December 10, 198

Docket Nos. 50-280
and 50-281

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Mr. W. L. Stewart
Vice President - Nuclear Operations
Virginia Electric and Power Company
Post Office Box 26666
Richmond, Virginia 23261

Dear Mr. Stewart:

SUBJECT: SURRY UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS
(TAC NOS. 64841 AND 64842)

The Commission has issued the enclosed Amendment No. 115 to Facility Operating License No. DPR-32 and Amendment No. 115 to Facility Operating License No. DPR-37 for the Surry Power Station, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated February 23, 1987.

These amendments modify the Technical Specifications to allow the accumulator water volume to vary between 975 and 1025 ft^3 per accumulator instead of 975 and 989 ft^3 per accumulator, and the height dependent heat flux hot channel factor ($F_Q(z)$) to be as high as 2.32 instead of 2.18 as presently stated in the Technical Specifications.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by

Chandu P. Patel, Project Manager
Project Directorate II-2
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 115 to DPR-32
2. Amendment No. 115 to DPR-37
3. Safety Evaluation

cc: w/enclosures
See next page

LA:PDII-2	PM:PDII-2	D:PDII-2	OGC
DMiller	CPatel:bg	HBerkow	MYoung
12/1 /87	12/1 /87	12/1 /87	12/3 /87

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P PDR

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

December 10, 1987

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and 50-281

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Vice President - Nuclear Operations
Virginia Electric and Power Company
Post Office Box 26666
Richmond, Virginia 23261

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Chandu P. Patel, Project Manager
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2. Amendment No. 115 to DPR-37
3. Safety Evaluation

cc: w/enclosures
See next page

Mr. W. L. Stewart
Virginia Electric and Power Company

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Richmond, Virginia 23219

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-280

SURRY POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 115
License No. DPR-32

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

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated February 23, 1987, 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:

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(B) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 115 , 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

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 10, 1987

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 115
License No. DPR-37

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

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated February 23, 1987, 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-37 is hereby amended to read as follows:

(B) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 115 , 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

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 10, 1987

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 115 FACILITY OPERATING LICENSE NO. DPR-32

AMENDMENT NO. 115 FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NOS. 50-280 AND 50-281

Revise Appendix A as follows:

Remove Pages	Insert Pages
TS 3.3-1	TS 3.3-1
TS 3.3-8	TS 3.3-8
TS 3.12-3	TS 3.12-3
TS 3.12-15	TS 3.12-15
TS Figure 3.12-8	TS Figure 3.12-8

3.3 SAFETY INJECTION SYSTEM

Applicability

Applies to the operating status of the Safety Injection System.

Objective

To define those limiting conditions for operation that are necessary to provide sufficient borated cooling water to remove decay heat from the core in emergency situations.

Specifications

A. A reactor shall not be made critical unless the following conditions are met:

1. The refueling water storage tank contains not less than 387,100 gallons of borated water. The boron concentration shall be at least 2000 ppm and not greater than 2200 ppm.
2. Each accumulator system is pressurized to at least 600 psia and contains a minimum of 975 ft^3 and a maximum of 1025 ft^3 of borated water with a boron concentration of at least 1950 ppm.

Time After Shutdown	Decay Heat, % of Rated Power
1 hour	1.3
8 hours	0.75
48 hours	0.48

Thus, the requirement for core cooling, in case of a postulated loss-of-coolant accident while in the hot shutdown condition is reduced by orders of magnitude below the requirements for handling a postulated loss-of-coolant accident occurring during power operation. Placing and maintaining the reactor in the hot shutdown condition significantly reduces the potential consequences of a loss-of-coolant accident, allows access to some of the Safety Injection System components in order to effect repairs, and minimizes the exposure to thermal cycling.

Failure to complete repairs within 48 hours of going to hot shutdown condition is considered indicative of unforeseen problems, i.e., possibly the need of major maintenance. In such a case the reactor is to be put into the cold shutdown condition.

The accumulators are able to accept leakage from the Reactor Coolant System without any effect on their availability. Allowable inleakage is based on the volume of water that can be added to the initial amount without exceeding the volume given in Specification 3.3.A.2. The maximum acceptable inleakage is 50 cubic feet per tank.

B. Power Distribution Limits

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

$F_Q(Z) \leq 2.32/P \times K(Z)$ for $P > 0.5$

$F_Q(Z) \leq 4.64 \times K(Z)$ for $P \leq 0.5$

$F^N_{\Delta H} \leq 1.55\ [1 + 0.3\ (1-P)]$ for three loop operation

$\leq 1.55\ [1 + 0.2\ (1-P)]$ for two loop operation

where P is the fraction of rated power at which the core is operating, K(Z) is the function given in TS Figure 3.12-8, and Z is the core height location of F_Q.

2. Prior to exceeding 75% power following each core loading and during each effective full power month of operation thereafter, power distribution maps using the movable detector system shall be made to confirm that the hot channel factor limits of this specification are satisfied. For the purpose of this confirmation:

   a. The measurement of total peaking factor F_Q^{Meas} shall be increased by eight percent to account for manufacturing tolerances, measurement error and the effects of rod bow. The measurement of enthalpy rise hot channel factor $F_{\Delta H}$ shall be increased by four percent to account for measurement error. If any measured hot channel factor exceeds its limit specified under Specification 3.12.B.1, the reactor power and high neutron flux trip setpoint shall be reduced until the limits under Specification 3.12.B.1 are met. If the hot channel factors cannot be brought to within the limits of $F_Q(Z) \leq 2.32 \times K(Z)$ and $F^N_{\Delta H} \leq 1.55$ within 24 hours, the Overpower ΔT and Overtemperature ΔT trip setpoints shall be similarly reduced.

It should be noted that the enthalpy rise factors are based on integrals and are used as such in the DNB and LOCA calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in radial (x-y) power shapes throughout the core. Thus, the radial power shape at the point of maximum heat flux is not necessarily directly related to the enthalpy rise factors. The results of the loss of coolant accident analyses are conservative with respect to the ECCS acceptance criteria as specified in 10 CFR 50.46 using the upper bound $F_Q(Z)$ times the hot channel factor normalized operating envelope given by TS Figure 3.12-8.

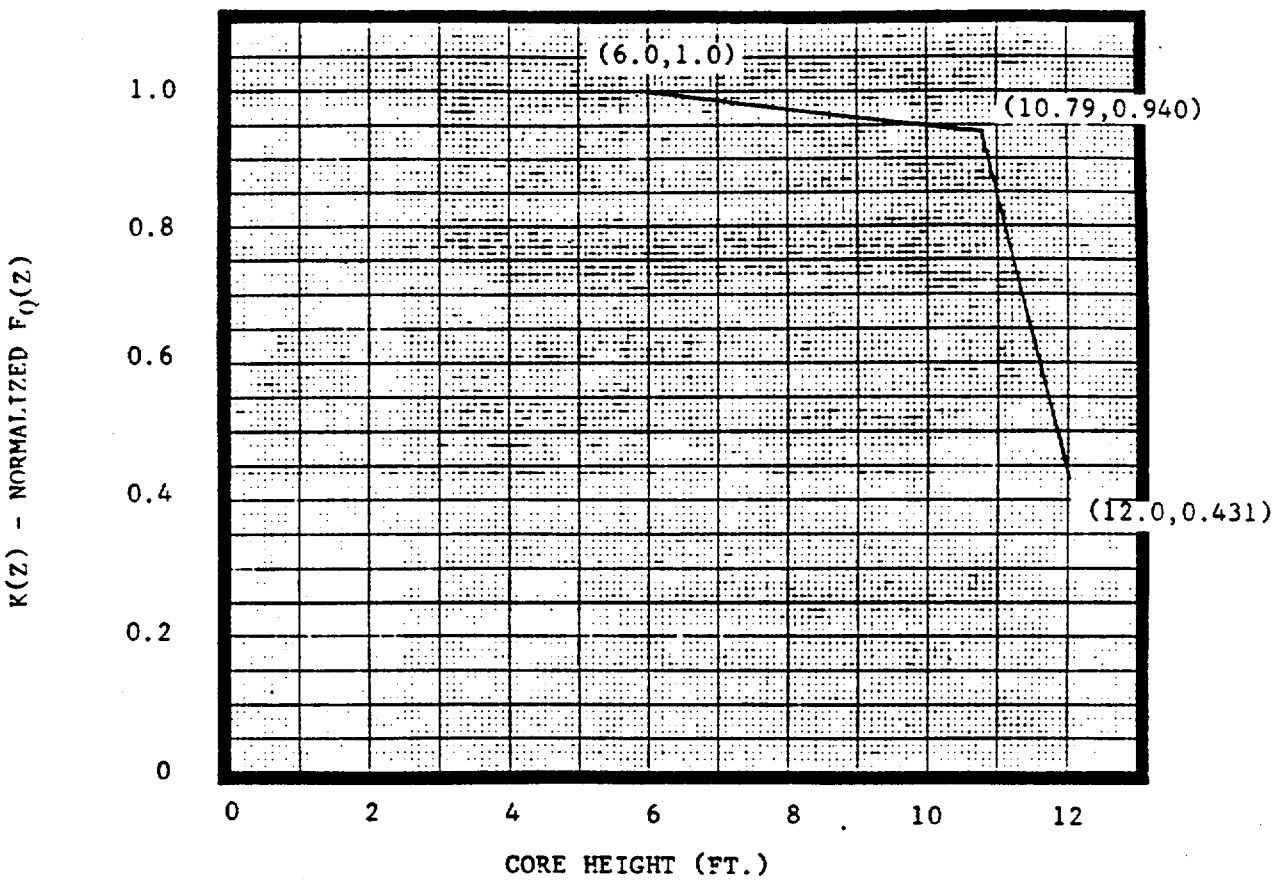
When an F_Q measurement is taken, measurement error, manufacturing tolerances, and the effects of rod bow must be allowed for. Five percent is the appropriate allowance for measurement error for a full core map (greater than or equal to 38 thimbles, including a minimum of 2 thimbles per core quandrant, monitored) taken with the movable incore detector flux mapping system, three percent is the appropriate allowance for manufacturing tolerances, and five percent is appropriate allowance for rod bow. These uncertainties are statistically combined and result in a net increase of 1.08 that is applied to the measured value of F_Q.

In the specified limit of $F^N_{\Delta H}$, there is an eight percent allowance for uncertainties, which means that normal operation of the core is expected to result in $F^N_{\Delta H} \leq 1.55\ [1 + 0.3\ (1-P)]/1.08$. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape (e.g., rod misalignment) affect $F^N_{\Delta H}$, in most cases without necessarily affecting F_Q, (b) the operator has a direct influence on F_Q through movement of rods and can limit it to the desired value; he has no direct control over $F^N_{\Delta H}$, and (c) an error in the predictions for radial power shape, which may be detected during startup physics tests and which may influence F_Q, can

HOT CHANNEL FACTOR NORMALIZED

OPERATING ENVELOPE

SURRY POWER STATION



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 115 TO FACILITY OPERATING LICENSE NO. DPR-32

AND AMENDMENT NO. 115 TO FACILITY OPERATING LICENSE NO. DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-280 AND 50-281

INTRODUCTION

By letter dated February 23, 1987, Virginia Electric and Power Company (the licensee) requested amendments to Facility Operating License Nos. DPR-32 and DPR-37, issued to the licensee for operation of the Surry Nuclear Power Station, Units 1 and 2, located in Surry County, Virginia.

The licensee proposed to modify the Technical Specifications (TS) to allow the accumulator water volume to vary between 975 and 1025 ft^3 per accumulator instead of 975 and 989 ft^3 per accumulator, and the height dependent heat flux hot channel factor ($F_Q(z)$) to be as high as 2.32 instead of 2.18 as presently stated in the TS for Surry Units 1 and 2. The licensee proposed these changes by using the results of a large-break loss-of-coolant accident (LOCA) reanalysis which was required due to recently approved Westinghouse Emergency Core Cooling Systems (ECCS) evaluation models (Refs. 2, 3, and 4). The following evaluation examines the analysis codes, the assumptions regarding the input parameters and the results of the analysis with respect to the corresponding requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50.

The analysis was performed with an NRC-approved code, the 1981 version of the Westinghouse LOCA-ECCS evaluation model BART (Refs. 4 and 5), including the evaluation model revisions approved by the NRC (Refs. 2 and 3). The code revisions refer to: (a) a modeling change in a calculation model called WREFLOOD, which was found to increase the peak cladding temperature by about 20°F, and (b) a systematic input error in the BART code, which caused low values of hot assembly bundle power to be used. This last error was found to increase the peak cladding temperature by about 100°F. The computational model and codes used for this analysis comply with the requirements of Appendix K to 10 CFR 50.

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DISCUSSION

Applicable Criteria for a Large-Break LOCA

The criteria to be satisfied in a large-break LOCA analysis are described in 10 CFR 50.46 and are:

1. peak cladding temperature shall not exceed 2,200° F;

2. localized maximum cladding oxidation must not exceed 17% during or after quenching;

3. cladding chemical interaction with water and steam (maximum hydrogen generation) must not exceed 1.0% of all the metal in the cladding cylinders surrounding the fuel;

4. calculated changes in core geometry shall be such that the core remains amenable to cooling; and

5. after the successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long lived radioactivity remaining in the core.

Description of the Transient

During a large-break LOCA, depressurization of the Reactor Coolant System (RCS) will result in a reactor trip and safety injection signals. The injection of borated water will complement void formation to shutdown the fission process, although the presence of boron is not accounted for in the LOCA analysis. Similarly, no credit is taken for control rod insertion, leaving void formation as the only credible mechanism to terminate the fission process in the early phase of the transient. Injection of the borated water provides for heat transfer from the core and prevents excessive clad temperature. When the primary coolant depressurizes to 600 psia, the accumulators will begin injecting borated water. The analysis assumes loss of offsite power, hence, reactor coolant pumps are assumed to trip and pump coast down is assumed. After the depressurization (blowdown) phase of the transient ends, refill of the reactor vessel begins with emergency core cooling water which was not assumed to be operational up to this time. The refill is completed when the water level reaches the bottom of the fuel rods. Reflood then takes place, i.e., the period of the transient when the water covers the core to the extent that the core temperature rise has been terminated. Continued operation of the ECCS pumps supplies water for the long-term cooling. The boric acid concentration in the primary coolant is such as to prevent recriticality.

The blowdown, refill and reflood stages of the transient are analyzed with the methodologies described in Reference 6. This document also describes the interfaces among the computer codes and the features of the codes that ensure compliance with 10 CFR 50, Appendix K. More specifically:

- SATAN-VI - Analyzes the thermal hydraulic transient in the RCS during blowdown. It includes RCS, pressure enthalpy, density and mass and energy flow (Ref. 7).

- COCO: - Calculates the containment pressure transient during all three phases of the LOCA analysis (Ref. 8).

- WREFLOOD: - Determines the core flooding rate, the coolant pressure and temperature and the quench front height during the refill and reflood phases of the LOCA (Ref. 9).

- LOCTA-IV: - Computes the thermal transient of the hottest fuel rod during the three phases (Ref. 10).

Initial Conditions and Parameters

The initial conditions and the numerical values of the input parameters have been conservatively determined (Ref. 11). The main core parameters and initial conditions assumed for a double-ended cold leg guillotine break (which was shown to be the limiting break; Reference 12) are listed in the following table:

Initial Conditions for Double-Ended Cold Leg Guillotine Break

Parameter	Calculational Input
Core power (MWt) 102% of	2441
Peak linear power (KW/ft) 102% of	14.39
Heat flux hot-channel factor (F_Q)	2.32
Enthalpy rise hot-channel factor ($F_{\Delta H}^N$)	1.62
Accumulator water volume (ft^3, each)	1000
Reactor vessel upper head temperature	T_{hot}
Guillotine break discharge coefficient C_D	0.4, 0.6

In addition, this LOCA reanalysis evaluated plant operation at steam generator tube plugging levels up to 7% based on the 10 CFR 50.46 acceptance criteria.

Results

The analysis at the 7% steam generator tube plugging level has shown that primary flow rate will be maintained at levels which will not be limiting in departure from nucleate boiling (DNB) limiting events (Ref. 13). The results of the analysis reported in Reference 13 are supported by measurements taken (prior to steam generator replacement) at several levels of steam generator tube plugging. Analysis and measurements support the conclusion that the 7% plugging level flow is considerably larger than the Surry thermal design flow. The impact of the 7% tube plugging on the dilution times in the uncontrolled boron dilution events is bounded by the analyses documented in Reference 12 and by the Surry Updated Final Safety Analysis Report (UFSAR), which were found acceptable.

The limiting double-ended cold leg guillotine break is the one with a discharge coefficient of $C_D = 0.4$. The results of the LOCA analysis show that:

- the limiting peak clad temperature = 1942.5°F
- maximum local cladding oxidation level = 4.44%
- total core metal reaction = 0.3%

A sensitivity analysis was performed with the accumulator volume as a parameter for the range 975 ft^3 to 1025 ft^3. For the low accumulator water volume, the peak clad temperature = 1968.1°F, and the maximum local cladding oxidation level = 4.22%. The low accumulator volume is the limiting case. Therefore, for the limiting LOCA transient (double-ended cold leg guillotine break) and assuming conservative initial conditions, analysis based on codes which satisfy the 10 CFR 50 Appendix K requirements showed that the reactor will meet the 10 CFR 50.46 acceptance criteria. Specifically:

1. the calculated peak cladding temperature does not exceed 2,200° F;
2. the localized maximum cladding oxidation does not exceed 17% during or after quenching;
3. cladding chemical interaction with water and steam does not exceed 1.0% of all the metal in the cladding cylinders surrounding the fuel;
4. the core remains amenable to cooling during and after the transient;
5. after the successful initial operation of the ECCS, the calculated core temperature is reduced and the long-term decay heat is removed for an extended period of time;
6. the accumulator water volume in the range of 975 ft^3 per accumulator has no significant impact on the calculated results of the LOCA analysis and the reactor meets the 10 CFR 50.46 acceptance criteria; and
7. the analysis assumed heat flux hot channel factor $F_Q = 2.32$ and enthalpy rise hot channel factor $F_{\Delta H}^N = 1.62$ and the calculational results are conservatively within the range of the acceptance criteria.

EVALUATION

The staff has reviewed the submittal by the licensee regarding the large-break LOCA for Surry Units 1 and 2. The methods and codes used meet the requirement of Appendix K to 10 CFR 50 and the calculated results are conservatively within the acceptance criteria specified in 10 CFR 50.46. Conservative values for the input data were assumed. Parametric studies show that the calculated LOCA results are insensitive for the accumulator capacity from 975 ft^3 to 1,025 ft^3. Similarly, assumed values of $F_Q(z) = 2.32$ and $F_{\Delta H}^N = 1.62$ yield LOCA analysis results within the acceptance criteria. Therefore, we conclude that the results of the large-break LOCA analysis and proposed changes in the TS are acceptable.

ENVIRONMENTAL CONSIDERATION

These amendments involve a change in the installation or use of the facilities components located within the restricted areas as defined in 10 CFR 20. The staff has determined that these amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

CONCLUSION

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

REFERENCES

1. Letter from W. L. Stewart, Virginia Electric and Power Company to USNRC, dated February 23, 1987 (Serial No. 87-001)

2. Letter from Charles E. Rossi, NRC to E. P. Rahe, Westinghouse, "Acceptance for Referencing of Licensing Topical Report WCAP-9561, Addendum 3, Revision 1," dated August 25, 1986.

3. Young, M. Y., "Addendum to BART-A1: A Computer Code for the Best Estimate Analysis of Reload Transients" (Special Report: Thimble Modeling in Westinghouse ECCS Evaluation Model): WCAP-9561-P, Addendum 3, Revision 1, dated July 1986.

4. Letter from J. R. Muller, NRC, to E. P. Hahe, Westinghouse, "Acceptance for Referencing of the 1981 Version of the Westinghouse Large Break ECCS Evaluation Model," December 1, 1981.

5. Letter from C. O. Thomas, NRC to E. P. Rahe, Westinghouse, "Acceptance for Referencing of Licensing Topical Report WCAP-9561, BART A-1: A Computer Code for Best Estimate Analyses of Reflood Transients," dated December 21, 1983.

6. Bordelon, F. M., et al., "Westinghouse ECCS Evaluation Model-Summary," WCAP-8339, dated July 1974.

7. Bordelon, F. M. et al., "SATAN-VI Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant," WCAP-8306, dated June 1974.

8. Bordelon, F. M., et al., "Containment Pressure Analysis Code COCO," WCAP-8326, dated June 1974.

9. Kelly, R. D., et al., "Calculational Model for Core Reflooding After a Loss-of-Coolant Accident (WREFLOOD)," WCAP-8121, dated June 1974.

10. Bordelon, F. M., et al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," WCAP-8305, dated June 1974.

11. Letter from C. M. Stallings, Virginia Electric and Power Company to E. G. Case, NRC, Serial No. 092, dated February 17, 1978.

12. Salvatori, R., "Westinghouse ECCS Sensitivity Studies," WCAP-8356, dated July 1974.

13. Letter from C. M. Stallings, Virginia Electric and Power Company to E. G. Case, NRC, Serial No. 344, August 9, 1977.

14. Letter from H. Denton to PWR Project Managers, "Westinghouse ECCS Evaluation Models," dated October 29, 1986.

Dated: December 10, 1987

Principal Contributor:

L. Lois