October 8, 1977

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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 Nos. 33 and 32 to Facility Operating Licenses Nos. DPR-32 and DPR-37 for the Surry Power Station Units Nos. 1 and 2, respectively. These amendments consist of changes to the Technical Specifications for each license in response to your application dated September 9, 1977, as supplemented September 30 and October 6, 1977.

These amendments relate to the replacement of 81 of the 157 fuel assemblies in the reactor core of Surry Unit No. 2 constituting refueling of the core for fourth cycle operation.

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

Sincerely,

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

Enclosures:
1. Amendment No. 33 to DPR-32
2. Amendment No. 32 to DPR-37
3. Safety Evaluation
4. Notice

cc w/enclosures:
See next page

OFFICE➤	ORB #4	ORB #4	OELD	ORB #4	
SURNAME➤	RIngram	MFairtile		RReid	
DATE➤	10/ /77	10/ /77	10/ /77	10/8/77	

NRC FORM 318 (9-76) NRCM 0240 ☆ U.S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

Virginia Electric & Power Company

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

Chief, Energy Systems
Analyses Branch (AW-459)
Office of Radiation Programs
U. S. Environmental Protection Agency
Room 645, East Tower
401 M Street, S.W.
Washington, D.C. 20460

U. S. Environmental Protection Agency
Region III Office
ATTN: EIS COORDINATOR
Curtis Building (Sixth Floor)
6th and Walnut Streets
Philadelphia, Pennsylvania 19106

Mr. James C. Dunstan
State Corporation Commission
Commonwealth of Virginia
Blandon Building
Richmond, Virginia 23209

cc w/enclosures and incoming
dtd: 9/9/77, 9/30/77, & 10/6/77
Commonwealth of Virginia
Council on the Environment
903 9th Street Office Building
Richmond, Virginia 23219

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

VIRGINIA ELECTRIC & POWER COMPANY

DOCKET NO. 50-280

SURRY POWER STATION UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 33
License No. DPR-32

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

   A. The application for amendment by Virginia Electric & Power Company (the licensee) dated September 9, 1977, as supplemented September 30 and October 6, 1977, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

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

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

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-32 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 33, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: October 8, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 33

FACILITY OPERATING LICENSE NO. DPR-32

DOCKET NO. 50-280

Revise the Technical Specifications as follows:

Remove Pages	Insert Pages
2.1-2	2.1-2
2.1-6	2.1-6
3.12-4a	3.12-4a
3.12-13	3.12-13
3.12-17	3.12-17
TS Table 3.12-1B	TS Table 3.12-1B
TS Figure 3.12-1B	TS Figure 3.12-1B

Changes on the revised pages are shown by marginal lines.

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 the rated power.

DELETED

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.

DELETED

References

1) FSAR Section 3.4

2) FSAR Section 3.3

3) FSAR Section 14.2

b. $F_Q(Z)$ shall be evaluated for normal (Condition I) operation of each unit by combining the measured values of $F_{xy}(Z)$ with the design Condition I axial peaking factor values, $F_Z(Z)$, as listed in TS Table 3.12-1A and TS Table 3.12-1B. For the purpose of this specification $F_{xy}(Z)$ shall be determined between 1.5 feet and 10.5 feet elevations of the core exclusive of grid strap locations. The measured values of $F_{xy}(Z)$ shall be increased by three percent to account for radial xenon redistribution effects associated with normal (Condition I) operation. In addition, the value of $F_{xy}(Z)$ shall be increased by two and one half percent to account for the predicted increase in the values of $F_{xy}(Z)$ during each effective full power month. This additional percent penalty on the values of $F_{xy}(Z)$ shall be applicable up to 9000 MWD/MTU burnup. The resulting $F_Q(Z)$ shall then be increased by three percent to account for manufacturing tolerances and further increased by five percent to account for measurement error. If the results of this evaluation predict that $F_Q(Z)$ could potentially violate its limiting values as established in Specification 3.12.B.1, either:

(1) the thermal power and high neutron flux trip setpoint shall be reduced at least 1% for each 1% of the potential violation (for the purpose of this specification, this power level shall be called $P_{THRESHOLD}$), or

(2) movable detector surveillance shall be required for operation when the reactor thermal power exceeds $P_{THRESHOLD}$. This surveillance shall be performed in accordance with the following:

(a) The normalized power distribution, $F_Q(Z)\Big|^{j}_{APDM}$, from thimble j at core elevation Z shall be measured utilizing at least two thimbles of the movable incore flux systen for 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 or 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 21.1 kw/ft for both units. 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.

DELETED

For normal (Condition I) operation, it may be necessary to perform surveillance to insure that the heat flux hot channel factor, $F_Q(Z)$, limit is met. To determine whether and at what power level surveillance is required, the potential (Condition I) values of $F_Q(Z)$ shall be evaluated monthly by combining the values of $F_{xy}(Z)$ obtained from the analysis of the monthly incore flux map with the flux of the design Condition I axial peaking factors, $F_Z(Z)$. The product of these shall be increased by five percent to account for measurement uncertainty, three percent to account for manufacturing tolerances, three percent to account for the effects of the radial redistribution of xenon during normal (Condition I) operation, and two and one half percent to account for the increase in the value of $F_{xy}(Z)$ as a function of burnup out to 9000 MWD/MTU burnup. $P_{THRESHOLD}$ is defined as the value of rated power minus one percent power for each percent of potential $F_Q(Z)$ violtion. If the potential values of $F_Q(Z)$ for normal (Condition I) operation are greater than the $F_Q(Z)$ limit, then surveillance shall be performed at all power levels above $P_{THRESHOLD}$.

Movable incore instrumentation thimbles for surveillance are selected so that the measurements are representative of the peak core power density. By limiting the core average axial power distribution, the total power peaking factor $F_Q(Z)$ can be limited since all other components remain relatively fixed. The remaining part of the total power peaking factor can be derived based on incore measurements, i.e., an effective radial peaking factor, $\overline{R}$, can be determined as the ratio of the total peaking

SURRY UNIT 2

CYCLE 4

CORE HEIGHT (FEET)	$F_Z(Z)$
1.5	1.279
2.0	1.308
2.5	1.313
3.0	1.360
3.5	1.375
4.0	1.400
4.5	1.404
5.0	1.402
5.5	1.390
6.0	1.366
6.5	1.330
7.0	1.302
7.5	1.277
8.0	1.237
8.5	1.221
9.0	1.248
9.5	1.269
10.0	1.265
10.5	1.202

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

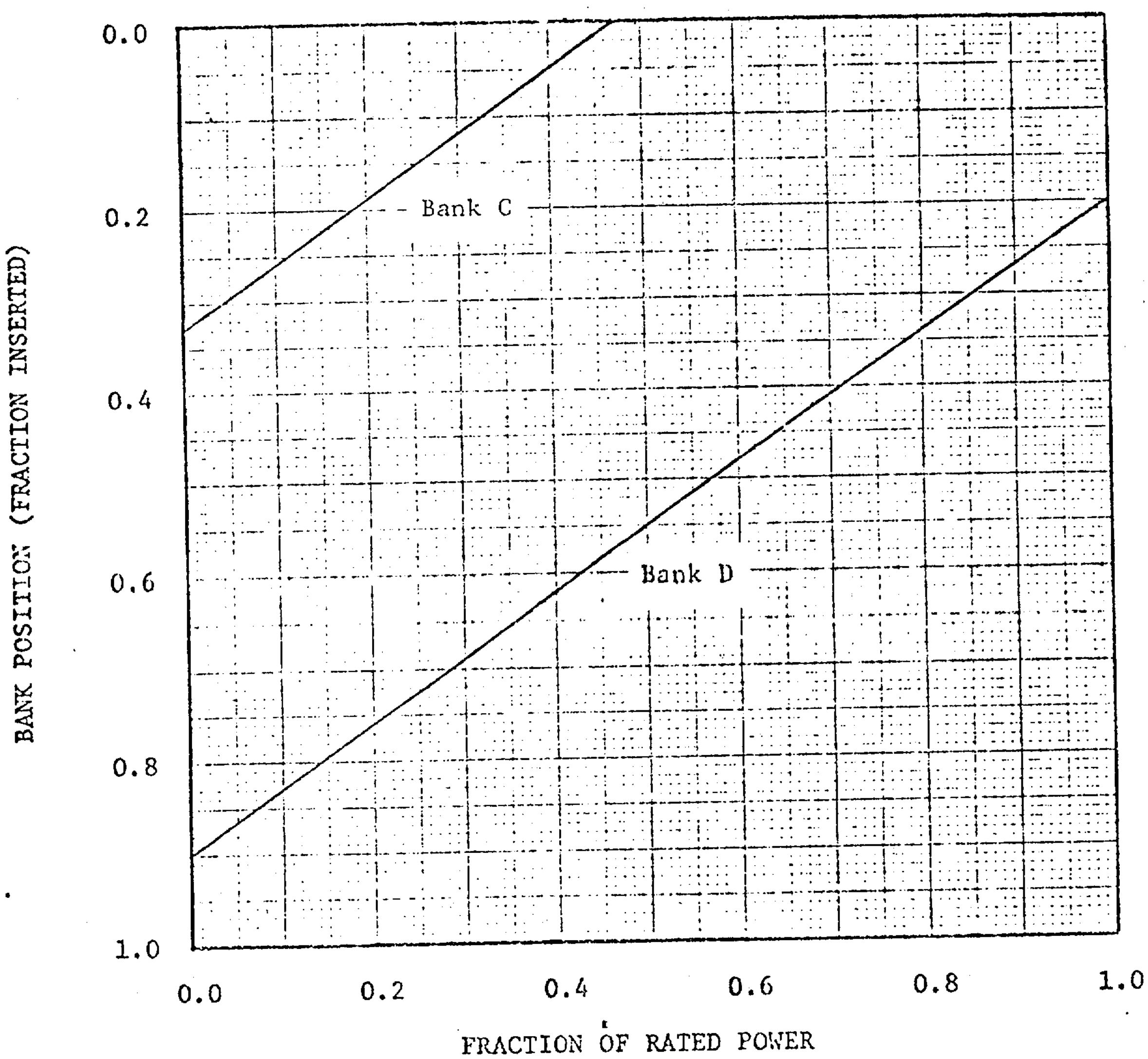

FIGURE 3.12-1B CONTROL BANK INSERTION LIMITS FOR NORMAL 3 LOOP OPERATION - UNIT 2

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

VIRGINIA ELECTRIC & POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 32
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 9, 1977, as supplemented September 30 and October 6, 1977, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

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

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

   E. The issuance of this amendment is in accordance with 10 CPR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 32, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: October 8, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 32

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
2.1-6	2.1-6
3.12-4a	3.12-4a
3.12-13	3.12-13
3.12-17	3.12-17
TS Table 3.12-1B	TS Table 3.12-1B
TS Figure 3.12-1B	TS Figure 3.12-1B

Changes on the revised pages are shown by marginal lines.

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 the rated power.

DELETED

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.

DELETED

References

1) FSAR Section 3.4

2) FSAR Section 3.3

3) FSAR Section 14.2

b. $F_Q(Z)$ shall be evaluated for normal (Condition I) operation of each unit by combining the measured values of $F_{xy}(Z)$ with the design Condition I axial peaking factor values, $F_Z(Z)$, as listed in TS Table 3.12-1A and TS Table 3.12-1B. For the purpose of this specification $F_{xy}(Z)$ shall be determined between 1.5 feet and 10.5 feet elevations of the core exclusive of grid strap locations. The measured values of $F_{xy}(Z)$ shall be increased by three percent to account for radial xenon redistribution effects associated with normal (Condition I) operation. In addition, the value of $F_{xy}(Z)$ shall be increased by two and one half percent to account for the predicted increase in the values of $F_{xy}(Z)$ during each effective full power month. This additional percent penalty on the values of $F_{xy}(Z)$ shall be applicable up to 9000 MWD/MTU burnup. The resulting $F_Q(Z)$ shall then be increased by three percent to account for manufacturing tolerances and further increased by five percent to account for measurement error. If the results of this evaluation predict that $F_Q(Z)$ could potentially violate its limiting values as established in Specification 3.12.B.1, either:

(1) the thermal power and high neutron flux trip setpoint shall be reduced at least 1% for each 1% of the potential violation (for the purpose of this specification, this power level shall be called $P_{THRESHOLD}$), or

(2) movable detector surveillance shall be required for operation when the reactor thermal power exceeds $P_{THRESHOLD}$. This surveillance shall be performed in accordance with the following:

(a) The normalized power distribution, $F_Q(Z)\Big|^{j}_{APDM}$, from thimble j at core elevation Z shall be measured utilizing at least two thimbles of the movable incore flux systen for 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 or 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 21.1 kw/ft for both units. 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.

DELETED

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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

SURRY UNIT 2

CYCLE 4

CORE HEIGHT (FEET)	$F_Z(Z)$
1.5	1.279
2.0	1.308
2.5	1.313
3.0	1.360
3.5	1.375
4.0	1.400
4.5	1.404
5.0	1.402
5.5	1.390
6.0	1.366
6.5	1.330
7.0	1.302
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8.5	1.221
9.0	1.248
9.5	1.269
10.0	1.265
10.5	1.202

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

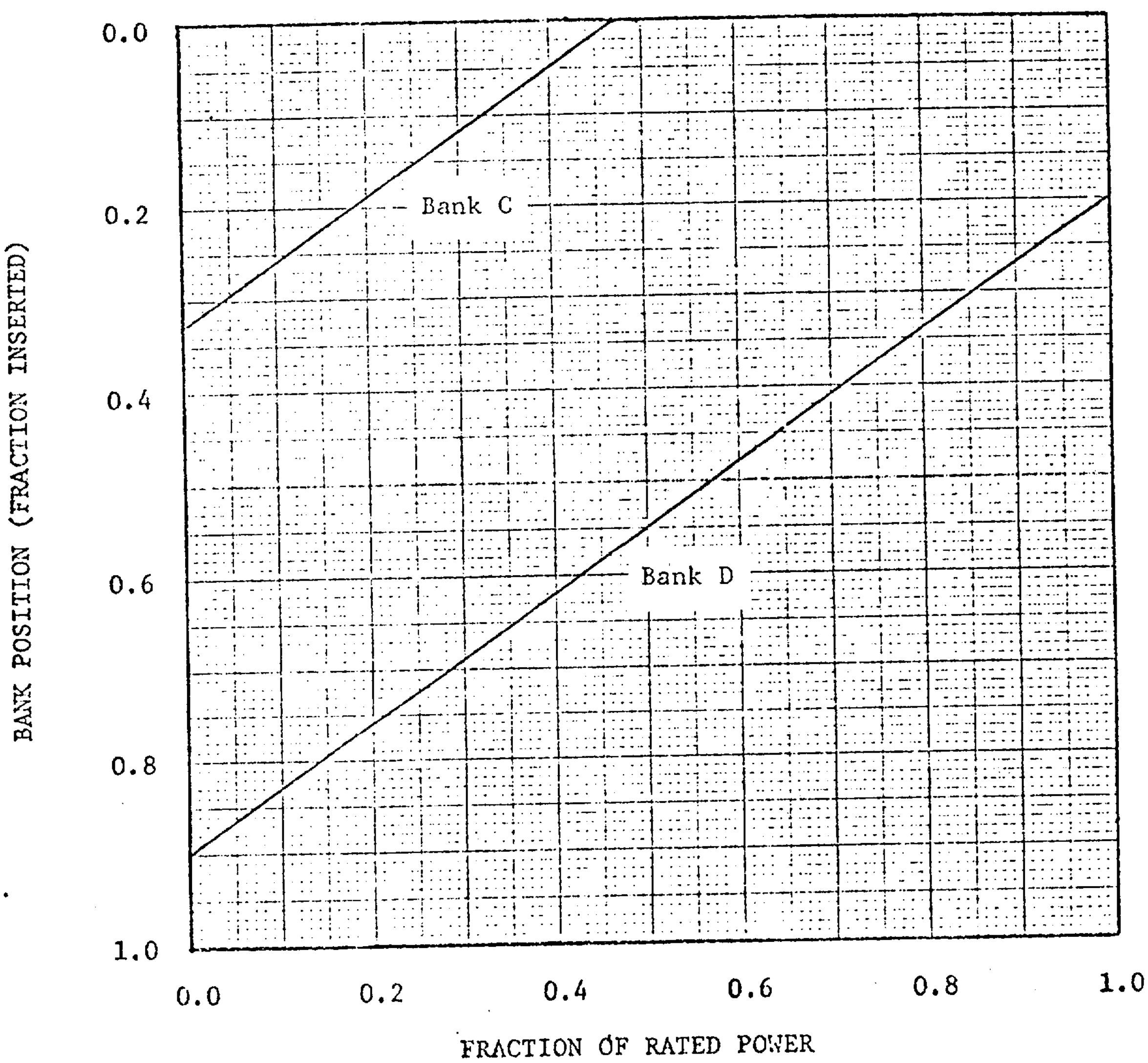

FIGURE 3.12-1B CONTROL BANK INSERTION LIMITS FOR NORMAL 3 LOOP OPERATION - UNIT 2

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENTS NOS. 33 AND 32 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 the application dated September 9, 1977,(1) as supplemented September 30, 1977(2) and October 6, 1977, 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 replace 81 of 157 fuel assemblies in the reactor core of Surry Unit No. 2 constituting refueling of the core for fourth cycle operation. This refueling consists of the replacement of 81 burned fuel assemblies by 80 fresh assemblies and one once-burned assembly. Cycle 4 will nominally extend 18 months commencing in mid-October, 1977 producing approximately 13,200 MWD/MTU. Possible operation at reduced power beyond this burnup (coastdown mode) was considered with allowance for a total cycle burnup of approximately 14,200 MWD/MTU.

Analyses performed for the Cycle 4 reload core design were based on the following assumptions.

1) Cycle 3 operation is terminated after 9200 (+300, -1060) MWD/MTU

2) Cycle 4 operation will not exceed 14,200 MWD/MTU

The licensee has proposed the following changes to the Technical Specifications which are assumed in the analysis of Cycle 4:

1) Delete requirement restricting fuel residence time.
2) Increase fuel rod peak linear power density to 21.1 KW/ft for Unit 2.
3) Revise axial peaking factor envelope, $F_z(z)$, to correspond with revised constant axial offset control analysis.
4) Revise power dependent control rod insertion limits for Unit 2 to show only currently applicable values.
5) Revise prescription for computing uncertainties and ajustments applied to the peak linear heat rate so that the same prescription applies to both Units 1 and 2.

EVALUATION

FUEL MECHANICAL DESIGN

The mechanical design of the fresh region 6 fuel assemblies is identical to the region 5 fuel loaded in the last core reload except for a modification to the top nozzle. The region 6 fuel has double leaf hold-down nozzle springs instead of the previously used single leaf springs. Double leaf springs are superior to single leaf springs in their shipping, storage, and handling characteristics. Either single or double leaf springs provide adequate hold-down force for reactor operation. We find the mechanical design acceptable.

REACTOR DESIGN

Core Loading

The Cycle 4 core loading will consist of the following regions:

Region	Number of Assemblies	Density % Theoretical	Burnup BOC 4 MWD/MTU	Burnup EOC 4 MWD/MTU
1	1	93.8	16,500	28,500
4B	50	94.6	18,200	32,600
4B*	2	94.6	18,200	32,600
5	24	95.0	7,000	22,700
6A	28	94.5	0	15,600
6B	52	95.0	0	12,600

With the exception of the single region 1 assembly, all assemblies are pre-pressurized, high pellet density assemblies.

With the exception of region 4B*, all assemblies are 15x15. Region 4B* consists of two 17x17 demonstration assemblies. Both the region 1 and region 4B* assemblies are predicted to sustain power densities that are non-limiting relative to other assemblies in Cycle 4.

The core will contain 848 fresh borosilicate burnable poison rods and 48 depleted burnable poison rods. The burnable poison rods are used to control local and gross core reactivity, control the moderator temperature coefficient and tailor radial power distributions.

Kinetics Characteristics

The Cycle 4 core is predicted to have the following kinetics characteristics:

Parameter	Cycle 3 Current Limit	Cycle 4
Moderator Temperature Coefficient Range (PCM/^{o}F)	BOC: +3.0 EOC: -35.0	BOC: 0.0 EOC: -35.0
Doppler-Only Power Coefficient, Zero to Full Power, Least Negative (PCM/Percent Power)	HZP: -11.5 HFP: -6.0	HZP: -12.2 HFP: -8.4
Minimum Delayed Neutron Fraction (Percent)	BOC: 0.55 EOC: 0.48	BOC: 0.58 EOC: 0.48
Prompt Neutron Lifetime (μ sec)	26	19
Maximum Positive Reactivity Insertion Rate from Subcritical (PCM/sec)	65	66
Most Negative Doppler Temperature Coefficient (PCM/^{o}F)	-1.6	-2.4

The Cycle 4 values are not within the bounds of the current analyses for the last three entries. A discussion of the effect of changes in the value of these parameters is given in the section on ACCIDENT ANALYSIS.

Shutdown Margin

The hot full power shutdown margin is predicted to be 4.80% $\Delta\rho$ at Beginning of Cycle and 3.40% $\Delta\rho$ at End of Cycle compared to an assumed shutdown margin of 1.77% $\Delta\rho$ in the steamline break analyses.

Basic Design Parameters

The basic design parameters for Cycle 4 are a core average power of 2441 MWT, a system pressure of 2250 PSIA, a reactor average temperature of 574.4^{o}F, a fuel rod average linear power density of 6.2 KW/ft, and a core flow of 265,000 GPM.

Thermal and Hydraulic Design

In regard to the maximum value of internal fuel rod pressure, the currently approved design criteria require that internal fuel rod pressure never exceed reactor coolant system operating pressure. The currently approved generic Westinghouse model which didnot consider enhanced fission gas release at higher exposures shows compliance with the above design criteria. A revised Westinghouse model which considers enhanced fission gas release is under current staff review. The licensee states that by using the revised model the fuel also meets the design criteria. Furthermore, the staff has recently approved a design criterion that permits internal W fuel pressure to exceed system pressure. On these bases the staff finds the fuel rod internal pressure throughout cycle 4 to be acceptable.

The Departure from Nuclear Boiling (DNB), fuel temperature, and internal pressure evaluations for the cycle 4 reload core were performed by VEPCO using the same models as were previously used. The present DNB core limits were found to be adequate and conservative. The potential effects of rod bow on DNB has been accommodated in accordance with the interim safety evaluation report issued by the NRC on March 22, 1977.

Normal Operation and Control

Operation of Unit 2 using the Westinghouse Constant Axial Offset Control Strategy, (CAOC)[3] during cycle 4 with a (+6%, -9%) flux difference target band has been supported by recomputation of anticipated axial power distributions and corresponding axial peaking factors. The axial peaking factors $F_z(z)$ are computed as a function of axial elevation and burnup. The maximum bounding values of $F_z(z)$, called the $F_z(z)$ envelope, are incorporated in the proposed Technical Specifications, TS Table 3.12-1B. This envelope represents the maximum anticipated values of $F_z(z)$ when the flux difference is maintained within the CAOC target band.

Operation within the $F_z(z)$ envelope, will ensure that Departure from Nucleate Boiling Ratio (DNBR) will be greater than 1.3 during steady state, load follow, and upset conditions.

Accident Analysis

The licensee has reviewed all postulated accidents which were reported in the Final Safety Analysis Report and states that those transients and accidents which were found to be potentially affected were reanalyzed. These were identified by the licensee to be steamline break and control rod withdrawal from subcritical conditions. The licensee states that all other transients and accidents are bounded by the reference analyses and on this basis we accept their conclusion that those transients and accidents need not be reanalyzed.

The reanalysis of the Steamline Break (SLB) accident was required by a predicted increase of the limiting enthalpy rise, $F_{\Delta H}$, during the transient. DNBR calculations were performed by the licensee using these revised values of $F_{\Delta H}$ and statepoints (pressure, temperature, average heat flux) obtained from previous analyses submitted for Unit 1, cycle 4. The licensee states that the minimum predicted DNBR is greater than 1.30.

SLB analysis methods are currently being generically reviewed by the NRC staff and the analysis for this cycle has not been reviewed by the staff.

The hypothetical steamline break is a design bases event for which limited clad failure is permitted. Staff scoping calculations show that approximately as much as 7% of the fuel rods could be failed without exceeding the site boundary dose rate limits. The relative power density predicted during the course of steamline break with all control rods except the most reactive rod inserted is highly non-uniform. The predicted minimum DNBR during the transient would occur near the region of the stuck rod and be restricted to a small region of the core. Even if departure from nucleate boiling were to occur, and even if clad failure were to occur, the staff is of the opinion that less than 7% of the fuel rods would fail and hence site boundary dose rate limits will not be violated.

On this basis operation of unit 2 during cycle 4 is deemed acceptable by the staff.

The reanlaysis of the rod withdrawal from subcritical conditions was due to an increase of the predicted reactivity insertion worth of withdrawal of two RCCA control banks moving together in their highest worth region. The base analysis was performed by VEPCO using a maximum positive reactivity insertion rate from subcritical of 65 pcm/sec. The maximum predicted value for cycle 4 is 66 pcm/sec. The reanalysis presumes a withdrawal rate of 75 pcm/sec. The reanalysis results in an increase of 17% of the predicted heat flux obtained during the transient. The licensee states that the conclusion presented in the FSAR that DNBR remains above 1.30 is still applicable. It is noted that substantial margin to a minimum DNBR of 1.30 exists for this hypothetical accident. On this basis the analysis is acceptable. The value of the bounding most negative Doppler temperature coefficient considered in the safety analysis has been increased from -1.6 pcm/^{o}F to -2.4 pcm/^{o}F. The licensee states that this change is primarily a result of a more conservative method of predicting the Doppler coefficient adopted by Westinghouse. The assumed more negative value adversely affects cooldown transients such as idle loop startup and steam line break. The licensee states that these transients were reanalyzed with the more negative Doppler temperature coefficient and it was concluded that safety limits would not be violated. The effect of the more negative Doppler temperature coefficient is in part mitigated by the more negative values of Doppler power coefficient applicable to cycle 4 which were used in the reanalysis. We conclude that since the most limiting cooldown transient, idle loop startup, is administratively preclude in the Technical Specifications that the cycle 4 parameters are acceptable.

The predicted prompt neutron lifetime for cycle 4 is less than that used in previous safety analyses. The effect of the reduced value of the prompt neutron lifetime is an increase in the conservatism of the ejected rod safety analysis. In transients other than the ejected rod, the prompt neutron lifetime plays a role only in its contribution to the effective neutron lifetime which is dominated by the delayed neturon contribution which remains invarient. Thus these transients are insensitive to prompt neutron lifetime. We find the change in prompt neutron lifetime does not change the acceptability of the reference analyses.

Technical Specification Changes

Fuel Residence Time (T.S. Page 2.1-2, T.S. Page 2.1-6)

Technical Specification restrictions on fuel residence time have been deleted. The restrictions were originally imposed to guarantee that fuel did not suffer sufficient exposure to experience fuel densification and clad weakening to the point of clad collapse. The removal of the restrictions are based on the use of high density pre-pressurized fuel in all except one assembly in cycle 4 in both units and in all assemblies in subsequent cycles. Such fuel can tolerate considerably more than three full cycles of exposure and thus requires no Technical Specification restrictions on residence time. (See page 5 of SE).

The single region 1 assembly (core location H-8) which will reside in the core during cycle 4 is limiting with respect to clad collapse. It is a once-burned assembly. The Westinghouse Evaluation Model predicts that this assembly can tolerate an additional 15,399 EFPH before clad collaspe sets in. This assembly is predicted not to exceed 10,045 EFPH during cycle 4 operation, hence substantial margins to predicted time of clad collapse exist obviating the need for explicit Technical Specification restriction.

Peak Linear Power Density (PLPD)(T.S. Page 3.12-13)

The PLPD limits are set as the design basis for fuel performance related to fission gas release, pellet temperature, and cladding material properties. Previously, the design basis PLPD's were 21.1 KW/FT and 20.4 KW/FT for Units 1 and 2, respectively. The proposed design basis and Technical Specifications relax these limits to 21.1 KW/FT for both units. This change is based on the Unit 2 cycle 4 fuel inventory of 156 pre-pressurized high pellet density fuel assemblies and one once-burned region 1 assembly from cycle 1. The pre-pressurized high density assemblies can all tolerate 21.1 KW/FT.

The single region 1 assembly in location H-8 generates relatively low power;

Relative power density: (assy) =0.791, (rod)=0.801, All Rods Out, 150 MWD/MTU

Relative power density: (assy) =0.561, (rod)=0.641, bank D-IN, 150 MTU/MTU

and is thus expected to have a PLPD well below 20.4 KW/FT for any anticipated mode of operation. Thus, the single restriction that monitored PLPD be limited to 21.1 KW/FT guarantees the integrity of all assemblies.

Control Rod Insertion Limits (T.S. Figure 3.12-1B)

The Power Dependent Insertion Limits (PDIL) proposed for Unit 2 cycle 4 are identical to those used for Unit 2 cycle 3. They were generated for Unit 2 cycle 3 assuming a cycle 2 burnup greater than 8700 MWD/MTU. The considerations used to determine these limits are the following:

1. Shutdown Margin
2. Operating statepoints assumed in rod ejection.
3. Values of limiting enthalpy rise.
4. Consequences of misaligned rod.
5. Consequences of uncontrolled rod withdrawal.

The PDIL was reviewed by the licensee to ensure that these criteria were met. It was concluded by the licensee that inputs to the cycle 3 PDIL analysis bounded the corresponding input parameters for cycle 4, and that the cycle 3 PDIL is adequate for cycle 4.

The staff concludes that since the licensee's analysis falls within the bounds of previously approved analyses the PDIL is acceptable.

Envelope Of Axial Peaking Factor (T.S. Table 3.12-13)

The $F_Z(Z)$ envelope is determined by examination of (18 cases) the CAOC study. The $F_Z(Z)$ envelope value is the highest value of $F_Z(Z)$ expected (in any of the 18 cases) at any time in life. For the purpose of predicting the potentially largest value of F_Q, the F_Z (Z) envelope is multiplied by the measured Fxy(Z) from the monthly incore map, and this product, multiplied by adjustments and uncertainties, is taken to be the potentially largest value of F_q in the subsequent month.

Changes to Make Units 1 and 2 Consistent (T.S. Page 3.12-4a, T.S. Page 3.12-7)

By letter of September 30, 1977, the licensee proposed four changes to the Technical Specifications which would result in making uncertainties and adjustments applied to measured values of the peak linear heat rate identical in Units 1 and 2. These changes do not impact any analyses performed or conflict with any other portions of the Technical Specifications and are acceptable.

ECCS Performance

On March 4, 1977, the licensee submitted information on LOCA reanalysis with 20% of the steam generator tubes plugged.[(5)] We have reviewed the current reload and have determined that analysis parameters assumed in the March 4, 1977, analysis bound the Unit 2, cycle 4 parameters. The steam generator tube plugging level will be less than 20% for the upcoming cycle. The analysis for the worst case break (C_D=0.4 DECLG) with 20%

of the steam generator tubes plugged indicates a peak clad temperature of 2186^{o}F, a local metal-water reaction of 7.9 percent and a total metal-water reaction of less than 0.3 percent. We conclude that operation of Surry Unit 2, for Cycle 4, within the constraints of the Technical Specifications is acceptable and in conformance with paragraph 50.46 of 10 CFR 50.

Environmental Conclusions

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

Conclusion

The licensee's reload submittal stated that some of the parameters listed in Table 15-2 of the Standard Format (Reg Guide 1.70, Rev. 2) were outside of the range used in the reference safety analyses. As a result the submittal indicated that certain transients such as bank withdrawal from subcriticality, and steam line break have changed and reanalyzed, and are intended to become the new reference safety anlayses for future cycles. However, detailed results of these analyses as described in Section 15 of the Standard Format were not submitted.

Unless this information and the analyses and supporting information, as described in Section 4 of the Standard Format, are submitted, the analyses submitted for Cycle 4 will not be considered acceptable as a reference safety analysis for future cycles.

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

Dated: October 8, 1977

REFERENCES

1. Letter from C. M. Stallings (VEPCO) to E. G. Case (NRC), September 9, 1977.

2. Letter from C. M. Stallings (VEPCO) to E. G. Case (NRC), September 30, 1977.

3. WCAP-8385, Topical Report - Power Distribution Control and Load Follow Procedures, Westinghouse Electric Corporation, September 1974.

4. WCAP-859., Topical Report - Axial Power Distribution Monitoring System, Westinghouse Electric Corporation, August 1975.

5. Letter from C. M. Stallings (VEPCO) to B. C. Rusche (NRC), March 4, 1977.

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

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 33 and 32 to Facility Operating Licenses Nos. DPR-32 and DPR-37, issued to Virginia Electric & Power Company (the licensee), which revised Technical Specifications for operation of the Surry Power Stations, Units Nos. 1 and 2 (the facilities) located in Surry County, Virginia. The amendments are effective as of the date of issuance.

These amendments relate to the replacement of 81 of the 157 fuel assemblies in the reactor core of Surry Unit No. 2 constituting refueling of the core for fourth cycle operation..

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

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

For further details with respect to this action, see (1) the application for amendments dated September 9, 1977, as supplemented September 30 and October 6, 1977, (2) Amendments Nos. 33 and 32 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 8th day of October 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

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