JUNE 3 0 1978

Docket Nos. 50-280
and ~~50-281~~

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

Gentlemen:

The Commission has issued the enclosed Amendment Nos. 42 and 41 to Facility Operating License Nos. DPR-32 and DPR-37 for the Surry Power Station Unit Nos. 1 and 2. These amendments consist of changes to the Technical Specifications common to each license in response to your application dated March 15, 1978 as supplemented May 11, 1978.

These amendments, which affect only Surry Unit No. 1, relate to Cycle 5 operation of that reactor. Also as provided for in 10 CFR 50.12 of the Code of Federal Regulations we have granted the enclosed specific exemption from the requirements of 10 CFR 50.46(a)(1) for Surry Unit No. 1. We have determined that this exemption is authorized by law, will not endanger life or property or the common defense and security and is otherwise in the public interest. This specific exemption is limited to the time period necessary to complete computer calculations, acceptable to the NRC staff, that have been corrected for the errors described in Section II of the exemption.

Copies of the Safety Evaluations related to the license amendments, and the exemption, the Notice of Issuance of the license amendment and the granting of the Exemption are also enclosed. The Notice and the Exemption are being forwarded to the Office of the Federal Register for publication.

Sincerely,

Original signed by

A. Schwencer, Chief
Operating Reactors Brach #1
Division of Operating Reactors

Const. I

~~NRR~~
ECase
6/30/78

Enclosures and cc's:
See next page

x27433:tsb
6/23/78

OFFICE ➤	ORB#1	RSB	OELD	ORB#1	AD/DOR	DIR/DOR
SURNAME ➤	GZech	~~RBaer~~ PCheck		ASchwencer	DEisenhut	VStello
DATE ➤	6/ /78	6/23/78	6/28/78	6/30/78	6/28/78	6/29/78

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

June 30, 1978

Docket Nos. 50-280
and 50-281

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

Gentlemen:

The Commission has issued the enclosed Amendment Nos. 42 and 41 to Facility Operating License Nos. DPR-32 and DPR-37 for the Surry Power Station Unit Nos. 1 and 2. These amendments consist of changes to the Technical Specifications common to each license in response to your application dated March 15, 1978 as supplemented May 11, 1978.

These amendments, which affect only Surry Unit No. 1, relate to Cycle 5 operation of that reactor. Also as provided for in 10 CFR 50.12 of the Code of Federal Regulations we have granted the enclosed specific exemption from the requirements of 10 CFR 50.46(a)(1) for Surry Unit No. 1. We have determined that this exemption is authorized by law, will not endanger life or property or the common defense and security and is otherwise in the public interest. This specific exemption is limited to the time period necessary to complete computer calculations, acceptable to the NRC staff, that have been corrected for the errors described in Section II of the exemption.

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Sincerely,

A. Schwencer

A. Schwencer, Chief
Operating Reactors Brach #1
Division of Operating Reactors

Enclosures and cc's:
See next page

Enclosures:
1. Amendment No. 42 to DPR-32
2. Amendment No. 41 to DPR-37
3. Safety Evaluation
4. Exemption w/associated
   Safety Evaluation
5. Notice of Issuance

cc w/enclosures:
Mr. Michael W. Maupin
Hunton & Williams
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Richmond, Virginia 23213

Swem Library
College of William & Mary
Williamsburg, Virginia 23185

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

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Council on the Environment
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Commonwealth of Virginia
State Corporation Commission
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Richmond, Virginia 23209

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Analyses Branch (AW-459)
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US Environmental Protection Agency
Region III Office
ATTN: EIS COORDINATOR
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6th and Walnut Streets
Philadelphia, Pennsylvania 19106

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. 42
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 March 15, 1978, as supplemented May 11, 1978, 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 License No. DPR-32 is hereby amended to read as follows:

   B <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 42 , 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

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: June 30, 1978

ATTACHMENT TO LICENSE AMENDMENT NOS. 42 AND 41

FACILITY OPERATING LICENSE NOS. DPR-32 AND DPR-37

DOCKET NOS. 50-280 AND 50-281

Revise the Technical Specifications as follows:

Remove Pages	Insert Pages
3.12-5	3.12-5
3.12-13	3.12-13
3.12-20	3.12-20
TS Table 3.12-1A	TS Table 3.12-1A
TS Figure 3.12-1A	TS Figure 3.12-1A

Changes on the revised pages are shown by marginal lines.

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 ±5% band about the target flux difference (defines the target band on axial flux difference).

   a. At a power level greater than 90 percent of rated power, if the indicated axial flux difference deviates from its target band, 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 target band for a maximum of one hour (cumulative) in any 24 hour period provided the flux difference does not exceed an envelope bounded

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. Belcw 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 5C% steady state power does not result in exceeding core limits.

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

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

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

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

The technical specifications on power distribution control given in 3.12.B.4 together with the surveillance requirements given in 3.12.B.2.b assure that the Limiting Condition for Operation for the heat flux hot channel factor is met.

