11/28/75

DISTRIBUTION:
[redacted]
NRC PDR (2)
Local PDR
ORB-4 Reading
Attorney, OELD
OI&E (3)
NDube
BJones (w/4 encls)
JMcGough
RIngram
RWReid
MFairtile
KRGoler
SKari
BScharf (15)
TJCarter
EP Licensing Assistant
EP Project Manager
PCollins
SVarga
CHebron
AESteen
DEisenhut
ACRS (16)
TBAbernathy

Dockets Nos. 50-280
and 50-281

Virginia Electric & Power Company
ATTN: Mr. Stanley Ragone
Senior Vice President
Post Office Box 26666
Richmond, Virginia 23261

Gentlemen:

The Commission has issued the enclosed Amendments No. 11 to Facility Operating Licenses Nos. DPR-32 and DPR-37 for the Surry Power Station, Units 1 and 2. The amendments include Change No. 26 to your Technical Specifications for each license and are in response to your request dated September 8, 1975, as supplemented October 22, 1975. and October 30, 1975.

The amendments revise the provisions in the Technical Specifications relating to the replacement of 81 of 157 fuel assemblies in the reactor core, constituting refueling of the core for third cycle operation of Unit 1.

We have evaluated the potential for environmental impact of plant operation in accordance with the enclosed amendments, and 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 pursuant to 10 CFR 51.5 (d)(4) that an environmental statement, negative declaration or environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

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

Sincerely,

Fairtile for

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

Enclosures:
See next page

OFFICE	RL:ORB-4	RL:ORB-4	OELD	TR	RL:ORB-4	
x7435 SURNAME	RIngram	MFairtile:esp			RWReid	
DATE	10/ /75	10/ /75	10/ /75	10/ /75	10/ /75	

Enclosures:
1. Amendment No. 11 to DPR-32
2. Amendment No. 11 to DPR-37
3. Safety Evaluation
4. Federal Register Notice

cc w/enclosures:
Michael W. Maupin, Esquire
Hunton, Williams, Gay & Gibson
P. O. Box 1535
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/enclosures & incoming dtd.
9/8/75 and 10/22/75 and 10/30/75
Ms. Susan T. Wilburn
Commonwealth of Virginia
Council on the Environment
P. O. Box 790
Richmond, Virginia 23206

OFFICE ➤						
SURNAME ➤						
DATE ➤						

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

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments No. 11 to Facility Operating Licenses Nos. DPR-32 and DPR-37 issued to Virginia Electric & Power Company (VEPCO) which revised Technical Specifications for operation of the Surry Power Station, Units 1 and 2, located in Surry County, Virginia. The amendments are effective as of the date of issuance.

The amendments revise the provisions in the Technical Specifications relating to the replacement of 81 of 157 fuel assemblies in the reactor core, constituting refueling of the core for third cycle operation of Unit 1.

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 is not required since the amendments do not involve a significant hazards consideration.

OFFICE ➤						
SURNAME ➤						
DATE ➤						

For further details with respect to this action, see (1) the application for amendments dated September 8, 1975, as supplemented October 22, 1975 and October 30, 1975, (2) Amendments No. 11 to Licenses Nos. DPR-32 and DPR-37, with Change No. 26, 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 & 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 Reactor Licensing, Office of Nuclear Reactor Regulation.

Dated at Bethesda, Maryland, this 26th day of November, 1975

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

Morton Fairtile, Acting Chief
~~Robert W. Reid, Chief~~
Operating Reactors Branch #4
Division of Reactor Licensing

MBF

OFFICE ➤	RL:ORB-4	RL:ORB-4 MBF	OELD	RL:ORB-4		
x7330 SURNAME ➤	RIngram	MFairtile	[illegible]	RWReid		
DATE ➤	~~10~~ 11/3/75	~~10~~ 11/29/75	11/24/75	11/25/75		

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

ATTACHMENT TO LICENSE AMENDMENT NO. 11

CHANGE NO. 26 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-32

DOCKET NO. 50-280

Revise Appendix A as follows:

Remove Pages	Insert Pages
2.1-2	2.1-2
2.1-6	2.1-6
3.12-3 - 3.12-22	3.12-3 - 3.12-27
--	Table 3.12-1
Figure 3.12-1A	Figure 3.12-1A
--	Figure 3.12-9

NOV 2 5 1975

4. The reactor thermal power level shall not exceed 118% of rated power.

B. The safety limit is exceeded if the combination of Reactor Coolant System average temperature and thermal power level is at any time above the appropriate pressure line in TS Figures 2.1-1, 2.1-2 or 2.1-3; or the core thermal power exceeds 118% of rated power.

C. The fuel residence time shall be limited to 7600 effective full power hours (EFPH) for Cycle 3 of Unit 1 and to 17,000 EFPH for Cycles 1 and 2 of Unit 2. 26

Basis

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the reactor coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed Departure From Nucleate Boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters; thermal power, reactor coolant temperature and pressure have been related to DNB through the W-3 correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially to this limiting criterion. Additional peaking factors to account for local peaking due to fuel rod axial gaps and reduction in fuel pellet stack length have been included in the calculation of this limit.

The fuel residence time is limited to 7600 EFPH for Cycle 3 of Unit 1 and to 17,000 EFPH for Cycles 1 and 2 of Unit 2 to assure no fuel clad flattening will occur in the cores without prior review by the Regulatory Staff. 26

4. Whenever the reactor is subcritical, except for physics test, 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. Operation with part length rods shall be restricted such that except during physics tests, the part length rod banks are withdrawn from the core at all times.

6. Insertion limits do not apply during physics tests or during periodic exercise of individual rods. However, the shutdown margin indicated in TS Figure 3.12-7 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 and part length rods fully withdrawn.

7. For Surry Unit 1, after 5000 MWD/MTU of burnup in Cycle 3, the total cumulative cycle energy-weighted average D bank insertion should not be greater than 9%. Should this energy-weighted D bank insertion limit be violated, movable detector surveillance is required for operation when the thermal power is in excess of 95% power. This surveillance will be performed in accordance with the following: 26

a. The normalized axial power distribution, $F_j(Z)$, from thimble j at core elevation Z shall be measured utilizing at least two thimbles of the movable incore flux system for which $\overline{R}$, 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.

(1) Raising the thermal power above 95% rated power, or

(2) Movement of the control bank of rods more than an accumulated total of five steps in any one direction.

b. If $F_j(Z)$ exceeds its limit, $(F_j(Z))_L$ as defined in the basis, the reactor power shall be reduced until the limit, $(F_j(Z))_L$, is met.

When the thermal power is in excess of 95%, surveillance, in accordance with a. and b. above, will continue until the total cumulative cycle energy-weighted D bank insertion is within the prescribed limit.

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:

$F_Q(Z) \leq (2.10/P) \times K(Z)$ for $P > .5$

$F_Q(Z) \leq (4.20) \times K(Z)$ for $P \leq .5$

$F^N_{\Delta H} \leq 1.52 \; (1 + 0.2(1 - P))$ 26

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, 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 three percent to account for manufacturing tolerances and further increased by five percent to account for measurement error.

   b. 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

$F_Q \leq 2.10 \times K(Z)$ and $F^N_{\Delta H} \leq 1.52$ within 24 hours, the overpower ΔT and overtemperature ΔT trip setpoints shall be similarly reduced. 25

3. For Cycle 3 of Surry Unit 1, $F_{xy}(Z)$ shall be limited to the values shown in TS Table 3.12 -1 for the unrodded core plane region located between a core plane elevation 2.5 feet from the top of the core and a core plane elevation 1.5 feet from the bottom of the core, excluding grid strap locations. $F_{xy}(Z)$ shall be determined to be within its limit by using the moveable incore detectors to obtain a power distribution map after each fuel loading, and at least once every full power month. With $F_{xy}(Z)$ exceeding its limit:

   (a) Operation is restricted to a maximum permitted power level, P_{RB}, which is reduced from 100% power one percent for every one percent $F_{xy}(Z)$ exceeds the $F_{xy}(Z)$ limit evaluated at 100% power, or

   (b) Demonstrate through manual surveillance using the moveable incore detector system that the axial power distribution limits $(F_j(Z))_L$, are not violated. For the purpose of this section of the Technical Specifications 26

$$(F_j(Z))_L = \frac{2.10 \qquad (K(Z))}{(P) \quad (\bar{R}_j) \quad (1.03) \quad (1 + \sigma_j) \quad (1.07) \quad (B(Z))}$$

   where B(Z) is the rod bow penalty as a function of axial core elevation shown in TS Figure 3.12 -9, and the other parameters in the above equation are defined in the basis. The surveillance on $F_j(Z)$ will be performed by measuring the normalized axial power distribution, $F_j(Z)$, from thimble j at core elevation Z utilizing at least two timbles of the moveable incore flux system for which $\bar{R}$, as defined in the basis, has been determined. This shall be done immediately following and as a minimum at

NOV 8 1978

30, 60, 90, 120, 240, and 480 minutes, following the events listed below and every eight hours thereafter:

1. Raising the thermal power above P_{RB} or
2. Movement of the control bank of rods more than an accumulated total of five steps in any one direction. 26

If $F_j(Z)$ exceeds its limit, $(F_j(Z))_L$ as defined in this section, the reactor power shall be reduced until the limit, $(F_j(Z))_L$ is met, or reduce power to a power level below P_{RB}.

4. The reference equilibrium indicated axial flux difference (called 26 the target flux difference) at a given power level P_o, is that indicated axial flux difference with the core in equilibrium xenon conditions (small or no oscillation) and the control rods more than 190 steps withdrawn. The target flux difference at any other power level, P, is equal to the target value of P multiplied by the ratio, P/P_o. The target flux difference shall be measured at least once per equivalent full power quarter. The target flux difference must be updated during each effective full power month of operation either by actual measurement, or by linear interpolation using the most recent value and the value predicted for the end of the cycle life.

5. Except during physics tests, during excore detector calibration 26 and except as modified by 3.12.B.5.a, b, or c below, the indicated axial flux difference shall be maintained within a +6 to -9% band about the target flux difference (defines the target band on axial flux difference).

NOV 25 1975

a. At a power level greater than 90 percent of rated power, if the indicated axial flux difference deviates from its target band, the flux difference shall be returned to the target band, or the reactor power shall immediately be reduced to a level no greater 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 +6 to -9% target band for a maximum of one hour (cumulative) in any 24 hour period provided the flux difference does not exceed an envelope bounded by -18 percent and +11.5 percent at 90% 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.5.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. 26

(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. One half of the time the indicated axial flux difference is out of its target band up to 50 percent of rated power is to be counted as contributed to the one hour cumulative maximum the flux difference deviates from its target band at a power level less than or equal 90 percent of rated power.

Alarms shall normally be used to indicate the deviations from the axial flux difference requirements in 3.12.B.5.a and the flux difference time limits in 3.12.B.5.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. 26 26

6. The allowable quadrant to average power tilt is

$$T = 2.0 + 50\ (1.40\ /F_{xy} - 1) \leq 10\%$$

where F_{xy} is 1.40 , or the value of the unrodded horizontal plane peaking factor appropriate to F_Q as determined by a movable incore detector map taken on at least a monthly basis; and T is the percentage operating quadrant tilt limit, having a value of 2% if F_{xy} is 1.40 or a value up to 10% if the option to measured F_{xy} is in effect. 26 26

NOV 2 5 1973

7. If the quadrant to average power tilt exceeds a value T% as selected in 3.12.B.6, except for physics and rod exercise testing, then: 26

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 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% except for physics tests, the power level and high neutron flux trip setpoint will be reduced from rated power, 2% for each percent of quadrant tilt.

8. If after a further period of 24 hours, the power tilt in 3.12.B.7 above is not corrected to less than +T%: 26

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 an abnormal 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. Inoperable Control Rods

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 of 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 3.12.C.1 and 3.12.C.2 shall not apply during physics test in which the assemblies are intentionally misaligned.

5. If an inoperable full-length rod is located below the 200 step level and is capable of being tripped, or if the full-length rod is located below the 30 step level whether or not it is capable of being tripped, then the insertion limits in TS Figure 3.12-2 apply.

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. No insertion limit changes are required by an inoperable part-length rod.

8. 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. 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.

1. Once per day, and

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

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

1. Movable detectors (at least two per quadrant)

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

E. Inoperable Rod Position Indicator Channels

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. Misaligned or Dropped Control Rod

1. If the Rod Position Indicator Channel is functional and the associated part length or 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 Section 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 part-length or 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. Early in core life, less shutdown margin is required, and TS Figure 3.12-7 shows the shutdown margin equivalent to 1.77% reactivity at end-of-life with respect to an uncontrolled cooldown. All other accident analyses 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 and part-length rods) 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 (part-length of 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 linear power density must not exceed 20.4 kw/ft. 26
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 must not exceed the limiting Kw/ft 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 on power distribution the following hot channel factors are defined.

$F_Q(Z)$, 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 $F^N_{\Delta H}$ 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 $F^N_{\Delta H}$.

An upper bound envelope of 2.10 times the normalized peaking factor axial dependent of TS Figure 3.12-8 has been determined from extensive analyses considering all operating maneuvers consistent with the technical specifications on power distribution control given in Section 3.12.B.4. 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.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map ($\geq$ 40 thimbles monitored) taken with the movable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerances.

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 $F^N_{\Delta H} \leq 1.52/1.08$. 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 affect F_Q, (b) the operator has a direct influence on F_Q through movement of rods, and can 26 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 can be compensated for the F_Q by tighter axial control, but compensation for $F^N_{\Delta H}$ is taken, experimental error must be allowed for and four percent is the appropriate allowance for a full core map ($\geq$ 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 is not necessary to measure these quantities. Instead it has been determined that, provided certain conditions are observed, the hot channel factor limits 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 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.

4. For Surry Unit 1, the total cumulative cycle energy weighted average D bank insertion limit is observed. 26

5. 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 5 are observed, these hot channel factors limits are met. In Specification 3.12.B.1, F_Q is arbitrarily limited for $P \leq .5$ (except for physics tests).

For Surry Unit 1, the total cumulative cycle energy weighted average D bank insertion limit referred to above is designed to ensure that long-term core depletion with significant D bank insertion does not occur, since such depletion could produce an axial burnup distribution which could cause the 26 total peaking factor to potentially exceed the LOCA limiting $F_Q(Z)$ for certain plant maneuvers near the end of Cycle 3. However, it has been determined that for these plant maneuvers, the $F_Q(Z)$ upper band envelope will not be violated if after 5000 MWD/MTU, the core is depleted with the cumualtive energy weighted D bank insertion from the beginning of cycle no greater than 9%. If this total cumulative cycle energy weighted average D bank insertion limit is violated, additional axial power distribution surveillance using the movable detector system is implemented in order to assure that the power peaking factor, $F_Q(Z)$, is maintained at or below the limiting value. Flux shape surveillance is not required below 95% power, since it has been determined that for the worst case, including plant maneuvers following core depletion with significant D bank insertion, the calculated $F_Q(Z)$ peaking factor at 100% power is at the most 5% above the LOCA limiting $F_Q(Z)$ envelope.

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 limit on the axial distribution function $F_j(Z)$ is derived as follows:

$$(F_j(Z))_L = \frac{2.10\ (K(Z))}{(P)(\overline{R}_j)(1.03)(1 + \sigma_j)(1.07)}$$

where:

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

b. P is the fraction of thermal power.

c. K(Z) is the reduction in limit as a function of core elevation Z as determined from TS Figure 3.12-8.

d. $(F_j(Z))_L$ is the operational limit on $F_j(Z)$.

e. $\overline{R}_j$, for thimble j, is determined from at least n=6 incore flux maps covering the full configuration of permissible rod patterns at the thermal power 95% of rated power.

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

where

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

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

26

The full incore flux map used to update $\bar{R}$ and for monitoring $F_j(Z)$ shall be taken at least once 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 $\bar{R}$ for each representative thimble.

f. σ_j is standard deviation of R_j and is derived from n flux maps from the relationship below, or 0.02, whichever is greater. 26

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

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

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 expresses as axial offset it varies only with burnup.

The technical specifications on power distribution control given in 3.12.B.4 along with the cycle energy weighted D bank insertion limit given in 3.12.A.7 26 assure that the F_Q upper bound envelope of 2.10 times Figure 3.12-8 is not exceeded and xenon distributions are not developed which at a later time, would cause greater local power peaking even though the flux difference is then within the limits specified by the procedure.

For Cycle 3 of Surry Unit 1, a limit on $F_{xy}(Z)$ has been imposed to insure that with the inclusion of the rod bow power peaking pealty, the LOCA $F_Q(Z)/P$ envelope will not be violated. If, by core flux mapping, the $F_{xy}(z)$ limit is determined to be violated, the minimum allowable power level will be reduced from 100% rated power by one percent for each one percent violation of the $F_{xy}(Z)$ limit, or manual moveable detector surveillance will be implemented for the period that the violation occurs. The imposition of the limit of $F_{xy}(Z)$ is an interim measure taken to conservatively include the potential effects of rod bowing on core power capability until the NRC has further evaluated the problem. 25

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 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 possible during certain physics tests or during required, periodic, excore detector calibrations which require larger flux differences than permitted. Therefore, the specifications on power distribution control are not applied during physics tests or excore detector calibrations; this is acceptable due to the low probability of a significant accident occuring during these operations.

In some instances of rapid unit power reduction automatic rod motion will cause the flux difference to deviate from the target band when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target band, however to simplify the specification, a limitation of one hour in any period of 24 hours is placed on operation outside the band. This ensures that the resulting xenon distributions are not significantly different from those resulting from operation within the target band. The instantaneous consequences of being outside the band, provided rod insertion limits are observed, is not worse than a 10 percent increment in peaking factor for the allowable flux difference at 90% power, in the range +14.5 to -21 percent (+11.5 percent to -18 percent indicated) where for every 4 percent below rated power, the permissible positive flux difference boundary is extended by 1 percent, and for every 5 percent below rated power, the permissible negative flux difference boundary is extended by 2 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. At the option of the operator, credit may be taken for measured decreases in the unrodded horizontal plane peaking factor, F_{xy}. This credit may take the form of an expansion of permissible quadrant tilt limits over tilt limits over the 2% value, up to a value of 10%, at which point specified power reductions are prudent. Monthly surveillance of F_{xy} bounds the quantity because it decreases with burnup. (WCAP-7912 L).

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 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.

Core Height, Z (feet)	$F_{xy}(Z)$ limit	
1.5	1.406/P	
2.0	1.445/P	
2.5	1.471/P	
3.0	1.535/P	
3.5	1.570/P	
4.0	1.527/P	
4.5	1.510/P	
5.0	1.492/P	
5.5	1.485/P	
6.0	1.478/P	
6.5	1.494/P	
7.0	1.461/P	
7.5	1.431/P	26
8.0	1.421/P	
8.5	1.395/P	
9.0	1.418/P	
9.5	1.422/P	

where $F_{xy}(Z)$ = ratio of peak power density to average power density in the horizontal plane at elevation Z

P = fraction of rated power at which the core is operating

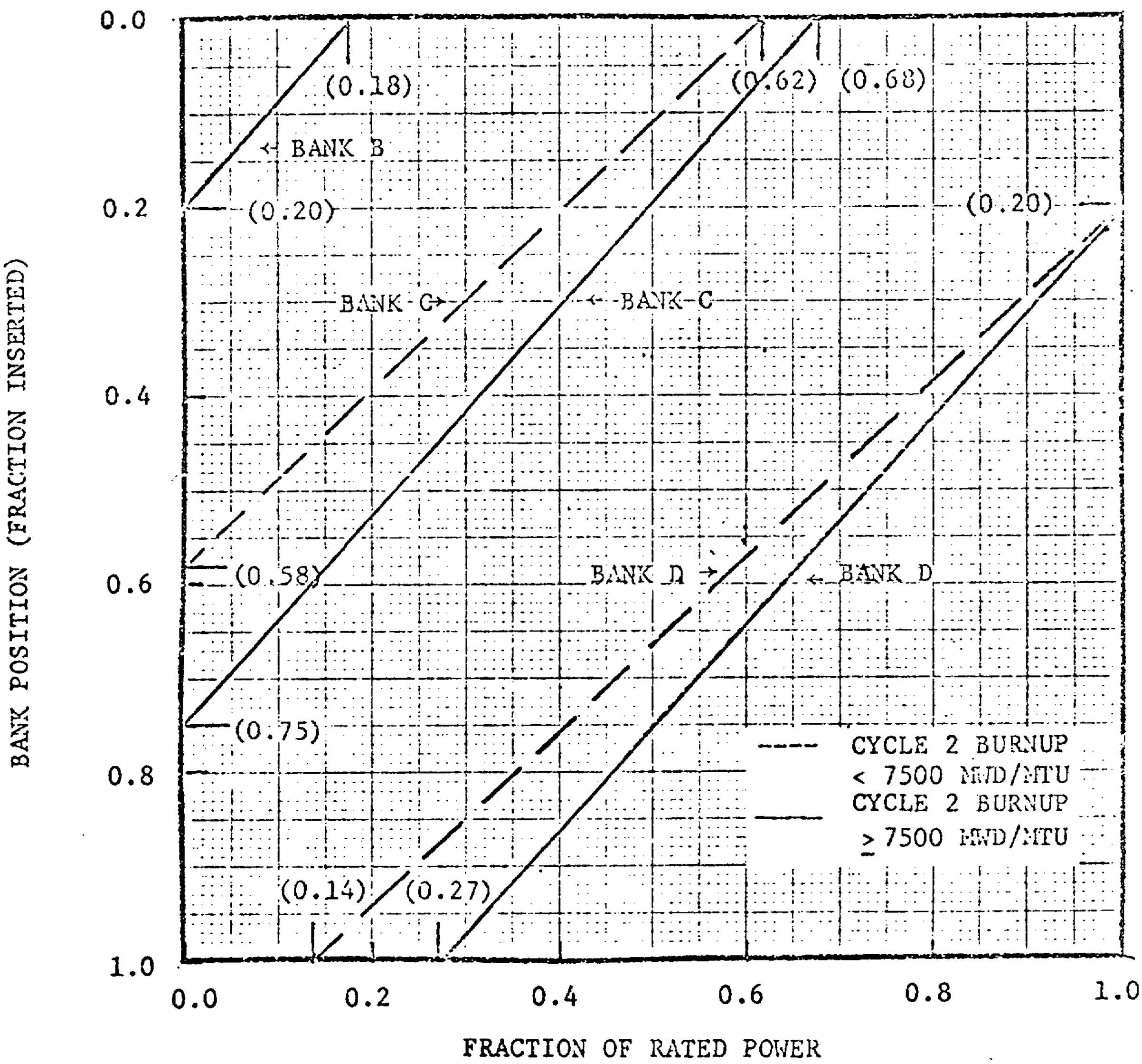
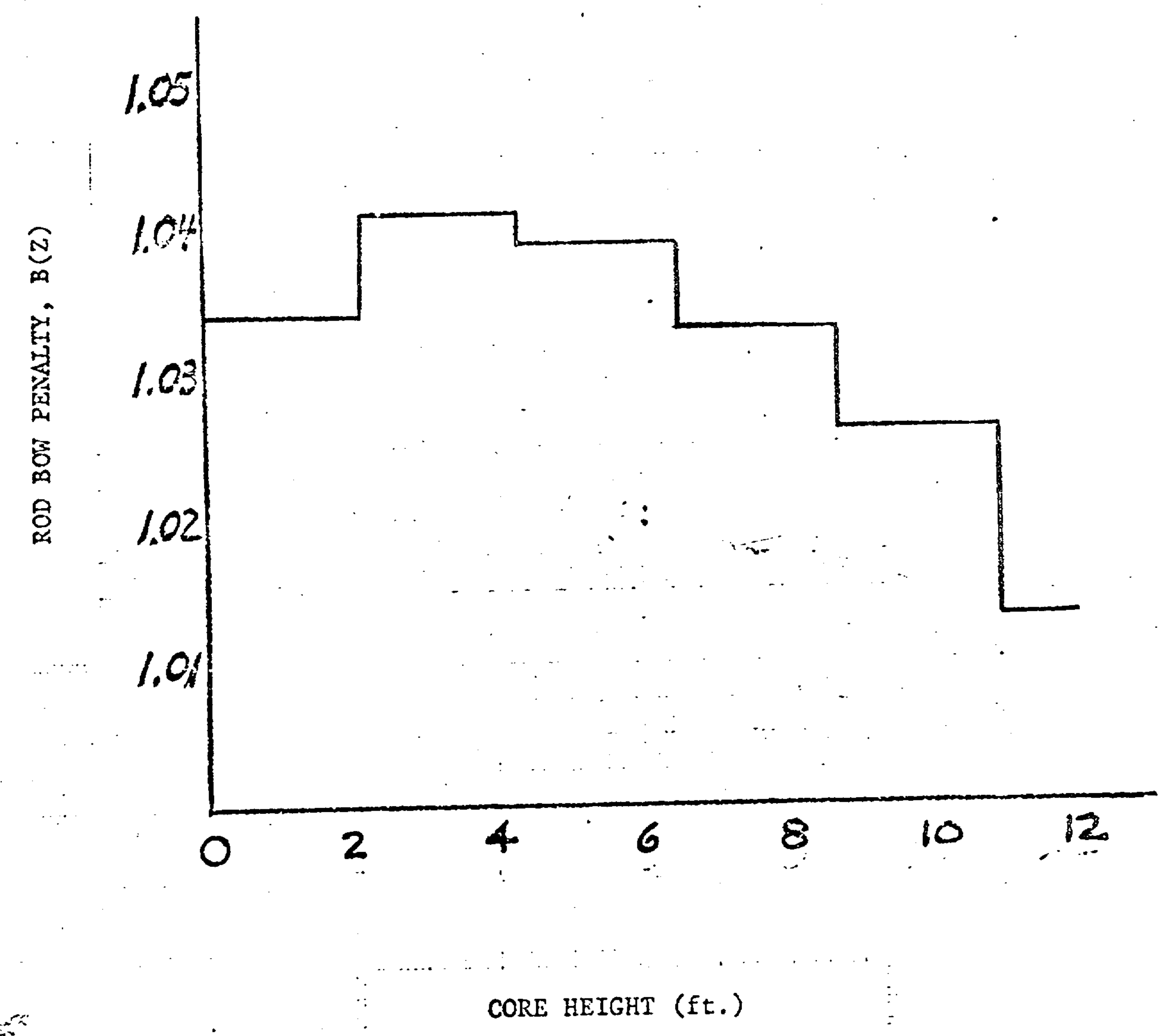


FIGURE 3.12-1A CONTROL BANK INSERTION LIMITS FOR 3-LOOP NORMAL OPERATION-UNIT 1

26

ROD BOW PENALTY
SURRY POWER STATION
UNIT 1



25

NOV 25 1976

UNITED STATES
NUCLEAR REGULATORY COMMISSIO
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. 11
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 8, 1975, as supplemented October 22, 1975 and October 30, 1975, 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; and

   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.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 3.B of Facility License No. DPR-37 is hereby amended to read as follows:

"3.B Technical Specifications

The Technical Specifications contained in Appendix A, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 27."

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

FOR THE NUCLEAR REGULATORY COMMISSION

Morton B. Fairtile for

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

Attachment:
Change No. 26 to the
Technical Specifications

ATTACHMENT TO LICENSE AMENDMENT NO. 11

CHANGE NO. 26 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NO. 50-281

Revise Appendix A as follows:

Remove Pages	Insert Pages
2.1-2	2.1-2
2.1-6	2.1-6
3.12-3 - 3.12-22	3.12-3 - 3.12-27
--	Table 3.12-1
Figure 3.12-1A	Figure 3.12-1A
--	Figure 3.12-9

NOV 25 1975

4. The reactor thermal power level shall not exceed 118% of rated power.

B. The safety limit is exceeded if the combination of Reactor Coolant System average temperature and thermal power level is at any time above the appropriate pressure line in TS Figures 2.1-1, 2.1-2 or 2.1-3; or the core thermal power exceeds 118% of rated power.

C. The fuel residence time shall be limited to 7600 effective full power hours (EFPH) for Cycle 3 of Unit 1 and to 17,000 EFPH for Cycles 1 and 2 of Unit 2. 26

Basis

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the reactor coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed Departure From Nucleate Boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters; thermal power, reactor coolant temperature and pressure have been related to DNB through the W-3 correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially to this limiting criterion. Additional peaking factors to account for local peaking due to fuel rod axial gaps and reduction in fuel pellet stack length have been included in the calculation of this limit.

The fuel residence time is limited to 7600 EFPH for Cycle 3 of Unit 1 and to 26
17,000 EFPH for Cycles 1 and 2 of Unit 2 to assure no fuel clad flattening will occur in the cores without prior review by the Regulatory Staff.

4. Whenever the reactor is subcritical, except for physics test, 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. Operation with part length rods shall be restricted such that except during physics tests, the part length rod banks are withdrawn from the core at all times.

6. Insertion limits do not apply during physics tests or during periodic exercise of individual rods. However, the shutdown margin indicated in TS Figure 3.12-7 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 and part length rods fully withdrawn.

7. For Surry Unit 1, after 5000 MWD/MTU of burnup in Cycle 3, the total cumulative cycle energy-weighted average D bank insertion should not be greater than 9%. Should this energy-weighted D bank insertion limit be violated, movable detector surveillance is required for operation when the thermal power is in excess of 95% power. This surveillance will be performed in accordance with the following: 26

a. The normalized axial power distribution, $F_j(Z)$, from thimble j at core elevation Z shall be measured utilizing at least two thimbles of the movable incore flux system for which $\overline{R}$, 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.

(1) Raising the thermal power above 95% rated power, or 26

(2) Movement of the control bank of rods more than an accumulated total of five steps in any one direction.

b. If $F_j(Z)$ exceeds its limit, $(F_j(Z))_L$ as defined in the basis, the reactor power shall be reduced until the limit, $(F_j(Z))_L$, is met.

When the thermal power is in excess of 95%, surveillance, in accordance with a. and b. above, will continue until the total cumulative cycle energy-weighted D bank insertion is within the prescribed limit.

B. <u>Power Distribution Limits</u>

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

$F_Q(Z) \leq (2.10/P) \times K(Z)$ for $P > .5$

$F_Q(Z) \leq (4.20) \times K(Z)$ for $P \leq .5$

$F^N_{\Delta H} \leq 1.52\ (1 + 0.2(1 - P))$ 26

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, 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 three percent to account for manufacturing tolerances and further increased by five percent to account for measurement error.

   b. 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

$F_Q \leq 2.10 \times K(Z)$ and $F^N_{\Delta H} \leq 1.52$ within 24 hours, the overpower ΔT and overtemperature ΔT trip setpoints shall be similarly reduced. 26

3. For Cycle 3 of Surry Unit 1, $F_{xy}(Z)$ shall be limited to the values shown in TS Table 3.12 -1 for the unrodded core plane region located between a core plane elevation 2.5 feet from the top of the core and a core plane elevation 1.5 feet from the bottom of the core, excluding grid strap locations. $F_{xy}(Z)$ shall be determined to be within its limit by using the moveable incore detectors to obtain a power distribution map after each fuel loading, and at least once every full power month. With $F_{xy}(Z)$ exceeding its limit:

   (a) Operation is restricted to a maximum permitted power level, P_{RB}, which is reduced from 100% power one percent for every one percent $F_{xy}(Z)$ exceeds the $F_{xy}(Z)$ limit evaluated at 100% power, or

   (b) Demonstrate through manual surveillance using the moveable incore detector system that the axial power distribution limits, $(F_j(Z))_L$, are not violated. For the purpose of this section of the Technical Specifications 26

$$(F_j(Z))_L = \frac{2.10 \qquad (K(Z))}{(P) \quad (\bar{R}_j) \quad (1.03) \quad (1+\sigma_j) \quad (1.07) \quad (B(Z))}$$

   where B(Z) is the rod bow penalty as a function of axial core elevation shown in TS Figure 3.12 -9, and the other parameters in the above equation are defined in the basis. The surveillance on $F_j(Z)$ will be performed by measuring the normalized axial power distribution, $F_j(Z)$, from thimble j at core elevation Z utilizing at least two timbles of the moveable incore flux system for which $\bar{R}$, 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:

1. Raising the thermal power above P_{RB} or
2. Movement of the control bank of rods more than an accumulated total of five steps in any one direction.

If $F_j(Z)$ exceeds its limit, $(F_j(Z))_L$ as defined in this section, the reactor power shall be reduced until the limit, $(F_j(Z))_L$ is met, or reduce power to a power level below P_{RB}. 26

4. The reference equilibrium indicated axial flux difference (called the target flux difference) at a given power level P_o, is that indicated axial flux difference with the core in equilibrium xenon conditions (small or no oscillation) and the control rods more than 190 steps withdrawn. The target flux difference at any other power level, P, is equal to the target value of P multiplied by the ratio, P/P_o. The target flux difference shall be measured at least once per equivalent full power quarter. The target flux difference must be updated during each effective full power month of operation either by actual measurement, or by linear interpolation using the most recent value and the value predicted for the end of the cycle life. 26

5. Except during physics tests, during excore detector calibration and except as modified by 3.12.B.5.a, b, or c below, the indicated axial flux difference shall be maintained within a +6 to -9% band about the target flux difference (defines the target band on axial flux difference). 26

a. At a power level greater than 90 percent of rated power, if the indicated axial flux difference deviates from its target band, the flux difference shall be returned to the target band, or the reactor power shall immediately be reduced to a level no greater 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 +6 to -9% target band for a maximum of one hour (cumulative) in any 24 hour period provided the flux difference does not exceed an envelope bounded by -18 percent and +11.5 percent at 90% 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.5.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. 26

(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. One half of the time the indicated axial flux difference is out of its target band up to 50 percent of rated power is to be counted as contributed to the one hour cumulative maximum the flux difference deviates from its target band at a power level less than or equal 90 percent of rated power.

Alarms shall normally be used to indicate the deviations from the
axial flux difference requirements in 3.12.B.5.a and the flux 26
difference time limits in 3.12.B.5.b. If the alarms are out of 26
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.

6. The allowable quadrant to average power tilt is 26

$$T = 2.0 + 50\ (1.40\ /F_{xy} - 1) \leq 10\%$$

where F_{xy} is 1.40 , or the value of the unrodded horizontal plane peaking factor appropriate to F_Q as determined by a movable incore detector map taken on at least a monthly basis; and T is the percentage operating quadrant tilt limit, having a value of 2% if
F_{xy} is 1.40 or a value up to 10% if the option to measured F_{xy} 26
is in effect.

NOV 2 5 1975

7. If the quadrant to average power tilt exceeds a value T% as selected in 3.12.B.6, except for physics and rod exercise testing, then: 26

   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 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% except for physics tests, the power level and high neutron flux trip setpoint will be reduced from rated power, 2% for each percent of quadrant tilt.

8. If after a further period of 24 hours, the power tilt in 3.12.B.7 above is not corrected to less than +T%: 26

   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 an abnormal 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. Inoperable Control Rods

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 of 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 3.12.C.1 and 3.12.C.2 shall not apply during physics test in which the assemblies are intentionally misaligned.

5. If an inoperable full-length rod is located below the 200 step level and is capable of being tripped, or if the full-length rod is located below the 30 step level whether or not it is capable of being tripped, then the insertion limits in TS Figure 3.12-2 apply.

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. No insertion limit changes are required by an inoperable part-length rod.

8. 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. 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.

1. Once per day, and

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

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

1. Movable detectors (at least two per quadrant)

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

NOV 2 5 1975

E. Inoperable Rod Position Indicator Channels

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. Misaligned or Dropped Control Rod

1. If the Rod Position Indicator Channel is functional and the associated part length or 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 Section 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 part-length or 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. Early in core life, less shutdown margin is required, and TS Figure 3.12-7 shows the shutdown margin equivalent to 1.77% reactivity at end-of-life with respect to an uncontrolled cooldown. All other accident analyses 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 and part-length rods) 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 (part-length of 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 linear power density must not exceed 20.4 kw/ft. 25
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 must not exceed the limiting Kw/ft 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 on power distribution the following hot channel factors are defined.

$F_Q(Z)$, 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.

NOV 25 1975

It should be noted that $F^N_{\Delta H}$ 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 $F^N_{\Delta H}$.

An upper bound envelope of 2.10 times the normalized peaking factor axial dependent of TS Figure 3.12-8 has been determined from extensive analyses considering all operating maneuvers consistent with the technical specifications on power distribution control given in Section 3.12.B.4. 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.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map ($\geq$ 40 thimbles monitored) taken with the movable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerances.

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 $F^N_{\Delta H} \leq 1.52/1.08$. 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 affect F_Q, (b) the operator has a direct influence on F_Q through movement of rods, and can 26 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 can be compensated for the F_Q by tighter axial control, but compensation for $F^N_{\Delta H}$ is taken, experimental error must be allowed for and four percent is the appropriate allowance for a full core map ($\geq$ 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 is not necessary to measure these quantities. Instead it has been determined that, provided certain conditions are observed, the hot channel factor limits 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 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.

4. For Surry Unit 1, the total cumulative cycle energy weighted average D bank insertion limit is observed. 26

5. 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 5 are observed, these hot channel factors limits are met. In Specification 3.12.B.1, F_Q is arbitrarily limited for $P \leq .5$ (except for physics tests).

For Surry Unit 1, the total cumulative cycle energy weighted average D bank insertion limit referred to above is designed to ensure that long-term core depletion with significant D bank insertion does not occur, since such depletion could produce an axial burnup distribution which could cause the 26 total peaking factor to potentially exceed the LOCA limiting $F_Q(Z)$ for certain plant maneuvers near the end of Cycle 3. However, it has been determined that for these plant maneuvers, the $F_Q(Z)$ upper band envelope will not be violated if after 5000 MWD/MTU, the core is depleted with the cumualtive energy weighted D bank insertion from the beginning of cycle no greater than 9%. If this total cumulative cycle energy weighted average D bank insertion limit is violated, additional axial power distribution surveillance using the movable detector system is implemented in order to assure that the power peaking factor, $F_Q(Z)$, is maintained at or below the limiting value. Flux shape surveillance is not required below 95% power, since it has been determined that for the worst case, including plant maneuvers following core depletion with significant D bank insertion, the calculated $F_Q(Z)$ peaking factor at 100% power is at the most 5% above the LOCA limiting $F_Q(Z)$ envelope.

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 limit on the axial distribution function $F_j(Z)$ is derived as follows: 26

$$(F_j(Z))_L = \frac{2.10\ (K(Z))}{(P)(\bar{R}_j)(1.03)(1 + \sigma_j)(1.07)}$$

where:

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

b. P is the fraction of thermal power.

c. K(Z) is the reduction in limit as a function of core elevation Z as determined from TS Figure 3.12-8.

d. $(F_j(Z))_L$ is the operational limit on $F_j(Z)$.

e. $\bar{R}_j$, for thimble j, is determined from at least n=6 incore flux maps covering the full configuration of permissible rod patterns at the thermal power 95% of rated power.

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

where

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

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

26

The full incore flux map used to update $\bar{R}$ and for monitoring $F_j(Z)$ shall be taken at least once 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 $\bar{R}$ for each representative thimble.

f. σ_j is standard deviation of R_j and is derived from n flux maps from the relationship below, or 0.02, whichever is greater. 26

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

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

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 expresses as axial offset it varies only with burnup.

The technical specifications on power distribution control given in 3.12.B.4 along with the cycle energy weighted D bank insertion limit given in 3.12.A.7 26 assure that the F_Q upper bound envelope of 2.10 times Figure 3.12-8 is not exceeded and xenon distributions are not developed which at a later time, would cause greater local power peaking even though the flux difference is then within the limits specified by the procedure.

For Cycle 3 of Surry Unit 1, a limit on $F_{xy}(Z)$ has been imposed to insure that with the inclusion of the rod bow power peaking pealty, the LOCA $F_Q(Z)/P$ envelope will not be violated. If, by core flux mapping, the $F_{xy}(z)$ limit is determined to be violated, the minimum allowable power level will be reduced from 100% rated power by one percent for each one percent violation of the $F_{xy}(Z)$ limit, or manual moveable detector surveillance will be implemented for the period that the violation occurs. The imposition of the limit of $F_{xy}(Z)$ is an interim measure taken to conservatively include the potential effects of rod bowing on core power capability until the NRC has further evaluated the problem. 26

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 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 possible during certain physics tests or during required, periodic, excore detector calibrations which require larger flux differences than permitted. Therefore, the specifications on power distribution control are not applied during physics tests or excore detector calibrations; this is acceptable due to the low probability of a significant accident occuring during these operations.

In some instances of rapid unit power reduction automatic rod motion will cause the flux difference to deviate from the target band when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target band, however to simplify the specification, a limitation of one hour in any period of 24 hours is placed on operation outside the band. This ensures that the resulting xenon distributions are not significantly different from those resulting from operation within the target band. The instantaneous consequences of being outside the band, provided rod insertion limits are observed, is not worse than a 10 percent increment in peaking factor for the allowable flux difference at 90% power, in the range +14.5 to -21 percent (+11.5 percent to -18 percent indicated) where for every 4 percent below rated power, the permissible positive flux difference boundary is extended by 1 percent, and for every 5 percent below rated power, the permissible negative flux difference boundary is extended by 2 percent.

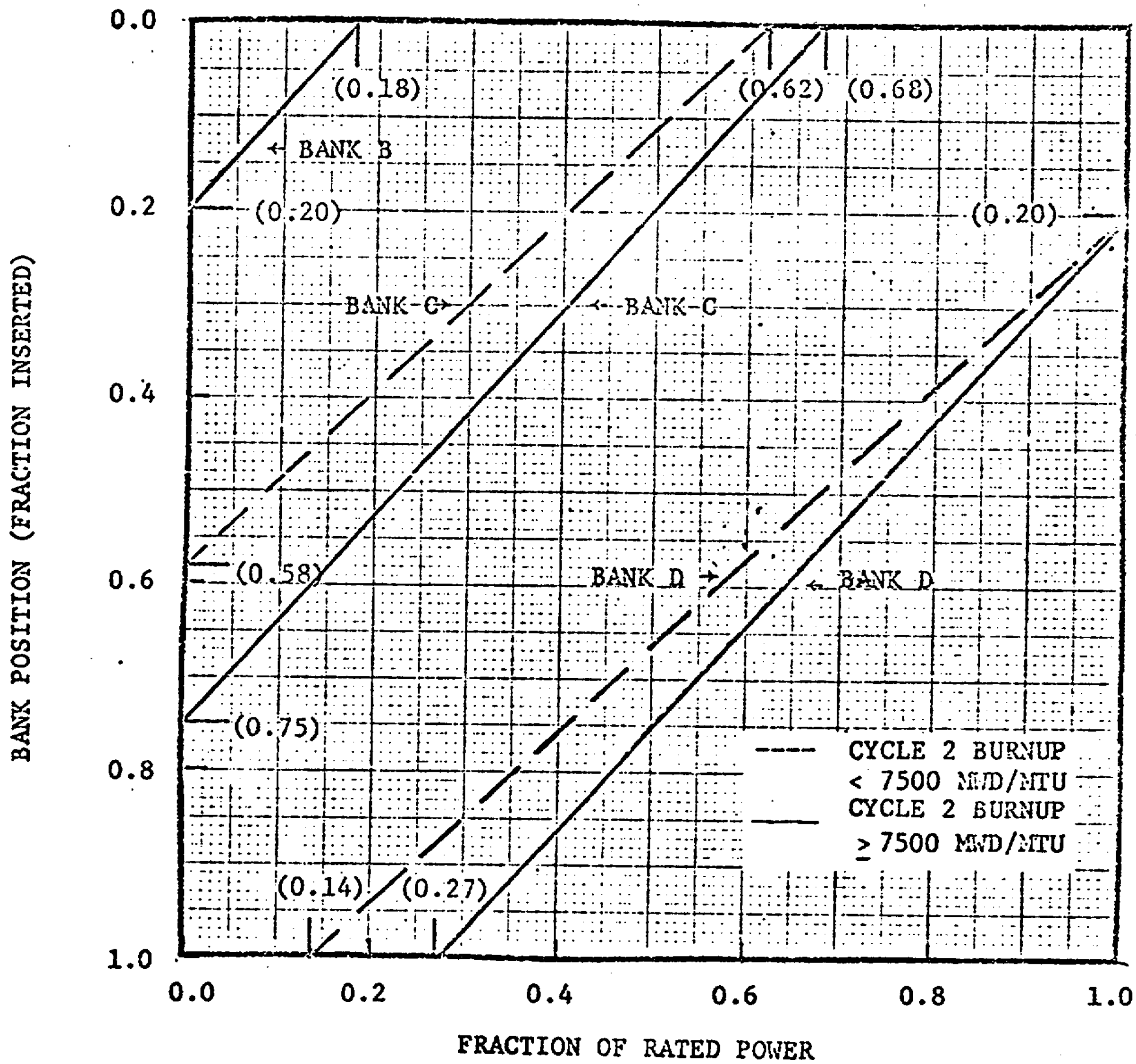
As discussed above, the essence of the procedure is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This is accomplished, by using the boron system to position the full length control rods to produce the required indicated flux difference. At the option of the operator, credit may be taken for measured decreases in the unrodded horizontal plane peaking factor, F_{xy}. This credit may take the form of an expansion of permissible quadrant tilt limits over tilt limits over the 2% value, up to a value of 10%, at which point specified power reductions are prudent. Monthly surveillance of F_{xy} bounds the quantity because it decreases with burnup. (WCAP-7912 L).

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 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.

Core Height, Z (feet)	$F_{xy}(Z)$ limit
1.5	1.406/P
2.0	1.445/P
2.5	1.471/P
3.0	1.535/P
3.5	1.570/P
4.0	1.527/P
4.5	1.510/P
5.0	1.492/P
5.5	1.485/P
6.0	1.478/P
6.5	1.494/P
7.0	1.461/P
7.5	1.431/P
8.0	1.421/P
8.5	1.395/P
9.0	1.418/P
9.5	1.422/P

where $F_{xy}(Z)$ = ratio of peak power density to average power density in the horizontal plane at elevation Z

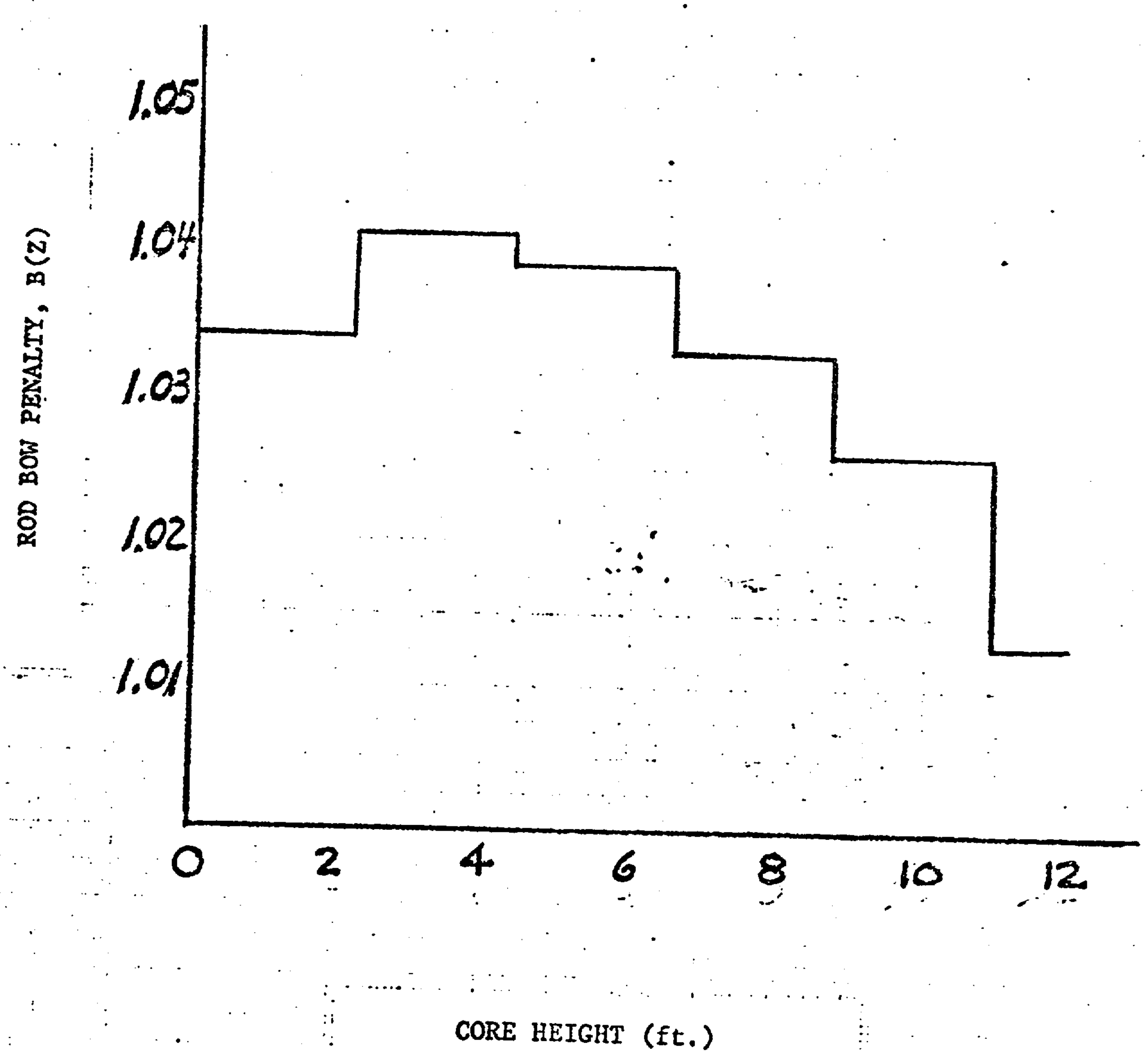
P = fraction of rated power at which the core is operating



26

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

ROD BOW PENALTY
SURRY POWER STATION
UNIT 1



NOV 2 [illegible] 1975

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENTS NO. 11 TO LICENSES NOS. DPR-32 AND DPR-37

CHANGE NO. 26 TO TECHNICAL SPECIFICATIONS

VIRGINIA ELECTRIC & POWER COMPANY

SURRY POWER STATION UNITS 1 & 2

DOCKETS NOS. 50-280 AND 50-281

I. Introduction

By a letter dated September 8, 1975, and supplemented by letters dated October 22, 1975 and October 30, 1975, Virginia Electric & Power Company (the licensee) requested changes to the Technical Specifications appended to Facility Operating Licenses Nos. DPR-32 and DPR-37 for the Surry Power Station Units 1 and 2. The purpose of the request is to revise the Surry 1 Technical Specifications as required to operate within the appropriate fuel and core design limits during the third fuel cycle.

II. Discussion

The reloading of the core for fuel cycle 3 will involve the replacement of 81 assemblies with 65 once-burned assemblies from cycle 1 and 16 fresh assemblies of the 157 fuel assemblies in the core. The third cycle core will consist of six regions of fuel; three that are carried over from the second cycle, Regions 4, 4A and 4B; two that are once burned from cycle 1, Regions 1 and 3; and one that is fresh, Region 5. The fuel to be added to the core is not significantly different in design or in operating characteristics from the fuel it replaces. The rearrangement of fuel assemblies in the reloaded core affects core physics calculations and, as a result, changes to the Technical Specifications are required. Rod bowing effects described by Westinghouse (reference 3) are also accommodated in these Technical Specification changes.

III. Safety Evaluation

A. Clad Flattening

Clad flattening time is predicted to be 17,000 effective full power hours (EFPH) for the limiting region, Region 3, using the NRC approved evaluation model - WCAP 8377 (Proprietary) and WCAP 8381 (Non-Proprietary), entitled, "Revised Clad Flattening Model" July 1974. Region 3 had a Cycle 1 fuel residence time of 9400 EFPH; therefore, Region 3 has a Cycle 3 allowable remaining residence time of 7600 EFPH. Cycle 3 has a predicted operational time of only 6400 EFPH. We conclude clad flattening will not occur during Cycle 3.

B. Nuclear Design

1. Core Characteristics

The Cycle 3 core loading will consist of 65 once-burned fuel assemblies from Cycle 1, 76 once-burned assemblies from Cycle 2, and 16 fresh assemblies. Two 17 x 17 test assemblies loaded in Cycle 2 will remain in the core. The presence of these assemblies does not affect the core nuclear characteristics adversely relative to an all 15 x 15 assembly core.

For the cycle 3 core loading, the worth of two control rod banks moving together was predicted to be 65 pcm/sec [1 pcm - $10^{-5}\Delta K$] for Cycle 3 compared to a value of 60. pcm/sec for the FSAR. Ejected control rod worths for the Beginning of Cycle - Hot Full Power (BOC-HFP) and End of Cycle - Hot Full Power (EOC-HFP) rod ejection incidents are greater for Cycle 3 than the corresponding analyses performed previously. These cases were reanalyzed, and are discussed below under accident analysis.

Other nuclear characteristics of the Cycle 3 core fall within the range used in accident analyses accepted for previous cycles. These analyses remain applicable.

2. Power Distribution

The licensee has provided predictions of the maximum peaking factor as a function of core axial height, $F_Q(Z)$, for the Cycle 3 core characteristics. The $F_Q(Z)$ calculations were performed using constant axial offset control (CAOC) procedures. The predictions consider various load following maneuvers as a function of extremes in possible depletion modes of the reactor, control strategies, and magnitude of the load follow. The maximum $F_Q(Z)$ calculated is compared with the $F_Q(Z)$ limit, which must be maintained to avoid exceeding the linear power density used for the LOCA analysis.

For Surry Unit 1 Cycle 3 the results of the calculations indicate that the $F_Q(Z)$ limit will not be violated under the present constant axial offset control Technical Specifications with the following single exception.

Some of the load follow maneuvers allowed under CAOC were analyzed for near end of cycle life condition and found to result in power peaking in the upper portion of the core in excess of the $F_Q(Z)$ limit. This potential power peaking was less than 5% above the $F_Q(Z)$ limit. To ensure that this will not occur however, the licensee has proposed to augment CAOC procedures by including a technical specification* which limits the energy-weighted average insertion over the cycle of control bank D to no greater than 9%. He has furnished results of an analysis (Reference 1) which shows the effectiveness of the energy-weighted bank D insertion limit in avoiding the $F_Q(Z)$ limit violations. We find this technique acceptable, as $F_Q(Z)$ limit violations will be avoided, and approve Technical Specification 3.12.4.7 proposed by the licensee to limit D bank insertion.

The licensee has also furnished results of departure from nucleate boiling ratio (DNBR) analysis for limiting axial power shapes generated by the $F_Q(Z)$ calculations. These show greater conservatism than the 1:55 axial cosine shape employed as a design basis for departure from nucleate boiling (DNB) protection setpoints. We conclude that the $F_Q(Z)$ limit and DNB design basis will not be violated in normal operation of Cycle 3.

3. Control Rod Insertion Limits

The licensee proposed to change the control rod insertion limits for Cycle 3 to provide more flexibility in control rod bank positioning of the Hot Zero Power (HZP) critical position and in going from HZP to power operation. He has evaluated these insertion limits for conformance with the following design limiting criteria:

1) The required shutdown margin must be maintained throughout the cycle.

2) The enthalpy rise hot channel factor, $F^N_{\Delta H}$ must be maintained within limits.

3) The consequences of an ejected control rod assembly must be within the accepted limits.

4) The trip reactivity assumed in the accident analysis must be available.

5) Statically misaligning a control assembly will not violate the thermal design basis with respect to DNBR.

6) The uncontrolled withdrawal of a control assembly bank will not result in a peak power density that exceeds the center line melting criterion.

$$*9\% \leq \frac{\int^{time} (\%\ insertion) \times Power\ dt}{\int^{time} Power\ dt}$$

The second criterion was limiting for the shortened Cycle 2 core prior to Cycle 3. The other criteria were satisfied with margin. We find these criteria acceptable for Cycle 3, particularly since there is an extra margin of 4% in the uncertainty allowance for $F^N_{\Delta H}$ in Cycle 3 over Cycle 2.

4. Rod Bowing

The licensee has employed the rod bow peaking factor penalties calculated by Westinghouse Electric Company (Reference 3). The Westinghouse calculations are based upon a characterization of all their bowing experience to date. The characterization is inferred from the inspection of 24 different regions of fuel (about 25,000 fuel rods) including more than 70 assemblies at burnups beyond 27,000 MWD/MTU. The bowing is characterized by a bow variance at each spacer span (6 elevation increments). The licensee has properly assessed span-wise bow penalties (for burn up of 25,000 MWD/T) for Surry 1, Cycle 3 from the Westinghouse data.

Combining the axially-dependent maximum peaking factors, $F_Q(Z)$, that were presented in Fig. 1 of Reference 1 with the axially-dependent rod bow penalties described above led to several potential violations of the LOCA limiting $F_Q(Z)$ envelope. The largest potential violation was 2.1%. This is less than the 2.5 to 3.5% rod bow penalty factors because there was adequate margin between the calculated $F_Q(Z)$ points and the limit envelope.

The licensee used a horizontal plane peaking factor, F_{xy} of 1.435 at the controlling axial elevations in order to predict the maximum $F_Q(Z)$. This value of F_{xy} is conservative by 2.1% as indicated by 3-dimensional calculations. The licensee is imposing a rod bow penalty, at the controlling axial elevation by reducing the allowable F_{xy}. The necessary reduction never exceeds the existing 2.1% conservatism thus the rod bowing penalty is accommodated by the conservatism in the F_{xy} calculations.

At our request the licensee has proposed a Technical Specification which verifies this accommodation of the rod bow penalty as follows: The F_{xy} in unrodded planes shall be measured and compared to the allowable F_{xy} for each axial location at core startup and at monthly intervals thereafter. Should the F_{xy} exceed the allowable value, in-core surveillance of $F_Q(Z)$ or a core power decrease is required to assure that the $F_Q(Z)$ limit is met. The in-core surveillance is a manual application of the axial power distribution monitoring system (APDMS). This in-core surveillance has been and continues to be included in Technical Specifications as an acceptable method for limiting peaking factors at Surry.

We find the above Technical Specification acceptable as it requires confirmation by measurement of the calculation of $F_Q(Z)$ regarding the value of F_{xy} and provides alternate procedures to assure that the $F_Q(Z)$ limits are observed even if the F_{xy} limit is not met. This procedure is acceptable as it provides assurance that peaking factor limits will not be exceeded in Cycle 3 operation.

5. Accident Analysis

Results of the analysis for the rod withdrawal from sub-critical incident showed that the peak heat flux increased by only 4% due to the higher reactivity insertion rate of 65 pcm/sec for Cycle 3. Since the peak heat flux for the analysis presented in the FSAR reached only 67% of the nominal full power value, the increased reactivity insertion rate does not affect the conclusions presented in the FSAR.

While ejected rod worths for BOC-HFP & EOC-HFP are greater for Cycle 3 than for the original analysis; reanalyses show fuel and clad temperatures and the number of fuel pins in DNB to be less than applicable limits for this accident (Reference 2). We conclude the reanalysis for the ejected rod accident is acceptable, thus the consequences are within limits.

C. Reactor System Design

1. Moderator Temperature Coefficient

The Surry Units 1 and 2 proposed Technical Specification change has been reviewed with the understanding that the positive moderator temperature coefficient referenced in the licensee's September 8, 1975 letter would not be used during Cycle 3. Technical Specification Section 3.1.E.1 requires that the moderator temperature coefficient be negative or zero.

2. Transient and Accident Analyses

The transients and accidents previously reported have been reevaluated for the cycle 3 core. We find such limiting transients as boron dilution and rod withdrawal (which was analyzed with a peaking factor $F_{\Delta H}$=1.55) to be within acceptable limits. That is, there is sufficient time for operator action before loss of shutdown margin and the minimum DNB ratio does not fall below 1.30. We find the steam line break accident to have been analyzed with the appropriate parameters applying to the cycle 3 core.

3. Rod Bowing Effects on DNBR

The analyses previously referenced were performed with a pitch reduction factor which results in a 3.3 percent margin in DNBR to allow for rod-to-rod bowing. Recent discussions with Westinghouse indicate that this penalty is inadequate. New data on 15 x 15 rod bundles with up to 27,000 MWd/MTu burnup show that the bowing model presented in WCAP-8346, "An Evaluation of Fuel Rod Bowing," underestimates the extent of rod bowing. The 15 x 15 bowing data indicate that a penalty of approximately 4.2 percent in DNBR should be applied to the Surry design to account for rod bowing during Cycle 3. We will require that a total penalty of 6.2 percent in DNBR (including Surry design pitch reduction penalty) be used to account for rod bowing. A suitably conservative value of 6.2 percent was chosen instead of the 4.2 percent penalty because the review of the Westinghouse approach for 15 x 15 geometry has not been completed. Once the review is complete the 6.2 percent penalty may be modified to conform to the data.

As stated previously, the Surry core design offers approximately 3.3 percent margin in DNBR due to pitch reduction in the analyses. The remaining 2.9 percent of the 6.2 percent penalty is equivalent to a 1.7 percent heat flux penalty. To achieve a 1.7 percent heat flux reduction Technical Specification 3.12 has been changed to limit operation of the Surry 1 cycle 3 core to an enthalpy rise peaking factor, $F_{\Delta H}$, of 1.52 rather than 1.55. With this limitation rod bowing effects on DNBR will be accommodated in an acceptable manner.

IV Conclusion

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

Date: NOV 25 1975

References

1. Letter from C. M. Stallings (VEPCO) to R. W. Reid (NRC) (Proprietary), Serial number 746, October 22, 1975.

2. Risher, D. H. "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods", WCAP-7588, Rev. 1-A (Proprietary), January 1975.

3. Letter from C. Eicheldinger (Westinghouse) to D. Vassallo (NRC) (proprietary), Number NS-CE-828, October 28, 1975.

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

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments No. 11 to Facility Operating Licenses Nos. DPR-32 and DPR-37 issued to Virginia Electric & Power Company (VEPCO) which revised Technical Specifications for operation of the Surry Power Station, Units 1 and 2, located in Surry County, Virginia. The amendments are effective as of the date of issuance.

The amendments revise the provisions in the Technical Specifications relating to the replacement of 81 of 157 fuel assemblies in the reactor core, constituting refueling of the core for third cycle operation of Unit 1.

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 is not required since the amendments do not involve a significant hazards consideration.

For further details with respect to this action, see (1) the application for amendments dated September 8, 1975, as supplemented October 22, 1975 and October 30, 1975, (2) Amendments No. 11 to Licenses Nos. DPR-32 and DPR-37, with Change No. 26, 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 & 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 Reactor Licensing, Office of Nuclear Reactor Regulation.

Dated at Bethesda, Maryland, this 26th day of November, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Morton Fairtile

Morton Fairtile, Acting Chief
Operating Reactors Branch #4
Division of Reactor Licensing