

DCS MS-016

JAN 27 1983

Docket Nos. 50-338
and 50-339

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SECY

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. 45 and 28 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 are effective as of the date of issuance.

The amendments consist of changes to the NA-1&2 Technical Specifications (TS) in response to your application transmitted by letter dated August 16, 1982 (Serial No. 490) and in our discussions with you regarding these matters.

The changes amend the NA-1&2 TS based on your reanalysis of the LOCA-ECCS performance assuming seven (7) percent steam generator tube plugging. These changes revise the heat flux hot channel factor, F_Q , from 2.14 to 2.20.

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. 45 to NPF-4
2. Amendment No. 28 to NPF-7
3. Safety Evaluation
4. Notice of Issuance

cc: w/enclosures
See next page

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OFFICE	ORB#3:DL	ORB#3:DL	ORB#3:DL	AD:ORB:DL	OELD	
SURNAME	PMKreutzer	LBEngle/ph	RAClark	GCLainas	D SWANSON	
DATE	1/19/83	1/19/83	1/19/83	1/19/83	1/25/83	

*Ad - me legal
objection to notice
and*



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,
Unit Nos. 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).

Other: Amendment Nos. 45 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/pm					
DATE →	1/28/83					

Virginia Electric and Power Company

cc:

Richard M. Foster, Esquire
Musick, Williamson, Schwartz,
Leavenworth & Cope, P.C.
P. O. Box 4579
Boulder, Colorado 80306

Michael W. Maupin, Esquire
Hunton, Williams, Gay and Gibson
P. O. Box 1535
Richmond, Virginia 23212

Mr. Paul W. Purdom
Environmental Studies Institute
Drexel University
32nd and Chestnut Streets
Philadelphia, Pennsylvania 19104

Atomic Safety and Licensing
Appeal Board Panel
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Ellyn R. Weiss, Esquire
Sheldon, Harman, Roisman and Weiss
1725 I Street, N.W. Suite 506
Washington, D. C. 20006

Mr. W. R. Cartwright, Station Manager
P. O. Box 402
Mineral, Virginia 23117

Mr. Anthony Gambardella
Office of the Attorney General
11 South 12th Street - Room 308
Richmond, Virginia 23219

Resident Inspector/North Anna
c/o U.S.N.R.C.
Route 2, Box 78A
Mineral, Virginia 23117

Mr. J. H. Ferguson
Executive Vice President - Power
Virginia Electric and Power Company
Post Office Box 26666
Richmond, Virginia 23261

Mrs. Margaret Dietrich
Route 2, Box 568
Gordonsville, Virginia 22042

Mr. James C. Dunstance
State Corporation Commission
Commonwealth of Virginia
Blandon Building
Richmond, Virginia 23209

Mrs. June Allen
North Anna Environmental Coalition
8720 Lockmoor Circle
Wichita, Kansas 67207

U.S. Environmental Protection Agency
Region III Office
ATTN: Regional Radiation Representative
Curtis Building
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. 45
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 August 16, 1982 (Serial No. 490), 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. 45, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Within 90 days after the effective date of this amendment, or such later time as the Commission may specify, the licensee shall satisfy any applicable requirement of P.L. 97-425 related to pursuing an agreement with the Secretary of Energy for the disposal of high-level radioactive waste and spent nuclear fuel.
4. The license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 27, 1983

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 45 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:

$$F_Q(Z) \leq \frac{[2.20]}{P} [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq [4.40] [K(Z)] \text{ for } P \leq 0.5$$

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

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 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.

POWER DISTRIBUTION LIMITS

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

POWER DISTRIBUTION LIMITS

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 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.

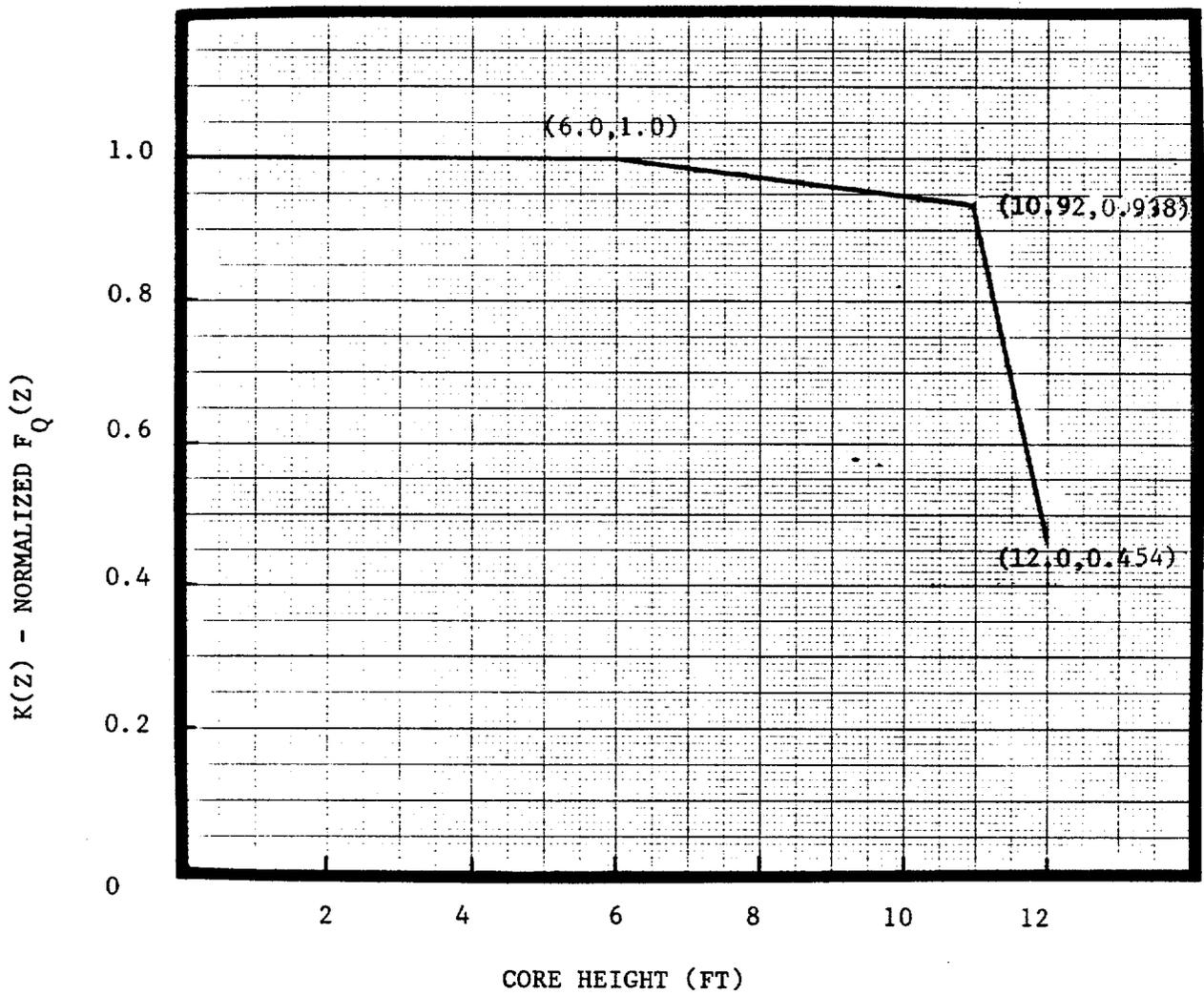


Figure 3.2-2 $K(Z)$ - Normalized $F_Q(Z)$ as a Function of Core Height

TABLE 3.2-1

DNB PARAMETERS

<u>PARAMETER</u>	<u>LIMITS</u>		
	<u>3 Loops In Operation</u>	<u>2 Loops In Operation** & Loop Stop Valves Open</u>	<u>2 Loops In Operation** & Isolated Loop Stop Valves Closed</u>
Reactor Coolant System T_{avg}	$\leq 587^{\circ}\text{F}$		
Pressurizer Pressure	$\geq 2205 \text{ psig}^*$		
Reactor Coolant System Total Flow Rate	$\geq 278,400 \text{ gpm}$		

*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:

$$[F_j(Z)]_S = \frac{[2.20] [K(Z)]}{(\bar{R}_j)(P_L)(1.03)(1 + \sigma_j)(1.07)}$$

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. \bar{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:

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

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°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.
- $F_{\Delta H}^N$ 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, as given in Specification 3.2.2., 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.

POWER DISTRIBUTION LIMITS

BASES

Although it is intended that the plant will be operated with the AXIAL FLUX DIFFERENCE within the + 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. 28
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 August 16, 1982 (Serial No. 490), 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.

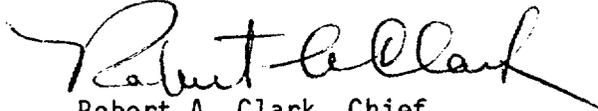
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-7 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 28, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Within 90 days after the effective date of this amendment, or such later time as the Commission may specify, the licensee shall satisfy any applicable requirement of P.L. 97-425 related to pursuing an agreement with the Secretary of Energy for the disposal of high-level radioactive waste and spent nuclear fuel.
4. The license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 27, 1983

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 28 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

3/4 2-5
3/4 2-8
3/4 2-17
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:

$$F_Q(Z) \leq \frac{[2.20]}{P} [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq [4.40] [K(Z)] \text{ for } P \leq 0.5$$

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

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 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.

POWER DISTRIBUTION LIMITS

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}^C 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

POWER DISTRIBUTION LIMITS

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}^C 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}^L .
- 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.

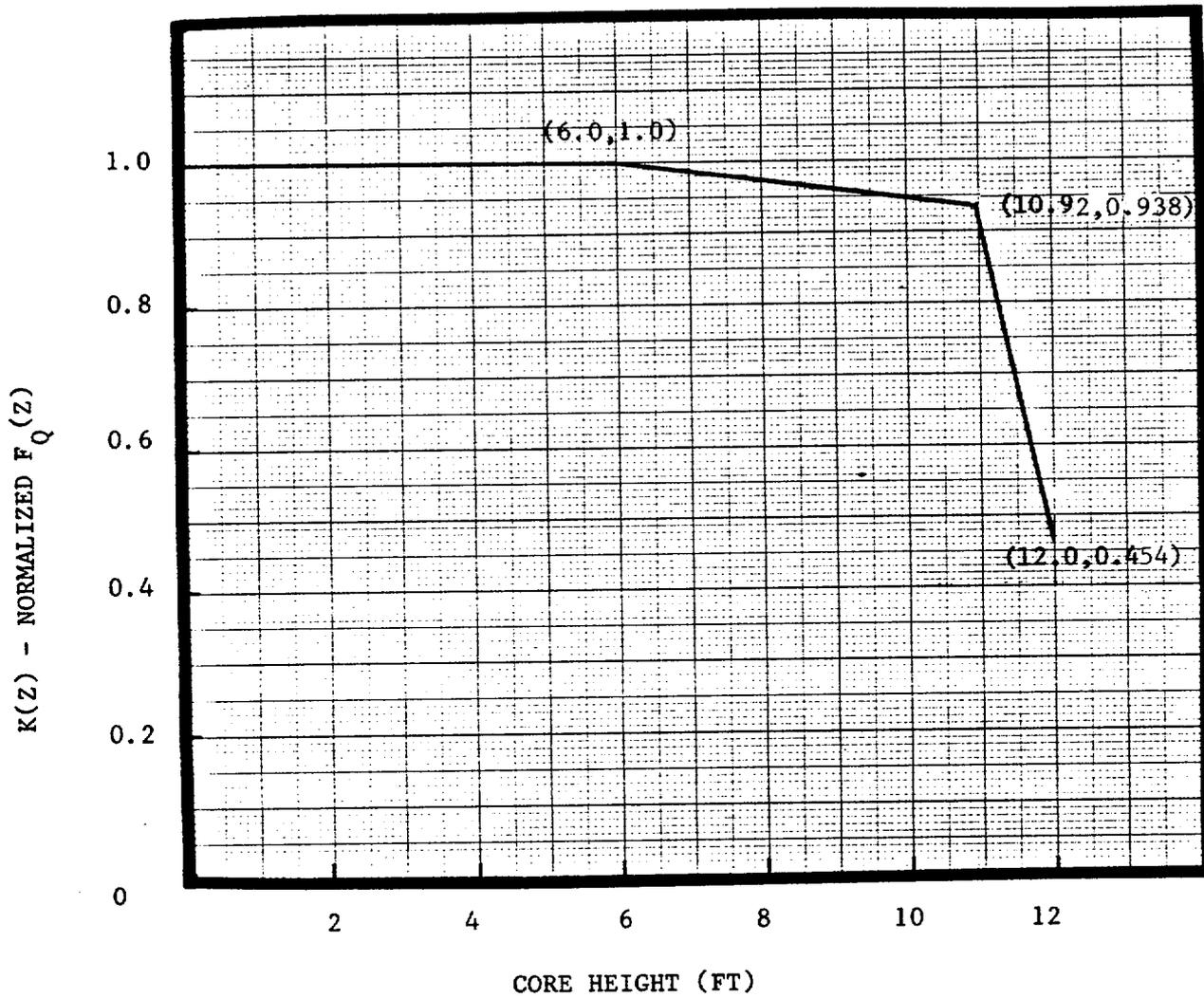


Figure 3.2-2 $K(Z)$ - Normalized $F_Q(Z)$ as a Function of Core Height

POWER DISTRIBUTION LIMITS

AXIAL POWER DISTRIBUTION

LIMITING CONDITION FOR OPERATION

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

$$[F_j(Z)]_S = \frac{[2.20] [K(Z)]}{(\bar{R}_j)(P_L)(1.03)(1 + \sigma_j)(1.07)}$$

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

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

Where:

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

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

POWER DISTRIBUTION LIMITS

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 \bar{R}_j , and is derived from n flux maps from the relationship below, or 0.02, (2%) whichever is greater.

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

- 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.

$F_{\Delta H}^N$ 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, as given in Specification 3.2.2, 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

POWER DISTRIBUTION LIMITS

BASES

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 AMENDMENT NOS. 45 AND 28 TO
FACILITY OPERATING LICENSE NOS. NPF-4 AND NPF-7
VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2
DOCKET NOS. 50-338 AND 50-339

Introduction:

By letter dated August 16, 1982, the Virginia Electric and Power Company (the licensee) requested an amendment in the form of changes to the Technical Specifications (TS), to Operating License Nos. NPF-4 and NPF-7 for the North Anna Power Station, Unit Nos. 1 and 2 (NA-1&2).

The TS changes, as requested by the licensee, are based upon a revised large break Loss-of-Coolant Accident (LOCA)-Emergency Core Cooling System (ECCS) reanalysis for seven (7) percent steam generator tube plugging.

Discussion:

A LOCA-ECCS reanalysis has been performed by the licensee using the NRC approved 1981 version of the Westinghouse LOCA-ECCS Model. The analysis was conducted in compliance with Appendix K to 10 CFR 50 and meets the acceptance criteria stipulated in 10 CFR 50.46. The reanalysis was performed by the licensee under supervision by Westinghouse and the results support continued full power operation for NA-1&2 at steam generator tube plugging levels up to seven (7) percent. In addition, the results of the reanalysis support a new F_Q limit of 2.20. Consistent with the reanalysis, the licensee has proposed applicable changes to the NA-1&2 TS.

As required by Appendix K of 10 CFR 50, certain conservative assumptions were made for the LOCA-ECCS analysis. The assumptions pertain to the condition of the reactor and associated safety system equipment at the time a LOCA is assumed to occur and include such items as the core peaking factors, the containment pressure and the performance of the emergency core cooling system.

All assumptions and initial operating conditions used in the reanalysis were the same as those used in the licensee's previously NRC approved non-LOCA transient analysis for seven (7) percent steam generator tube plugging except as follows: (1) the 17 x 17 generic fuel parameters were updated to reflect current values and (2) the 1981 NRC approved Westinghouse LOCA-ECCS model was used which incorporates the impact of the fuel rod burst and blockage models required by NUREG-0630.

For the LOCA-ECCS reanalysis with seven (7) percent steam generator tube plugging, the assumed worst single failure was loss of the largest low pressure injection pump. Assuming a peaking factor (F_0) of 2.20, the limiting large break LOCA ($C_D = 0.4$) yielded a calculated peak clad temperature of 2194.7°F, a maximum local cladding oxidation of 7.88 percent, and a total metal-water reaction of less than 0.3 percent.

For breaks up to and including the double-ended severance of a reactor coolant pipe and for operating conditions previously reviewed and approved, the ECCS meets the Acceptance Criteria specified in 10 CFR 50.46.

Evaluation:

The results of the licensee's LOCA-ECCS reanalysis supports a new F_0 limit of 2.20. We have reviewed the TS changes due to changing F_0 from 2.14 to 2.20. As part of this change, a new $K(Z)$ curve was generated in Figure 3.2-2 of the NA-1&2 TS. We have independently verified the accuracy of the curve by recalculation using Westinghouse methodology. Our calculations show that the $K(Z)$ curve as submitted by the licensee is correct.

Also, we have determined that the licensee has provided acceptable evaluations of both LOCA and non-LOCA events for operation with F_0 equal to 2.20 with steam generator tube plugging up to seven (7) percent.

Based on the above, we find that the results of the LOCA-ECCS reanalysis with seven (7) percent steam generator tube plugging and a F_0 value equal to 2.20 meets the criteria of 10 CFR Part 50.46 and the analysis was performed in accordance with 10 CFR 50, Appendix K. Therefore, we find the proposed NA-1&2 TS changes to be acceptable.

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: January 27, 1983

Principal Contributors:

L. B. Engle, ORB-3/DL
M. Chatterton, CPB/DSI
R. Barrett, RSB/DSI

UNITED STATES NUCLEAR REGULATORY COMMISSION
DOCKET NOS. 50-338 AND 50-339
VIRGINIA ELECTRIC AND POWER COMPANY
NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY
OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 45 and 28 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, Unit Nos. 1 and 2 based upon the licensee's reanalysis for the Loss-of-Coolant Accident-Emergency Core Cooling System performance assuming seven (7) percent steam generator tube plugging. The licensee's reanalysis has been determined to meet the criteria of 10 CFR 50.46 and the reanalysis was performed in accordance with 10 CFR Part 50, Appendix K.

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 these license amendments. Prior public notice of these amendments was not required since these amendments do not involve a significant hazards consideration.

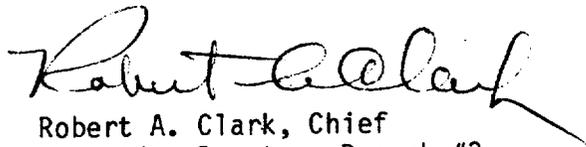
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The Commission has determined that the issuance of the amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal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

For further details with respect to this action, see (1) the application for amendment dated August 16, 1982 (Serial No. 490); (2) Amendment Nos. 45 and 28 to Facility Operating License Nos. 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 Supervisors 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 27th day of January, 1983.

FOR THE NUCLEAR REGULATORY COMMISSION


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