The target (or reference) value of flux difference is determined as follows. At any time that equilibrium xenon conditions have been established, the indicated flux difference is noted with the full length rod control bank more than 190 steps withdrawn (i.e. normal full power operating position appropriate for the time in life, usually withdrawn farther as burnup proceeds). This value, divided by the fraction of full power at which the core was operating is the full power value of the target flux difference. Values for all other core power levels are obtained by multiplying the full power value by the fractional power. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and indicated deviation of ±5% ΔI are permitted from the indicated reference value. During periods where extensive load following is required, it may be impractical to establish the required core conditions for measuring the target flux difference every month. For this reason, the specification provides two methods for updating the target flux difference.

Strict control of the flux difference (and rod position) is not as necessary during part power operation. This is because xenon distribution control at part power is not as significant as the control at full

SURRY UNIT 1

CYCLE 5

CORE HEIGHT (FEET)	$F_Z(Z)$
1.5	1.272
2.0	1.280
2.5	1.252
3.0	1.247
3.5	1.264
4.0	1.280
4.5	1.289
5.0	1.294
5.5	1.294
6.0	1.287
6.5	1.283
7.0	1.282
7.5	1.272
8.0	1.255
8.5	1.227
9.0	1.185
9.5	1.189
10.0	1.202
10.5	1.180

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

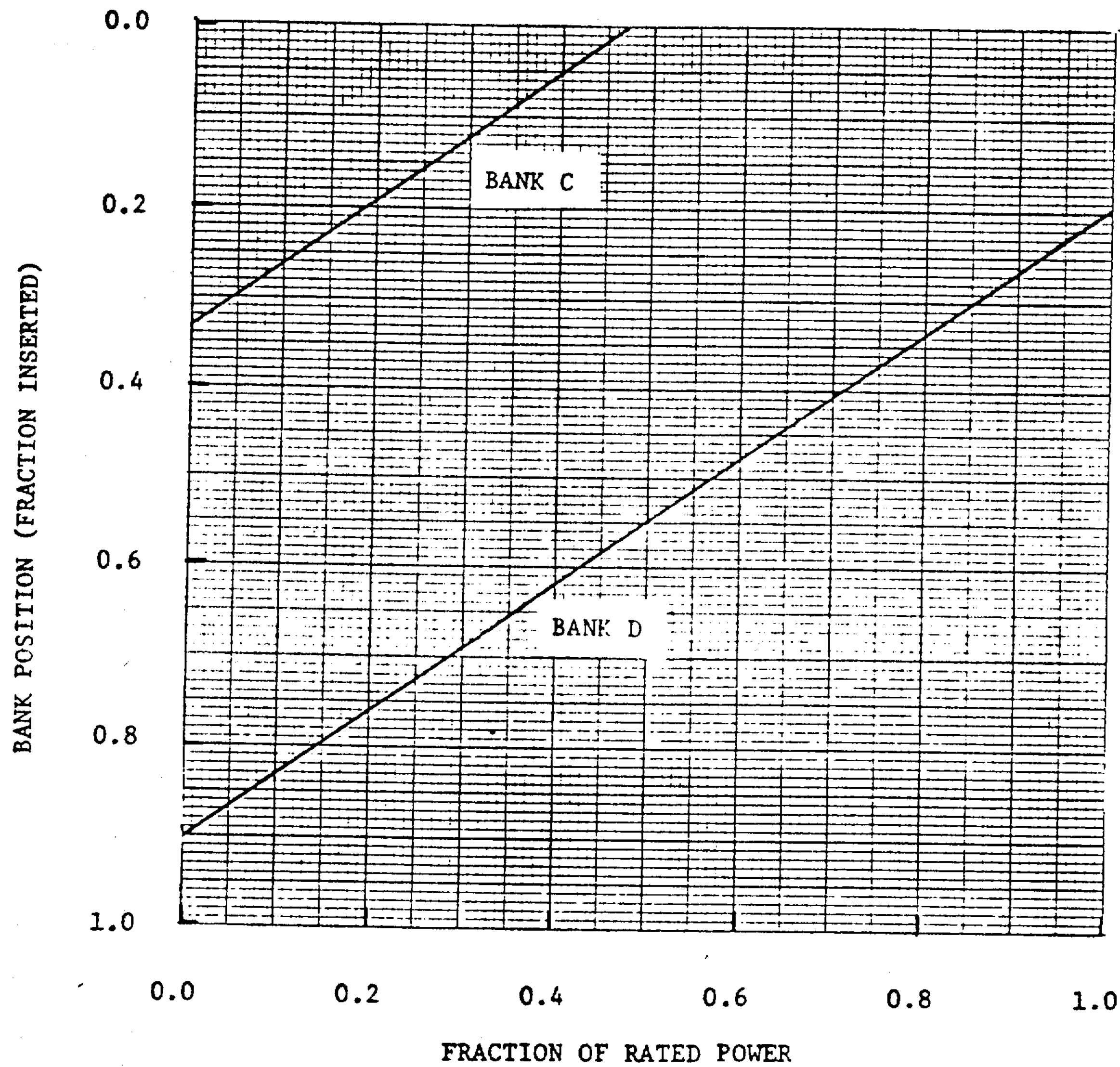

FIGURE 3.12-1A CONTROL BANK INSERTION LIMITS FOR 3-LOOP NORMAL OPERATION-UNIT 1

VIRGINIA ELECTRIC & POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 41
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 March 15, 1978, as supplemented May 11, 1978, 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 License No. DPR-37 is hereby amended to read as follows:

   B <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 41, 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

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 30, 1978

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENTS NOS. 42 AND 41 TO LICENSE NOS. DPR-32 AND DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION UNIT NOS. 1 AND 2

DOCKET NOS. 50-280 AND 50-281

Introduction

By application dated March 15, 1978[1], as supplemented May 11, 1978[2], Virginia Electric and Power Company (VEPCO or the licensee) proposed to change the Technical Specifications for the Surry Power Station Unit Nos. 1 and 2 to permit Cycle 5 operation of Unit No. 1.

Discussion

The refueling for Cycle 5 consists of the replacement of 73 burned fuel assemblies by 64 fresh assemblies and nine previously burned assemblies. The previously burned assemblies are: one which was last irradiated in Cycle 2 of Surry Unit No. 2 and eight which were last irradiated in Cycle 3 of Surry Unit No. 1.

Cycle 5 will nominally extend 18 months commencing about July 1978 and will produce approximately 13,700 megawatt days per metric ton of uranium (MWD/MTU). Because of this long cycle, each fresh fuel assembly will contain 0, 8, 12, 16 or 20 fresh borosilicate burnable poison rods, depending on location. A total of 768 borosilicate poison rods will be used. Possible operation at reduced power beyond this burnup (coastdown mode) was considered with allowance for a total cycle burnup of approximately 14,700 MWD/MTU.

Analyses performed for the Cycle 5 reload core design were based on the following assumptions:

1) Cycle 4 operation is terminated after 12,400 (+1000,-1000) MWD/MTU

2) Cycle 5 operation will not exceed 14,700 MWD/MTU

VEPCO has proposed the following changes to the Technical Specifications for Unit 1 which are assumed in the analysis of Cycle 5.

1) Reduce the axial flux difference operating band from +6 to -9% to +5%.

2) Revise the table of Design Condition Axial Peaking Factors, $F_Z(Z)$ vs Core Height.

3) Revise the power dependent control rod insertion limits.

Evaluation

Fuel Mechanical Design

The mechanical design of the fresh fuel assemblies (Region 7) is identical to the Region 6 fuel loaded in the last core reload except for a modification to the top nozzle. The Region 7 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 because they provide increased hold-down force margin. Either single or double leaf springs provide adequate hold-down force for reactor operation. Double leaf springs were used in all 68 fresh assemblies of the Surry Unit 2 reload in October 1977. We find the mechanical design acceptable.

Clad flattening will not occur during Cycle 5. Clad flattening time is predicted to be greater than 30,000 effective full power hours (EFPH) for all fuel regions being irradiated during Cycle 5 using the approved Westinghouse Evaluation Model[3]. The most limiting region, Region 6B, currently has an accumulated fuel residence time of 9439 EFPH. Therefore, Region 6B could be exposed for over 20,000 additional EFPH before clad flattening would be predicted to occur. Since Cycle 5 operation will not exceed approximately 10,300 EFPH, a significant safety margin will exist.

The revised Westinghouse fuel performance analysis code, PAD 3.3[4], which models enhanced fission gas release at high burnups, was used to show conformance to the design basis. This code is considered by Westinghouse to provide a conservative estimate of fuel performance. We are currently reviewing PAD 3.3 and expect to find it acceptable with modifications.

VEPCO states that the minimum burnup at which the fuel diametral gap is predicted to increase is 48,000 MWD/MTU rod average burnup. Exposure to the point where the gap begins to open up is considered acceptable in the approved Westinghouse revised fuel internal pressure criteria report[5]. The licensee estimated the peak rod average burnup for the Surry 1 Cycle 5 operation to be 35,000 MWD/MTU. Thus, there is a substantial margin to accommodate modifications to PAD 3.3 which we may require. From the current review, we conclude this margin is sufficient.

Thermal and Hydraulic Design

VEPCO states that the DNB evaluation for the Cycle 5 reload was performed using the same models as were previously used. The DNB limits defined by the current Technical Specification safety limit curves were found to be adequate and conservative for Cycle 5 operation. The potential effect of rod bow on DNB was accommodated in accordance with our interim safety evaluation report[6]. We, therefore, conclude that the thermal and hydraulic design for Cycle 5 is acceptable.

Power Distribution Control and Monitoring

VEPCO submitted the results of an analysis of the maximum axial peaking factors $F_Z(Z)$ as a function of core axial elevation expected for normal operation of the power plant during Cycle 5 operation. The analysis used Westinghouse constant axial offset control (CAOC) strategy. It included load follow maneuvers considered to produce bounding values of the axial peaking factor. The analysis assumed a $\pm 5\%$ operating band of flux difference rather than the +6 to -9% band used for Cycle 4. The tighter flux difference is more restrictive in terms of operating convenience, but produces slightly lower $F_Z(Z)$ values. These are used in the Surry Technical Specifications to determine the threshold power level at which surveillance of $F_Z(Z)$ is initiated in order to ensure that the peaking factors assumed as input for the LOCA analysis are not exceeded in normal operation of the power plant. The reduced $F_Z(Z)$ values lead to a higher power level at which the surveillance is required.

There are no changes to the Technical Specifications for power distribution control and monitoring for Cycle 5 operation other than the revised $F_Z(Z)$ curve produced by the CAOC analysis with the tighter ($\pm 5\%$) flux difference operating band. We have accepted[7] use of this analysis on a plant-specific basis. We, therefore, conclude the power distribution control and monitoring procedures and related Technical Specification changes proposed for Cycle 5 operation of Surry Unit 1 are acceptable.

Control Rod Insertion Limits

VEPCO has proposed to raise the control rod insertion limits to provide adequate margin to the $F_{\Delta H}$ limits. There are a number of criteria which the control rod insertion limits are checked against each cycle. The most important of these are shutdown margin, ejected control rod worth, and $F_{\Delta H}$. VEPCO found the latter to be the most restrictive for Cycle 5 operation and the insertion limits are proposed to be adjusted accordingly. We find the proposed control rod insertion limit Technical Specification change acceptable.

Shutdown Margin

The hot full power shutdown margin is predicted by the licensee to be 4.91% $\Delta\rho$ at beginning of cycle (BOC) and 3.97% $\Delta\rho$ at end of cycle (EOC) compared to a shutdown margin requirement of 1.77% $\Delta\rho$ as assumed in the steam line break analysis. This is acceptable because of extra margin between predicted and required shutdown margin throughout cycle life. In addition, in determining the predicted shutdown margin, a 10% calculational uncertainty is subtracted for the case where all rods are inserted except for the stuck highest worth rod. Further, confirmation of the validity of the prediction will be made during the startup physics test program by measuring the regulating banks (which contain about half of the total control rod worth) during startup tests. These measured worths are compared with predictions for the measurement conditions made with the same model used for calculating the shutdown margin.

Radial Power Distributions

During our review, we questioned whether the measurement uncertainty of 5% is appropriate relative to the maximum Cycle 5 unrodded radial peaking factor. This question arose because the predicted hot fuel assembly for Cycle 5 was an edge assembly in Cycle 4. Since there has been a greater uncertainty in measurement to calculation comparisons for edge assemblies, we were concerned about the appropriateness of the 5% measurement uncertainty for Cycle 5. The licensee has provided an uncertainty analysis(2) which has satisfied us that this 5% measurement uncertainty is acceptable. The licensee's submittal indicated the results of a number of expiremental and calculated power distribution comparisons that support this conclusion.

Accident Analysis

Calculated values of the Cycle 5 kinetics parameters compared with those from the currently applicable transient and accident analyses have been provided by the licensee. Where the Cycle 5 values are within the bounds of the applicable analyses, no reanalysis of the transient or accident is necessary. However, the most negative Doppler temperature coefficient is more negative than the current limit. Also the beginning of life (BOL) and end of life (EOL) value of the delayed neutron fraction range are lower than the current limit.

An increase in the most negative Doppler coefficients has an impact on cooldown events. Since the limiting cooldown event for Surry Unit 1 is inadvertent startup of an inactive loop*, the licensee submitted results of reevaluation of this transient. These show an increase in the positive reactivity insertion of approximately 3%, and an increase in the maximum power level reached during the transient by less than one-half percent over the currently applicable safety analysis. Therefore, the conclusions in the FSAR for the cooldown class of accidents remain appropriate for the Cycle 5 reload core.

VEPCO reanalyzed the rod ejection accident since both BOL and EOL minimum delayed neutron fractions are less than the current limits. The BOL and EOL hot full power and zero power cases were reanalyzed with conservative input values to ensure that the fuel and clad limits were not exceeded. This reanalysis was performed using the method documented in the approved Reference 8. The results of the reanalysis show that the fuel does not exceed the limiting criteria of Reference 8, even at the hot spot. Therefore, the conclusions presented in the FSAR for the rod ejection accident remain valid.

VEPCO reanalyzed the rod withdrawal from subcritical accident due to an increase in the calculated positive reactivity insertion rate from the withdrawal of two RCCA control banks moving together in their highest worth region. The reanalysis assumed a positive insertion rate of 75 pcm/sec. All other input conditions were identical to those used in the previously applicable analysis, except for the trip reactivity which is more conservative. The reanalysis results in an increase of 17% in the maximum heat flux obtained during the transient. This increase in maximum heat flux is still below the nominal full power value. Based on our evaluation of the results of the reanalysis provided by the licensee, we conclude that the consequences of this accident remain acceptable.

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*At present, this accident is precluded since the Technical Specifications do not allow operation with an inactive loop.

Startup Tests

The startup physics test program for Cycle 5 has been reviewed. Basically this program is identical to the program for Cycle 4. The tests can be grouped into five categories:

1) Control rod worth measurements
2) Critical boron concentration measurements and boron worth coefficient
3) Isothermal temperature coefficient measurements
4) Core power distribution measurements
5) Power coefficient measurement

The acceptance criteria will be the same as for Cycle 4 except that the criterion on individual rod bank measurements will remain ±15%. The criterion on the sum of the measured rod banks will remain ±10%. The actions to be taken if these acceptance criteria are not met are explained in Reference 2.

We have reviewed this entire program and have found it to be acceptable. However, due to the nature of the data obtained from these tests (it confirms values used in the safety analysis) a physics startup test report will be submitted to the NRC by VEPCO within 45 days of completion of the test program.

ECCS Analysis

On March 23, 1978 the Westinghouse Electric Corporation informed us of the discovery that an error had been made in the WEST-ECCS evaluation model which resulted in incorrectly calcuated peak clad temperatures for all LOCA analyses previously submitted by customers. Preliminary estimates indicated that several plants would not meet the 2200°F limit of 10 CFR 50.46 with present maximum overall peaking factor limits. Westinghouse and several of its customers met with us on March 29, 1978 to discuss the error and its impact on specific plant analyses. Subsequent to that meeting, Westinghouse provided information through the licensees of operating reactors to justify continued operation at interim peaking factor Technical Specification limits proposed by the NRC staff on April 3, 1978. On May 26, 1978, VEPCO submitted an interim ECCS analysis which showed that the performance requirements of 10 CFR 50.46 would not be exceeded. This subject is dealt with in a specific exemption to 10 CFR 50.46(a)(1) granted the same date as this safety evaluation.

Environmental Consideration

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

Conclusion

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

Date: June 30, 1978

References

1. Letter from C. M. Stallings (VEPCO) to E. G. Case (NRC), Serial No. 108, March 15, 1978.

2. Letter from C. M. Stallings (VEPCO) to E. G. Case (NRC), Serial No. 272 May 11, 1978.

3. R. A. George, et al., "Revised Clad Flattening Model," WCAP-8377 (Proprietary) and WCAP-8381 (Non-proprietary), July 1974.

4. Miller, J. V. (Ed), "Improved Analytical Model Used In Westinghouse Fuel Rod Design Computation," WCAP-8785 (Non-proprietary) and WCAP-8720 (Proprietary), October 1976.

5. Risher, D. H, et al., "Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis," WCAP-8963 (Propietary) and WCAP-8964 (Non-proprietary), June 1977.

6. "Interim Safety Evaluation Report on Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors (Revision 1)," NRC, February 16, 1977.

7. Letter from D. Vassallo (NRC) to C. Eicheldinger (W), April 4, 1976.

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

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of VIRGINIA ELECTRIC & POWER COMPANY) Surry Power Station, Unit No. 1	Docket No. 50-280

EXEMPTION

I.

Virginia Electric & Power Company (the licensee), is the holder of Facility Operating License No. DPR-32 which authorizes the operation of the nuclear power reactor known as Surry Power Station, Unit No. 1 (the facility) at steady reactor power levels not in excess of 2441 megawatts thermal (rated power). The facility consists of a Westinghouse Electric Corporation designed pressurized water reactor (PWR) located at the licensee's site in Surry County, Virginia.

II.

In accordance with the requirements of the Commission's ECCS Acceptance Criteria 10 CFR 50.46, the licensee submitted on August 9, 1977 as supplemented on August 26, October 19 and November 16, 1977 an ECCS evaluation for proposed operation using 15 x 15 fuel manufactured by the Westinghouse Electric Corporation. This evaluation included limits on the peaking factor. The ECCS performance evaluation submitted by the licensee was based upon an ECCS evaluation developed by the Westinghouse Electric Corporation (Westinghouse), the designer of the Nuclear Steam Supply System for this facility. The Westinghouse ECCS Evaluation Model had been previously found to conform to the requirements of the Commission's ECCS Acceptance Criteria, 10 CFR Part 50.46 and Appendix K. The evaluation indicated that with the peaking factor limit as set forth in the evaluation, and with other limits set forth in the facility's Technical Specifications, the ECCS cooling performance for the facility would conform with the criteria contained in 10 CFR 50.46(b) which govern calculated peak clad temperature, maximum cladding oxidation, maximum hydrogen generation, coolable geometry and long-term cooling.

On March 23, 1978 Westinghouse informed the Nuclear Regulatory Commission (NRC) that an error had been discovered in the fuel rod heat balance equation involving the incorrect use of only half of the volumetric heat generation due to metal-water reaction in calculating the cladding temperature. Thus, the LOCA analyses previously submitted to the Commission by licensees of Westinghouse reactors were in error. The staff promptly determined that no immediate action was required to assure safe operation of these plants.

The error identified would result in an increase in calculated peak clad temperature, which, for some plants, could result in calculated temperatures in excess of 2200°F unless the allowable peaking factor was reduced somewhat. Westinghouse identified a number of other areas in the approved model which Westinghouse indicated contained sufficient conservatism to offset the calculated increase in peak clad temperature resulting from the correction of the error noted above. Four of these areas were generic, applicable to all plants, and a number of others were plant-specific. As outlined in the attached Safety Evaluation Report (SER), the staff determined that some of these modifications would be appropriate to offset to some extent the penalty resulting from correction of the error.

Revised computer calculations correcting the error, noted above, and incorporating the model modifications and plant-specific input modifications described in the SER have been run for Surry. Since some of these model modifications have not been approved by the staff, the licensee adjusted the results in a conservative fashion using the various parametric studies that have been made for various aspects of the approved Westinghouse model over the course of time, these studies provide a reasonable basis for concluding that when final revised calculations for the facility are submitted using the revised and corrected model, they will demonstrate that with the peaking factors set forth in the SER operation will conform to the criteria of 10 CFR 50.46(b). Such revised calculations fully conforming to 10 CFR 50.46 are to be provided for the facility as soon as possible.

Since the staff has not completed its review of all aspects of the model used in making these computer calculations submitted by the licensee, the staff cannot determine that the evaluation is wholly in conformance with the requirements of 10 CFR 50.46(a). However, operation as proposed in the licensee's submittal of May 26, 1978, at the peaking factor limit specified in the Exemption will assure that the ECCS system will conform to the performance criteria of 50.46. Accordingly, while the full compliance with 10 CFR 50.46 operation of the facility will not endanger life or property or the common defense and security.

In the absence of any safety problem associated with operation of the facility during the period until the computer computations are completed, there appears to be no public interest consideration favoring restriction of the operation of the captioned facility. Accordingly, the Commission has determined that an exemption in accordance with 10 CFR 50.12 is appropriate. This exemption will be terminated by the staff upon completion of its review of the evaluation model used by the licensee.

III.

Copies of the Safety Evaluation and the following documents are available for inspection at the Commission's Public Document Room at 1717 H Street, Washington, D. C. 20555, and are being placed in the Commission's local public document room at the Swem Library, College of William and Mary, Williamsburg, Virginia.

(1) Letter from Westinghouse to NRC dated April 7, 1978.

(2) Letters from Virginia Electric & Power Company dated April 7, 1978 and May 26, 1978.

(3) This Exemption in the matter of Surry Power Station, Unit No. 1.

IV.

Wherefore, in accordance with the Commission's regulations as set forth in 10 CFR Part 50, the licensee is hereby granted an exemption from the requirements of 10 CFR 50.46(a)(1) that ECCS performance be calculated in accordance with an acceptable calculational model which conforms to the provisions in Appendix K, without errors discussed herein. This exemption is conditioned as follows:

(1) Until further authorization by the Commission, the Technical Specification limit for total nuclear peaking factor (F_Q) for the facility shall be limited to 1.94.

(2) Minimum accumulator water volume shall be decreased to 975 ft^3 as specified in the May 26, 1978 letter from VEPCO.

FOR THE NUCLEAR REGULATORY COMMISSION

Victor Stello, Jr., Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland
this 30th day of June 1978

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING A SPECIFIC EXEMPTION TO 10 CFR 50.46(a)(1)

RELATED TO ERROR IN WESTINGHOUSE ECCS EVALUATION MODEL

FOR

SURRY POWER STATION UNIT NO. 1

DOCKET NO. 50-280

Introduction

Westinghouse was informed on March 21, 1978 by one of its customers that an error had been discovered in its ECCS Evaluation Model. This error was common to both the blowdown and heatup codes. Westinghouse determined by analyses that the fuel rod heat balance equation in the LOCTA IV & SATAN VI codes was in error and that the LOCA analyses previously submitted by its customers were incorrect and predicted peak clad temperatures (PCT's) which were too low. Westinghouse determined that only half of the volumetric heat generation due to metal-water reaction was used in calculating the cladding temperatures. Thus an unreviewed safety question existed since preliminary estimates indicated that some plants would not meet the 2200°F limit of 10 CFR 50.46 at the calculated maximum overall peaking factor limit. Westinghouse notified its customers and NRC on March 23, 1978 while the customers as licensees notified NRC through the regional Offices of Inspection and Enforcement.

Promptly upon notification by Westinghouse, the NRC staff assessed the immediate safety significance of this information. We noted certain points that indicated no immediate action was required to assure safe operation of the plants. First, most plants operate at a peaking factor significantly below the maximum peaking factor used for safety calculations. By making safety computations at factors higher than actual operation levels the facility has a wide range of flexibility, without the need for hour to hour recomputations of core status. The difference between the actual peaking factors and the maximum calculated peaking factors, for most plants, would offset the penalty resulting from the correction of the error. Second, for most reactors there are a number of very plant-specific parameters which bear upon aspects of the ECCS performance calculations. Licensees do not generally take credit for these plant-specific parameters preferring to provide a simpler computation which conservatively disregards these individually small credits. Third, the error in the Westinghouse computations relates to the zirconium-water reaction heat source. This is an aspect of Appendix K, which is generally recognized to be very conservative. New experimental data indicate that the methods required by Appendix K appreciably over estimate the heat source. Thus, while the error in fact entails a deviation from a specific requirement of Appendix K, it does not entail a matter of immediate safety significance.

Westinghouse continued to evaluate the impact of the error on previous plant-specific LOCA analyses and performed scoping calculations, sensitivity studies and some plant-specific reanalyses. In addition, Westinghouse investigated several modifications to the previously approved methods which if approved by the NRC staff would offset some of the immediate impact of the error on Technical Specifications limits and on the plants operating flexibility.

On March 29, 1978, Westinghouse and several of its customers met with members of the NRC staff in Bethesda. Westinghouse described in detail the origin of the error, explained how it affected the LOCA analyses, and how the error had been corrected and characterized its effect on current plant specific analyses. In order to minimize reduction in the overall peaking factor (F_Q), Westinghouse presented a description of three proposed ECCS-LOCA evaluation model modifications which would contribute a compensating reduction of PCT. They were characterized as follows:

1. <u>Revised FLECHT 15 x 15 Heat Transfer Correlation</u>

   This new reflood heat transfer correlation which had been recently developed and submitted by Westinghouse in Reference (1) was proposed as a replacement for the currently approved FLECHT correlation. To determine the benefit, the proposed correlation was incorporated into the LOCTA IV heatup code and was found to result in improved heat transfer during the reflood portion of the LOCA.

2. <u>Revised Zircaloy Emissivity</u>

   Based on recent EPRI data (Reference 2), Westinghouse proposed to modify the presently approved equation for Zircaloy cladding emissivity to a constant value of 0.9. The higher emissivity (previously below 0.8) provides increased radiative heat transfer from the hot fuel pin during the steam cooling period of reflood.

3. Post-CHF Heat Transfer

   Westinghouse proposed to replace its present post-CHF transition boiling heat transfer correlation with the Dougall-Rohsenow film boiling correlation (Reference 3) which was included in Appendix K to 10 CFR Part 50 as an acceptable post-CHF correlation.

These three model modifications were classified as generic, applicable to all plant analyses. Subsequently, as discussed below, these changes were rejected by the NRC staff as providing generic benefit. However, a portion of the credit proposed by Westinghouse was approved by the NRC staff for certain specific plants, which had provided specific calculations with the new 15 x 15 correlation. Since March 29, 1978, Westinghouse has provided us with additional sensitivity analyses and plant-specific analysis in which they evaluated the effects of some changes to plant-specific inputs in the LOCA analyses. These were as follows:

1. Assumed Plant Power Level

   A reduction of the plant power level assumed in the SATAN VI blowdown analyses from 102% of the Engineered Safeguards Design Power (ESDR) level to 102% of rated power was proposed. Previously, analyses had been performed at approximately 4.5% over the rated power. This change was worth approximately 0.01 in F_Q.

2. COCO Code Input

   A modification to the COCO code input (Reference 3) to more realistically model the painted containment walls was proposed. Since the paint on containment walls provides additional resistance to heat loss into the walls, the COCO code calculates an increase in containment back pressure, which results in a benefit to the calcuated peak cladding temperature of 0 to 40°F, during the reflooding transient. The magnitude of the benefit is dependent on the type of plant and the heat transfer properties of the paint, and results in up to 0.03 benefit in F_Q.

3. Initial Fuel Pellet Temperature

A modification of the initial fuel pellet temperature from the design basis to the actual as-built pellet temperature was proposed. In the present LOCA calculations, Westinghouse has assumed margins in the initial pellet temperature. The margin available is plant-specific and ranges from 28°F to 55°F. Use of the actual pellet temperature rather than the assumed value results in a reduction in pellet temperature (stored energy) at the end of blowdown, as calculated by the SATAN code, of approximately 1/3 of the initial pellet temperature margin. Westinghouse has provided sensitivity analyses which indicate that a 37°F reduction in fuel pellet temperature at end of blowdown is worth approximately 0.1 in F_Q.

4. Accumulator Water Volume Consideration

Westinghouse has evaluated the effect on ECCS performance of reducing the accumulator water volume, and has determined that for those plants for which the downcomer is refilled before the accumulators are emptied, there is a benefit in PCT. The sensitivity studies have indicated that this benefit in F_Q is plant-specific.

5. Steam Generator Tube Plugging Consideration

In previous analyses, Westinghouse has assumed values of steam generator tube plugging which were greater than the actual plant-specific degree of plugging. Sensitivity analyses submitted in Reference 4 were used to evaluate the benefit available by realistically representing the plant-specific data. For the plants affected, the benefit in PCT ranged from 7 to 66°F which was conservatively worth from 0.01 to 0.066 in F_Q.

The information provided by Westinghouse was separated into two categories; the generic evaluation model modifications and the plant-specific sensitivity studies and reanalyses. The NRC staff reviewed the peaking factor limits proposed by Westinghouse to verify their conservatism.

The metal-water reaction heat generation error in the Westinghouse ECCS evaluation model was evaluated by us to determine an appropriate interim penalty. Westinghouse provided two preliminary separate effects calculations which indicated that a maximum penalty of from 0.14 to 0.17 was appropriate to compensate for the model error. The staff conservatively rounded this penalty up to 0.20. (Reference 5)

Westinghouse also proposed several compensating generic changes in their evaluation model to offset any necessary reductions in peaking factor due to the error. These changes were assessed by us as follows: (Reference 5)

1. No credit would be given at this time for the changes in the post-CHF heat transfer correlation and new Zircaloy emissivity data.

2. Partial credit (70%) would be given at this time for the use of the new 15 x 15 FLECHT correlation only for plants which had provided a specific calculation demonstrating that such credit was appropriate.

Based on this review we developed recommended interim peaking factor limits for all the operating plants and decided that any other plant-specific interim factors (benefits) not related to the generic review should be considered separately.

Discussion and Evaluation

Virginia Electric and Power Company (VEPCO) in the letter of April 7, 1978 (Reference 6) has indicated that a new ECCS-LOCA analysis will be performed with a LOCA evaluation model fully in compliance with 10 CFR 50, Appendix K and with a corrected metal-water reaction input. In the meantime, by the letter dated May 26, 1978 (Reference 7), VEPCO provided an interim analysis which was performed with the October 1975 version of the Westinghouse LOCA evaluation model, which was modified to include an improved 15 x 15 FLECHT heat transfer correlation which the staff is presently reviewing and corrected the model for the full effect of the metal-water reaction and included the following plant-specific input modifications:

1. The minimum value of the accumulator water volume was decreased from 1075 ft^3 to 975 ft^3.

2. The value of initial fuel temperature was based on as-built values of fuel characteristics rather than generic values.

3. The ECCS containment parameters were modified to reflect the containment response in a more realistic, but still conservative, manner.

4. The effect of painted containment wall surfaces was incorporated into the containment analysis.

5. Plugging of 25% of the steam generator tubes was assumed. (NRC approval was granted for 25% tube plugging on December 2, 1977.)

Since the new FLECHT correlation is still under review by the staff after discussions with the staff the licensee adjusted the results of its calculations to take only partial credit (70%) for use of the new FLECHT correlation. By using various parametric studies previously made, the reduction in FLECHT correlation credit resulted in a reduction in peaking factor by 0.03 to 1.94, maintaining peak clad temperature below 2200°F. The results of the licensee's calculations and adjustments thereto are listed below:

Peak Cladding Temperature:	2146°F
Local Zr/Water Reaction:	6.94%
Total Zr/Water Reaction:	<0.3%
Peaking Factor:	$F_Q = 1.94$

The critical break remained the same as in the previous analysis (Reference 8). It is a double ended cold leg guillotine break (DECLG) with C_D=0.4. The predicted oxidation and the amount of fuel cladding that reacts chemically with water or steam are below the limits set by 10 CFR 50.46. Nevertheless, the use of such external adjustments to the computer calculations, while adequate to demonstrate available safety margins, does not wholly satisfy 10 CFR 50.46.

The licensee stated in its letter dated April 7, 1978 (Reference 6), that compliance with the F_Q limit will be assured by the performance of axial power distribution monitoring (APDM) as described in the Technical Specifications.

Conclusion

We conclude that when final revised calculations for the facility are submitted using the revised and corrected model, they will demonstrate that with the peaking factors set forth herein, operation will conform to the criteria of 10 CFR 50.46(b). Such revised calculations fully conforming to 10 CFR 50.46 are to be provided for the facility as soon as possible.

As discussed herein, the peaking factor limits specified by the licensee, in combination with any necessary operating surveillance requirements, will assure that the ECCS will conform to the performance requirements of 10 CFR 50.46(b). Accordingly, limits on calculated peak clad temperature, maximum cladding oxidation maximum hydrogen generation, coolable geometry and long term cooling provide reasonable assurance that the public health and safety will not be endangered.

References

1. WCAP-9220, "Westinghouse ECCS Evaluation Model, February 1978 Version", February 1978.

2. EPRI Report NP-525, "High Temperature Properties of Zircaloy-Oxygen Alloy", March 1977.

3. R. S. Dougall, W. M. Rohsenow, "Film Boiling on the Inside of Vertical Tubes with Upward Flow of the Fluid at Low Qualities", MIT Report 9079-26, September 1963.

4. WCAP-8986 - "Perturbation Technique For Calculating ECCS Cooling Performance", February 1977.

5. Memorandum: Rosztoczy to Eisenhut and Ross, "Metal-Water Reaction Heat Generation Error in Westinghouse ECCS Evaluation Model Computer Program," April 7, 1978.

6. Letter from VEPCO (C. M. Stallings) to NRC (E. G. Case), dated April 7, 1978, Serial No. 197.

7. Letter from VEPCO (C. M. Stallings) to NRC (E. G. Case), dated May 26, 1978, Serial No. 303.

8. Letter from VEPCO (C. M. Stallings) to NRC (B. C. Rusche), dated March 4, 1977, Serial No. 219/082776.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-280 AND 50-281

VIRGINIA ELECTRIC AND POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

AND SPECIFIC EXEMPTION TO 10 CFR 50.46(a)(1) FOR SURRY UNIT NO. 1

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 42 and 41 to Facility Operating License Nos. DPR-32 and DPR-37, issued to Virginia Electric & Power Company (the licensee), which revised Technical Specifications common to the Surry Power Station, Unit Nos. 1 and 2 (the facilities) located in Surry County, Virginia and has granted a specific exemption to 10 CFR 50.46(a)(1) for Surry Unit No. 1. The amendments and exemptions are effective as of the date of issuance.

The amendments consist of Technical Specification changes required for Cycle 5 operation of Surry Unit No. 1.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments 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.

The exemption to 10 CFR 50.46(a)(1) is limited to the time period necessary to complete computer calculations acceptable to the NRC staff, that correct for errors in the Westinghouse ECCS models. The NRC has determined that this exemption is authorized by law, will not endanger life or property or the common defense and security and is otherwise in the public interest.

For further details with respect to these actions see (1) application for amendments dated March 15, 1978, as supplemented May 11, 1978, (2) Amendment Nos.42 and 41 to License Nos. DPR-32 and DPR-37, (3) Specific Exemptions to 10 CFR 50.46(a)(1) dated June 30, 1978, and (4) 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 30th day of June 1978.

FOR THE NUCLEAR REGULATORY COMMISISON

Victor Stello, Jr., Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation