MAY 9 1979

Docket Nos. ~~50-280~~
and 50-281

Mr. W. L. Proffitt
Senior Vice President - Power
Virginia Electric and Power Company
Post Office Box 26666
Richmond, Virginia 23261

Dear Mr. Proffitt:

The Commission has issued the enclosed Amendment Nos. 49 and 48 to Facility Operating License Nos. DPR-32 and DPR-37 for the Surry Power Station, Unit Nos. 1 and 2. The amendment consists of changes to the Technical Specifications in response to your application dated December 26, 1978, as supplemented January 9, 1979.

These amendments revise the Technical Specification limits for total nuclear peaking factor (F_Q) for Surry Units 1 and 2. The new F_Q was established using an approved ECCS model with a steam generator tube plugging limit of 28%. However, allowable Unit 2 steam generator tube plugging will be limited to 5% at this time because refurbished steam generators are being installed and there is no need for a greater limit.

These amendments also supersede the Exemption to the License for Surry Unit No. 1 dated June 30, 1978, and the Order for Modification of License for Surry Unit No. 2 dated April 28, 1978. Accordingly, that Exemption and Order have been terminated.

Gnotr
cep

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DATE➤						

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed. A notice of proposed issuance of this amendment was published in the Federal Register on January 19, 1979 (44 FR 4057).

Sincerely,

Original Signed By

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 49 to DPR-32
2. Amendment No. 48 to DPR-37
3. Notice of Issuance

cc: w/enclosures
See next page

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

May 9, 1979

Docket Nos. 50-280
and 50-281

Mr. W. L. Proffitt
Senior Vice President - Power
Virginia Electric and Power Company
Post Office Box 26666
Richmond, Virginia 23261

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These amendments revise the Technical Specification limits for total nuclear peaking factor (F_Q) for Surry Units 1 and 2. The new F_Q was established using an approved ECCS model with a steam generator tube plugging limit of 28%. However, allowable Unit 2 steam generator tube plugging will be limited to 5% at this time because refurbished steam generators are being installed and there is no need for a greater limit.

These amendments also supersede the Exemption to the License for Surry Unit No. 1 dated June 30, 1978, and the Order for Modification of License for Surry Unit No. 2 dated April 28, 1978. Accordingly, that Exemption and Order have been terminated.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed. A notice of proposed issuance of this amendment was published in the Federal Register on January 19, 1979 (44 FR 4057).

Sincerely,

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 49 to DPR-32
2. Amendment No. 48 to DPR-37
3. Notice of Issuance

cc: w/enclosures
See next page

cc: Mr. Michael W. Maupin
Hunton and Williams
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Richmond, Virginia 23213

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Donald J. Burke
U. S. Nuclear Regulatory Commission
Region II
Office of Inspection and Enforcement
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

Mr. Sherlock Holmes, Chairman
Board of Supervisors of Surry County
Surry County Courthouse, Virginia 23683

Commonwealth of Virginia
Council on the Environment
903 Ninth Street Office Building
Richmond, Virginia 23219

Mr. James R. Wittine
Commonwealth of Virginia
State Corporation Commission
Post Office Box 1197
Richmond, Virginia 23209

Director, Technical Assessment Division
Office of Radiation Programs (AW-459)
U. S. Environmental Protection Agency
Crystal Mall #2
Arlington, Virginia 20460

U. S. Environmental Protection Agency
Region III Office
ATTN: EIS COORDINATOR
Curtis Building - 6th Floor
6th and Walnut Streets
Philadelphia, Pennsylvania 19106

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-280

SURRY POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 49
License No. DPR-32

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 26, 1978, as supplemented January 9, 1979, 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 the license amendment, and paragraph 3.B of Facility Operating License No. DPR-32 is hereby amended to read as follows:

   B. Technical Specification

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 49 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. License Exemption dated June 30, 1978 is hereby terminated.

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

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer / for

R. H. Vollmer, Assistant Director
for Systems and Projects
Division of Operating Reactors

Attachment:
Changes to the
Technical Specifications

Date of Issuance: May 9, 1979

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 48
License No. DPR-37

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 26, 1978, as supplemented January 9, 1979, 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 the license amendment, and paragraph 3.B of Facility Operating License No. DPR-37 is amended and 3.H. is added to read as follows:

   B. Technical Specification

      The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 48 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   H. The steam generator for Unit 2 shall not exceed a tube plugging level of 5% until approved by the NRC.

3. The Order for Modification of License dated April 28, 1978 is hereby terminated.

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

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer /for

R. H. Vollmer, Assistant Director
for Systems and Projects
Division of Operating Reactors

Attachment:
Changes to the
Technical Specifications

Date of Issuance: May 9, 1979

ATTACHMENT TO LICENSE AMENDMENT NOS. 49 AND 48

FACILITY OPERATING LICENSE NOS. DPR-32 AND DPR-37

DOCKET NOS. 50-280 AND 50-281

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
3.3-1	3.3-1
3.12-4	3.12-4
3.12-4a	3.12-4a
3.12-5	3.12-5
3.12-6	3.12-6
3.12-7	3.12-7
3.12-14	3.12-14
3.12-14a	
3.12-17	3.12-17
3.12-21	3.12-21
Table 3.12-1a	Table 3.12-1a
Table 3.12-2	Table 3.12-2
Figure 3.12-8	Figure 3.12-8
	Figure 3.12-10

3.3 SAFETY INJECTION SYSTEM

Applicability

Applies to the operating status of the Safety Injection System.

Objective

To define those limiting conditions for operation that are necessary to provide sufficient borated cooling water to remove decay heat from the core in emergency situations.

Specifications

A. A reactor shall not be made critical unless the following conditions are met:

1. The refueling water tank contains not less than 350,000 gal. of borated water with a boron concentration of at least 2000 ppm.

2. Each accumulator system is pressurized to at least 600 psia and contains a minimum of 975 ft^3 and a maximum of 989 ft^3 of borated water with a boron concentration of at least 1950 ppm.

3. The boron injection tank and isolated portion of the inlet and outlet piping contains no less than 900 gallons of water with a boron concentration equivalent to at least 11.5% to 13% weight boric acid solution at a temperature of at least 145°F. Additionally, recirculation between a unit's Boron Injection Tank and the Boric Acid Tank(s) assigned to the unit shall be maintained.

$F_Q(Z) \leq 2.05/P \times K(Z)$ for $P > .5$

$F_Q(Z) \leq 4.10 \times K(Z)$ for $P \leq .5$

$F^N_{\Delta H} \leq 1.55\ (1 + 0.2(1-P)) \times T(BU)$

$F^N_{\Delta H}\Big|^{LOCA}_{Assm.} \leq 1.38/P$

$F^N_{\Delta H}\Big|^{LOCA}_{Rod} \leq 1.45/P$

where P is the fraction of rated power at which the core is operating, K(Z) is the function given in TS Figure 3.12-8, Z is the core height location of F_Q, and T(BU) is the interim thimble cell rod bow penalty on $F^N_{\Delta H}$ given in TS Figure 3.12-9.

2. Prior to exceeding 75% power following each core loading, and during each effective full power month of operation thereafter, power distribution maps using the movable detector system, shall be made to confirm that the hot channel factor limits of this specification are satisfied. For the purpose of this confirmation:

   a. The measurement of total peaking factor, $F_Q^{Meas.}$, shall be increased by eight percent to account for manufacturing tolerances, measurement error, and the effects of rod bow. The measurement of enthalpy rise hot channel factor, the hot assembly enthalpy rise factor, $F^N_{\Delta H}\Big|^{LOCA}_{Assm.}$, and the hot rod enthalpy rise factor, $F^N_{\Delta H}\Big|^{LOCA}_{Rod}$, shall be increased by four percent to account for measurement error. If any measured hot channel factor exceeds its limit specified under 3.12.B.1, the reactor power and high neutron flux trip setpoint shall be reduced until the limits under 3.12.B.1 are met. If the hot channel factors cannot be brought to within the limits $F_Q \leq 2.05 \times K(Z)$, $F^N_{\Delta H} \leq 1.55 \times T(BU)$, $F^N_{\Delta H}\Big|^{LOCA}_{Rod} \leq 1.45$, and $F^N_{\Delta H}\Big|^{LOCA}_{Assm.} \leq 1.38$ within 24 hours, the Overpower ΔT and Overtemperature ΔT trip setpoints shall be similarly reduced.

Deleted

b. $F_Q(Z)$ shall be evaluated for normal (Condition I) operation of Unit 2 by combining the measured values of $F_{xy}(Z)$ with the design Condition I axial peaking factor values, $F_Z(Z)$, as listed in TS Table 3.12-1B. For the purpose of this specification $F_{xy}(Z)$ shall be determined between 1.5 feet and 10.5 feet elevations of the core exclusive of grid plane regions located at 25.9 ±3.2 inches, 52.1 ±3.2 inches, 78.3 ±3.2 inches, and 104.5 ±3.2 inches. The measured values of $F_{xy}(Z)$ shall be increased by nine percent to account for manufacturing tolerances, measurement error, rod bow, xenon redistribution, and any burnup dependent peaking factor increases. If the results of this evaluation predict that $F_Q(Z)$ could potentially violate its limiting values as established in Specification 3.12.B.1, either:

(1) the thermal power and high neutron flux trip setpoint shall be reduced at least 1% for each 1% of the potential violation (for the purpose of this specification, this power level shall be called $P_{THRESHOLD}$), or

(2) movable detector surveillance shall be required for operation when the reactor thermal power exceeds $P_{THRESHOLD}$. This surveillance shall be performed in accordance with the following:

(a) The normalized power distribution, $F_Q(Z)\Big|^{j}_{APDM}$, from thimble j at core elevation Z shall be measured utilizing at least two thimbles of the movable incore flux system for

3. The reference equilibrium indicated axial flux difference (called the target flux difference) at a given power level P_o, is that indicated axial flux difference with the core in equilibrium xenon conditions (small or no oscillation) and the control rods more than 190 steps withdrawn. The target flux difference at any other power level, P, is equal to the target value of P multiplied by the ratio, P/P_o. The target flux difference shall be measured at least once per equivalent full power quarter. The target flux difference must be updated during each effective full power month of operation either by actual measurement, or by linear interpolation using the most recent value and the value predicted for the end of the cycle life.

4. Except as modified by 3.12.B.4.a, b, c, or d below, the indicated axial flux difference shall be maintained within a ±5% band about the target flux difference (defines the target band on axial flux difference).

   a. At a power level greater than 88 percent of rated power, if the indicated axial flux difference deviates from its target band, within 15 minutes either restore the indicated axial flux difference to within the target band, or reduce the reactor power to less than 88 percent of rated power.

   b. At a power level no greater than 88 percent of rated power,

      (1) The indicated axial flux difference may deviate from its target band for a maximum of one hour (cumulative) in any 24-hour period provided the flux difference is within the limits shown on Figure 3.12-10.

One minute penalty is accumulated for each one minute of operation outside of the target band at power levels equal to or above 50% of rated power.

(2) If 3.12.B.4.b(1) is violated, then the reactor power shall be reduced to less than 50% power within 30 minutes and the high neutron flux setpoint shall be reduced to no greater than 55% power within the next four hours.

(3) A power increase to a level greater than 88 percent of rated power is contingent upon the indicated axial flux difference being within its target band.

(4) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Table 4.1-1 provided the indicated AFD is maintained within the limits of Figure 3.12-10. A total of 16 hours of operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

c. At a power level no greater than 50 percent of rated power,

(1) The indicated axial flux difference may deviate from its target band.

(2) A power increase to a level greater than 50 percent of rated power is contingent upon the indicated axial flux difference not being outside its target band for more than one hour accumulated penalty during the preceding 24-hour period. One half minute penalty is accumulated for each one minute of operation outside of the target band at power levels between 15% and 50% of rated power.

d. The axial flux difference limits of Specifications 3.12.B.4.a, b, and c may be suspended during the performance of physics tests provided:

(1) The power level is maintained at or below 85% of rated power, and

(2) The limits of Specification 3.12.B.1 are maintained.

The power level shall be determined to be $\leq$ 85% of rated power at least once per hour during physics tests. Verification that the limits of Specification 3.12.B.1 are being met shall be demonstrated through in-core flux mapping at least once per 12 hours.

Alarms shall normally be used to indicate the deviations from the axial flux difference requirements in 3.12.B.4.a and the flux difference time limits in 3.12.B.4.b and c. If the alarms are out of service temporarily, the axial flux difference shall be logged, and conformance to the limits assessed, every hour for the first 24 hours, and half-hourly thereafter.

The indicated axial flux difference for each excore channel shall be monitored at least once per 7 days when the alarm is operable and at least once per hour for the first 24 hours after restoring the alarm to operable status.

5. The allowable quadrant to average power tilt is 2.0%.

6. If, except for physics and rod exercise testing, the quadrant to average power tilt exceeds 2%, then:
   a. The hot channel factors shall be determined within 2 hours and the power level adjusted to meet the specification of 3.12.B.1, or
   b. If the hot channel factors are not determined within two hours, the power level and high neutron flux trip setpoint shall be reduced from rated power, 2% for each percent of quadrant tilt.
   c. If the quadrant to average power tilt exceeds ±10%, the power level and high neutron flux trip setpoint will be reduced from rated power, 2% for each percent of quadrant tilt.

$F_Q(Z)$, Height Dependent 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_Q^E, Engineering Heat Flux Hot Channel Factor, is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

$F_{\Delta H}^N$, 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 for both LOCA and non-LOCA considerations.

$F_{\Delta H}^N \Big|_{\text{Assm.}}^{\text{LOCA}}$, Hot Assembly Nuclear Enthalpy Rise Factor, is defined as the ratio of the integral of linear power along the assembly with the highest integrated power to the average assembly power.

It should be noted that the enthalpy rise factors are based on integrals and are used as such in the DNB and LOCA calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in radial (x-y) power shapes throughout the core. Thus the radial power shape at the point of maximum heat flux is not necessarily directly related to the enthalpy rise factors. The results of the loss of coolant accident analyses are conservative with respect to the ECCS acceptance criteria as specified in 10 CFR 50.46 using an upper bound envelope of 2.05 times the hot channel factor normalized operating envelope given by TS Figure 3.12-8.

For normal (Condition I) operation of Unit 2, it may be necessary to perform surveillance to insure that the heat flux hot channel factor, $F_Q(Z)$, limit is met. To determine whether and at what power level surveillance is required, the potential (Condition I) values of $F_Q(Z)$ shall be evaluated monthly by combining the measured values of $F_{xy}(Z)$ obtained from the analysis of the monthly incore flux map with the values of the design Condition I axial peaking factors, $F_Z(Z)$. The product of these shall be increased by nine percent to account for measurement uncertainty, manufacturing tolerances, rod bow, radial redistribution of xenon during normal (Condition I) operation, and any burnup dependent peaking factor increases. $P_{THRESHOLD}$ is defined as the value of rated power minus one percent power for each percent of potential $F_Q(Z)$ violation. If the potential values of $F_Q(Z)$ for normal (Condition I) operation are greater than the $F_Q(Z)$ limit, then surveillance shall be performed at all power levels above $P_{THRESHOLD}$.

Movable incore instrumentation thimbles for surveillance are selected so that the measurements are representative of the peak core power density. By limiting the core average axial power distribution, the total power peaking factor $F_Q(Z)$ can be limited since all other components remain relatively fixed. The remaining part of the total power peaking factor can be derived based on incore measurements, i.e., an effective radial peaking factor, $\overline{R}$, can be determined as the ratio of the total peaking power and allowance has been made in predicting the heat flux peaking factors for less strict control at part power. Strict control of the flux difference is not always possible during certain physics tests or during excore detector calibrations. Therefore, the specifications on power distribution control are less restrictive during physics tests and excore detector calibrations; this is acceptable due to the low probability of a significant accident occurring during these operations.

In some instances of rapid unit power reduction automatic rod motion will cause the flux difference to deviate from the target band when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target band; however to simplify the specification, a limitation of one hour in any period of 24 hours is placed on operation outside the band. This ensures that the resulting xenon distributions are not significantly different from those resulting from operation within the target band. The instantaneous consequences of being outside the band, provided rod insertion limits are observed, is not worse than a 10 percent increment in peaking factor for the allowable flux difference at 88% power, in the range ±13.5 percent (±10.5 percent indicated) where for every 2 percent below rated power, the permissible flux difference boundary is extended by 1 percent.

As discussed above, the essence of the procedure is to maintain the xenon distribution in the core as close to the equilibrium full power condition

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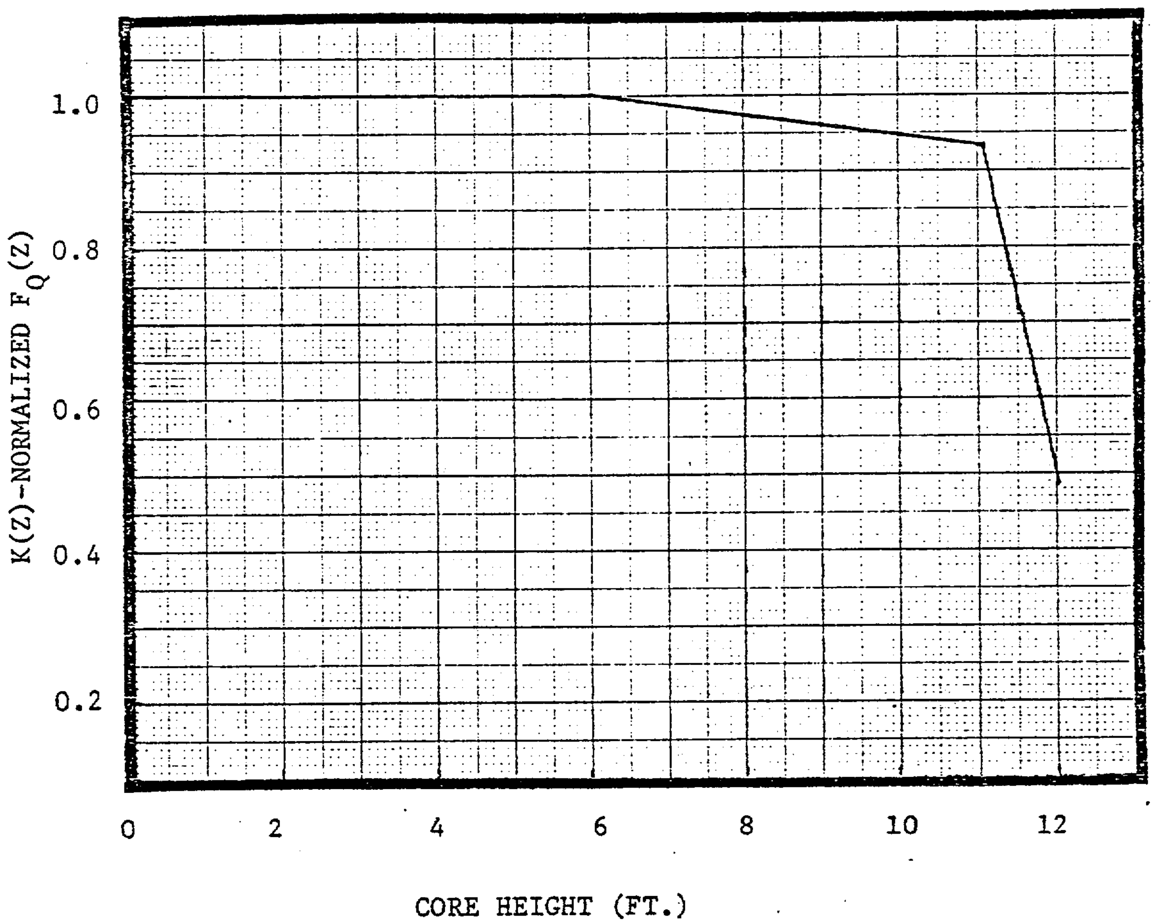
HOT CHANNEL FACTOR NORMALIZED

OPERATING ENVELOPE

SURRY POWER STATION

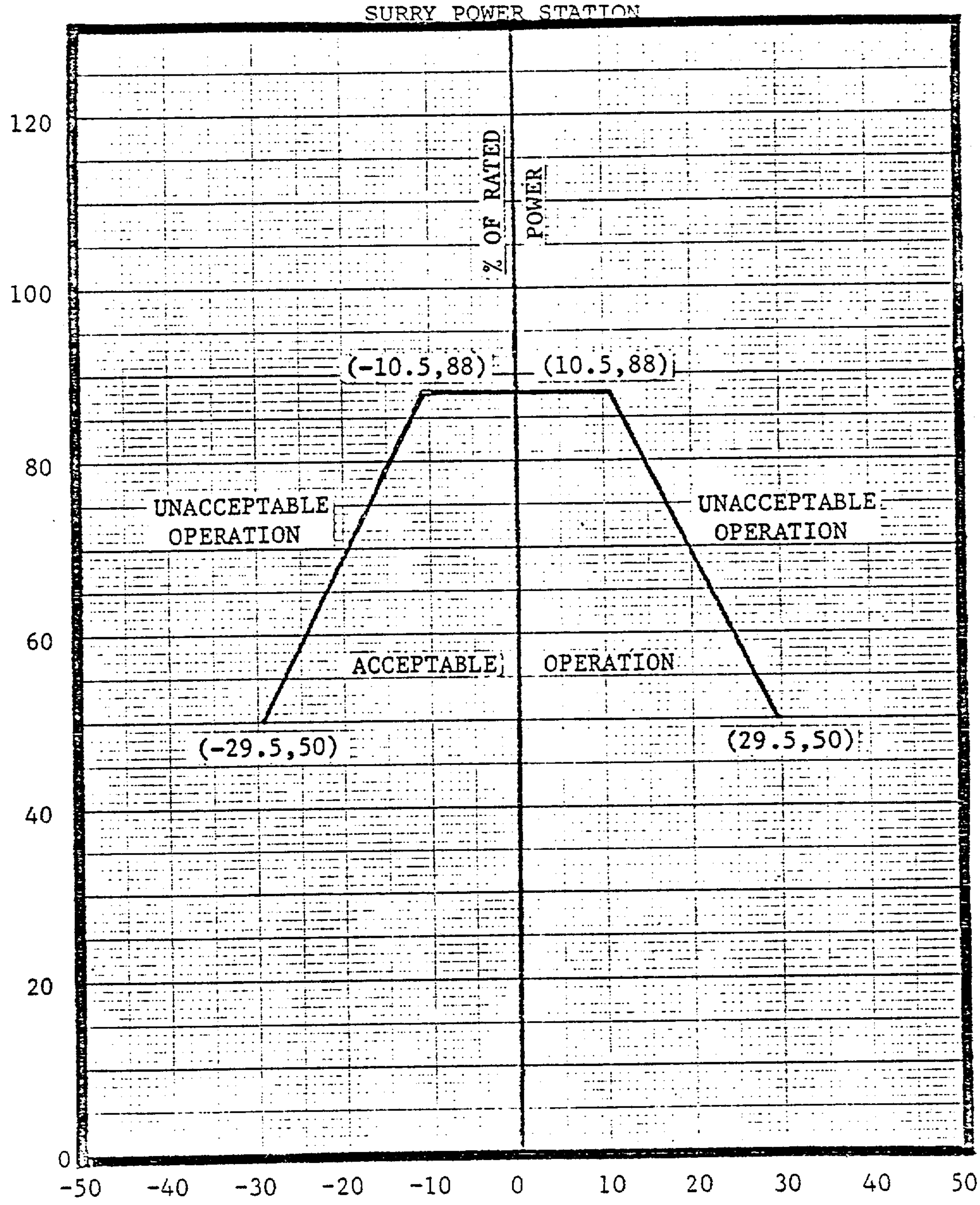
UNIT NOS. 1 AND 2



AXIAL FLUX DIFFERENCE LIMITS

AS A FUNCTION OF RATED POWER

SURRY POWER STATION



SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 49 AND 48 TO

FACILITY OPERATING LICENSE NOS. DPR-32 AND DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY NUCLEAR POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-280 AND 50-281

Introduction

By letter dated December 26, 1978 (Reference 1) as supplemented January 9, 1979 (Reference 2) Virginia Electric and Power Company (the Licensee) requested changes to the Technical Specifications to Operating License Nos. DPR-32 and DPR-37 for the Surry Nuclear Power Station, Unit Nos. 1 and 2. The proposed changes are in response to the Exemption and Order for Modification of License issued on June 30, 1978 (Reference 3) and April 28, 1978 (Reference 4) for Unit Nos. 1 and 2, respectively, which were issued as a result of the discovery of an error in the Zr-water reaction model in the evaluation model computer codes used in the LOCA analysis (Reference 5). The proposed Technical Specification changes are supported by the LOCA reanalysis performed with the approved February 1978 version of the ECCS evaluation model (Reference 6) which included the correction of the error and other approved model changes. The reanalysis also included input assumptions of 28 percent of steam generator tubes plugged and reduced accumulator water volume. It was performed with the total peaking factor, F_Q of 2.05. In addition, the licensee has presented several other related changes to the Technical Specifications dealing with the new limits for core power distribution. Most of these changes were in the conservative direction and, for the one change which decreased the degree of conservatism, the licensee has provided an appropriate justification.

Evaluation

On March 21, 1978, an error was discovered in the Westinghouse October 1975 ECCS evaluation model. The error involved the calculated heat generation resulting from the Zr-water reaction and affected the calculated cladding temperatures after a LOCA. Following discovery of this error, the licensee administratively reduced the total peaking factor limits for Unit Nos. 1 and 2 from F_Q=1.85 to F_Q=1.79. This new value of F_Q was intended to conservatively accommodate the error and was applicable for up to 25 percent of the steam generator tubes plugged (Reference 7). The licensee also committed to provide a new LOCA-ECCS analysis, which was to be performed with an acceptable evaluation model. As noted in the Order for Modification of License, issued for the Surry Power Station, Unit 2 (Reference 4) the NRC conditionally approved the total peaking factor limit of F_Q=1.79, but requested the licensee to provide, as soon as possible, a valid ECCS reanalysis to confirm the conservatism of this limit. In response to this request the licensee submitted an interim LOCA analysis applicable to Surry Unit Nos. 1 and 2 (Reference 8) performed with the October 1975 version of the Westinghouse evaluation model corrected for the Zr-water reaction error. In that analysis, in addition to the error correction, the licensee assumed an F_Q of 1.94 and reduced the accumulator water volume to 975 cu ft. The analysis also included consideration of several other plant specific input assumptions which partially offset the penalty resulting from correction of the Zr-water reaction error. The submittal was reviewed by us and approved for Unit No. 2 operation subject to the licensee submitting LOCA reanalysis performed with a fully approved version of the ECCS evaluation model. Also, based on the results of this interim analysis, an Exemption was granted for the Surry Power Station, Unit No. 1 to operate with F_Q=1.94 and with the new (reduced) value of accumulator water volume (Reference 3). On September 13, 1978 in a letter to the licensee (Reference 10) we reiterated our request that the ECCS analysis, performed with a fully corrected and approved model, be provided. On October 11, 1978 (Reference 11) the licensee committed to provide such an analysis by January 1979. On December 26, 1978 the requested analysis was submitted (Reference 1). The analysis was performed with the NRC approved February 1978 version of the Westinghouse evaluation model (Reference 6) which in addition to including the correction of the Zr-water reaction error and several code maintenance and analytical improvements, contained the following changes: modification of the input to the containment code, modified accumulator model, steam dynamic cooling and an improved 15 x 15 FLECHT heat transfer correlation. The submitted analysis was based on the assumptions listed below:

(1) Limiting value of hot channel peaking factor of F_Q=2.05

(2) Core inlet temperature of 534.5°F

(3) Initial fuel temperature based on generic values of fuel characteristics

(4) Modified containment parameters

(5) 28 percent of steam generator tubes plugged

(6) Accumulator water volume of 975 cu ft.

The proposed value of F_Q was justified by the LOCA analysis which indicated that with F_Q=2.05 and with the assumptions listed above the peak cladding temperature and the local and total Zr-water reactions were within the limits set forth in 10 CFR 50.46. The values of these parameters are listed below:

Peak Cladding Temperature:	2172°F
Local Zr-water Reaction:	7.81%
Total Zr-Water Reaction:	<0.3%

The core inlet temperature assumed in the analysis represented the best estimate value and was lower by 4.5°F from the value used in the interim analysis (Reference 8). The use of the best estimate inlet temperatures is consistent with our position which accepts the use of these temperatures in LOCA analyses. In addition, the current analysis was based on generic values of fuel characteristics which were more conservative than the as-built values previously used and therefore would permit the analysis to be referenced for future cores loaded with fuel having similar fuel characteristics.

Although the containment heat sinks used in the analysis were somewhat different from those assumed in the previous interim analysis, the licensee has shown that they represent in a more realistic manner the plant specific characteristics of the containment. Also, in the analysis the licensee took credit for paint on some of the containment components, the existence of which reduces the flow of heat to the containment structure. This assumption decreases the steam condensation rates and results in higher calculated containment backpressures after a LOCA.

We have reviewed the results of the LOCA analysis submitted by the licensee and have concluded that the safe operation of the plants with $F_Q \leq 2.05$ and steam generator tube plugging levels of ≤ 28 percent has been adequately demonstrated. We concur therefore with the following proposed changes to the Technical Specifications:

(1) Peaking factor change from F_Q-1.94 for Unit No. 1 and F_Q=1.79 for Unit No. 2 to F_Q=2.05 for both units, and

(2) Accumulator water volume change from 1075 cu ft minimum and 1089 cu ft maximum to 975 cu ft minimum and 989 cu ft maximum for both units.

Although the ECCS analysis demonstrates compliance with the regulations, the amount of steam generator plugging for Unit No. 2 steam generators, which are being refurbished, will be limited to 5% at this time.

Since the new value of total nuclear peaking factor would remain below 2.32, the licensee has provided for Unit No. 1 an applicable "18 case FAC analysis" (Reference 12) which indicates that the total nuclear peaking factor would not exceed the value of 2.05 during normal plant operation, including load follow maneuvers and therefore has justified that the use of APDMS is not required. For Unit No. 2 the licensee is committed to use APDMS surveillance.

In addition to the changes resulting directly from the LOCA reanalysis, the licensee has proposed other Technical Specification changes which are related to the power distribution limits in the core. These changes are listed below:

(1) Removal of the steam generator tube plugging limits below which values specified for the maximum assembly and rod enthalpy rise factors $(F_{\Delta H}/^{LOCA}_{assm.}$ and $F_{\Delta H}/^{LOCA}_{rod})$ are not applicable.

(2) Change in the procedures for evaluating $F_Q(Z)$ during plant operation. For Unit No. 2, where the APDMS surveillance is required, the procedure for obtaining $F_Q(Z)$ from the measured values of $F_{xy}(Z)$ and the values for $F_Z(Z)$ specified in Table 3.12-1B would be retained. For Unit No. 1 which we conclude does not require APDMS surveillance, this procedure would be deleted.

(3) New limits for the axial flux difference. This proposed change would specify the new limits for the allowable axial flux difference by: (a) assigning lower value (88%) for the reactor power level above which the axial flux difference must be maintained within a $\pm$5% target band; (b) defining the operational conditions for which axial flux difference could remain outside the prescribed limits without reduction of reactor power, and (c) specifying new allowable axial flux difference limits for reactor operation below 88% of its rated power.

With the exception of the new allowable axial flux difference limits, all the changes listed above would make the Technical Specifications more restrictive. The last change extends the allowable axial flux difference limits and results from a plant specific analysis (Reference 2) performed by the licensee to justify this change. The current allowable axial flux difference limits were derived by a Westinghouse generic analysis which resulted in limits more restrictive than the proposed limits resulting from the analysis performed specifically for Surry Units 1 and 2. The proposed limits are based on plant specific parameters and are conservative. The proposed limits, derived by previously approved methods, assure that the power distribution in the core will be maintained within specific bounds (F_Q, $F_{\Delta H}$) and all safety limits and criteria will be met. We conclude that because this change does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, it does not involve a significant hazards consideration.

Based on the review of the submitted documents, we conclude from the results of the ECCS reanalysis performed with the previously approved February 1978 version of the Westinghouse evaluation model, that operation of Surry Unit Nos. 1 and 2 at a peaking factor limit of 2.05 with reduced accumulator water volume will be in conformance with the 10 CFR 50.46 criteria. We consider the ECCS analysis and all the changes to the plant Technical Specifications resulting from this analysis and from the proposed modifications of the core power distribution limits acceptable for operating the plant with up to a maximum of 28 percent of steam generator tubes plugged. Unit No. 2, which will return to operation with refurbished steam generators that have no plugged tubes, will be limited for the present to 5% plugging.

Environmental Impacts of Proposed Action

The plugging of 3 to 5 percent of the steam generator tubes as would be allowed by the proposed action would impact the total plant occupational exposure. We have estimated the total exposure resulting from plugging each additional 1 percent of the tubes to be 92 man-rems per reactor. Our estimate is based on the licensee's historical data regarding total number of tubes plugged and occupational exposure. For 3 to 5 percent, the estimated exposure would be 275 to 459 man-rems/per reactor. We have examined other PWRs that have plugged a significant number of tubes and find that experience, as a function of percent of tubes plugged, is typical. Therefore, we conclude that there will be no significant environmental impact associated with the proposed action.

Environmental Consideration

We have determined that these 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) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) 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.

References

1. Letter from Vepco (C. M. Stallings) to NRC (H. R. Denton), dated December 26, 1978, Serial No. 736, transmitting: ECCS Analysis (attachment 1); Proposed Technical Specifications (attachment 2).

2. Letter from Vepco (C. M. Stallings) to NRC (H. R. Denton), dated January 9, 1979, Serial No. 019.

3. Letter from NRC (A. Schwencer) to Vepco (W. L. Proffitt), dated June 30, 1978 transmitting Exemption to 10 CFR 50.46 (a) (1) for Surry Power Station, Unit 1.

4. Letter from NRC (A. Schwencer) to Vepco (W. L. Proffitt), dated April 28, 1978, transmitting Order for Modification of License for Surry Power Station, Unit 2.

5. Letter from Westinghouse Electric Corporation NS-CE-1751 (C. Eicheldinger) to NRC (J. F. Stolz) dated April 7, 1978, transmitting LOCA-ECCS Analysis with Zirc/Water Reaction Correction.

6. WCAP-9220-P-A, Westinghouse ECCS Evaluation Model, February 1978 Version, February 1978.

7. Letter from Vepco (C. M. Stallings) to NRC (A. Schwencer) dated April 7, 1978, Serial No. 197.

8. Letter from Vepco (C. M. Stallings) to NRC (E. G. Case), dated May 26, 1978 Serial No. 303, transmitting Large Break LOCA-ECCS Reanalysis for Surry Power Station, Units 1 and 2.

9. WCAP-8622 (Proprietary), WCAP-8623 (Nonproprietary), "Westinghouse ECCS Evaluation Model-October 1975 Version," November 1975.

10. Letter from NRC (A. Schwencer) to Vepco (W. L. Profitt), dated September 13, 1978.

11. Letter from Vepco (C. M. Stallings) to NRC (H. R. Denton), dated October 11, 1978.

12. Letter from Vepco (C. M. Stalling) to NRC (E. G. Case), dated April 4, 1978, Serial No. 171-032-78.

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-280 AND 50-281

VIRGINIA ELECTRIC AND POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 49 and 48 to Facility Operating License Nos. DPR-32 and DPR-37 issued to Virginia Electric and Power Company, which revised Technical Specifications for operation of the Surry Power Station, Unit Nos. 1 and 2 (the facility) located in Surry County, Virginia. The amendments are effective as of the date of issuance.

These amendments revise the Technical Specification limits for total nuclear peaking factor (F_Q) for Surry Units 1 and 2. The new F_Q was established using an approved ECCS model with a steam generator tube plugging limit of 28%. However, allowable Unit 2 steam generator tube plugging will be limited to 5% at this time because refurbished steam generators are being installed and there is no need for a greater limit.

These amendments also supersede the Exemption to the License for Surry Unit No. 1 dated June 30, 1978, and the Order for Modification of License for Surry Unit No. 2 dated April 28, 1978. Accordingly, that Exemption and Order have been terminated.

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 amendment. A Notice of Proposed Issuance of Amendment to Facility Operating License in connection with portions of this action was published in the FEDERAL REGISTER on January 19, 1979 (44 FR 4057). No request for a hearing or petition for leave to intervene was filed following that notice of proposed action. The amendments also approve requests for changes to the Technical Specifications which did not involve significant hazards considerations and thus prior public notice of those proposed changes was not required or given.

The Commission has determined that the issuance of these 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 December 26, 1978 as supplemented January 9, 1979, (2) Amendment Nos. 49 and 48 to License Nos. DPR-32 and DPR-37, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C. and at the Swem Library, College of William and Mary, Williamsburg, Virginia. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 9th day of May, 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors