Distribution
Docket File
NRC PDR
Local PDR
ORB 1 File
D. Eisenhut
C. Parrish
D. Neighbors
OELD
OI&E (5)
G. Deegan (8)
B. Scharf
J. Wetmore
ACRS (10)
OPA (Clare Miles)
R. Diggs
NSIC
TERA
Chairman, ASLAB

JUN 16 1981

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JUN 19 1981
U.S. NUCLEAR REGULATORY COMMISSION

Docket Nos. 50-280
and 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. 70 to Facility Operating License No. DPR-32 and Amendment No. 70 to Facility Operating License No. DPR-37 for the Surry Power Station, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated April 28, 1981, as supplemented May 15, 1981.

These amendments revise the Technical Specifications to change the heat flux hot channel factor (F_Q) to 2.18 for Units 1 and 2. These amendments also make editorial changes to the Technical Specifications.

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

Sincerely,

Original Signed By:

Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Enclosures:
1. Amendment No. 70 to DPR-32
2. Amendment No. 70 to DPR-37
3. Safety Evaluation
4. Notice of Issuance

cc: w/enclosures
See next page

No legal objection to form of amendment & notice. SER is not reviewed

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OFFICE	ORB 1	ORB 1	ORB 1	AD:DL	OELD		
SURNAME	CParrish	DNeighbors/rs	SVarga	TNovak	Cutchin		
DATE	6/ /81	6/5/81	6/5/81	6/5/81	6/9/81		

NRC FORM 318 (10-80) NRCM 0240

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USGPO: 1981—335-960

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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. 70
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 April 28, 1981, as supplemented May 15, 1981, 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 3.B of Facility Operating License No. DPR-32 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 70, 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

Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: JUN 16 1981

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. 70
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 April 28, 1981, as supplemented May 15, 1981, 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 3.B of Facility Operating License No. DPR-37 is hereby amended to read as follows:

   B. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 70, 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

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Attahcment:
Changes to the Technical
Specifications

Date of Issuance: JUN 1 6 1981

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 70 TO FACILITY OPERATING LICENSE NO. DPR-32

AMENDMENT NO. 70 TO FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NOS. 50-280 AND 50-281

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.12-1	3.12-1
3.12-2	3.12-2
3.12-3	3.12-3
3.12-4	3.12-4
3.12-4a	-
3.12-4b	-
3.12-5	3.12-5
3.12-6	3.12-6
3.12-7	3.12-7
3.12-8	3.12-8
3.12-9	3.12-9
3.12-10	3.12-10
3.12-11	3.12-11
3.12-12	3.12-12
3.12-13	3.12-13
3.12-14	3.12-14
3.12-15	3.12-15
3.12-15a	-
3.12-16	3.12-16
3.12-16a	-
3.12-17	3.12-17
3.12-18	3.12-18
3.12-19	3.12-19
3.12-20	-
3.12-21	-
3.12-22	-
6.6-9	6.6-9
TS Table 3.12-1	-
TS Table 3.12-1A	-
TS Table 3.12-1B	-
TS Table 3.12-2	-
TS Figure 3.12-8(Unit 1)	TS Figure 3.12-8(Units 1 and 2)
TS Figure 3.12-8a(Unit 2)	-
TS Figure 3.12-8b(Unit 2)	-
TS Figure 3.12-10	TS Figure 3.12-10

3.12 CONTROL ROD ASSEMBLIES AND POWER DISTRIBUTION LIMITS

Applicability

Applies to the operation of the control rod assemblies and power distribution limits.

Objective

To ensure core subcriticality after a reactor trip, a limit on potential reactivity insertions from hypothetical control rod assembly ejection, and an acceptable core power distribution during power operation.

Specification

A. Control Bank Insertion Limits

1. Whenever the reactor is critical, except for physics tests and control rod assembly exercises, the shutdown control rods shall be fully withdrawn.
2. Whenever the reactor is critical, except for physics tests and control rod assembly exercises, the full length control rod banks shall be inserted no further than the appropriate limit determined by core burnup shown on TS Figures 3.12-1A, 3.12-1B, 3.12-2, or 3.12-3 for three-loop operation and TS Figures 3.12-4A, 3.12-4B, 3.12-5 or 3.12-6 for two-loop operation.
3. The limits shown on TS Figures 3.12-1A through 3.12-6 may be revised on the basis of physics calculations and physics data obtained during unit startup and subsequent operation, in accordance with the following:
   a. The sequence of withdrawal of the controlling banks, when going from zero to 100% power, is A, B, C, D.
   b. An overlap of control banks, consistent with physics cal-

culations and physics data obtained during Unit Startup and subsequent operation, will be permitted.

c. The shutdown margin with allowance for a stuck control rod assembly shall be greater than or equal to 1.77% reactivity under all steady-state operation conditions, except for physics tests, from zero to full power, including effects of axial power distribution. The shutdown margin as used here is defined as the amount by which the reactor core would be subcritical at hot shutdown conditions (T_{avg} ≥547°F) if all control rod assemblies were tripped, assuming that the highest worth control rod assembly remained fully withdrawn, and assuming no changes in xenon or boron.

4. Whenever the reactor is subcritical, except for physics tests, the critical rod position, i.e., the rod position at which criticality would be achieved if the control rod assemblies were withdrawn in normal sequence with no other reactivity changes; shall not be lower than the insertion limit for zero power.

5. Insertion limits do not apply during physics tests or during periodic exercise of individual rods. However, the shutdown margin indicated above must be maintained except for the low power physics test to measure control rod worth and shutdown margin. For this test the reactor may be critical with all but one full length control rod, expected to have the highest worth, inserted.

B. Power Distribution Limits

1. At all times except during low power physics tests, the hot channel factors defined in the basis must meet the following limits:

$$F_Q(Z) \leqq 2.18/P \times K(Z) \text{ for } P > 0.5$$

$$F_Q(Z) \leqq 4.36 \times K(Z) \text{ for } P \leqq 0.5$$

$$F_{\Delta H}^N \leqq 1.55\ (1+0.2(1-P))$$

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, and Z is the core height location of F_Q.

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 $F_{\Delta H}$ shall be increased by four percent to account for measurement error. If any measured hot channel factor exceeds its limit specified under Specification 3.12.B.1, the reactor power and high neutron flux trip setpoint shall be reduced until the limits under Specification 3.12.B.1 are met. If the hot channel factors cannot be brought to within the limits of $F_Q(Z) \leqq 2.18 \times K(Z)$ and $F_{\Delta H}^N \leqq 1.55$ within 24 hours, the Overpower ΔT and Overtemperature ΔT trip setpoints shall be similarly reduced.

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 measurements 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 Specifications 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 TS 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 Specification 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 TS Table 4.1-1 provided the indicated axial flux difference is maintained within the limits of TS Figure 3.12-10. A total of 16 hours of operation may be accumulated with the axial flux difference 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 for 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 less than or equal to 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 Specification 3.12.B.4.a and the flux difference time limits in Specifications 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 requirement of Specification 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.

7. If, except for physics and rod exercise testing, after a further period of 24 hours, the power tilt in Specification 3.12.B.5 above is not corrected to less than 2%:

a. If design hot channel factors for rated power are not exceeded, an evaluation as to the cause of the discrepancy shall be made and reported as a reportable occurrence to the Nuclear Regulatory Commission.

b. If the design hot channel factors for rated power are exceeded and the power is greater than 10%, the Nuclear Regulatory Commission shall be notified and the Nuclear Overpower, Nuclear Overpower ΔT, and Overtemperature ΔT trips shall be reduced one percent for each percent the hot channel factor exceeds the rated power design values.

c. If the hot channel factors are not determined the Nuclear Regulatory Commission shall be notified and the Overpower

ΔT and Overtemperature ΔT trip settings shall be reduced by the equivalent of 2% power for every 1% quadrant to average power tilt.

C. <u>Inoperable Control Rods</u>

1. A control rod assembly shall be considered inoperable if the assembly cannot be moved by the drive mechanism or the assembly remains misaligned from its bank by more than 15 inches. A full-length control rod shall be considered inoperable if its rod drop time is greater than 1.8 seconds to dashpot entry.
2. No more than one inoperable control rod assembly shall be permitted when the reactor is critical.
3. If more than one control rod assembly in a given bank is out of service because of a single failure external to the individual rod drive mechanism, i.e. programming circuitry, the provisions of Specifications 3.12.C.1 and 3.12.C.2 shall not apply and the reactor may remain critical for a period not to exceed two hours provided immediate attention is directed toward making the necessary repairs. In the event the affected assemblies cannot be returned to service within this specified period the reactor will be brought to hot shutdown conditions.
4. The provisions of Specifications 3.12.C.1 and 3.12.C.2 shall not apply during physics tests in which the assemblies are intentionally misaligned.
5. The insertion limits in TS Figure 3.12-2 apply:
   a. If an inoperable full-length rod is located below the 200 step level and is capable of being tripped, or

b. If the full-length rod is located below the 30 step level, whether or not it is capable of being tripped.

6. If an inoperable full-length rod cannot be located or if the inoperable full-length rod is located above the 30 step level and cannot be tripped, then the insertion limits in TS Figure 3.12-3 apply.

7. If a full-length rod becomes inoperable and reactor operation is continued, the potential ejected rod worth and associated transient power distribution peaking factors shall be determined by analysis within 30 days. The analysis shall include due allowance for non-uniform fuel depletion in the neighborhood of the inoperable rod. If the analysis results in a more limiting hypothetical transient than the cases reported in the safety analysis, the unit power level shall be reduced to an analytically determined part power level which is consistent with the safety analysis.

D. <u>Core Quadrant Power Balance</u>:

1. If the reactor is operating above 75% of rated power with one excore nuclear channel out of service, the core quadrant power balance shall be determined:

   a. Once per day, and

   b. After a change in power level greater than 10% or more than 30 inches of control rod motion.

2. The core quadrant power balance shall be determined by one of the following methods:

   a. Movable detectors (at least two per quadrant)

   b. Core exit thermocouples (at least four per quadrant)

E. <u>Inoperable Rod Position Indicator Channels</u>

1. If a rod position indicator channel is out of service, then:

   a. For operation between 50% and 100% of rated power, the position of the RCC shall be checked indirectly by core instrumentation (excore detector and/or thermocouples and/or movable incore detectors) every shift or subsequent to motion of the non-indicating rod exceeding 24 steps, whichever occurs first.

   b. During operation below 50% of rated power, no special monitoring is required.

2. Not more than one rod position indicator (RPI) channel per group nor two RPI channels per bank shall be permitted to be inoperable at any time.

F. <u>Misaligned or Dropped Control Rod</u>

1. If the Rod Position Indicator Channel is functional and the associated full length control rod is more than 15 inches out of alignment with its bank and cannot be realigned, then unless the hot channel factors are shown to be within design limits as specified in Specification 3.12.B.1 within 8 hours, power shall be reduced so as not to exceed 75% of permitted power.

2. To increase power above 75% of rated power with a full-length control rod more than 15 inches out of alignment with its bank, an analysis shall first be made to determine the hot channel factors and the resulting allowable power level based on Section 3.12-B.

Basis

The reactivity control concept assumed for operation is that reactivity changes accompanying changes in reactor power are compensated by control rod assembly motion. Reactivity changes associated with xenon, samarium, fuel depletion, and large changes in reactor coolant temperature (operating temperature to cold shutdown) are compensated for by changes in the soluble boron concentration. During power operation, the shutdown groups are fully withdrawn and control of power is by the control groups. A reactor trip occurring during power operation will place the reactor into the hot shutdown condition. The control rod assembly insertion limits provide for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod assembly remains fully withdrawn, with sufficient margins to meet the assumptions used in the accident analysis. In addition, they provide a limit on the maximum inserted rod worth in the unlikely event of a hypothetical assembly ejection and provide for acceptable nuclear peaking factors. The limit may be determined on the basis of unit startup and operating data to provide a more realistic limit which will allow for more flexibility in unit operation and still assure compliance with the shutdown requirement. The maximum shutdown margin requirement occurs at end of core life and is based on the value used in the analysis of the hypothetical steam break accident. The rod insertion limits are based on end of core life conditions. The shutdown margin for the entire cycle length is established at 1.77% reactivity. All other accident analyses with the exception of the chemical and volume control system malfunction analysis are based on 1% reactivity shutdown margin.

Relative positions of control rod banks are determined by a specified control rod bank overlap. This overlap is based on the consideration of axial power shape control.

The specified control rod insertion limits have been revised to limit the potential ejected rod worth in order to account for the effects of fuel densification.

The various control rod assemblies (shutdown banks, control banks A, B, C, and D) are each to be moved as a bank; that is, with all assemblies in the bank within one step (5/8 inch) of the bank position. Position indication is provided by two methods: a digital count of actuating pulses which shows the demand position of the banks, and a linear position indicator, Linear Variable Differential Transformer, which indicates the actual assembly position. The position indication accuracy of the Linear Differential Transformer is approximately ±5% of span (± 7.5 inches) under steady state conditions. The relative accuracy of the linear position indicator is such that, with the most adverse errors, an alarm is actuated if any two assemblies within a bank deviate by more than 14 inches. In the event that the linear position indicator is not in service, the effects of malpositioned control rod assemblies are observable from nuclear and process information displayed in the Main Control Room and by core thermocouples and in-core movable detectors. Below 50% power, no special monitoring is required for malpositioned control rod assemblies with inoperable rod position indicators because, even with an unnoticed complete assembly misalignment (full length control rod assembly 12 feet out of alignment with its bank), operation at 50% steady state power does not result in exceeding core limits.

The specified control rod assembly drop time is consistent with safety analyses that have been performed.

An inoperable control rod assembly imposes additional demands on the operators. The permissible number of inoperable control rod assemblies is limited to one in order to limit the magnitude of the operating burden, but such a failure would not prevent dropping of the operable control rod assemblies upon reactor trip.

Two criteria have been chosen as a design basis for fuel performance related to fission gas release, pellet temperature, and cladding mechanical properties. First, the peak value of fuel centerline temperature must not exceed 4700°F. Second, the minimum DNBR in the core must not be less than 1.30 in normal operation or in short term transients.

In addition to the above, the peak linear power density and the nuclear enthalpy rise hot channel factor must not exceed their limiting values which result from the large break loss of coolant accident analysis based on the ECCS acceptance criteria limit of 2200°F on peak clad temperature. This is required to meet the initial conditions assumed for the loss of coolant accident. To aid in specifying the limits of power distribution, the following hot channel factors are defined:

$F_Q(Z)$, <u>Height Dependent Heat Flux Hot Channel Factor</u>, 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 tolerance on fuel pellets and rods.

F_Q^E, <u>Engineering Heat Flux Hot Channel Factor</u>, 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.

$F_{\Delta H}^N$, <u>Nuclear Enthalpy Rise Hot Channel Factor</u>, 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.

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 2.18 times the hot channel factor normalized operating envelope given by TS Figure 3.12-8.

When an F_Q measurement is taken, measurement error, manufacturing tolerances, and the effects of rod bow must be allowed for. Five percent is the appropriate allowance for measurement error for a full core map (≧40 thimbles monitored) taken with the movable incore detector flux mapping system, three percent is the appropriate allowance for manufacturing tolerances, and five percent is the appropriate allowance for rod bow. These uncertainties are statistically combined and result in a net increase of 1.08 that is applied to the measured value of F_Q.

In the specified limit of $F^N_{\Delta H}$ there is an eight percent allowance for uncertainties, which means that normal operation of the core is expected to result in $F^N_{\Delta H} \leqq 1.55\ (1+0.2\ (1-P))/1.08$. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape (e.g., rod misalignment) affect $F^N_{\Delta H}$, in most cases without necessarily affecting F_Q, (b) the operator has a direct influence on F_Q through movement of rods and can limit it to the desired value; he has no direct control over $F^N_{\Delta H}$, and (c) an error in the predictions for radial power shape, which may be detected during startup physics tests and which may influence F_Q, can be compensated for by tighter axial control. Four percent is the appropriate allowance for measurement uncertainty for $F_{\Delta H}^{N}$ obtained from a full core map (≥40 thimbles monitored) taken with the movable incore detector flux mapping system.

Measurement of the hot channel factors are required as part of startup physics tests, during each effective full power month of operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following core loading provides confirmation of the basic nuclear design bases including proper fuel loading patterns. The periodic incore mapping provides additional assurance that the nuclear design bases remain inviolate and identify operational anomalies which would, otherwise, affect these bases.

For normal operation, it has been determined that, provided certain conditions are observed, the enthalpy rise hot channel factor $F_{\Delta H}^{N}$ limit will be met. These conditions are as follows:

1. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the bank demand position. An indicated misalignment limit of 13 steps precludes a rod misalignment no greater than 15 inches with consideration of maximum instrumentation error.
2. Control rod banks are sequenced with overlapping banks as shown in TS Figures 3.12-1A, 3.12-1B, and 3.12-2.
3. The full length control bank insertion limits are not violated.
4. Axial power distribution control procedures, which are given in terms of flux difference control and control bank insertion limits are observed. Flux difference refers to the difference

between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined as the difference in normalized power between the top and the bottom halves of the core.

The permitted relaxation in $F^N_{\Delta H}$ with decreasing power level allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, this hot channel factor limit is met.

A recent evaluation of DNB test data obtained from experiments of fuel rod bowing in thimble cells has identified that the reduction in DNBR due to rod bowing in thimble cells is more than completely accommodated by existing thermal margins in the core design. Therefore, it is not necessary to continue to apply a rod bow penalty to $F^N_{\Delta H}$.

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 Specification 3.12.B.4 together with the surveillance requirements given in Specification 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 deviations 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 sufficently 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 $\pm$ 13.8 percent ($\pm$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 as possible. This is accomplished, by using the boron system to position the full length control rods to produce the required indicated flux difference.

A 2% quadrant tilt allows that a 5% tilt might actually be present in the core because of insensitivity of the excore detectors for disturbances near the core center such as misaligned inner control rod and an error allowance. No increase in F_Q occurs with tilts up to 5% because misaligned control rods producing such tilts do not extend to the unrodded plane, where the maximum F_Q occurs.

The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

(1) Reactor protection system or engineering safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

(2) Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

Note: Routine surveillance testing, instrument calibration, or preventative maintenance which require system configurations as described in items 2.b(1) and 2.b(2) need not be reported except where test results themselves reveal a degraded mode as described above.

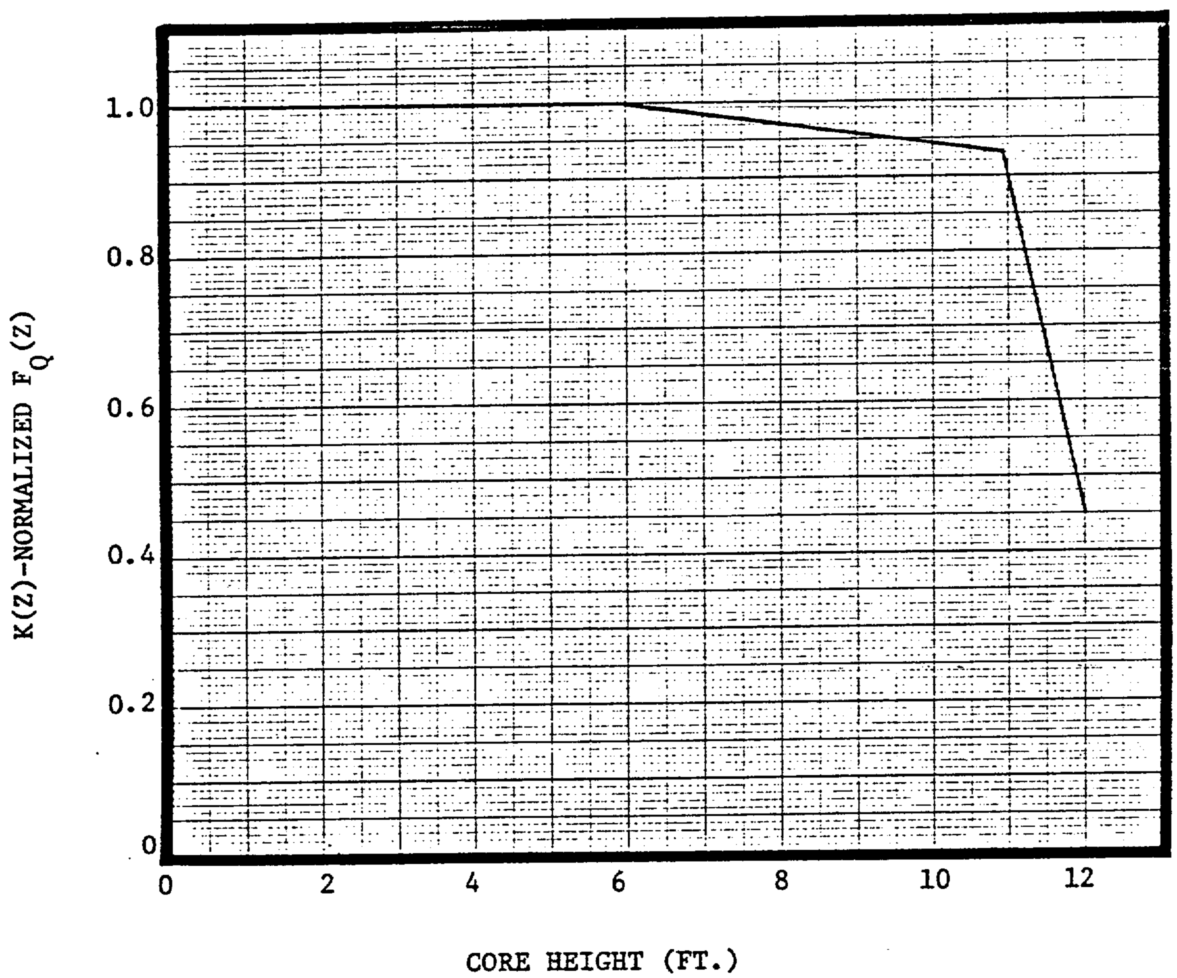
(3) Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

(4) Abnormal degradation of systems other than those specified in item 2.a(3) above designed to contain

HOT CHANNEL FACTOR NORMALIZED

OPERATING ENVELOPE

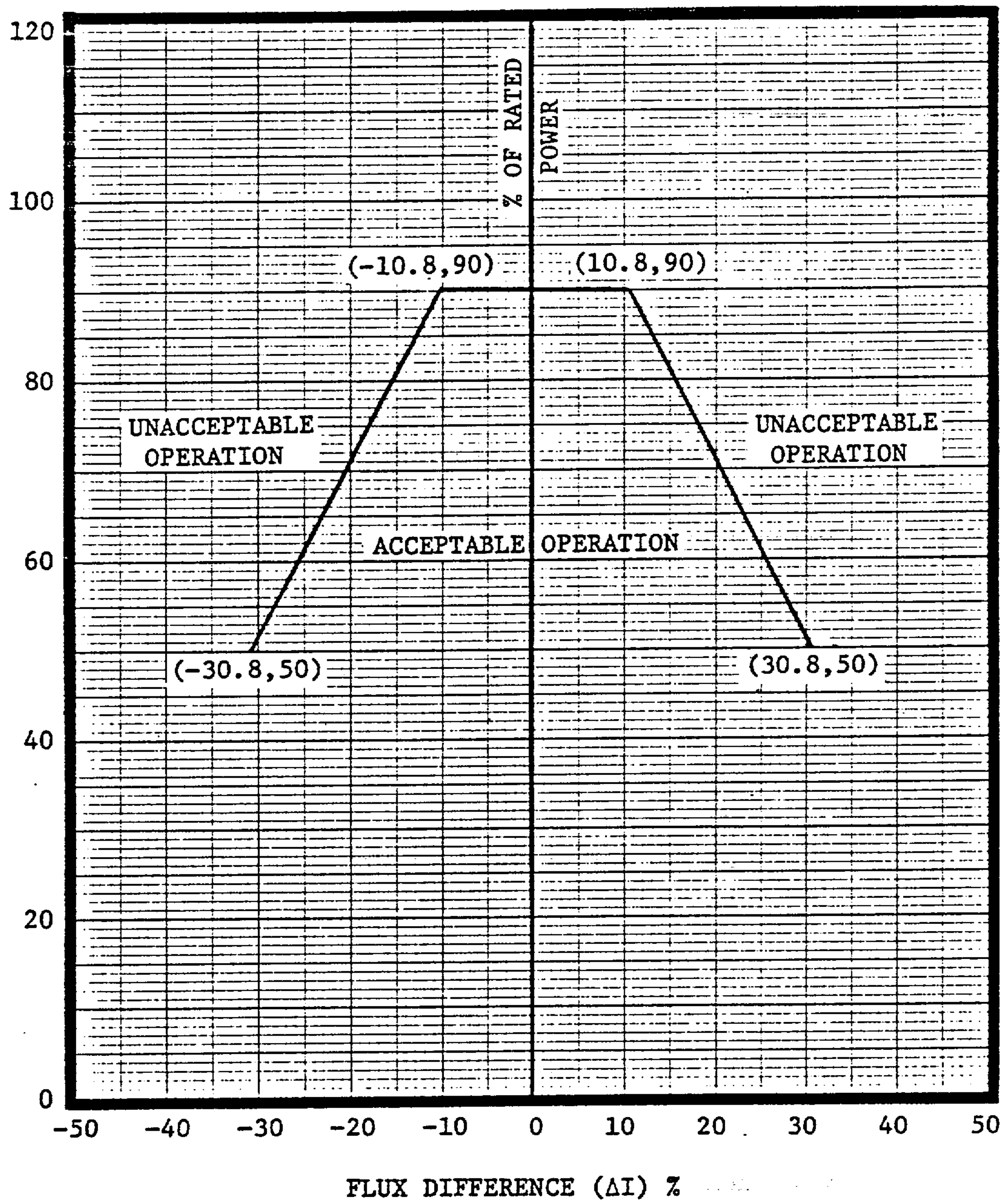
SURRY POWER STATION



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

RELATED TO AMENDMENT NO. 70 TO FACILITY OPERATING LICENSE NO. DPR-32

AND AMENDMENT NO. 70 TO FACILITY OPERATING LICENSE NO. DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-280 AND 50-281

INTRODUCTION

By letter dated April 28, 1981, as supplemented May 15, 1981, Virginia Electric and Power Company (the licensee) requested amendments to License Nos. DPR-32 and DPR-37 for the Surry Power Station, Units Nos. 1 and 2. A letter dated July 28, 1980, also provided information pertinent to these changes. These changes would revise the total peaking factor, F_Q, to a value of 2.18. Editorial changes have also been made as requested in the licensee's request.

DISCUSSION AND EVALUATION

License Amendment No. 58 to Facility Operating License No. DPR-37 for Surry Power Station, Unit 2, dated May 16, 1980, changed the value of F_Q to 2.19 based on a LOCA-ECCS analysis with a steam generator tube plugging limit of 3%. In our Safety Evaluation (SE), supporting the Unit 2 change, we stated that the evaluation could be extended to Unit 1 after the unit is suitably modified to comply with the assumptions made in the ECCS-LOCA analysis.

Since Unit 1 is being modified as indicated in the Unit 2 SE, the SE for Unit 2 applies to Unit 1. Therefore, we conclude that the change to F_Q=2.19 for Unit 2 is applicable to Unit 1.

By letter dated July 28, 1980, the licensee reduced the F_Q value of 2.19 to 2.18 by administrative action. This change resulted from an assessment of the fuel rod modeling concerns raised in NUREG-0630. This assessment indicated that a total peaking factor penalty of 0.007 remained after application of approved benefits giving an F_Q of 2.183. An F_Q of 2.18 will more than account for the penalty. This change is in a conservative direction and an F_Q of 2.18 is acceptable for both Units 1 and 2.

Environmental Consideration

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

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: JUN 16 1981

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-280 AND 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. 70 to Facility Operating License No. DPR-32 and Amendment No. 70 to Facility Operating License No. DPR-37 issued to Virginia Electric and Power Company (the licensee), which revised Technical Specifications for operation of the Surry Power Station, Unit Nos. 1 and 2, respectively, (the facilities), located in Surry County, Virginia. The amendments are effective as of the date of issuance.

These amendments revise the Technical Specifications to change the heat flux hot channel factor (F_Q) to 2.18 for Units 1 and 2. These amendments also make editorial changes to the Technical Specifications.

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

The Commission has determined that the issuance of 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 amendments dated April 28, 1981, as supplemented May 15, 1981, (2) Amendment Nos. 70 and 70 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 23185. 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 Licensing.

Dated at Bethesda, Maryland, this 16 day of June, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing