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

May 25, 1976

Dockets Nos. 50-280
and 50-281

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

Gentlemen:

The Commission has issued the enclosed Amendments No. 20 to Facility Operating Licenses Nos. DPR-32 and DPR-37 for the Surry Power Station Units Nos. 1 and 2. These amendments consist of changes to the Technical Specifications in response to your two applications dated March 11, 1976, as supplemented May 12, and 14, 1976.

These amendments relate to both the increase in the limiting nuclear enthalpy hot channel factor ($F^N_{\Delta H}$) for Surry Units Nos. 1 and 2 and to the replacement of 81 of 157 fuel assemblies in the reactor core of Surry Unit No. 2 constituting refueling of the core for third cycle operation.

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

Sincerely,

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch No. 4
Division of Operating Reactors

Enclosures:
1. Amendment No. 20 to DPR-32
2. Amendment No. 20 to DPR-37
3. Safety Evaluation
4. Federal Register Notice

cc w/enclosures:
See next page

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
dated: 5/12/76, 5/14/76
Ms. Susan T. Wilburn
Commonwealth of Virginia
Council on the Environment
P. O. Box 790
Richmond, Virginia 23206

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 20
License No. DPR-37

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

   A. The two applications for amendment by Virginia Electric and Power Company (the licensee) dated March 11, 1976, as supplemented May 12 and 14, 1976, comply 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 applications, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. After weighing the environmental aspects involved, the issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Coller

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

Attachment:
Changes to the
Technical Specifications

Date of Issuance:

May 25, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 20

FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NO. 50-281

Revise the Technical Specifications as follows:

REMOVE PAGES	INSERT PAGES
2.1-2	2.1-2
2.1-3	2.1-3
2.1-6	2.1-6
3.12-1 thru	3.12-1 thru
3.12-27	3.12-22
TS Table 3.12-1	TS Table 3.12-1
TS Figure 3.12-1B	TS Figure 3.12-1B
TS Figure 3.12-9	TS Figure 3.12-9

The changed areas on the pages are shown by a marginal line.

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 6699 EFPH for Cycle 3 of Unit 2.

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

uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNB ratio (DNBR) during steady state operation, normal operational transients and anticipated transients, is limited to 1.30. A DNBR of 1.30 corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.[1]

The curves of TS Figure 2.1-1 which show the allowable power level decreasing with increasing temperature at selected pressures for constant flow (three loop operation) represent limits equal to, or more conservative than, the loci of points of thermal power, coolant system average temperature, and coolant system pressure for which the DNB ratio is equal to 1.30 or the average enthalpy at the exit of the core is equal to the saturation value. The area where clad integrity is assured is below these lines. The temperature limits are considerably more conservative than would be required if they were based upon a minimum DNB ratio of 1.30 alone but are such that the plant conditions required to violate the limits are precluded by the self-actuated safety valves on the steam generators. The three loop operation safety limit curve has been revised to allow for heat flux peaking effects due to fuel densification. The effects of rod bowing are also considered in the DNBR analyses.

The curves of TS Figures 2.1-2 and 2.1-3 which show the allowable power level decreasing with increasing temperature at selected pressures for constant flow (two loop operation), represent limits equal to, or more conservative,

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 6699 EFPH for Cycle 3 of Unit 2 to assure no fuel clad flattening will occur in the cores without prior review by the Regulatory Staff.

References

1) FSAR Section 3.4

2) FSAR Section 3.3

3) FSAR Section 14.2

3.12 CONTROL ROD ASSEMBLIES AND POWER DISTRIBUTION LIMITS

Applicability

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

Objective

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

Specification

A. Control Bank Insertion Limits

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

culations and physics data obtained during unit startup and subsequent operation, will be permitted.

c. The shutdown margin with allowance for a stuck control rod assembly shall exceed the applicable value shown on TS Figure 3.12-7 under all steady-state operation conditions, except for physics tests, from zero to full power, including effects of axial power distribution. The shutdown margin as used here is defined as the amount by which the reactor core would be subcritical at hot shutdown conditions ($T_{avg} \geq 547^{o}F$) if all control rod assemblies were tripped, assuming that the highest worth control rod assembly remained fully withdrawn, and assuming no changes in xenon, boron, or part-length rod position.

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

5. 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 in burnup of 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:

   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 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 $P \leq .5$

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

where P is the fraction of rated power at which the core is operating, K(Z) is the function given in Figure 3.12-8, 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 $F^N_{\Delta H} \leq 1.55$ within 24 hours, the overpower ΔT and overtemperature ΔT trip setpoints shall be similarly reduced.

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

4. Except during physics tests, during excore detector calibration and except as modified by 3.12.B.4.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).

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

(3) A power increase to a level greater than 90 percent of rated power is contingent upon the indicated axial flux difference being within its target band.

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

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

(2) A power increase to a level greater than 50 percent of rated power is contingent upon the indicated axial flux difference not being outside its target band for more than two hours (cumulative) out of the preceding 24 hour period. 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.4.a and the flux Difference time limits in 3.12.B.4.b. If the alarms are out of service temporarily, the axial flux difference shall be logged, and conformance to the limits assessed, every hour for the first 24 hours, and half-hourly thereafter.

5. The allowable quadrant to average power tilt is

$$T = 2.0 + 50\,(1.435\,/F_{xy} - 1) \leq 10\%$$

where F_{xy} is 1.435 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 1.435 or a value up to 10% if the option to measure F_{xy} is in effect.

6. If the quadrant to average power tilt exceeds a value T% as selected in 3.12.B.5, except for physics and rod exercise testing, then:

   a. The hot channel factors shall be determined within 2 hours and the power level adjusted to meet the specification of 3.12.B.1, or

   b. If the hot channel factors are not determined within two hours, the power 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.

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

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

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

   c. If the hot channel factors are not determined the Nuclear Regulatory Commission shall be notified and the overpower ΔT and overtemperature ΔT trip settings shall be reduced by the equivalent of 2% power for every 1% quadrant to average power tilt.

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

1. A control rod assembly shall be considered inoperable if the assembly cannot be moved by the drive mechanism, or the assembly remains misaligned from its bank by more than 15 inches. A full-length control rod shall be considered inoperable if its rod drop time is greater than 1.8 seconds to dashpot entry.

2. No more than one inoperable control rod assembly shall be permitted when the reactor is critical.

3. If more than one control rod assembly in a given bank is out of service because of a single failure external to the individual rod drive mechanisms, i.e. programming circuitry, the provisions

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

1. If a rod position indicator channel is out of service then:
   a. For operation between 50% and 100% of rated power, the position of the RCC shall be checked indirectly by core instrumentation (excore detector and/or thermocouples and/or movable incore detectors) every shift or subsequent to motion, of the non-indicating rod, exceeding 24 steps, whichever occurs first.
   b. During operation below 50% of rated power no special monitoring is required.
2. Not more than one rod position indicator (RPI) channel per group nor two RPI channels per bank shall be permitted to be inoperable at any time.

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

1. If the Rod Position Indicator Channel is functional and the associated 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. 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 Fuel Hot Channel Factor, is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

$F_{\Delta H}^N$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

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

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 $F^N_{\Delta H} \leq 1.55/1.08$. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape (e.g. rod misalignment) affect $F^N_{\Delta H}$, in most cases without necessarily affecting F_Q, (b) the operator has a direct influence on F_Q through movement of rods, and can limit it to the desired value, he has no direct control over $F^N_{\Delta H}$, and (c) an error in the predictions for radial power shape, which may be detected during startup physics tests 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.

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 factor 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 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 cumulative 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:

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

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 $F_{Q_i}^{meas}$

The full incore flux map used to update $\overline{R}$ 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 $\overline{R}$ for each representative thimble.

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

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

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

h. The factor 1.07 is the combined uncertainty associated with the measurement of F_{Q_i} and $F_{i_j}(Z)_{MAX}$

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

The technical specifications on power distribution control given in 3.12.B.4 along with the cycle energy weighted D bank insertion limit given in 3.12.A.7 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.

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 occurring during these operations.

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

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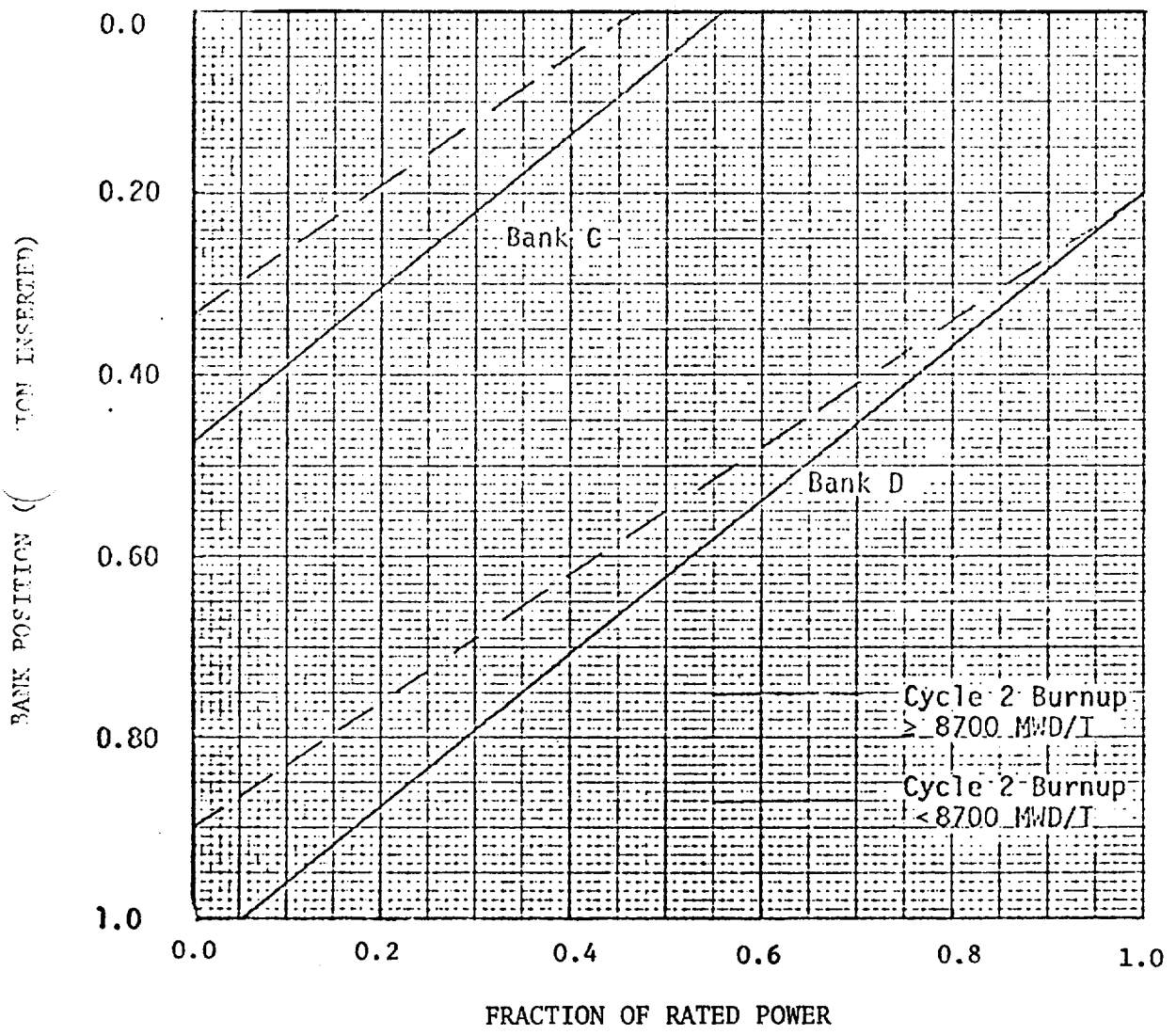

FIGURE 3.12-1B CONTROL BANK INSERTION LIMITS FOR NORMAL 3 LOOP OPERATION - UNIT 2

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENTS NO. 20 TO LICENSES NOS. DPR-32 AND DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION UNITS NOS. 1 AND 2

DOCKETS NOS. 50-280 AND 50-281

INTRODUCTION

By two applications dated March 11, 1976, as supplemented May 12, 1976 and May 14, 1976, Virginia Electric and Power Company (the licensee) proposed to change the Technical Specifications for the Surry Power Station Units Nos. 1 and 2. The proposed change would permit the licensee to increase the limiting nuclear enthalpy hot channel factor ($F^N_{\Delta H}$) for Surry Units Nos. 1 and 2, and replace 81 of 157 fuel assemblies in the reactor core of Surry Unit No. 2 constituting refueling of the core for third cycle operation.

EVALUATION

Limiting Nuclear Enthalpy Hot Channel Factor (Applicable to Units Nos. 1 and 2)

In April 1976 the staff issued an Interim Safety Evaluation Report on Westinghouse fuel rod bowing. The Westinghouse Topical Report WCAP 8691, "Fuel Rod Bowing," has been reviewed with regard to the effect of rod bowing on thermal margin (DNB) for 15 x 15 fuel. The staff evaluation concludes that for 15 x 15 fuel, the Westinghouse bowing factor of 5.7% used as the basis for their fuel design is not exceeded. Therefore no additional penalty needs to be assessed for rod bowing on 15 x 15 fuel.

Our review of the Surry Units Nos. 1 and 2 design indicates that the fuel design, 15 x 15, is within the scope of the Interim Safety Evaluation Report on bowing. Therefore we conclude that no additional penalty need be applied to Surry Units Nos. 1 and 2 over and above that used in the FSAR. We find VEPCO's proposed Amendment 36 acceptable and recommend return of the limiting nuclear enthalpy hot channel factor ($F\Delta H$) to the FSAR value of 1.55.

Fuel System Design (This and Following Sections Applicable to Unit No. 2 Only)

The mechanical design of the reload fuel assemblies (Region 5) is the same as the Region 4 assemblies. The rearrangement of the fuel assemblies affects core physics calculations and, as a result, changes to the Technical Specifications are required.

Clad flattening time is predicted to be greater than 17,000 effective full power hours (EFPH) for the limiting region, Region 3, using the NRC approved evaluation model [4]-WCAP 8377 (Proprietary) and WCAP 8381 [4] (Non-Proprietary), entitled "Revised Clad Flattening Model" dated July 1974. The irradiation time for the Region 3 fuel assemblies from Cycle 1 operation was 10,301 EFPH. The expected additional irradiation during Cycle 3 operation is 6446 EFPH which gives a total of 16,747 EFPH. However, the licensee requested an irradiation time of 6699 EFPH which would permit some of the fuel to attain the full allowable time of 17,000 EFPH. We agree with the licensee's conclusion that clad flattening will not occur through 17,000 EFPH.

Nuclear Design

Core Characteristics

The Cycle 3 core loading will consist of 25 fuel assemblies (15,500 MWD/MTU Burnup) in Region 1, 32 fuel assemblies (10,500 MWD/MTU Burnup) in Region 3, 24 fuel assemblies (10,500 MWD/MTU Burnup) in Region 4A, 52 fuel assemblies (7,800 MWD/MTU Burnup) in Region 4B, and 24 fresh fuel assemblies in Region 5. Depleted burnable poison rods will be inserted in 24 Region 3 fuel assemblies and in 20 Region 4B fuel assemblies to reduce the radial peaking factor. Two of the Region 4B assemblies contain secondary source rods and their associated burnable poison and will be symmetrically loaded. The two Region 4B assemblies symmetric (90°) to the secondary source rod assemblies have matching burnable poison inserted to preserve core symmetry.

The Cycle 3 reload kinetics parameters, control rod worths, and core peaking factors remain bounded by the values assumed in the Final Safety Analysis Report (FSAR) accident analyses with the exception of the rod ejection accident. The Cycle 3 highest control rod worth is greater and end of life delayed neutron fraction is lower than the value used for Cycle 2 analyses. As a result the rod ejection transient was reanalyzed. The results of this analysis are discussed below under the accident analysis section.

Control Rod Insertion Limits

The licensee has proposed changing the control rod insertion limits to maintain margin on $F_{\Delta H}$. The Bank D insertion limit has been evaluated dependent on Cycle 2 burnup. For burnup greater than or equal to 8,700 MWD/MTU, Bank D can only be fully inserted at zero power. For burnup less than 8,700 MWD/MTU, Bank D can be fully inserted at 5% power. The 100% power insertion limit is to be 20% for Bank D for either burnup limit. The licensee has evaluated these insertion limits and verified conformance with limiting criteria. We conclude that these criteria are acceptable for Cycle 3.

Accident Analysis

The safety analyses applicable to operation during Cycle 3 are based on previous Cycle 2 safety analyses (References 7 and 8) and those reported in the FSAR (Reference 9). As previously mentioned the exception to the above is the rod ejection accident. The rod ejection accident cases reanalyzed for deeper rod insertion and lower delayed neutron fraction are the Beginning-of-Cycle hot zero power and hot full power and End-of-Cycle hot zero power and hot full power. The results of the reanalyses indicate adequate margin in all cases between the limiting values and calculated values for the percentage of pellet melt, clad temperature and average fuel temperature.

Information was provided by the licensee on the effect of steam generator tube plugging on DNBR and peak clad temperature following a postulated LOCA. In the anlayses two possible plugging configurations are considered. First, the steam generators may be plugged uniformly, that is, the same number of tubes is plugged in each steam generator. Second, the steam generators are plugged non-uniformly, that is, a different number of tubes is plugged in each steam generator. The latter in the case for Surry Unit No. 2. The analyses indicate that a plugging level of 7.5% of the steam generator tubes could be permitted without exceeding a peak clad temperature of 2200°F. Plugging of 25% of the steam generator tubes could be permitted and still not reduce the primary system flow rate to below 4% (the uncertainty of the best estimate flow calculation) of the thermal design flow and/or increase the RCS temperature to within 4°F (the assumed uncertainly of the RCS temperature) of the thermal design temperature. At the present time 5.5% of the tubes in Surry Unit No. 2 have been plugged. We conclude that there is adequate margin provided in the level of steam generator tube plugging for Surry Unit No. 2. However, if more than a total of 7.5% of the tubes are plugged during forthcoming cycles, the licensee will be required to provide additional analyses.

Technical Specifications

The licensee has proposed (Reference 2) to modify the Technical Specifications to remove the limitations on the radial peaking factor $F_{xy}(z)$ for Units Nos. 1 and 2. The limitations were imposed pending review of the effect of rod bowing on the Westinghouse 15 x 15 fuel design. The staff review (Reference 6) of the convolution approval used in Reference 5 for the calculation of the uncertainty in the peaking factor $F_Q(z)$ finds this method to be acceptable. Therefore we find the proposed Technical Specification change to be acceptable.

The licensee has proposed for Unit No. 2 only (Reference 1) a Technical Specification change on the rod insertion limits for Cycle 3 operation. Those limits, as discussed in this report, are acceptable.

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

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

REFERENCES

1. Letter C. M. Stallings to B. C. Rusche dated March 11, 1976.

2. Letter C. M. Stallings to B. C. Rusche dated May 12, 1976.

3. Letter C. M. Stallings to B. C. Rusche dated May 14, 1976.

4. Westinghouse Topical Report, "Revised Clad Flattening Model", WCAP 8377 (Proprietary) and WCAP 8381 (Non-Proprietary), dated July, 1974.

5. J. R. Reavis, W. J. Leech, F. F. Cadek, S. Cerni, and J. M. Hellman, "Fuel Rod Bowing," Westinghouse Licensing Topical Report, WCAP 8691.

6. NRC Memorandum D. F. Ross to R. C. DeYoung and D. G. Eisenhut dated April 14, 1976.

7. Letter C. M. Stallings to K. R. Goller dated March 12, 1975.

8. Letter C. M. Stallings to K. R. Goller dated June 5, 1975.

9. Final Safety Analysis Report, Surry Power Station Units 1 and 2.

Dated:
May 25, 1976

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS NOS. 50-280 AND 50-281

VIRGINIA ELECTRIC AND 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. 20 to Facility Operating Licenses Nos. DPR-32 and DPR-37 issued to Virginia Electric and Power Company which revised Technical Specifications for operation of the Surry Power Station, Units Nos. 1 and 2, located in Surry County, Virginia. These amendments are effective as of the date of issuance.

These amendments relate to both the increase in the limiting nuclear enthalpy hot channel factor for Surry Units Nos. 1 and 2, and to the replacement of 81 of 157 fuel assemblies in the reactor core of Surry Unit No. 2 constituting refueling of the core for third cycle operation.

The applications for the amendments comply 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. Notices of Proposed Issuance of Amendments to Facility Operating Licenses in connection with this action were published in the FEDERAL REGISTER on April 1, 1976 (40 F.R. 14018 and 14019). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental statement, negative declaration or environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the two applications for amendments dated March 11, 1976, as supplemented May 12 and 14, 1976, (2) Amendments No. 20 to licenses Nos. DPR-32 and DPR-37, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C. and at the Swem Library, College of William and Mary, Williamsburg, Virginia.

A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 25th day of May, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch No. 4
Division of Operating Reactors