o = Indicated ΔT at RATED THERMAL POWER

T = Average temperature, °F

$T"$ = Indicated T_{avg} at RATED THERMAL POWER $\leq$ 580.3°F

K_4 = 1.086

K_5 = 0.02/°F for increasing average temperature

K_5 = 0 for decreasing average temperatures

K_6 = 0.00116/°F for T > T"; K_6 = 0 for T $\leq$ T"

$\frac{\tau_3 S}{1+\tau_3 S}$ = The function generated by the rate lag controller for T_{avg} dynamic compensation

τ_3 = Time constant utilized in the rate lag controller for T_{avg} τ_3 = 10 secs.

S = Laplace transform operator (sec^{-1})

$f_2(\Delta I)$ = 0 for all ΔI

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 2 percent span.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the W-3 correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figures 2.1-1, 2.1-2 and 2.1-3 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNRB is no less than 1.30, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

The curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.55 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.55\ [1+0.3\ (1-P)]$$

where P is the fraction of RATED THERMAL POWER

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f(delta I) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the setpoints to provide protection consistent with core safety limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Reactor Coolant System piping, valves and fittings, were initially designed to ANSI B 31.1 1967 Edition and ANSI B 31.7 1969 Edition (Table 5.2.1-1 of FSAR) which permits a maximum transient pressure of 120% (2985 psig) of component design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

SURVEILLANCE REQUIREMENTS (Continued)

(b) At least once per 31 EFPD, whichever occurs first.

2. When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limits for Rated Thermal Power (F_{xy}^{RTP}) shall be provided for all core planes containing Bank "D" control rods and all unrodded core planes, in a Core Surveillance Report per Technical Specification 6.9.1.10.

f. The F_{xy} limits of e, above, are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:

1. Lower core region from 0 to 15%, inclusive.

2. Upper core region from 85 to 100%, inclusive.

3. Grid plane regions at 17.8 ±2%, 32.1 ±2%, 46.4±2%, 60.6±2% and 74.9±2%, inclusive (17 x 17 fuel elements).

4. Core plane regions within ±2% of core height (±2.88 inches) about the bank demand position of the bank "D" control rods.

g. With F_{xy}^{C} exceeding F_{xy}^{L}:

1. The effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit.

4.2.2.3 When $F_Q(Z)$ is measured for other than F_{xy} determination, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

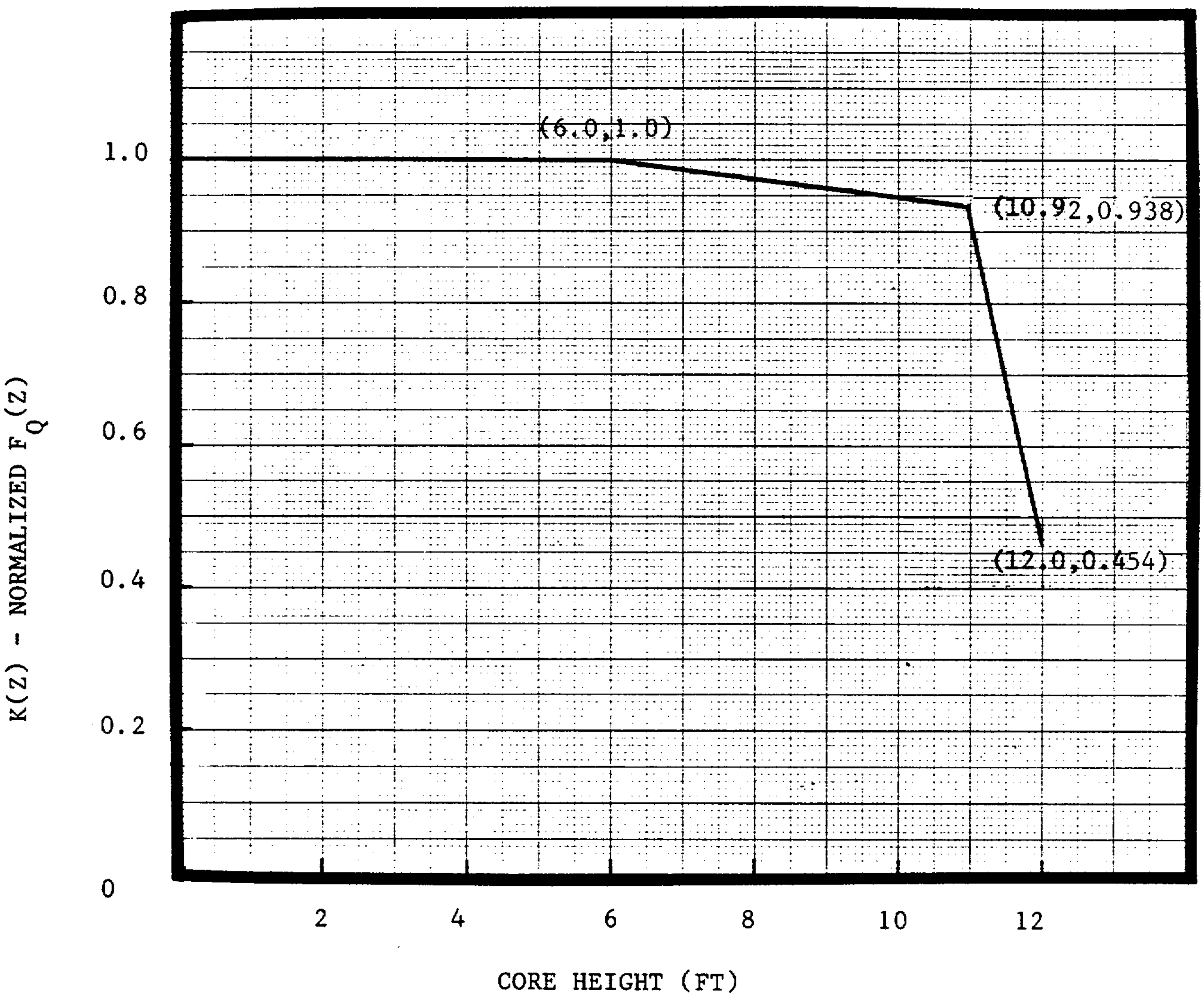

Figure 3.2-2 K(Z) - Normalized $F_Q(Z)$ as a Function of Core Height

NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$$F^N_{\Delta H} \leq 1.55 \, [1 + 0.3 \, (1-P)] \, [1-RBP \, (BU)]$$

where: $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

RBP (BU) = Rod Bow Penalty as a function of region average burnup as shown in Figure 3.2-3, where a region is defined as those assemblies with he same loading date (reloads) or enrichment (first cores).

APPLICABILITY: MODE 1.

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours,

b. Demonstrate through in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a. or b., above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to

ACTION Continued

exceeding this THERMAL POWER and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 $F^N_{\Delta H}$ shall be determined to be within its limit by using the movable incore detectors to obtain a power distribution map:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

c. The provisions of Specification 4.0.4 are not applicable.

4.2.3.2. The measured $F^N_{\Delta H}$ of 4.2.3.1 above, shall be increased by 4% for measurement uncertainty.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 46 AND 29 TO

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

VIRGINIA ELECTRIC AND POWER COMPANY

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

DOCKET NOS. 50-338 AND 50-339

Introduction:

By letter dated December 8, 1982 (Serial No. 622), the Virginia Electric and Power Company (the licensee) requested an amendment in the form of changes to the Technical Specifications (TS) for Operating License Nos. NPF-4 and NPF-7 for the North Anna Power Station, Unit Nos. 1 and 2 (NA-1&2), respectively.

The licensee's requested amendment would specifically change the existing F_{xy} and $F^N_{\Delta H}$ fractional power multiplier from 0.2 to 0.3. The proposed changes would allow optimization of core loading patterns and provide additional operating flexibility.

In addition, the licensee's requested change included an administrative change to the NA-2 TS in order to provide consistency between the NA-1 and NA-2 TS. Specifically, the one-for-one reduction of the $F_Q(z)$ limit, when the F_{xy} limit is exceeded would be eliminated since NA-2 does not use the Axial Power Distribution Monitoring System (APDMS) to ensure F_Q limit compliance.

A discussion and our evaluation of the licensee's proposed changes to the NA-1&2 TS is provided below.

Discussion:

(1) $F^N_{\Delta H}$ Technical Specification Change:

Historically, increasing the allowable $F^N_{\Delta H}$ with decreasing power has been permitted for all previously approved Westinghouse designs. The increase is permitted by the Departure from Nucleate Boiling (DNB) protection setpoints and allows for radial power distribution changes with rod insertion to the insertion limit. The change to a larger (0.2 to 0.3) partial power multiplier is requested for NA-1&2 to allow optimization of the core loading pattern by minimizing restrictions on $F^N_{\Delta H}$ at low power. This change will also minimize the probability of making rod insertion limit changes to satisfy peaking factor criteria at low power with the control rod banks at the insertion limit.

As a result of the multiplier change the thermal core limits at 2250 and 2400 psia are slightly more limiting below 100 percent power. Also new axial offset limits were calculated and there were minor changes to the $f(\Delta I)$ function. The new core limits and the changes to the $f(\Delta I)$ function were submitted. The analysis showed that no changes to the overpower ΔT and overtemperature ΔT setpoints or the K factors were necessary. Therefore no accident reanalysis was required.

We have approved the 0.3 partial power multiplier for $F^N_{\Delta H}$ for WCAP-9500 and other plants. The licensee's request for the NA-1&2 TS changes is similar. Based on our review we find this change acceptable.

(2) F_{xy} Technical Specification Change:

It was determined that there was not sufficient information available in the licensee's submittal for completing our review of the increase in the F_{xy} partial multiplier (0.2 to 0.3). The licensee, when advised that additional information was needed, requested that the other proposed TS changes be processed. Therefore, the licensee's proposed change in F_{xy} has been placed in abeyance until such time that the additional information is provided for our review.

(3) Technical Specification 4.2.2.2.g.2:

The licensee has requested the removal of TS 4.2.2.2.g.2 from the NA-2 TS. This TS is required for facilities that use the APDMS to ensure F_Q limit compliance. NA-2 does not use the APDMS, and therefore, we find this change to be acceptable.

Conclusion:

Based on our review of the licensee's submittal for the above noted changes, we find the change in the partial power multiplier $F^N_{\Delta H}$ (from 0.2 to 0.3) to be acceptable. In addition, we find the change to the NA-2 TS 4.2.2.2.g.2 acceptable. However, as stated above, we do not find the change in the partial multiplier for F_{xy} to be acceptable at this time.

Environmental Consideration

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated, do not create the possibility of an accident of a type different from any evaluated previously, and do not involve a significant reduction in a margin of safety, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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.

Date: April 22, 1983

Principal Contributors:
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7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-338 AND 50-339

VIRGINIA ELECTRIC AND POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 46 and 29 to Facility Operating License Nos. NPF-4 and NPF-7 issued to the Virginia Electric and Power Company (the licensee) which revised Technical Specifications for operation of the North Anna Power Station, Units No. 1 and No. 2 (the facility) located in Louisa County, Virginia. The amendments were effective as of the date of issuance.

The amendments revise the partial power multiplier from 0.2 to 0.3 for $F^N_{\Delta H}$. In addition, an administrative change has been made to the NA-2 TS 4.2.2.2.g. to provide consistency between the North Anna Power Station, Units No. 1 and No. 2 Technical Specifications.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since these amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of the amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the application for amendments dated December 8, 1982, (2) Amendment No. 46 and No. 29 to Facility Operating Licenses No. NPF-4 and NPF-7 and (3) the Commission's related Safety Evaluation. These items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C. 20555 and at the Board of Supervisor's Office, Louisa County Courthouse, Louisa, Virginia 23093 and at the Alderman Library, Manuscripts Department, University of Virginia, Charlottesville, Virginia 22901. A copy of items (2) and (3) may be obtained upon request to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland this 22nd day of April, 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing