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MAY 1 6 1980

Docket No. 50-281

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

Dear Mr. Ferguson:

The Commission has issued the enclosed Amendment No. 58 to Facility Operating License No. DPR-37 for the Surry Power Station, Unit No. 2. This amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated May 31, 1979, as supplemented October 16 and 25, 1979, and January 11 and February 20, 1980.

This amendment revises the Technical Specifications to change the heat flux hot channel factor (F_Q) to 2.19 based on a LOCA-ECCS analysis with a steam generator tube plugging limit of 3%.

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

Sincerely,

Original Signed By

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

Enclosures:
1. Amendment No. 58 to DPR-37
2. Safety Evaluation
3. Notice of Issuance

cc: w/enclosures
See next page

No legal objection to forms of amendment or notice SER not reviewed

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OFFICE	DOR:ORB1	DOR:ORB1	DOR:ORB1	DOR:AD:ORP	OELD	
SURNAME	JDNeighbors:jb	CSParrish	ASchwencer	~~WPGammill~~ T Novak	Cutchin	
DATE	05/8/80	05/7/80	05/8/80	05/8/80	05/12/80	

NRC FORM 318 (9-76) NRCM 0240 ☆U.S. GOVERNMENT PRINTING OFFICE: 1979-289-369

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

May 16, 1980

Docket No. 50-281

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

Dear Mr. Ferguson:

The Commission has issued the enclosed Amendment No. 58 to Facility Operating License No. DPR-37 for the Surry Power Station, Unit No. 2. This amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated May 31, 1979, as supplemented October 16 and 25, 1979, and January 11 and February 20, 1980.

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

Sincerely,

A. Schwencer

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

Enclosures:
1. Amendment No. 58 to DPR-37
2. Safety Evaluation
3. Notice of Issuance

cc: w/enclosures
See next page

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Richmond, Virginia 23219

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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. 50-281

SURRY POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 58
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 May 31, 1979, as supplemented October 16 and 25, 1979, and January 11 and February 20, 1980, 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.

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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 to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 58 , 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

A. Schwencer

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

Attachment:
Changes to the
Technical Specifications

Date of Issuance: May 16, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 58

FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NO. 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 certain vertical lines indicating the area of change.

Remove	Insert
3.12-4	3.12-4
3.12-4a	3.12-4a
3.12-4b	3.12-4b
3.12-5	3.12-5
3.12-6	3.12-6
3.12-14	3.12-14
3.12-17	3.12-17
3.12-18	3.12-18
3.12-19	3.12-19
3.12-20	3.12-20
3.12-21	3.12-21
TS Table 3.12-1B	TS Table 3.12-1B
TS Figure 3.12-8	TS Figure 3.12-8a
	TS Figure 3.12-8b
TS Figure 3.12-10	TS Figure 3.12-10

Unit 1	Unit 2
$F_Q(Z) \leq 2.05/P \times K(Z)$ for $P > 0.5$	$F_Q(Z) \leq 2.19/P \times K(Z)$ for $P > 0.5$
$F_Q(Z) \leq 4.10 \times K(Z)$ for $P \leq 0.5$	$F_Q(Z) \leq 4.38 \times K(Z)$ for $P \leq 0.5$
$F^N_{\Delta H} \leq 1.55\ (1 + 0.2(1-P)) \times T(BU)$	$F^N_{\Delta H} \leq 1.55\ (1+0.2(1-P)) \times T(BU)$
$F^N_{\Delta H}\Big\vert_{\substack{\text{LOCA}\\ \text{Assm.}}} \leq 1.38/P$	$F^N_{\Delta H}\Big\vert_{\substack{\text{LOCA}\\ \text{Assm.}}} \leq 1.476/P$
$F^N_{\Delta H}\ {}_{\substack{\text{LOCA}\\ \text{Rod}}} \leq 1.45/P$	$F^N_{\Delta H}\Big\vert_{\substack{\text{LOCA}\\ \text{Rod}}} \leq 1.55/P$

where P is the fraction of rated power at which the core is operating, K(Z) is the function given in TS Figure 3.12-8a for Unit 1 and Figure 3.12-8b for Unit 2, 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, $F^N_{\Delta H}\Big\vert_{\substack{\text{LOCA}\\ \text{Assm.}}}$, and the hot rod enthalpy rise factor, $F^N_{\Delta H}\Big\vert_{\substack{\text{LOCA}\\ \text{Rod}}}$, 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 listed below within 24 hours, the Overpower ΔT and Overtemperature ΔT trip setpoints shall be similarly reduced.

Unit 1	Unit 2
$F_Q \leq 2.05 \times K(Z)$	$F_Q \leq 2.19 \times K(Z)$
$F_{\Delta H}^N \leq 1.55$	$F_{\Delta H}^N \leq 1.55$
$F_{\Delta H}^N \Big\vert_{\text{Assm.}}^{\text{LOCA}} \leq 1.38$	$F_{\Delta H}^N \Big\vert_{\text{Assm.}}^{\text{LOCA}} \leq 1.476$
$F_{\Delta H}^N \Big\vert_{\text{Rod}}^{\text{LOCA}} \leq 1.45$	$F_{\Delta H}^N \Big\vert_{\text{Rod}}^{\text{LOCA}} \leq 1.55$

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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 at P_o 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 $\pm 5\%$ band about the target flux difference (defines the target band on axial flux difference).

   a. At a power level greater than 90 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 90 percent of rated power.

   b. At a power level no greater than 90 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 90 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.

$F_Q(Z)$, 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 (Unit 1) or 2.19 (Unit 2) times the hot channel factor normalized operating envelope given by TS Figures 3.12-8a and 3.12-8b.

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The procedures for axial power distribution control are designed to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers. Basically, control of flux difference is required to limit the difference between the current value of flux difference (ΔI) and a reference value which corresponds to the full power equilibrium value of axial offset (axial offset = ΔI/fractional power). The reference value of flux difference varies with power level and burnup, but expressed as axial offset it varies only with burnup.

The technical specifications on power distribution control given in 3.12.B.4 together with the surveillance requirements given in 3.12.B.2 assure that the Limiting Condition for Operation for the heat flux hot channel factor is met.

The target (or reference) value of flux difference is determined as follows. At any time that equilibrium xenon conditions have been established, the indicated flux difference is noted with the full length rod control bank more than 190 steps withdrawn (i.e. normal full power operating position appropriate for the time in life, usually withdrawn farther as burnup proceeds). This value, divided by the fraction of full power at which the core was operating is the full power value of the target flux difference. Values for all other core power levels are obtained by multiplying the full power value by the fractional power. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and indicated deviation of $\pm 5\%$ ΔI are permitted from the indicated reference value. During periods where extensive load following is required, it may be impractical to establish the required core conditions for measuring the target flux difference every month. For this reason, the specification provides two methods for updating the target flux difference.

Strict control of the flux difference (and rod position) is not as necessary during part power operation. This is because xenon distribution control at part power is not as significant as the control at full 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 90% power, in the range ± 13.8 percent (±10.8 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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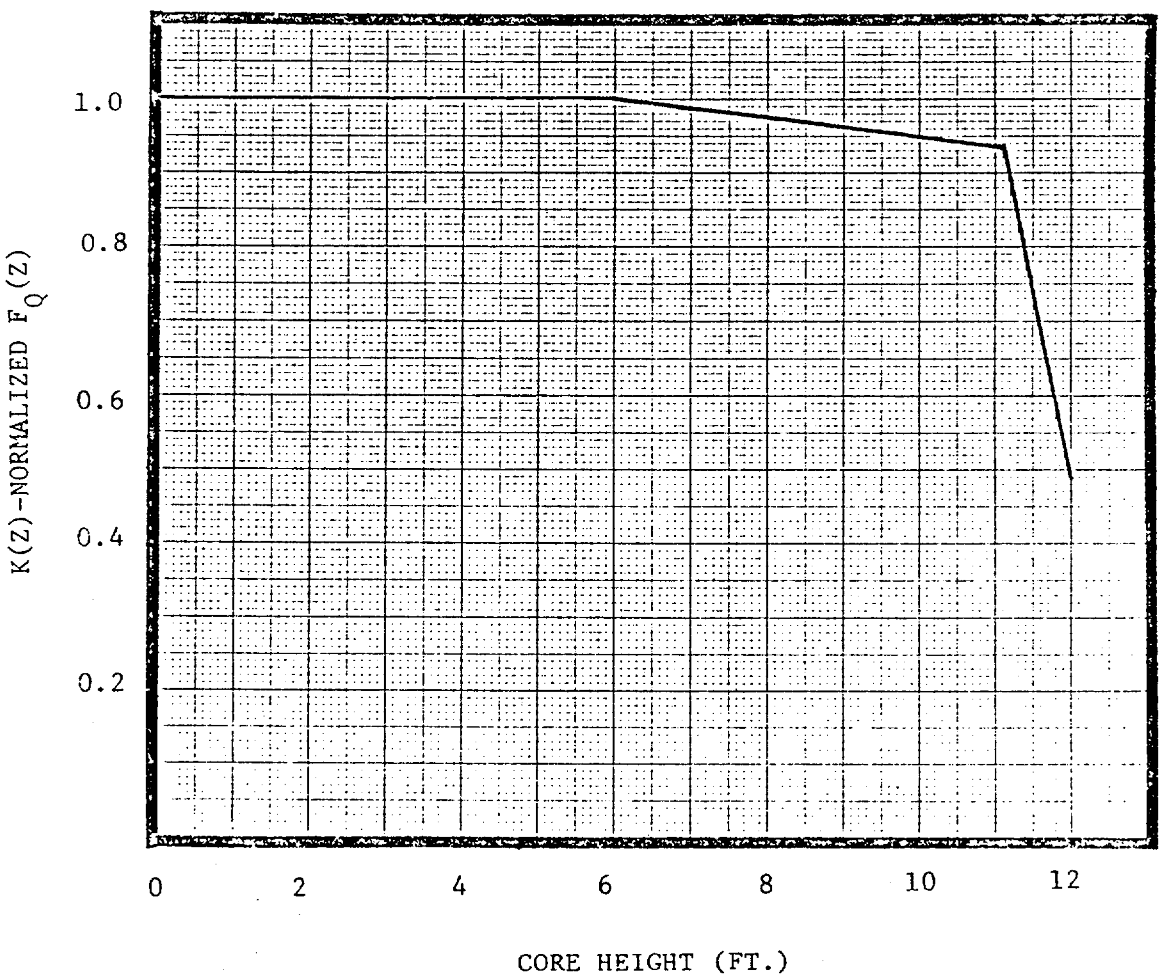
HOT CHANNEL FACTOR NORMALIZED

OPERATING ENVELOPE

SURRY POWER STATION

UNIT NO. 1

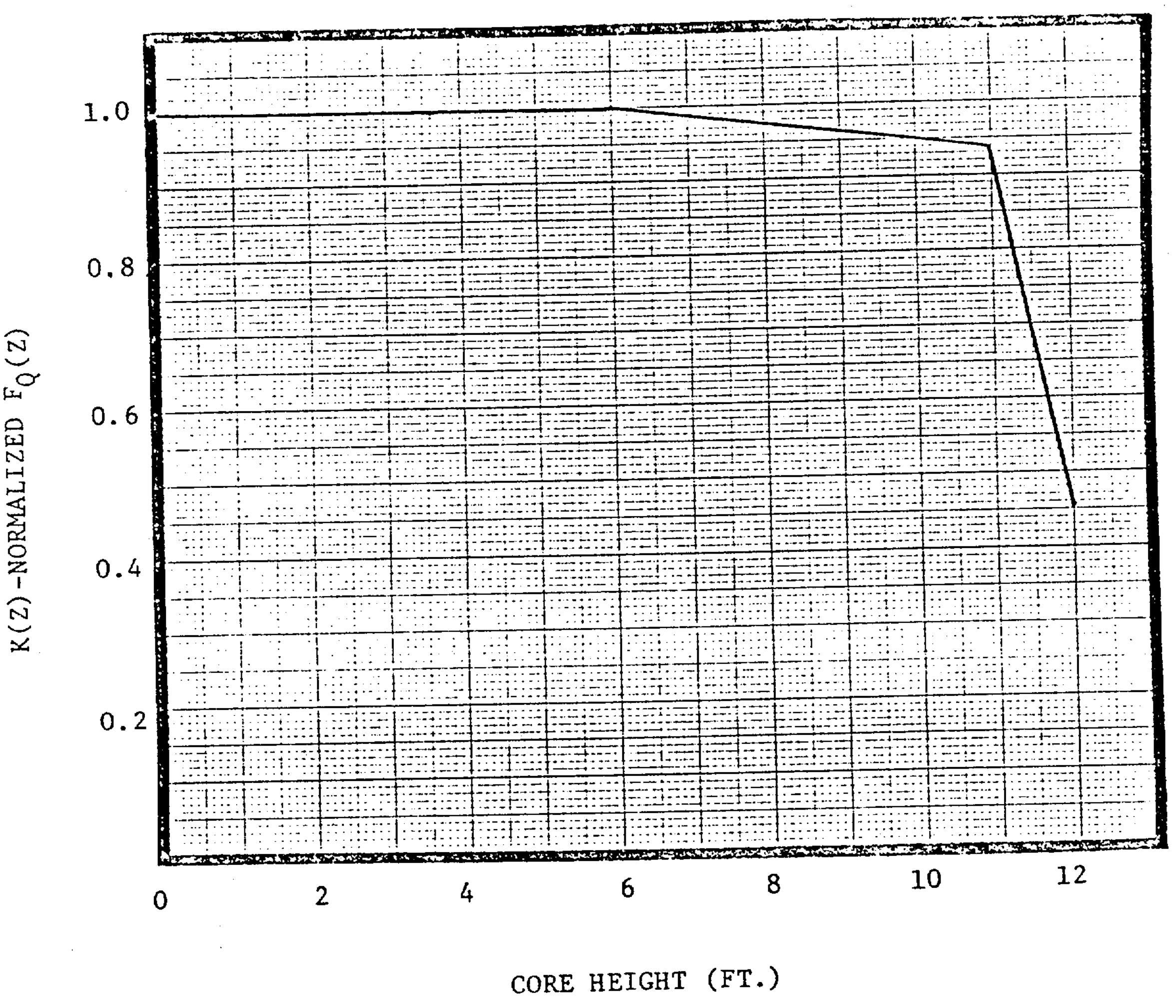


HOT CHANNEL FACTOR NORMALIZED

OPERATING ENVELOPE

SURRY POWER STATION

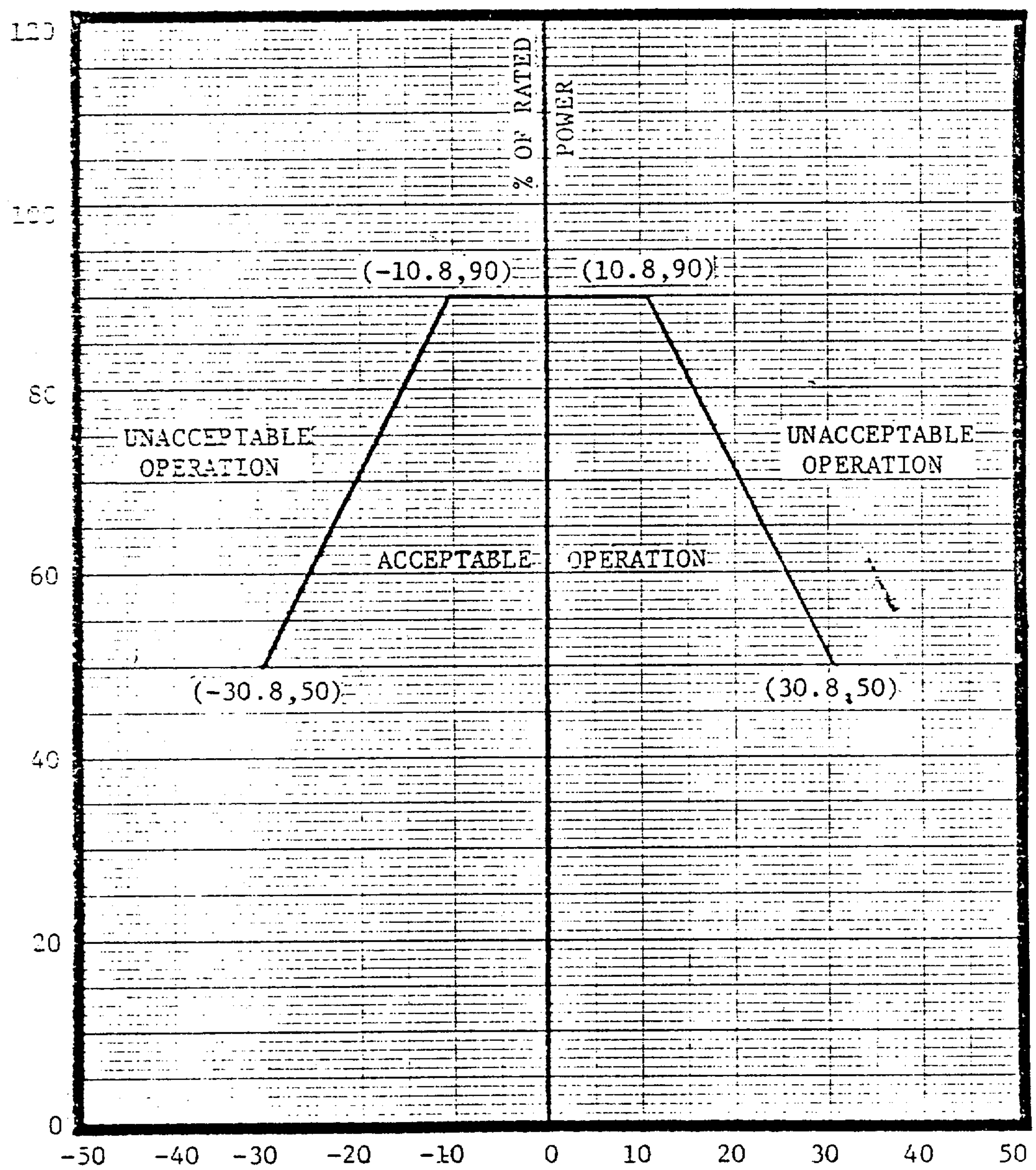
UNIT NO. 2



AXIAL FLUX DIFFERENCE LIMITS

AS A FUNCTION OF RATED POWER

SURRY POWER STATION



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 58 TO LICENSE NO. DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION UNIT 2

DOCKET NO. 50-281

Introduction

By letter dated May 31, 1979 (Reference 1), as supplemented October 16, 1979 (Reference 2), October 25, 1979 (Reference 3), January 11, 1980 (Reference 4) and February 20, 1980 (Reference 5), Virginia Electric and Power Company (the licensee) requested an amendment to Operating License No. DPR-37 for Surry Power Station, Unit 2. References 1 and 2 contain a LOCA analysis and proposed Technical Specification changes in connection with the operation of Unit 2 with 3 percent of steam generator tubes plugged, with modifications made during the steam generator replacement operation and with a peaking factor F_Q of 2.19.

The changes to the Technical Specifications for Unit 2 requested by the licensee are the following:

(a) Change of the heat flux hot channel factor, F_Q to 2.19, the limiting enthalpy rise hot channel factor to 1.55 and the corresponding limiting assembly enthalpy rise factor to 1.476 for plant operation with 3 percent of steam generator tubes plugged.

(b) Change of the Hot Channel Factor Normalized Operating Envelope for Unit 2 (TS Fig. 3.12-8b).

(c) Removal of the specifications for the augmented surveillance of core power distribution and change of the axial flux difference limits (TS Fig. 3.12-10).

Since the limiting value of F_Q is below the level at which the excore detectors could provide reliable readings, the licensee has analytically predicted the maximum values for the total peaking factor, $F_Q(Z)$ using the "3 case analysis" methodology (Reference 6). These predicted values lie below the LOCA predicted limits (Reference 5) and augmented power distribution surveillance is therefore not required during Cycle 5 operation.

Evaluation

The licensee has provided an evaluation of the performance of Emergency Core Cooling System (ECCS) for Unit 2 corresponding to the hot channel peaking factor value of F_Q=2.19 and assuming a steam generator plugging level of 3 percent. In addition, the following changes in plant operational parameters were introduced into the LOCA analysis:

(a) Change of the Low Head Safety Injection flow due to NPSH consideration.

(b) Modification of the Containment Spray System.

(c) Change of the containment initial temperature from 90°F to 80°F.

(d) Changes of the steam generator model parameters reflecting the modification caused by the replacement of the steam generators.

The change in the low safety injection flow was needed in order to meet the NPSH requirement of the LPSI pumps. The flow was limited by means of a venturi flow restrictor. This change resulted in a slightly lower safety injection flow. However, the licensee has demonstrated (Reference 3) that this flow is still higher than the value assumed in the LOCA analysis.

The modification of the containment spray system consisted of adding additional spray header capacity. This additional capacity increased the fill time and resulted in a greater time to actuation of the spray system. The time assumed in the analysis was therefore conservatively changed from 52 to 59 seconds. The additional modification consisted of removing the flow reducing device, which had been installed as a part of the interim NPSH solution. This required increasing the assumed pump runout flow rates from 2250 to 3500 gpm. Although this modification caused the actuation time to change from 410 to 365 seconds, the results of the analysis were not affected because peak cladding temperature occurred at a much earlier time.

The reduction of the containment initial temperature from 90 to 80°F is in itself a conservative change in LOCA analysis. It is consistent with the minimum value of the allowable containment temperature range restrictions required by the NPSH considerations.

The changes introduced to the steam generator model parameters were mostly in thermal-hydraulic area and were caused by the modified reactor coolant side pressure drop and by flow area and tube length changes (References 7 and 8). The magnitude of these changes was small and it did not significantly influence the results of the LOCA analysis.

The LOCA analysis was performed using the February 1978 version of the Westinghouse Evaluation Model (Reference 9) which was reviewed and approved by us. It was performed for a spectrum of three double ended cold leg guillotine breaks (DECLG) with discharge coefficients of C_D=0.4, 0.6 and 1.0. The input parameters assumed in the analysis are listed below:

Core Power: 102 percent of 2441 MWt (rated power).
Peak Linear Power: 102 percent of 13.59 Kw/ft.
Peaking Factor: 2.19
Accumulator Water Volume: 975 cu ft/each

The results of the analysis indicate a peak cladding temperature of 2190°F, a maximum local Zr-water reaction of 7.99 percent and a total Zr-water reaction of less than 0.3 percent, all these values occurring at the critical break size of C_D=0.4.

The licensee did not include a small break analysis since neither the change in the steam generator tube plugging level, nor the other modifications introduced to the plant affected significantly the results of the original analysis.

The licensee has addressed (Reference 4) the potential impact of the recent concerns related to fuel performance model changes included in draft report NUREG-0630 (Reference 10). The licensee has shown that although these model changes by themselves could cause quite significant peak cladding temperature increases, there are at least two compensating effects which could provide credits required to offset the penalties causing these increases. These effects are due to the changes involving the slip and break flow models which have been approved by us for UHI plants after an extensive review. As a result the effects produced by the fuel performance model change could be excluded from the present LOCA analysis without reducing its degree of conservatism.

The licensee has predicted the maximum values of the total peaking factor reached by the Surry Unit 2 plant during its Cycle 5 operation. The prediction was made using the NRC approved, "3 case FAC" methodology (Reference 11) and the axial flux difference limits specified in the licensee's submittal (Reference 2). The predicted peaking factor, $F_Q(Z)$ is below the LOCA determined limit and no augmented core power distribution surveillance is required during the Cycle 5 operation. The licensee is therefore justified in removing the augmented surveillance requirement from the plant's Technical Specifications.

Based on our review of the submittal documents, we conclude that the results of the LOCA analysis performed with F_Q=2.19 are conservative relative to the 10 CFR 50.46 criteria. We consider the resultant changes to the Technical Specifications acceptable for operating Unit 2 with up to a maximum 3 percent of steam generator tubes plugged.

The submitted LOCA analysis was reviewed for Unit 2, however, the evaluation could be extended to Unit 1 after the unit is suitably modified to comply with the assumptions made in the analysis.

Environmental Consideration

We have determined that the amendment does 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 amendment 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 this amendment.

Conclusion

We have concluded, based on the consideration discussed above, that: (1) because the amendment 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, the amendment does 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: May 16, 1980

References

1. Letter from C. M. Stallings (VEPCO) to H. R. Denton (NRC), Serial No. 388, dated May 31, 1979.

2. Letter from C. M. Stallings (VEPCO) to H. R. Denton (NRC), Serial No. 820, dated October 16, 1979.

3. Letter from C. M. Stallings (VEPCO) to H. R. Denton (NRC), Serial No. 388 A, dated October 25, 1979.

4. Letter from C. M. Stallings (VEPCO) to D. G. Eisenhut (NRC), Serial No. 039, dated January 11, 1980.

5. Letter from C. M. Stallings (VEPCO) to H. R. Denton (NRC), Serial No. 388 B, dated February 20, 1980.

6. Letter from C. Eicheldinger (Westinghouse) to J. F. Stolz (NRC), Serial No. NS-CE-1749, dated April 6, 1978.

7. Letter from C. M. Stallings (VEPCO) to E. G. Case (NRC), Serial No. 351, dated August 17, 1977.

8. Letter from A. Schwencer (NRC) to W. L. Proffitt (VEPCO), dated December 15, 1978.

9. WCAP-9220, Westinghouse ECCS Evaluation Model, February 1978 Version, February 1978.

10. NUREG-0630 (Draft), Cladding Swelling and Rupture Models for LOCA Analysis, November 1979.

11. Letter from R. A. Wiesemann (Westinghouse) to J. F. Stolz (NRC), Serial No. AW-78-34, dated April 6, 1978.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-281

VIRGINIA ELECTRIC AND POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 58 to Facility Operating License No. DPR-37 issued to Virginia Electric and Power Company, which revised Technical Specifications for operation of the Surry Power Station, Unit No. 2 (the facility) located in Surry County, Virginia. The amendment is effective as of the date of issuance.

This amendment revises the Technical Specifications to change the heat flux hot channel factor (F_Q) to 2.19 based on a LOCA-ECCS analysis with a steam generator tube plugging limit of 3%.

The application for the amendment 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. Prior public notice of this amendment was not required since it does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment 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 this amendment.

For further details with respect to this action, see (1) the application for amendment dated May 31, 1979, as supplemented October 16 and 25, 1979, and January 11 and February 20, 1980, (2) Amendment No. 58 to License No. 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 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 16th day of May, 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

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