November 26, 1976

Dockets Nos.: 50-280
and 50-281

Virginia Electric & Power Company
ATTN: Mr. W. L. Proffitt
Senior Vice President - Power
P. O. Box 26666
Richmond, Virginia 23261

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Gentlemen:

The Commission has issued the enclosed Amendments No. 26 to Facility Operating Licenses Nos. DPR-32 and DPR-37 for the Surry Power Station Units Nos. 1 and 2. The amendments consist of an added condition to the license for Surry Unit No. 2 and changes to the Technical Specifications in partial response to your application dated September 27, 1976, as supplemented October 29, 1976, and your submittals dated October 19, 1976, and November 15, 1976. We did not include any revisions in your Technical Specifications related to the Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$); hence, you should continue to operate Surry Unit No. 2 under the existing Technical Specifications for $F_{\Delta H}$ and the additional restrictions of your August 18, 1976 letter (Serial No. 194).

These amendments concern changes required as a result of the steam generator repair for Surry Unit No. 2 and the consequent need for revision to the emergency core cooling system evaluation and the power distribution and power distribution monitoring requirements. The revised emergency core cooling system evaluation also fulfills the requirements of our Order for Modification of License dated August 27, 1976.

Based on the conclusions of our enclosed Safety Evaluation, we concur that the repair program for the steam generators of Surry Unit No. 2 is adequate subject to the conditions of the amendment to the license of Surry Unit No. 2. Our evaluation of the repair program for the steam generators of Surry Unit No. 1 has not been completed.

You are requested to submit the details of the steam generator inspection program which you plan for Surry Unit No. 2 after two months

of operation. These details should be submitted no later than 30 days prior to the date you expect the inspection to commence.

Copies of the Federal Register Notice are also enclosed.

Sincerely,

Original Signed by

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Enclosures:
1. Amendment No. 26 to DPR-32
2. Amendment No. 26 to DPR-37
3. Safety Evaluation
4. Federal Register Notice

cc w/enclosures: See next page

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Virginia Electric & Power Company

cc w/enclosure(s):
Michael W. Maupin, Esq.
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Richmond, Virginia 23213

Swem Library
College of William & Mary
Williamsburg, Virginia 23185

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

cc w/enclosure(s) & incoming
dtd: 9/27/76, 10/29/76, 10/19/76 & 11/15/76
Commonwealth of Virginia
Council on the Environment
903 9th Street Office Building
Richmond, Virginia 23219

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

VIRGINIA ELECTRIC & POWER COMPANY

DOCKET NO. 50-280

SURRY POWER STATION UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 26
License No. DPR-32

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric & Power Company (the licensee) dated September 27, 1976, as supplemented October 29, 29, 1976, 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.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 26, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 26

FACILITY OPERATING LICENSE NO. DPR-32

DOCKET NO. 50-280

Revise Appendix A Technical Specifications as follows:

Remove Pages	Insert Pages
3.3-1	3.3-1
3.12-3 & 3.12-4	3.12-3 - 3.12-4b
3.12-6 - 3.12-8	3.12-6 - 3.12-8
3.12-14	3.12-14
3.12-16 - 3.12-20	3.12-16 - 3.12-20
3.12-22	3.12-22
-	Table 3.12-1A
-	Table 3.12-1B
Figure 3.12-1A	Figure 3.12-1A
Figure 3.12-8	Figure 3.12-8
4.10-1	4.10-1
5.3-1 - 5.3-3	5.3-1 - 5.3-3
6.6-9	6.6-9

Changes on the revised pages are shown by marginal lines.

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 1075 ft^3 and a maximum of 1089 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

7.

DELETED

B. Power Distribution Limits

1. At all times except during low power physics tests and implementation of 3.12.B.2.b.(2), the hot channel factors defined in the basis must meet the following limits:

$F_Q(Z) \leq (2.00/P) \times K(Z)$ for $P > .5$

$F_Q(Z) \leq (4.00) \times K(Z)$ for $P \leq .5$

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

where P is the fraction of rated power at which the core is operating, K(Z) is the function given in Figure 3.12-8, 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 three percent to account for manufacturing tolerances and further increased by five percent to account for measurement error, and the measurement of enthalpy rise hot channel factor, $F^N_{\Delta H}$, shall be increased by four percent to account for measurement error. If either 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 $F_Q \leq 2.00 \times K(Z)$ and $F^N_{\Delta H} \leq 1.55$ within 24 hours, the Overpower ΔT and Overtemperature ΔT trip setpoints shall be similarly reduced.

b. $F_Q(Z)$ shall be evaluated for normal (Condition I) operation of each unit 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 3.12-1A and 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 strap locations. The measured values of $F_{xy}(Z)$ shall be increased by three percent to account for radial xenon redistribution effects associated with normal (Condition I) operation. (In addition, the value of $F_{xy}(Z)$ for Unit 1 shall be increased by two and one half percent to account for the predicted increase in the values of $F_{xy}(Z)$ during each effective full power month. This additional percent penalty on the values of $F_{xy}(Z)$ for Unit 1 shall be applicable up to 9000 MWD/MTU burnup.) The resulting $F_Q(Z)$ shall then be increased by three percent to account for manufacturing tolerances and further increased by five percent to account for measurement error. 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

which $\overline{R}_j$, as defined in the Basis, has been determined. This shall be done immediately following and as a minimum at 30, 60, 90, 120, 240, and 480 minutes following the events listed below and every eight hours thereafter:

i. Raising the thermal power above $P_{THRESHOLD}$, or

ii. Movement of the control bank of rods more than an accumulated total of five steps in any one direction while reactor power is greater than $P_{THRESHOLD}$ except during control rod assembly exercises and excore detector calibrations.

(b) If $F_Q(Z)\Big|^{j}_{APDM}$ exceeds its limit, $F_Q(Z)$ as defined in 3.12.B.1, the reactor power shall be reduced until the limit is met or until thermal power is reduced to $P_{THRESHOLD}$.

by -18 percent and +11.5 percent at 90% power. (One half of the time the indicated axial flux difference is out of its target band at power levels up to 50 percent of rated power is to be counted as contributing to the one hour cumulative maximum flux difference deviation from its target band at a power level less than or equal 90 percent of rated power.) For every 4 percent below 90% power, the permissible positive flux difference boundary is extended by 1 percent. For every 5 percent below 90% power, the permissible negative flux difference boundary is extended by 2 percent.

(2) If 3.12.B.4.b(1) is violated then the reactor power shall be reduced to no greater than 50% power and the high neutron flux setpoint shall be reduced to no greater than 55% power.

(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.

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 two hours (cumulative) out of the preceding 24 hour period in which the power level is no greater than 50 percent of rated power.

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. 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.

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

DELETED

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.

7. If, except for physics and rod exercise testing, after a further period of 24 hours, the power tilt in 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, 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 mechanisms, i.e. programming circuitry, the provisions

$F_Q(Z)$, 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.

It should be noted that $F_{\Delta H}^N$ is based on an integral and is used as such in the DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes throughout the core. Thus the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F_{\Delta H}^N$.

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 2.00 times the hot channel factor normalized operating envelope of TS Figure 3.12-8.

For normal operation, it has been determined that, provided certain conditions are observed, the enthalpy rise hot channel factor, $F^N_{\Delta H}$, 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 and part length control bank insertion limits are not violated.

DELETED

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 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.

DELETED

For normal (Condition I) operation, 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 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 five percent to account for measurement uncertainty, three percent to account for manufacturing tolerances, three percent to account for the effects of the radial redistribution of xenon during normal (Condition I) operation, and for Unit 1, two and one half percent to account for the increase in the value of $F_{xy}(Z)$ as a function of burnup out to 9000 MWD/MTU burnup. $P_{THRESHOLD}$ 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

factor result from a full core flux map and the axial peaking factor in a selected thimble. Based on this approach, the operational value of the heat flux hot channel factor, $F_Q(Z)\Big|^{j}_{APDM}$ is derived as follows:

$$F_Q(Z)\Big|^{j}_{APDM} = F_j(Z)\ (\overline{R}_j)\ (1.03)\ (1 + \sigma_j)\ (1.07)$$

where:

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

DELETED

b. $F_Q(Z)\Big|^{j}_{APDM}$ is the operational value of the heat flux hot channel factor for the purpose of this surveillance.

c. $\overline{R}_j$, for thimble j, is determined from at least n=6 incore flux maps covering the full configuration of permissible rod patterns for power levels for which this surveillance is required.

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

where

$$R_{ij} = \frac{F_{Q_i}^{meas}}{(F_{ij}(Z))_{max}}$$

and $F_{ij}(Z)$ is the normalized axial power distribution from thimble at elevation Z in map i which had a measured peaking factor without uncertainties of densification allowance of $F_{Q_i}^{meas}$.

The full incore flux map used to update $\overline{R}_j$ shall be taken at least per every regular effective full power month. The continued accuracy and representativeness of the selected thimbles shall be verified by using the latest flux maps to update the $\overline{R}_j$ for each representative thimble.

e. σ_j is the standard deviation of $\overline{R}_j$ and is derived from n flux maps covering the full configuration of permissible rod patterns for power levels for which surveillance is required using the relationship below, or 0.02, whichever is greater:

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

f. The factor 1.03 reduction in the (kw/ft) limit is the engineering uncertainty factor.

g. The factor 1.07 is the combined uncertainty associated with the measurement of F_{Q_i} and $F_{ij}(Z)_{max}$.

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.b 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 +6 to -9% 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

as possible. This is accomplished, by using the boron system to position the full length control rods to produce the required indicated flux difference.

DELETED

A 2% quadrant tilt allows that a 5% tilt might actually be present in the core because of intensitivity of the excore detectors for disturbances near the core center such as misaligned inner control rods 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.

SURRY UNIT 1

CYCLE 4

CORE HEIGHT (Feet)	$F_Z(Z)$
1.5	1.318
2.0	1.318
2.5	1.309
3.0	1.362
3.5	1.391
4.0	1.408
4.5	1.416
5.0	1.415
5.5	1.401
6.0	1.375
6.5	1.336
7.0	1.300
7.5	1.274
8.0	1.240
8.5	1.212
9.0	1.218
9.5	1.258
10.0	1.269
10.5	1.231

TABLE 3.12-1A: DESIGN CONDITION I AXIAL PEAKING FACTORS, $F_Z(Z)$ VS. CORE HEIGHT FOR SURRY UNIT I

SURRY UNIT 2

CYCLE 3

CORE HEIGHT (Feet)	$F_Z(Z)$
1.5	1.340
2.0	1.321
2.5	1.280
3.0	1.293
3.5	1.264
4.0	1.282
4.5	1.296
5.0	1.306
5.5	1.306
6.0	1.259
6.5	1.289
7.0	1.273
7.5	1.273
8.0	1.268
8.5	1.253
9.0	1.231
9.5	1.202
10.0	1.221
10.5	1.225

TABLE 3.12-1B: DESIGN CONDITION I AXIAL PEAKING FACTORS, $F_Z(Z)$ VS. CORE HEIGHT FOR SURRY 2

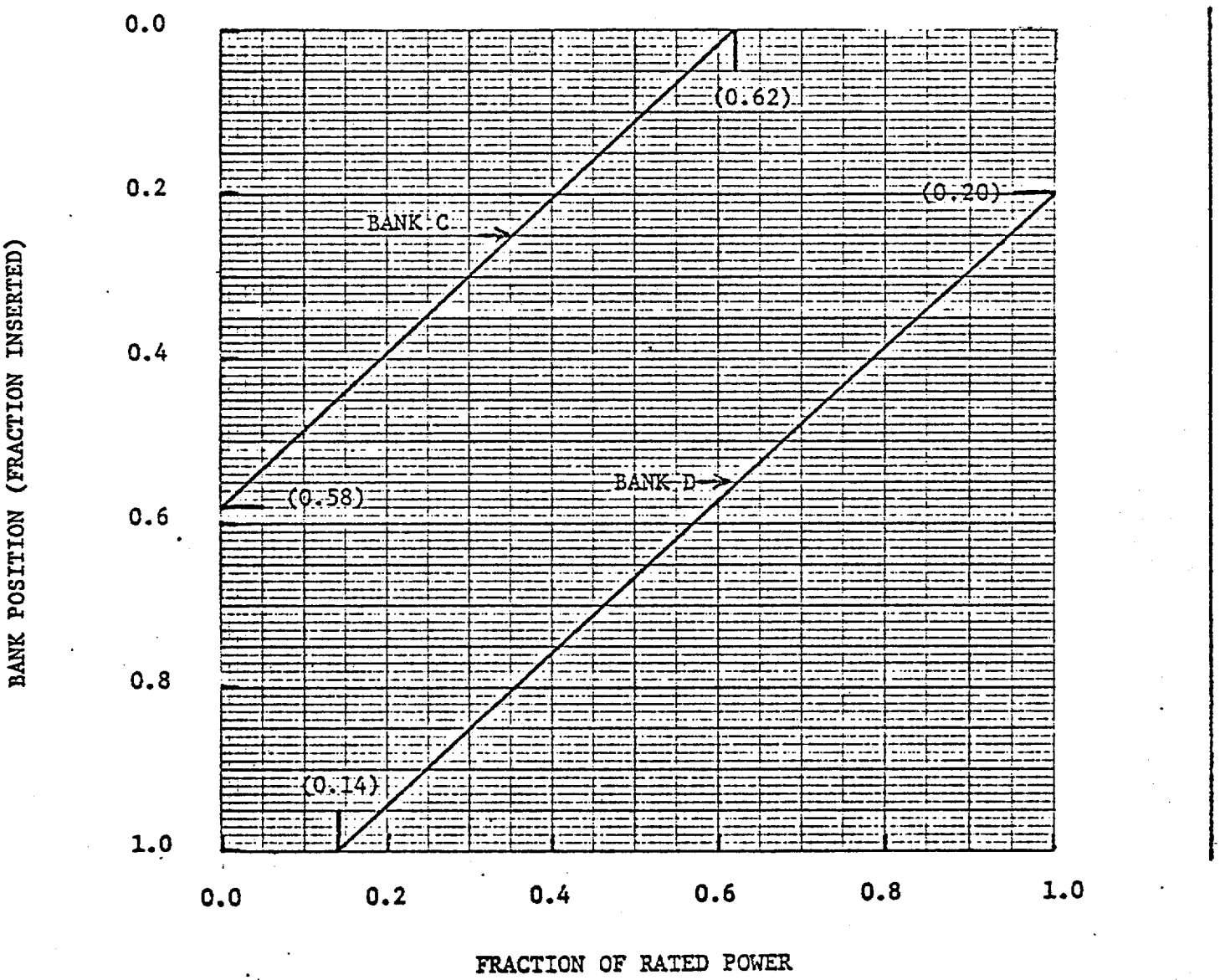
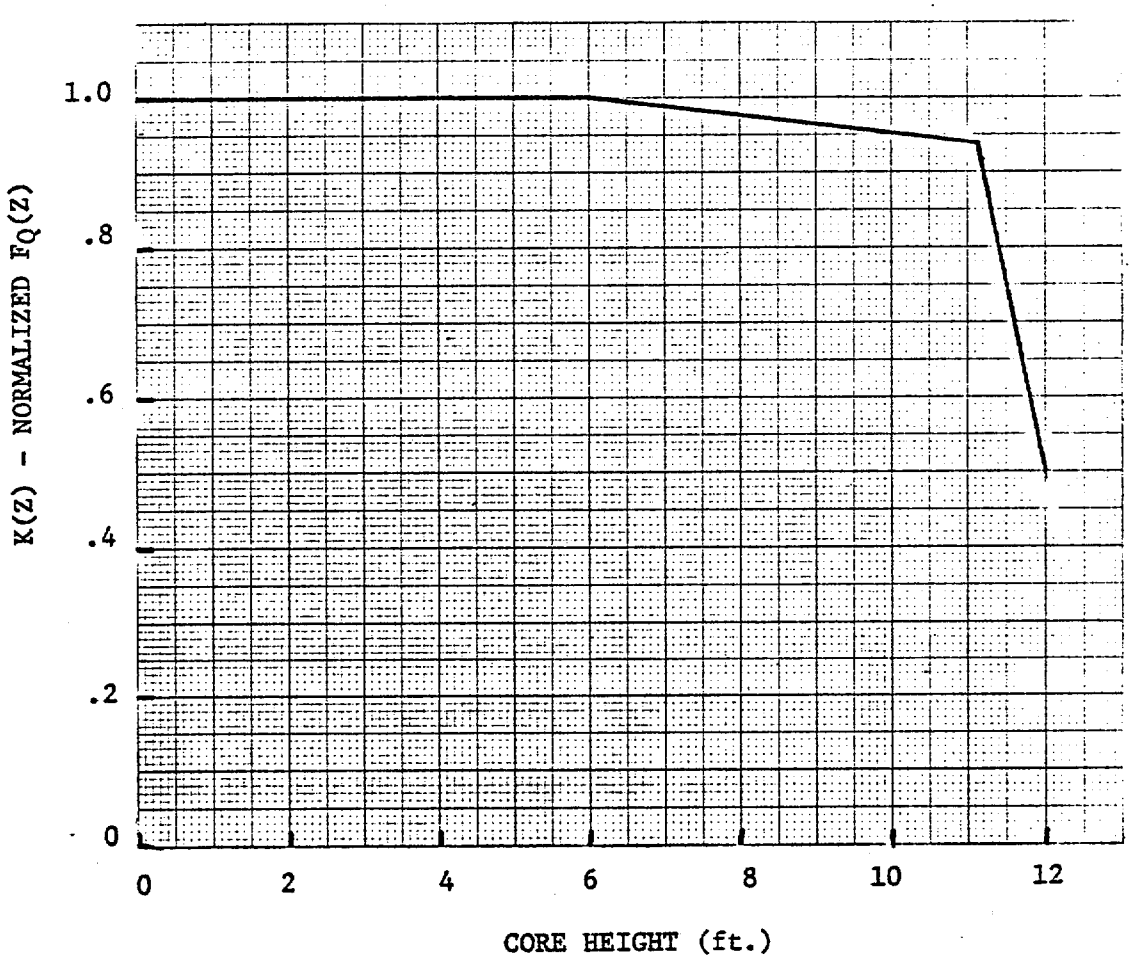


FIGURE 3.12-1A CONTROL BANK INSERTION LIMITS FOR 3-LOOP NORMAL OPERATION-UNIT 1

HOT CHANNEL FACTOR NORMALIZED
OPERATING ENVELOPE

SURRY POWER STATION
UNIT NOS. 1 AND 2



4.10 REACTIVITY ANOMALIES

Applicability

Applies to potential reactivity anomalies.

Objective

To require evaluation of applicable reactivity anaomalies within the reactor.

Specification

A. Following a normalization of the computed boron concentration as a function of burnup, the actual boron concentration of the coolant shall be compared monthly with the predicted value. If the difference between the observed and predicted steady-state concentrations reaches the equivalent of one percent in reactivity, an evaluation as to the cause of the discrepancy shall be made and reported to the Nuclear Regulatory Commission per Section 6.6 of these Specifications.

B. During periods of power operation at greater than 10% of power, the hot channel factors, F_Q and $F^N_{\Delta H}$ shall be determined during each effective full power month of operation using data from limited core maps. If these factors exceed values of

$$F_Q(Z) \leq (2.00/P) \times K(Z) \text{ for } P > .5$$

$$F_Q(Z) \leq (4.00) \times K(Z) \text{ for } P \leq .5$$

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

5.3 REACTOR

Applicability

Applies to the reactor core, Reactor Coolant System, and Safety Injection System.

Objective

To define those design features which are essential in providing for safe system operations.

Specifications

A. Reactor Core

1. The reactor core contains approximately 176,200 lbs of uranium dioxide in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 tubing to form fuel rods. All fuel rods are pressurized with helium during fabrication. The reactor core is made up of 157 fuel assemblies. Each fuel assembly contains 204 fuel rods except for two demonstration fuel assemblies in Unit 2 which are part of Region 4 fuel. The demonstration assemblies each contain 264 fuel rods.

2. The average enrichment of the initial core is 2.51 weight per cent of U-235. Three fuel enrichments are used in the initial core. The highest enrichment is 3.12 weight per cent of U-235.

3. Reload fuel will be similar in design to the initial core. The enrichment of reload fuel will not exceed 3.60 weight percent of U-235.

4. Burnable poison rods are incorporated in the initial core. There are 816 poison rods in the form of 12 rod clusters, which are located in vacant control rod assembly guide thimbles. The burnable poison rods consist of pyrex clad with stainless steel.

5. There are 48 full-length control rod assemblies and 5 part-length control rod assemblies in the reactor core. The full-length control rod assemblies contain a 144-inch length of silver-indium-cadmium alloy clad with stainless steel. The part-length control rod assemblies contain a 36-inch length of silver-indium-cadmium alloy with the remainder of the stainless steel sheath filled with Al_2O_3.

6. Surry Unit 1, Cycle 4, Surry Unit 2, Cycle 3, and subsequent cores will meet the following criteria at all times during the operating lifetime.

   a. Hot channel factors:

   $F_Q(Z) \leq (2.00/P) \times K(Z)$ for $P > 0.5$

   $F_Q(Z) \leq (4.00) \times K(Z)$ for $P \leq 0.5$

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

   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 of F_Q.

b. The moderator temperature coefficient in the power operating range is less than or equal to:

1) +3.0 pcm/°F at less than 50% of rated power, or

2) +3.0 pcm/°F at 50% of rated power and linearly decreasing to 0 pcm/°F at rated power.

c. Capable of being made subcritical in accordance with Specification 3.12 A.3.C

7. Up to 10 grams of enriched fissionable material may be used either in the core or available on the plant site, in the form of fabricated neutron flux detectors for the purposes of monitoring core neutron flux.

B. Reactor Coolant System

1. The design of the Reactor Coolant System complies with the code requirements specified in Section 4 of the FSAR.

2. All piping, components, and supporting structures of the Reactor Coolant System are designed to Class 1 seismic requirements, and have been designed to withstand:

a. Primary operating stresses combined with the Operational seismic stresses resulting from a horizontal ground acceleration of 0.07g and a simultaneous vertical ground acceleration of 2/3 the horizontal, with the stresses maintained within code allowable working stresses.

b. Primary operating stresses when combined with the Design Basis Earthquake seismic stresses resulting from a horizontal ground acceleration of 0.15g and a simulataneous vertical ground

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 engineered 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. Specifically, the implementation of 3.12.B.2.b.(2) is not reportable.

(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

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

VIRGINIA ELECTRIC & POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 26
License No. DPR-37

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric & Power Company (the licensee) dated September 27, 1976, as supplemented October 29, 1976, 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 adding a new Paragraph 3.E. as follows and by changes to the Technical Specifications as indicated in the attachment to this license amendment:

"E. Steam Generator Inspection

In order to perform an inspection of the steam generators, the plant shall be brought to the cold shutdown condition within 61 equivalent days of operation from the effective date of issuance of this amendment. For the purpose of this requirement, equivalent operation is defined as operation with a primary coolant temperature greater than 350°F. Nuclear Regulatory Commission approval shall be obtained before resuming power operation following this inspection.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 26, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 26

FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NO. 50-281

Revise Appendix A Technical Specifications as follows:

Remove Pages	Insert Pages
3.3-1	3.3-1
3.12-3 & 3.12-4	3.12-3 - 3.12-4b
3.12-6 - 3.12-8	3.12-6 - 3.12-8
3.12-14	3.12-14
3.12-16 - 3.12-20	3.12-16 - 3.12-20
3.12-22	3.12-22
-	Table 3.12-1A
-	Table 3.12-1B
Figure 3.12-1A	Figure 3.12-1A
Figure 3.12-8	Figure 3.12-8
4.10-1	4.10-1
5.3-1 - 5.3-3	5.3-1 - 5.3-3
6.6-9	6.6-9

Changes on the revised pages are shown by marginal lines.

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 1075 ft^3 and a maximum of 1089 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

7.

DELETED

B. Power Distribution Limits

1. At all times except during low power physics tests and implementation of 3.12.B.2.b.(2), the hot channel factors defined in the basis must meet the following limits:

$F_Q(Z) \leq (2.00/P) \times K(Z)$ for $P > .5$

$F_Q(Z) \leq (4.00) \times K(Z)$ for $P \leq .5$

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

where P is the fraction of rated power at which the core is operating, K(Z) is the function given in Figure 3.12-8, 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 three percent to account for manufacturing tolerances and further increased by five percent to account for measurement error, and the measurement of enthalpy rise hot channel factor, $F^N_{\Delta H}$, shall be increased by four percent to account for measurement error. If either 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 $F_Q \leq 2.00 \times K(Z)$ and $F^N_{\Delta H} \leq 1.55$ within 24 hours, the Overpower ΔT and Overtemperature ΔT trip setpoints shall be similarly reduced.

b. $F_Q(Z)$ shall be evaluated for normal (Condition I) operation of each unit 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 3.12-1A and 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 strap locations. The measured values of $F_{xy}(Z)$ shall be increased by three percent to account for radial xenon redistribution effects associated with normal (Condition I) operation. (In addition, the value of $F_{xy}(Z)$ for Unit 1 shall be increased by two and one half percent to account for the predicted increase in the values of $F_{xy}(Z)$ during each effective full power month. This additional percent penalty on the values of $F_{xy}(Z)$ for Unit 1 shall be applicable up to 9000 MWD/MTU burnup.) The resulting $F_Q(Z)$ shall then be increased by three percent to account for manufacturing tolerances and further increased by five percent to account for measurement error. 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

which $\overline{R}_j$, as defined in the Basis, has been determined. This shall be done immediately following and as a minimum at 30, 60, 90, 120, 240, and 480 minutes following the events listed below and every eight hours thereafter:

i. Raising the thermal power above $P_{THRESHOLD}$, or

ii. Movement of the control bank of rods more than an accumulated total of five steps in any one direction while reactor power is greater than $P_{THRESHOLD}$ except during control rod assembly exercises and excore detector calibrations.

(b) If $F_Q(Z)\Big|^{j}_{APDM}$ exceeds its limit, $F_Q(Z)$ as defined in 3.12.B.1, the reactor power shall be reduced until the limit is met or until thermal power is reduced to $P_{THRESHOLD}$.

by -18 percent and +11.5 percent at 90% power. (One half of the time the indicated axial flux difference is out of its target band at power levels up to 50 percent of rated power is to be counted as contributing to the one hour cumulative maximum flux difference deviation from its target band at a power level less than or equal 90 percent of rated power.) For every 4 percent below 90% power, the permissible positive flux difference boundary is extended by 1 percent. For every 5 percent below 90% power, the permissible negative flux difference boundary is extended by 2 percent.

(2) If 3.12.B.4.b(1) is violated then the reactor power shall be reduced to no greater than 50% power and the high neutron flux setpoint shall be reduced to no greater than 55% power.

(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.

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 two hours (cumulative) out of the preceding 24 hour period in which the power level is no greater than 50 percent of rated power.

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. 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.

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

DELETED

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.

7. If, except for physics and rod exercise testing, after a further period of 24 hours, the power tilt in 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, 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 mechanisms, i.e. programming circuitry, the provisions

$F_Q(Z)$, 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.

It should be noted that $F_{\Delta H}^N$ is based on an integral and is used as such in the DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes throughout the core. Thus the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F_{\Delta H}^N$.

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 2.00 times the hot channel factor normalized operating envelope of TS Figure 3.12-8.

For normal operation, it has been determined that, provided certain conditions are observed, the enthalpy rise hot channel factor, $F^N_{\Delta H}$, 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 and part length control bank insertion limits are not violated.

DELETED

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 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.

DELETED

For normal (Condition I) operation, 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 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 five percent to account for measurement uncertainty, three percent to account for manufacturing tolerances, three percent to account for the effects of the radial redistribution of xenon during normal (Condition I) operation, and for Unit 1, two and one half percent to account for the increase in the value of $F_{xy}(Z)$ as a function of burnup out to 9000 MWD/MTU burnup. $P_{THRESHOLD}$ 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

factor result from a full core flux map and the axial peaking factor in a selected thimble. Based on this approach, the operational value of the heat flux hot channel factor, $F_Q(Z)\Big|^{j}_{APDM}$ is derived as follows:

$$F_Q(Z)\Big|^{j}_{APDM} = F_j(Z)\ (\overline{R}_j)\ (1.03)\ (1 + \sigma_j)\ (1.07)$$

where:

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

DELETED

b. $F_Q(Z)\Big|^{j}_{APDM}$ is the operational value of the heat flux hot channel factor for the purpose of this surveillance.

c. $\overline{R}_j$, for thimble j, is determined from at least n=6 incore flux maps covering the full configuration of permissible rod patterns for power levels for which this surveillance is required.

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

where

$$R_{ij} = \frac{F_{Q_i}^{meas}}{(F_{ij}(Z))_{max}}$$

and $F_{ij}(Z)$ is the normalized axial power distribution from thimble at elevation Z in map i which had a measured peaking factor without uncertainties of densification allowance of $F_{Q_i}^{meas}$.

The full incore flux map used to update $\overline{R}_j$ shall be taken at least per every regular effective full power month. The continued accuracy and representativeness of the selected thimbles shall be verified by using the latest flux maps to update the $\overline{R}_j$ for each representative thimble.

e. σ_j is the standard deviation of $\overline{R}_j$ and is derived from n flux maps covering the full configuration of permissible rod patterns for power levels for which surveillance is required using the relationship below, or 0.02, whichever is greater:

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

f. The factor 1.03 reduction in the (kw/ft) limit is the engineering uncertainty factor.

g. The factor 1.07 is the combined uncertainty associated with the measurement of F_{Q_i} and $F_{ij}(Z)_{max}$.

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.b 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 +6 to -9% 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

as possible. This is accomplished, by using the boron system to position the full length control rods to produce the required indicated flux difference.

DELETED

A 2% quadrant tilt allows that a 5% tilt might actually be present in the core because of intensitivity of the excore detectors for disturbances near the core center such as misaligned inner control rods 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.

SURRY UNIT 1

CYCLE 4

CORE HEIGHT (Feet)	$F_Z(Z)$
1.5	1.318
2.0	1.318
2.5	1.309
3.0	1.362
3.5	1.391
4.0	1.408
4.5	1.416
5.0	1.415
5.5	1.401
6.0	1.375
6.5	1.336
7.0	1.300
7.5	1.274
8.0	1.240
8.5	1.212
9.0	1.218
9.5	1.258
10.0	1.269
10.5	1.231

TABLE 3.12-1A: DESIGN CONDITION I AXIAL PEAKING FACTORS, $F_Z(Z)$ VS. CORE HEIGHT FOR SURRY UNIT I

SURRY UNIT 2

CYCLE 3

CORE HEIGHT (Feet)	$F_Z(Z)$
1.5	1.340
2.0	1.321
2.5	1.280
3.0	1.293
3.5	1.264
4.0	1.282
4.5	1.296
5.0	1.306
5.5	1.306
6.0	1.259
6.5	1.289
7.0	1.273
7.5	1.273
8.0	1.268
8.5	1.253
9.0	1.231
9.5	1.202
10.0	1.221
10.5	1.225

TABLE 3.12-1B: DESIGN CONDITION I AXIAL PEAKING FACTORS, $F_Z(Z)$ VS. CORE HEIGHT FOR SURRY 2

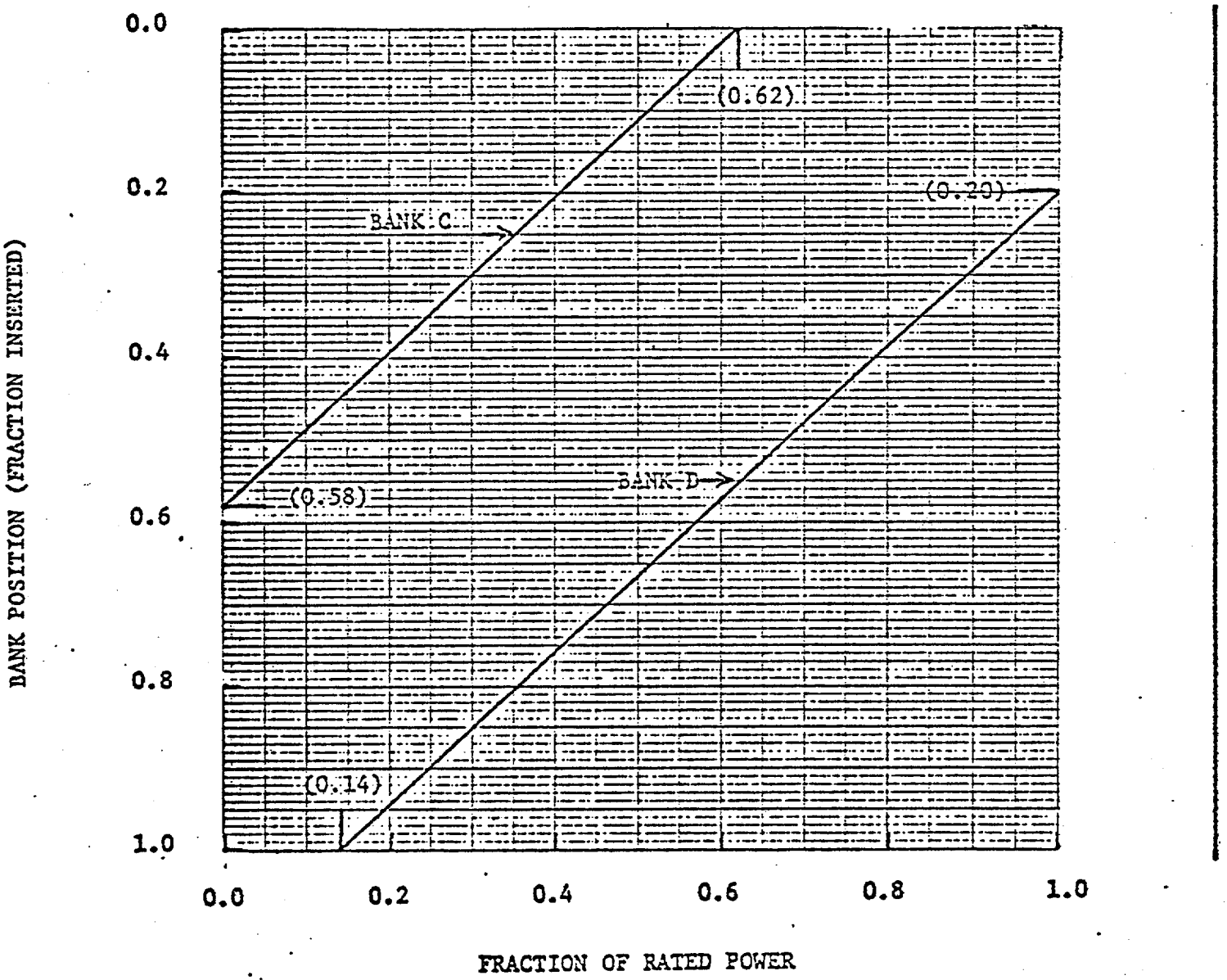
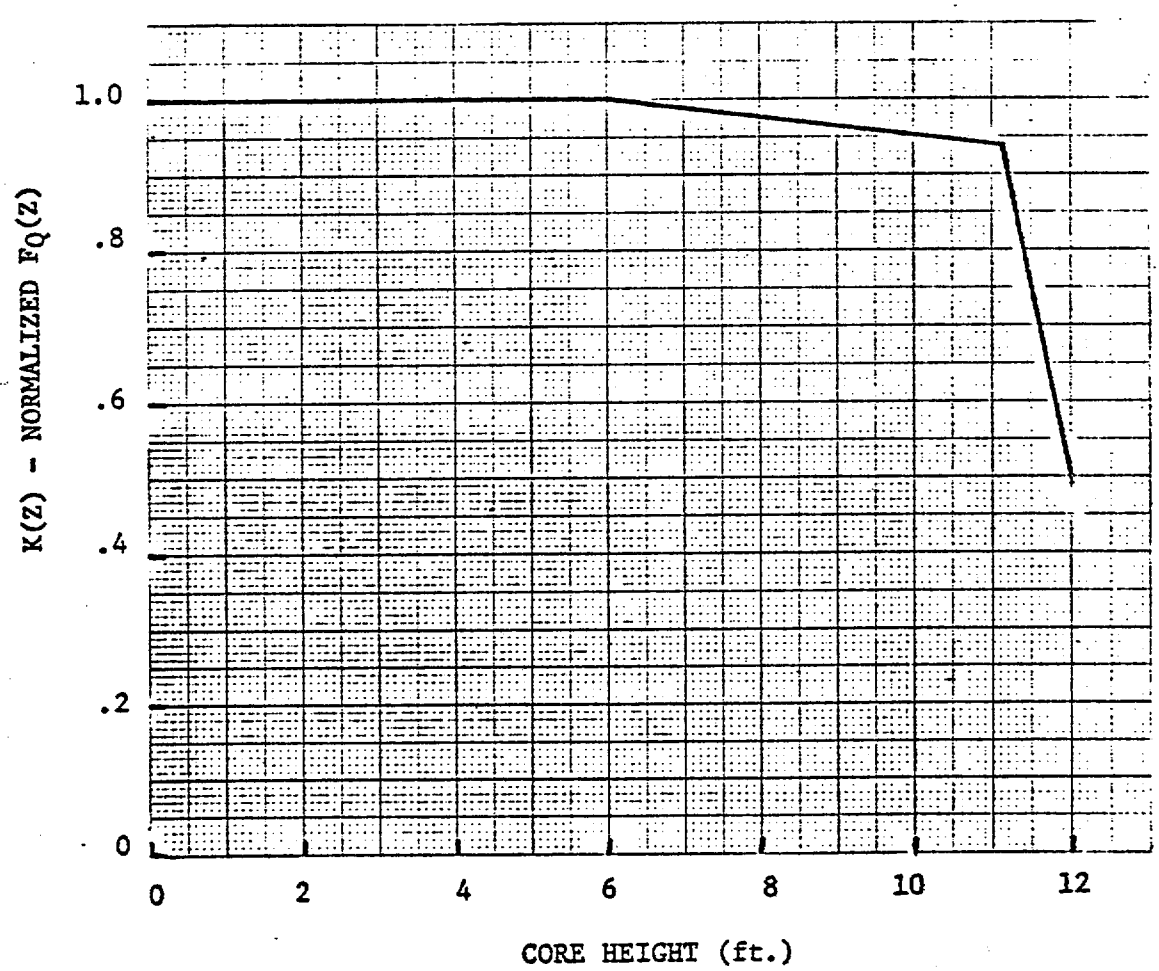


FIGURE 3.12-1A CONTROL BANK INSERTION LIMITS FOR 3-LOOP NORMAL OPERATION-UNIT 1

HOT CHANNEL FACTOR NORMALIZED
OPERATING ENVELOPE

SURRY POWER STATION
UNIT NOS. 1 AND 2



4.10 REACTIVITY ANOMALIES

Applicability

Applies to potential reactivity anomalies.

Objective

To require evaluation of applicable reactivity anaomalies within the reactor.

Specification

A. Following a normalization of the computed boron concentration as a function of burnup, the actual boron concentration of the coolant shall be compared monthly with the predicted value. If the difference between the observed and predicted steady-state concentrations reaches the equivalent of one percent in reactivity, an evaluation as to the cause of the discrepancy shall be made and reported to the Nuclear Regulatory Commission per Section 6.6 of these Specifications.

B. During periods of power operation at greater than 10% of power, the hot channel factors, F_Q and $F^N_{\Delta H}$ shall be determined during each effective full power month of operation using data from limited core maps. If these factors exceed values of

$$F_Q(Z) \leq (2.00/P) \times K(Z) \text{ for } P > .5$$

$$F_Q(Z) \leq (4.00) \times K(Z) \text{ for } P \leq .5$$

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

5.3 REACTOR

Applicability

Applies to the reactor core, Reactor Coolant System, and Safety Injection System.

Objective

To define those design features which are essential in providing for safe system operations.

Specifications

A. Reactor Core

1. The reactor core contains approximately 176,200 lbs of uranium dioxide in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 tubing to form fuel rods. All fuel rods are pressurized with helium during fabrication. The reactor core is made up of 157 fuel assemblies. Each fuel assembly contains 204 fuel rods except for two demonstration fuel assemblies in Unit 2 which are part of Region 4 fuel. The demonstration assemblies each contain 264 fuel rods.

2. The average enrichment of the initial core is 2.51 weight per cent of U-235. Three fuel enrichments are used in the initial core. The highest enrichment is 3.12 weight per cent of U-235.

3. Reload fuel will be similar in design to the initial core. The enrichment of reload fuel will not exceed 3.60 weight percent of U-235.

4. Burnable poison rods are incorporated in the initial core. There are 816 poison rods in the form of 12 rod clusters, which are located in vacant control rod assembly guide thimbles. The burnable poison rods consist of pyrex clad with stainless steel.

5. There are 48 full-length control rod assemblies and 5 part-length control rod assemblies in the reactor core. The full-length control rod assemblies contain a 144-inch length of silver-indium-cadmium alloy clad with stainless steel. The part-length control rod assemblies contain a 36-inch length of silver-indium-cadmium alloy with the remainder of the stainless steel sheath filled with Al_2O_3.

6. Surry Unit 1, Cycle 4, Surry Unit 2, Cycle 3, and subsequent cores will meet the following criteria at all times during the operating lifetime.

   a. Hot channel factors:

$$F_Q(Z) \leq (2.00/P) \times K(Z) \text{ for } P > 0.5$$

$$F_Q(Z) \leq (4.00) \times K(Z) \text{ for } P \leq 0.5$$

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

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 of F_Q.

b. The moderator temperature coefficient in the power operating range is less than or equal to:

1) +3.0 pcm/°F at less than 50% of rated power, or

2) +3.0 pcm/°F at 50% of rated power and linearly decreasing to 0 pcm/°F at rated power.

c. Capable of being made subcritical in accordance with Specification 3.12 A.3.C

7. Up to 10 grams of enriched fissionable material may be used either in the core or available on the plant site, in the form of fabricated neutron flux detectors for the purposes of monitoring core neutron flux.

B. Reactor Coolant System

1. The design of the Reactor Coolant System complies with the code requirements specified in Section 4 of the FSAR.

2. All piping, components, and supporting structures of the Reactor Coolant System are designed to Class 1 seismic requirements, and have been designed to withstand:

a. Primary operating stresses combined with the Operational seismic stresses resulting from a horizontal ground acceleration of 0.07g and a simultaneous vertical ground acceleration of 2/3 the horizontal, with the stresses maintained within code allowable working stresses.

b. Primary operating stresses when combined with the Design Basis Earthquake seismic stresses resulting from a horizontal ground acceleration of 0.15g and a simulataneous vertical ground

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 engineered 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. Specifically, the implementation of 3.12.B.2.b.(2) is not reportable.

(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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENTS NO. 26 TO LICENSES NOS. DPR-32 AND DPR-37

VIRGINIA ELECTRIC & POWER COMPANY

SURRY POWER STATION UNITS NOS. 1 AND 2

DOCKETS NOS. 50-280 AND 50-281

Introduction

As a consequence of an 80 GPM leak in a steam generator tube at the Virginia Electric & Power Company's (VEPCO) Surry Power Station Unit 2 on September 15, 1976, the reactor was shutdown for corrective action. The steam generator tubes form part of the primary coolant system boundary. A program to investigate the cause of the leak and to take corrective action to reduce the likelihood of such an event during future operation has been completed and by letter dated November 15, 1976, VEPCO has requested NRC concurrence to resume power operation.

Approximately 400 tubes in each steam generator at Surry Unit 2 were plugged in order to reduce the likelihood of future steam generator tube failures. The steam generator tube plugging increases the calculated peak clad temperature in the unlikely event of a loss-of-coolant accident. Prior to the shutdown for the steam generator tube leak, Surry Unit 2 was operating under the conditions of an NRC Order for Modification of License dated August 27, 1976, which included operating restrictions and required an analysis which compensated for the effects on the loss-of-coolant accident evaluation of a higher than expected temperature of the primary coolant in the upper head region. The effect of the plugging of the steam generator tubes results in the Order for Modification of License being invalid. Therefore, in response to our Order for Modification of License and also in support of the current reload of Surry Unit 1, VEPCO submitted an emergency core cooling system evaluation which included corrections for the effects of the higher temperature of the primary coolant in the upper head region and also included the effects of additional plugged steam generator tubes. This submittal dated September 27, 1976, and supplemented on October 29, 1976 requested amendments to Facility Operating Licenses Nos. DPR-32 and DPR-37.

As a result of the ECCS analysis VEPCO has also requested revisions to the Surry Units 1 and 2 Technical Specifications related to power distribution limits and power distribution monitoring requirements.

Steam Generator Tube Integrity

Introduction

By letters dated October 19 and 25, 1976 and November 15, 1976, VEPCO submitted inspection results concerning the Surry Units Nos. 1 and 2 steam generator tubes. These letters describe phenomena related to the steam generator tube leak that occurred in steam generator "A" of Unit No. 2 on September 15, 1976. In addition to the above submittals, we held meetings with VEPCO and Westinghouse, the steam generator manufacturer, to discuss (1) tube denting, (2) the tube leak in steam generator "A" of Surry Unit No. 2, (3) the required analysis for the corrective action, and (4) future inspection plans.

Discussion

Following the conversion from a sodium phosphate secondary water treatment to an all volatile treatment (AVT) during January 1975 for Surry Unit No. 1 and during February 1975 for Unit No. 2, steam generator tube denting was noted in Unit No. 2 in May 1975 and in Unit No. 1 in November 1975. In May 1976, dented tube samples and segments of the tube support plate were removed from Unit No. 2, which revealed that the tube support plates were cracked. The preliminary laboratory reports indicated that the annulus between tubes and support plates was filled with a hardened corrosion product that expanded volumetrically to exert sufficient forces to "dent" the tube diametrically and to crack the tube support plate ligaments between the tube holes and the water circulation flow holes. The phenomenon of denting may be attributed to residual phosphates that remained in the annulus when the phosphate treatment was converted to AVT. The corrosion product from the carbon steel support plate expands volumetrically to exert sufficient forces to dent the tube and crack the tube support plate ligaments between the tube holes and the water circulation flow holes. These dented tubes have otherwise generally retained their integrity; although there have been very small but detectable leaks at the dent locations, there have been no rapid failures.

On September 15, 1976, during normal operation, one U-tube in steam generator "A" at Surry Unit No. 2 rapidly developed a substantial primary to secondary leak (about 80 GPM). Subsequent investigation established that the leak resulted from an axial crack in the U-bend of the tube near the top, approximately 4 1/2 inches in length.

Eddy current tests of a substantial number (over 100) of nearby tubes proved inconclusive, but removal of the damaged tube and eight others along the same bundle row and subsequent laboratory analysis showed that the failure resulted from intergranular stress assisted attack (often called "intergranular stress corrosion") that initiated from the primary side. Eight additional tubes, adjacent to the leaking tube, were removed from row 1. Five of these eight showed significant ovalization. Radiographic examination showed that four of the eight had cracks on the inner surface only at the extrados and intrados of the U-bend apex. One tube had a "tight" crack, about 40% through wall, which was found only by metallographic examination on the inside diameter of the extrados at the U-bend apex. The tubes that had defect indications were located near the middle of the flow slot. Three tubes at the end of the flow slot showed no evidence of cracking. Presumably the strain at the U-bend apex in these corner tubes was not sufficient to cause cracking.

Along the chord of the innermost rows of tubes, in the tube support plate, there is a row of rectangular flow slots, consisting of six slots, approximately 16-inches long by 2-3/4 inches wide, each about 20 inches between centers. As a result of the previously described pressures built up in the tube support plate acting on these slots, the slots have been observed to show "hourglassing"; that is, the central portion of the parallel flow slot walls have moved closer so that some flow slots are now narrower in the center than at the ends. Since the initial parallel slot walls have moved closer, the tube support plate material supporting the tubes nearest this central portion of these flow slots has also moved inward, in turn forcing an inward displacement of the legs of the U-bends at these locations. This inward movement of the legs of the U-bends at these locations caused an increase in the dynamic strain at the U-bend. This additional strain results in an increase in ovalization of the tubes at the U-bend apex. It is this additional increase in dynamic strain at the apex of the U-bend which is believed to be the additional factor required to initiate and enhance the susceptibility to stress corrosion cracking of Inconel 600 Alloy tubing exposed to PWR primary coolant.

Westinghouse has examined tubes removed from rows 1, 2, and 3 from the Surry Unit No. 1 and Turkey Point Unit No. 4 steam generators which have also experienced denting, flow slot "hourglassing", and excess tube ovality and cracking at the U-bend. However cracking at the U-bend was found only in the Row 1 tubes of these steam generators.

Westinghouse has measured the hourglassing at the uppermost support plates in Surry Units Nos. 1 and 2 which is indicative of the U-bend leg displacement in row 1. The design flow slot opening is 2.75 inches. The dimension defined as "displacement" is the original slot width of 2.75 inches less the remaining slot width. In five months of operation the flow slot openings in Unit No. 2 have decreased to an average of 1.46 inches for all six flow slots, but the largest slot displacement has only been

1.37 inches. In Surry Unit No. 1, the flow slot opening has decreased to 0.50 inch at the location where 16 tubes were removed from row 1, 10 tubes from row 2, and 1 tube from row 3. No U-bend cracks were found in rows 2 and 3. The total slot displacement in Surry Unit No. 1 was 2.25 inches. Therefore, an additional 0.88 inch inward displacement of the flow slot would have to occur at Unit No. 2 to cause sufficient rise in the strain at the U-bend apex of the row 2 tubes of Unit No. 2 to be equivalent to the strain in the row 2 tubes of Unit No. 1 in which no cracking has been observed.

The VEPCO submittal of October 19, 1976, gave an estimated rate of flow slot closure of 0.19 inch per calendar month for the first (bottom) tube support plate above the tube sheet in the Unit No. 2 steam generators. This rate was determined from measurements taken on the bottom support plate during May and October 1976, where the maximum slot displacement was 2.31 inches. The maximum flow slot displacement is 1.37 inches in the upper (top) support plate. Therefore a more realistic flow slot closure rate in the top support plate would be 0.11 inch per calendar month. This closure rate is based on the ratio of the upper and lower support plate flow slot displacement times the lower support plate closure rate. Although there is the possibility that "hourglassing" may continue during the operation of Surry Unit No. 2 steam generators, the Westinghouse analysis indicates that only the tubes located adjacent to the flow slot openings in row 1 are susceptible to intergranular defects at the U-bend apex but these tubes are plugged. The row 1 tubes have the highest level of plastic deformation and residual stresses due to the smaller U-bend radius, and are subject to additional strain as a result of any continued closure of the flow slots. Even though "hourglassing" may continue, it was demonstrated that an equivalent strain to that which caused cracking in the row 1 tubes is not projected to occur in other rows, and the cumulative effect anticipated for tubes beyond row 1 would be substantially less because of the larger U-bend radius, less plastic deformation, and smaller residual stresses.

Based on the above observations, VEPCO estimates that it would take eight (8) months operating time before Surry Unit No. 2 steam generators would attain the same degree of flow slot closure as observed in Surry Unit No. 1. This can be verified by an inspection after two months of operation, as discussed later.

VEPCO has initiated selective plugging as the corrective action to prevent the recurrence of "intergranular stress corrosion cracking" at the U-bend apex of the small radius steam generator tubes. Consequently, all the tubes in row 1, approximately 2/3 of the tubes in row 2, and approximately 1/3 of the tubes in row 3 of all three Surry Unit No. 2 steam generators have already been plugged. This plugging pattern (described in the October 19 submittal) was based on an "ovality" thesis which was quite conservative considering the new results presented in the November 15, 1976 submittal. VEPCO has proposed to return Surry Unit No. 2 to power for two months and then perform a reinspection. The two months of proposed operation is defined as 61 effective days at primary system temperatures greater than 350°F.

Evaluation

VEPCO has submitted both analytical and experimental data in support of the revised plugging criteria and the proposed two months of Surry Unit No. 2 operation. We have reviewed these data and have performed independent evaluations to determine the adequacy of the revised plugging criteria and the conditions for the two months of operation.

Regarding the tube plugging criteria applied to Surry Unit No. 2 steam generators, the tube "denting" phenomenon, the two months operating period, and the potential for "intergranular cracking" at the U-bend apex of the tubes in rows 2, 3, 4 and etc., we have concluded the following in our safety evaluation of Surry Unit No. 2: (1) the strain in the steam generator tubes at the U-bend apex is displacement controlled by the tube support plate flow slot closure, (2) the total inward movement of the flow slots will not cause significant additional strain at the U-bend apex of the tubes in rows 2, 3, 4 and etc., during the proposed two-month operating period, (3) the closure of the flow slots in the top support plate will not progress to the distance observed in Surry Unit No. 1, (4) all of row 1 is plugged, 2/3 of row 2 is plugged and 1/3 of row 3 is plugged and there is a sufficiently low probability for initiation of intergranular cracking of the unplugged tubes in rows 2, 3, 4, etc., (5) although tube "denting" is associated with tube support plate corrosion, support plate cracking, and the "hourglassing" of the support plate flow slots, there have been only small leaks at the dent locations and no rapid failures have occurred, (6) no cracking has been observed in any tubes from rows 2, 3, etc., and (7) the cumulative damage anticipated for the unplugged tubes in rows 2, 3, 4, etc., as a result of continued hourglassing of the top support plate flow slots, will be substantially less than that incurred in the row 1 tubes, and row 1 has been plugged.

In addition, on the basis of the analytical and experimental data and the observations of 11 tubes removed from Surry Unit 1, nine tubes from Surry Unit 2, and nine tubes from Turkey Point Unit No. 4, we concur that the assumption of tube plugging based on tube "ovality" at the U-bend of tubes with small bend radius is conservative. Therefore, the tube plugging in the steam generators of Surry Unit No. 2 is acceptable. Furthermore, the proposed operating conditions for two months are acceptable. The top support plate "hourglassing" in the Surry Unit No. 2 steam generators is less than Surry Unit No. 1. Since Surry Unit No. 1 experienced a greater degree of "hourglassing" during an equivalent operating period and no intergranular cracking was evident at the U-bend apex of tubes in rows 2 and 3, we also believe that it is likely that the unplugged tubes in rows 2, 3, 4, etc. in the Surry Unit No. 2 steam generators are free of intergranular cracks at the U-bend apex as the result of flow slot hourglassing. We agree that the maximum rate of flow slot hourglassing of 0.11 inch/month predicted for Surry Unit No. 2 steam generators is a reasonable assumption.

However, the consideration of reactor operation beyond the proposed two months are dependent upon: (1) our assessment of additional information from the Surry Unit No. 1 inspection, (2) an evaluation of Westinghouse's analysis of other facilities with denting and intergranular cracking of steam generator tubes with small U-bend radius, and (3) the results of the inspection at Surry Unit No. 2 after the two months of operation.

We therefore conclude that:

a. Unplugged tubes in rows 2, 3, 4, etc. in all the steam generators of Surry Unit No. 2 retain sufficient integrity to withstand normal operating and postulated accident conditions for a two month operating period as defined above.

b. There is reasonable assurance of tube integrity for a limited period of two months to provide adequate protection to the public health and safety.

Emergency Core Cooling System (ECCS) Reanalysis

We have reviewed the ECCS reanalysis for Surry Units 1 and 2. The analysis was performed with the October 1975 version of the Westinghouse Evaluation Model. Assumptions and input data used in the reanalysis were the same as was used in the previous analysis submitted June 6, 1975 except for the following:

1) Limiting value of heat flux hot channel factor (F_Q) changed from 2.1 to 2.0.

2) Minimum allowable value of the containment atmosphere temperature changed from 75°F to 90°F. Changing the minimum containment temperature to 90°F instead of 75°F was based on temperature measurements made in the containment during the Cycle 3 operation which indicated an average bulk temperature of over 100°F during power operation.

3) The number of steam generator tubes assumed to be plugged was increased to a maximum of 12%. The actual number is currently less than 10%.

4) Temperature of the fluid in the upper head region was assumed to be equal to hot leg temperature.

5) Minimum water volume in the accumulators was changed from 975 ft^3 to 1075 ft^3 in each. There are three per plant.

A reanalysis of the small break LOCA was not required. The small break analysis submitted by VEPCO on June 6, 1975 remains valid. The small breaks are relatively insensitive to the fluid temperature in the upper head region and to steam generator tube plugging.

The highest peak clad temperature (PCT) predicted for the worst small break in the June 6, 1975 submittal was approximately 1800°F. We agree with the licensee that reanalysis of the small break cases is not required.

Analyses were conducted with varying percentages of steam generator tubes plugged and the worst case break parameters to determine the effect of the additional steam generator tubes plugged. At about 11% of tubes plugged, an apparent threshold effect was discovered. The base case assumed a steam generator tube plugging level of 7 percent tubes plugged in each steam generator and a Moody discharge coefficient of 0.4. This case yielded a maximum peak clad temperature of 2074°F. Increasing the percent tubes plugged to 10% and holding all other parameters the same resulted in predicted maximum clad temperature of 2091°F. However, when the percent of tubes plugged was increased to 11%, the maximum PCT increased to over 2200°F. It was initially determined by the licensee after consulting with the Nuclear Steam System Supplier (NSSS) that the threshold effect was due to predicted flow oscillations in the SATAN Model downcomer node where the "end of bypass" criteria is determined. Flow oscillations were observed following the initial flow reversal in the downcomer and were larger in magnitude as the percent of steam generator tubes plugged was increased. At 11% tubes plugged, a flow oscillation was noted which caused an additional flow reversal and therefore caused a new determination of "end of bypass". This, resulted in "end of bypass" being defined several seconds later in the blowdown and resulted in the removal of more ECCS water since Appendix K to 10 CFR 50 specifies that all ECCS water be assumed to bypass the reactor core and spill out of the break up to the time defined as end of bypass. The NSSS advised the licensee that the rate of accumulator water injection affected the oscillations and that they expected the oscillations to be less severe with lower accumulator injection rates. The licensee proposed increasing the accumulator water volume, therefore decreasing the flow rate. An analysis with 12% of the steam generator tubes plugged was then conducted with an increased volume in an accumulator (975 ft^3 to 1075 ft^3) and the flow oscillations decreased such that the initial flow reversal could be defined to be the end of bypass. The PCT for this case was 2107°F, well below the 2200°F maximum of 10 CFR 50.45 Appendix K.

Subsequent discussion of these observed flow oscillations between the NRC staff and the NSSS led us to a preliminary conclusion that the oscillations are a code anomaly and that the magnitude of the oscillations is probably a function of system volume which decreases as more steam generator tubes are plugged. We concluded that lowering the accumulator flow rate would decrease the magnitude of oscillations. We also determined that the same phenomena had been observed with RELAP and had been at least partially corrected by accumulator modeling changes.

Although the cause of the flow phenomena has not been completely defined, we conclude that the analysis of Surry Units 1 and 2 is acceptable based on the following:

1. With the increased accumulator volume and decreased discharge flow rate, 12% steam generator plugging, a maximum PCT of 2107°F, percent clad strain reached, and maximum clad oxidation all limits are well within Appendix K requirements.

2. The flow oscillations observed resulted in a reverse flow spike of insignificant magnitude and time duration and had no real effect on downcomer flow even if it were assumed to be real.

We therefore conclude that the licensee's "Large Break LOCA-ECCS Reanalysis for Surry Power Station Units Nos. 1 and 2," dated October 29, 1976, and based on the previously approved Westinghouse October 1975 ECCS Model, is acceptable for operation of both plants within the constraints of the analysis, and complies with 10 CFR 50.46.

Axial Power Distribution Monitoring System (APDMS)

Both the Surry Unit No. 1 Cycle 4 core and the current Surry Unit No. 2 core will be limited to peaking factors, $F_Q(Z)$, below the level where excore detectors can reliably assure the limit is not exceeded. They have therefore proposed an Axial Power Distribution Monitoring System (APDMS) to ensure $F_Q(Z)$ will not exceed the LOCA limiting value of $F_Q(Z)$. We have reviewed the monitoring procedure and calculation method for determining the maximum peaking factor and the plant conditions under which APDMS scanning is required.

We conclude that the VEPCO proposed APDMS system and procedures are adequate subject to the following revisions which will be implemented within two months.

1. Provide a programming change for the P-250 process computer in order to enable continuous monitoring and alarm when rod position and power level are such that APDMS surveillance is required,

2. Provide a software change which would alarm at the appropriate time intervals when APDMS monitoring is required following the initial alarm described in (1) above, and

3. Provide software changes that will alarm when the P-250 computer output determines that $F_Q(Z)$ exceeds the limiting value and a power reduction is required.

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.

Dated: November 26, 1976

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS NOS. 50-280 AND 50-281

VIRGINIA ELECTRIC & POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments No.26 to Facility Operating Licenses Nos. DPR-32 and DPR-37, issued to Virginia Electric & Power Company (the licensee), which added a condition to the license for Surry Unit 2 and revised Technical Specifications for operation of the Surry Power Station Units Nos. 1 and 2 (the facilities) located in Surry County, Virginia. The amendments are effective as of the date of issuance.

These amendments concern changes required as a result of the steam generator repair for Surry Unit 2 and the consequent need for revision to the emergency core cooling system evaluation and the power distribution and power distribution monitoring requirements. The revised emergency core cooling system evaluation also fulfills the requirements of the Commission's Order for Modification of License dated August 27, 1976.

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 the 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 September 27, 1976, as supplemented October 29, 1976, and the licensee's submittals dated October 19, 1976, and November 15, 1976, (2) Amendments No.26 to Licenses 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 26th day of November 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

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