DCS MS-016

APR 13 1982

Docket Nos. 50-338
50-339

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APR 19 1982
US NUCLEAR REGULATORY COMMISSION

Mr. R. H. Leasburg
Vice President - Nuclear Operations
Virginia Electric and Power Company
Post Office Box 26666
Richmond, Virginia 23261

Dear Mr. Leasburg:

The Commission has issued the enclosed Amendment Nos. 39 and 20 to Facility Operating License Nos. NPF-4 and NPF-7 for the North Anna Power Station, Unit Nos. 1 and 2 (NA 1&2). The amendments consist of changes to Technical Specifications (TS) as requested in your application dated November 12, 1981 (Serial No. 627) and as supplemented by your letter dated February 12, 1982 (Serial No. 080).

The changes amend the NA 1&2 TS based on your reanalysis of the Emergency Core Cooling System performance for the postulated large-break Loss-of-Coolant Accident (LOCA) assuming seven (7) percent steam generator uniform tube plugging. These TS changes revise the heat flux hot channel factor, F_Q, from 2.10 to 2.14.

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. 39 to NPF-4
2. Amendment No. 20 to NPF-7
3. Safety Evaluation
4. Notice of Issuance

cc w/encls:
See next page

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no legal objection to notice and.

OFFICE	DL:ORB#3	DL:ORB#3	DL:ORB#3	DL:OR	OELD		
SURNAME	PKreutzer	LEngle:ms	RAClark	TMNovak	D.Swanson		
DATE	3/31/82	4/1/82	4/1/82	4/5/82	4/9/82		

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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 ELECTRIC AND POWER COMPANY,
North Anna Power Station Units No. 1 and 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).
- X Other: Amendment Nos. 39 and 20
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→	4/16/82						

NRC FORM 102 7—79

Virginia Electric and Power Company

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Regional Administrator
Nuclear Regulatory Commission, Region II
Office of Executive Director for Operations
101 Marietta Street, Suite 3100
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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. 39
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 November 12, 1981 as supplemented February 12, 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. 39 , 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 13, 1982

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 39 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

3/4 2-5
3/4 2-8
3/4 2-16
B 3/4 2-1

POWER DISTRIBUTION LIMITS

HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \left[\frac{2.14}{P}\right] [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq [4.28] [K(Z)] \text{ for } P \leq 0.5$$

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

and K(Z) is the function obtained from Figure 3.2-2 for a given core height location.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Comply with either of the following ACTIONS:

1. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit. The Overpower ΔT Trip Setpoint reduction shall be performed with the reactor in at least HOT STANDBY.

2. Reduce THERMAL POWER as necessary to meet the limits of Specification 3.2.6 using the APDMS with the latest incore map and updated $\overline{R}$.

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a, above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 F_{xy} shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured F_{xy} component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.

c. Comparing the F_{xy} computed (F_{xy}^C) obtained in b, above to:

1. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in e and f, below, and

2. The relationship:

$$F_{xy}^L = F_{xy}^{RTP} [1 + 0.2(1-P)]$$

where F_{xy}^L is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP} and P is the fraction of RATED THERMAL POWER at which F_{xy} was measured.

d. Remeasuring F_{xy} according to the following schedule:

1. When F_{xy}^C is greater than the F_{xy}^{RTP} limit for the appropriate measured core plane but less than the F_{xy}^L relationship, additional power distribution maps shall be taken and F_{xy}^C compared to F_{xy}^{RTP} and F_{xy}^L:

a) Either within 24 hours after exceeding by 20% of RATED THERMAL POWER or greater, the THERMAL POWER at which F_{xy}^C was last determined, or

(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} 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.

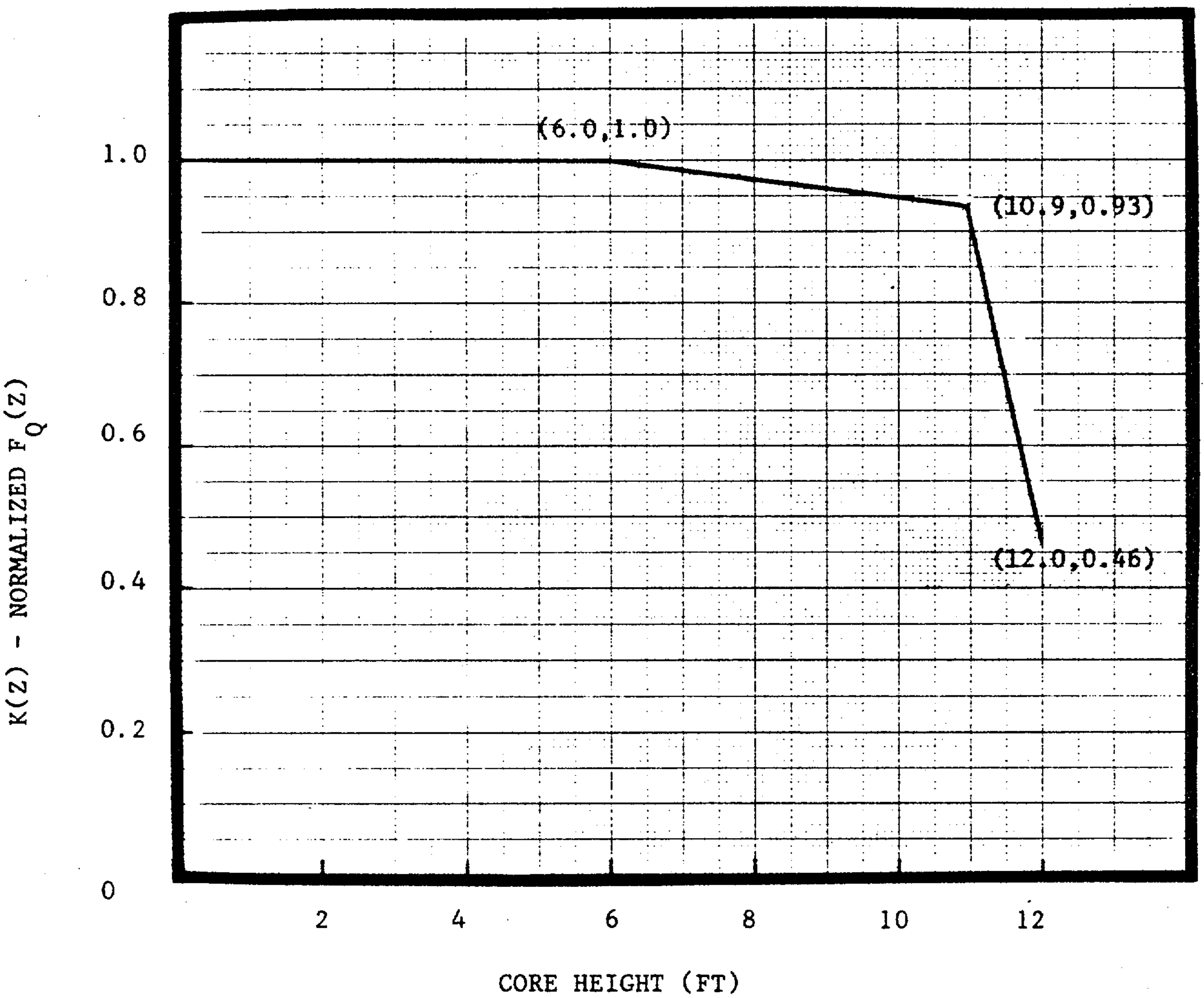

Figure 3.2-2 K(Z) - Normalized $F_Q(Z)$ as a Function of Core Height

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS: 3 Loops In Operation	LIMITS: 2 Loops In Operation** & Loop Stop Valves Open	LIMITS: 2 Loops In Operation** & Isolated Loop Stop Valves Closed
Reactor Coolant System T_{avg}	$\leq$ 585°F		
Pressurizer Pressure	$\geq$ 2205 psig*		
Reactor Coolant System Total Flow Rate	$\geq$ 278,400 gpm		

*Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of 10% RATED THERMAL POWER.

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

POWER DISTRIBUTION LIMITS

AXIAL POWER DISTRIBUTION

LIMITING CONDITION FOR OPERATION

3.2.6 The axial power distribution shall be limited by the following relationship:

$$[F_j(Z)]_S = \frac{[2.14]\ [K(Z)]}{(\overline{R}_j)(P_L)(1.03)(1 + \sigma_j)(1.07)}$$

Where:

a. $F_j(Z)$ is the normalized axial power distribution from thimble j at core elevation Z.

b. P_L is the fraction of RATED THERMAL POWER.

c. K(Z) is the function obtained from Figure 3.2-2 for a given core height location.

d. $\overline{R}_j$, for thimble j, is determined from at least n=6 incore flux maps covering the full configuration of permissible rod patterns above P_m% of RATED THERMAL POWER in accordance with:

$$\overline{R}_j = \frac{1}{n} \sum_{i=1}^{n} R_{ij}$$

Where:

$$R_{ij} = \frac{F_{Qi}^{Meas}}{[F_{ij}(Z)]_{Max}}$$

and $[F_{ij}(Z)]_{Max}$ is the maximum value of the normalized axial distribution at elevation Z from thimble j in map i which had a measured peaking factor without uncertainties or densification allowance of F_Q^{Meas}.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core ≥ 1.30 during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature & cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of $2200^{\circ}F$ is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope of 2.14 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

Although it is intended that the plant will be operated with the AXIAL FLUX DIFFERENCE within the $\pm$ 5% target band about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1 hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1 while at THERMAL POWER levels between 50% and P_f% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than P_f% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and P_f% and 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly target band.

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. 20
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 November 12, 1981 as supplemented February 12, 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.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. 20, 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 13, 1982

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 20 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 indentified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

3/4 2-5
3/4 2-8
3/4 2-17
3/4 2-18
B 3/4 2-1

HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \left[\frac{2.14}{P}\right] [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq [4.28] [K(Z)] \text{ for } P \leq 0.5$$

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

and K(Z) is the function obtained from Figure 3.2-2 for a given core height location.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Comply with either of the following ACTIONS:

1. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit. The Overpower ΔT Trip Setpoint reduction shall be performed with the reactor in at least HOT STANDBY.

2. Reduce THERMAL POWER as necessary to meet the limits of Specification 3.2.6 using the APDMS with the latest incore map and updated $\bar{R}$.

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a, above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 F_{xy} shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured F_{xy} component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.

c. Comparing the F_{xy} computed (F_{xy}^C) obtained in b, above to:

1. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in e and f, below, and

2. The relationship:

$$F_{xy}^L = F_{xy}^{RTP} [1 + 0.2(1-P)]$$

where F_{xy}^L is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP} and P is the fraction of RATED THERMAL POWER at which F_{xy} was measured.

d. Remeasuring F_{xy} according to the following schedule:

1. When F_{xy}^C is greater than the F_{xy}^{RTP} limit for the appropriate measured core plane but less than the F_{xy}^L relationship, additional power distribution maps shall be taken and F_{xy}^C compared to F_{xy}^{RTP} and F_{xy}^L:

a) Either within 24 hours after exceeding by 20% of RATED THERMAL POWER or greater, the THERMAL POWER at which F_{xy}^C was last determined, or

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, and
2. The $F_Q(Z)$ limit shall be reduced at least 1% for each 1% F_{xy}^{C} exceeds F_{xy}.

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.

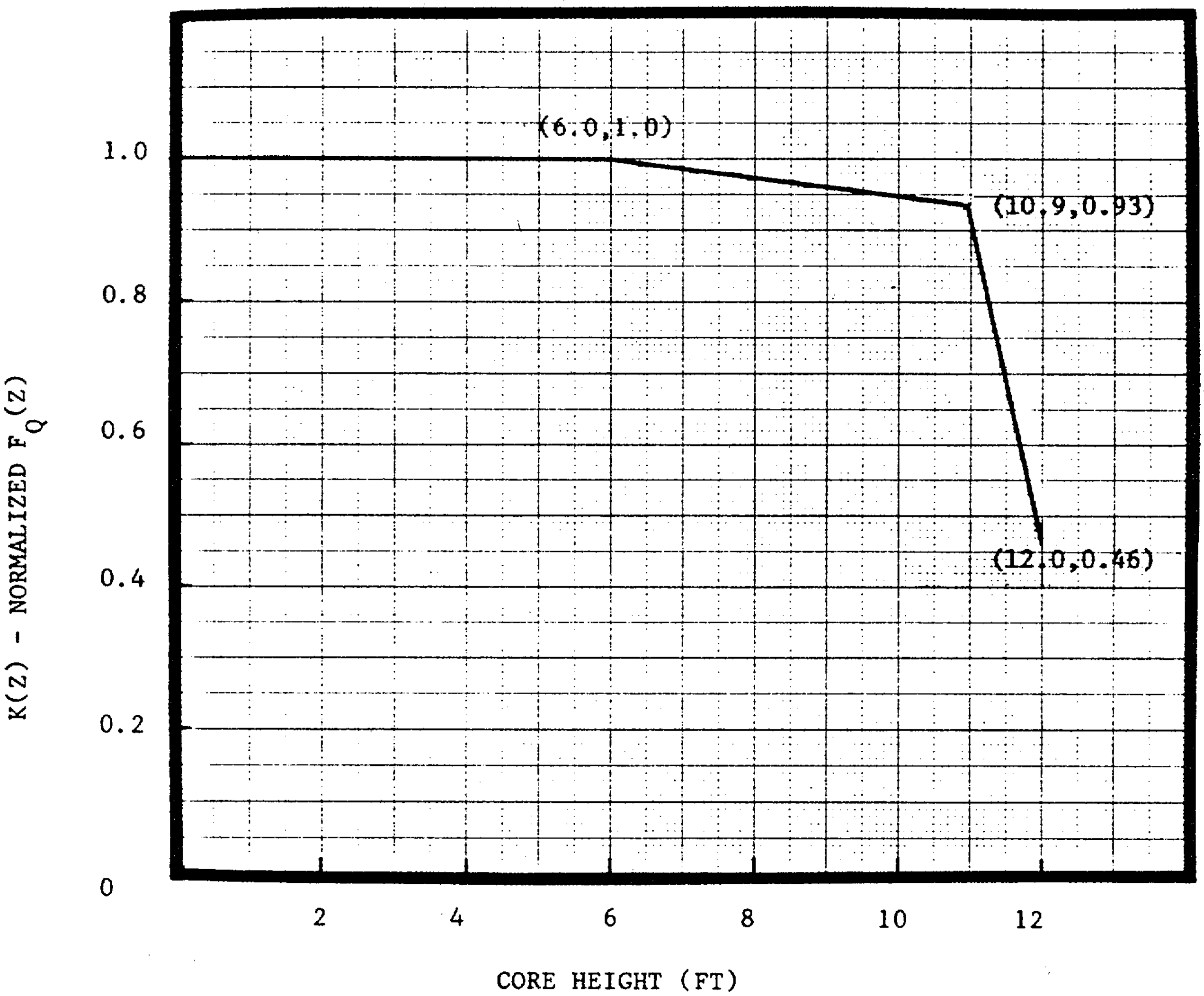

Figure 3.2-2 K(Z) - Normalized $F_Q(Z)$ as a Function of Core Height

AXIAL POWER DISTRIBUTION

LIMITING CONDITION FOR OPERATION

3.2.6 The axial power distribution shall be limited by the following relationship:

$$[F_j(Z)]_S = \frac{[2.14]\ [K(Z)]}{(\overline{R}_j)(P_L)(1.03)(1+\sigma_j)(1.07)}$$

Where:

a. $F_j(Z)$ is the normalized axial power distribution from thimble j at core elevation Z.

b. P_L is the fraction of RATED THERMAL POWER.

c. K(Z) is the function obtained from Figure 3.2-2 for a given core height location.

d. $\overline{R}_j$, for thimble j, is determined from at least n=6 incore flux maps covering the full configuration of permissible rod patterns above P_m % of RATED THERMAL POWER in accordance with:

$$\overline{R}_j = \frac{1}{n} \sum_{i=1}^{n} R_{ij}$$

Where:

$$R_{ij} = \frac{F_{Qi}^{Meas}}{[F_{ij}(Z)]_{Max}}$$

and $[F_{ij}(Z)]_{Max}$ is the maximum value of the normalized axial distribution at elevation Z from thimble j in map i

LIMITING CONDITION FOR OPERATION (Continued)

which had a measured peaking factor without uncertainties or densification allowance of F_Q^{Meas}.

e. σ_j is the standard deviation associated with thimble j, expressed as a fraction or percentage of $\overline{R}_j$, and is derived from n flux maps from the relationship below, or 0.02, (2%) whichever is greater.

$$\sigma_j = \frac{\left[\frac{1}{n-1}\sum_{i=1}^{n}(\overline{R}_j - R_{ij})^2\right]^{1/2}}{\overline{R}_j}$$

f. The factor 1.07 is comprised of 1.02 and 1.05 to account for the axial power distribution instrumentation accuracy and the measurement uncertainty associated with F_Q using the movable detector system, respectively.

g. The factor 1.03 is the engineering uncertainty factor.

APPLICABILITY: MODE 1 ABOVE P_m% of RATED THERMAL POWER#, where the value for P_m is established in the Core Surveillance Report per Technical Specification 6.9.1.10.

ACTION:

a. With a $F_j(Z)$ factor exceeding $[F_j(Z)_S]$ by less than or equal to 4 percent, reduce THERMAL POWER one percent for every percent by which

#The APDMS may be out of service when surveillance for determining power distribution maps is being performed.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core greater than or equal to 1.30 during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature & cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope of 2.14 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other

THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

Although it is intended that the plant will be operated with the AXIAL FLUX DIFFERENCE within the $\pm$ 5% target band about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1 hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1 while at THERMAL POWER levels between 50% and P_f% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of rated THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than P_f% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and P_f% and 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly target band.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENTS NO. 39 AND NO. 20 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 November 12, 1981, 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 2 (NA-1&2). The requested change to the TS is based on the licensee's reanalysis of the Emergency Core Cooling System (ECCS) performance for the postulated large break Loss-of-Coolant-Accident (LOCA) assuming seven (7) percent uniform plugging of steam generator tubes. The reanalysis was performed with the NRC approved February 1978 version of the Westinghouse LOCA-ECCS evaluation model.

By letter dated February 12, 1982 the licensee provided supplemental information regarding non-LOCA accidents and transients which could be affected by the seven (7) percent uniform steam generator tube plugging.

The above reanalysis results in a newly adjusted overall heat flux hot channel factor of F_Q equals 2.14 for which the licensee has requested a change to the NA-1 & 2 TS.

Discussion

Significant Input:

Certain conservative assumptions were made for the NA-1&2 LOCA-ECCS reanalysis as required by Appendix K to 10 CFR Part 50. The assumptions pertain to the conditions of the reactor and associated safety system equipment at the time that a LOCA is assumed to occur and includes such items as the core peaking factors, the containment pressure, and the performance of the ECCS. All previous LOCA-ECCS submittals for NA-1&2 have shown that the limiting double ended break size equals 0.4. For this reanalysis, the licensee has also explicitly determined that the limiting double ended break size is also 0.4.

All assumptions and initial operating conditions used in the licensee's reanalysis are the same as those used in the presently NRC approved LOCA-ECCS analysis for NA-1&2 with the following exceptions.

Significant changes in the reanalysis reflect the operational conditions and limits necessary to allow full power operation for steam generator tube plugging levels up to seven (7) percent. The currently approved analysis allows for five (5) percent tube plugging. A core inlet temperature of 548.6 degrees Fahrenheit (F) was used in the reanalysis. This core inlet temperature value was adjusted from NA-1&2 operational data to encompass the five (5) to seven (7) percent steam generator tube plugging increase.

Several changes were made to containment parameters. The thickness of one of the heat sinks was changed to better represent the as-built plant containment. Also, the previous value for the high-containment pressure setpoint was lowered to 18.5 pounds per square inch absolute to agree with the present value specified in the NA-1&2 TS.

The model calculations were performed assuming conservative generic 17 x 17 fuel parameters consistent with currently approved NRC methodology. The previously required 65 degree F uncertainty in fuel pellet temperature was removed. Also, a previous requirement of analysis using a spectrum of fuel heatup rates has been removed, which conforms with current NRC methodology for ECCS analysis.

When the above input changes were incorporated in the reanalysis, the assumed heat flux hot channel factor increased from 2.10 to 2.20. A value of 2.20 still ensures compliance with the 10 CFR Part 50.46 acceptance criteria. The increase from 2.10 to 2.20 is allowable from the NRC approved changes in the generic fuel parameters, the elimination of the fuel heatup rate spectrum, and the higher peak clad temperature resulting from the reanalysis. Finally, an adjustment penalty of minus 0.06 must be applied to the overall heat flux hot channel factor F_Q equals 2.20 which results in an adjusted heat flux hot channel factor F_Q equals 2.14.

The non-LOCA accidents and transients addressed in Chapter 15 of the NA-1&2 FSAR are affected in a variety of ways by increased steam generator tube plugging. The excess heat removal accidents tend to be slightly less severe because of the impaired heat transfer brought about by the two (2) percent increase in steam generator tube plugging. Other accidents, such as overpressurization events remain essentially the same. The licensee's review of non-LOCA accidents has concentrated on those events to be judged adversely affected by the two (2) percent increase in steam generator plugging.

Fuel Pellet Stored Energy

For LOCA analysis, Westinghouse methodology requires input be initialized with various steady state fuel parameters, one of which is a volumetric-average fuel temperature. To account for modeling uncertainties not explicity considered elsewhere, a 65 degree F increase in temperature had previously been applied to the steady state fuel performance calculated value. The licensee has deleted this uncertainty from the present LOCA reanalysis for 7 percent steam generator plugging.

We have previously approved the deletion of this uncertainty for this stored energy conservatism in our review of WCAP-8720, "Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations," dated March 27, 1980. Therefore, we find removal of the 65 degree uncertainty in the fuel pellet temperature for the licensee's reanalysis to be acceptable.

Supplemental ECCS Analysis:

We have been generically evaluating three cladding material models that are used in ECCS evaluations. These models predict cladding rupture temperature, cladding burst strain, and fuel assembly flow blockage. We have discussed our evaluation of these models with vendors and other industry representatives in our "Summary Minutes of Meeting on Cladding Rupture Temperature, Cladding Strain, and Assembly Flow Blockage," dated November 20, 1979, and in our published NRC report NUREG-0630, "Cladding Swelling and Rupture Models for LOCA Analysis," dated April 1980, wherein we concluded that licensing cladding models were, in some areas, non-conservative. Our letter dated November 9, 1979 required licensees to confirm that operating reactors would continue to be in conformance with 10 CFR Part 50.46 when substituting NUREG-0630 cladding material models in presently approved ECCS evaluations.

In our letter to Westinghouse dated December 1, 1981, we stated the completion of our generic review and approval of new acceptance criteria for Westinghouse cladding models. For licensees using old Westinghouse ECCS evaluation models, the ECCS analyses should be accompanied by supplemental calculations using the cladding material models specified in NUREG-0630.

The licensee has referenced the old Westinghouse ECCS evaluation and has provided the supplemental ECCS calculations specified in NUREG-0630 for its reanalysis with seven (7) percent steam generator tube plugging. The licensee's reanalysis also accounted for a non-conservatism identified by Westinghouse in its February 1978 ECCS evaluation model which used a fast-heatup-rate rupture-temperature correlation for slow transient analysis.

Based on a heat flux hot channel factor of F_Q =2.20, the licensee's reanalysis for seven (7) percent steam generator tube plugging assessed the combined impact of the fuel-heatup-rates and the NUREG-0630 models to be worth 855^{O}F peak cladding temperature above that previously calculated.

Subsequently, Westinghouse calculated that a reduction in the total peaking factor F_Q of 0.21 would offset the portion of the 855 degree F increase in peak cladding temperature that exceeded 2200 degrees F. However, Westinghouse also identified a margin in F_Q available through the use of upper-head-injection thermohydraulic models that we have generically approved for the NA-1&2 type of three-loop plant This margin is worth 0.15 in F_Q . Therefore, a F_Q reduction of minus 0.06 (0.15-0.21) is required and an overall heat flux channel factor F_Q of 2.14 (2.20-0.06) is determined to be applicable for NA-1&2. Based on the above, we conclude that the licensee's reanalysis has adequately addressed our concerns related to the clad swelling and rupture issue.

Non- LOCA Accidents and Transients:

The licensee has evaluated non-LOCA accidents and transients adversely impacted by increased steam generator tube plugging.

The licensee has determined that steam generator tube plugging up to seven (7) percent would not reduce the primary system flow below the thermal design limit. Therefore, analysis of departure from nucleate boiling (DNB) events, such as rod withdrawal at power, would not be affected.

Tube plugging affects pump coastdown characteristics and could adversely affect loss-of-flow accidents. The licensee has evaluated this matter, and the change in loop resistance is so small that the impact is negligible.

Boron dilution events could be affected by the reduced volume. However, the two (2) percent reduction is not considered significant since more than an hour is still available for diagnosis and correction of such an event.

Evaluation:

Based on our review of the above matters, we conclude that the results of the ECCS-LOCA analysis with a F_Q equal to 2.14 meets the criteria of CFR Part 50.46 and the analysis was performed in accordance with 10 CFR 50 Appendix K. In addition, we have reviewed the licensee's evaluation of non-LOCA transients that might be affected by tube plugging, and we find that these transients are not adversely affected by a steam generator tube plugging increase from five (5) to seven (7) percent. Also we have determined that the licensee has adequately addressed our concerns regarding the cladding material models addressed in NUREG-0630. We therefore conclude that the proposed technical specification changes for NA-1&2 are acceptable for seven (7)% steam generator tube plugging.

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 accidents previously considered and do not involve a significant decrease in a safety margin, 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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: April 13, 1982

Principal Contributors:

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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 No. 39 and No. 20 to Facility Operating License Nos. NPF-4 and NPF-7 issued to the Virginia Electric and Power Company (the licensee) for operation of the North Anna Power Station, Units No. 1 and No. 2 (the facility) located in Louisa County, Virginia. The amendments are effective as of the date of issuance.

The amendments revise the Technical Specifications for the North Anna Power Station, Units No. 1 and No. 2 based on the licensee's reanalysis of the Emergency Core Cooling System (ECCS) performance for the postulated large-break Loss-of-Coolant-Accident (LOCA) assuming seven (7) percent steam generator uniform tube plugging. The licensee's reanalysis has been determined to meet the criteria of 10 CFR Part 50.46 and the reanalysis was performed in accordance with 10 CFR Part 50, Appendix K and adequately addresses the cladding material models addressed in NUREG-0630.

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 November 12, 1981 as supplemented February 12, 1982, (2) Amendment No. 39 and No. 20 to Facility Operating Licenses No. NPF-4 and NPF-7, respectively, 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, NW., 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 13th day of April, 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
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