January 19, 1977

Dockets Nos.: 50-280
and 50-281

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

Gentlemen:

The Commission has issued the enclosed Amendments No. 28 to Facility Operating Licenses Nos. DPR-32 and DPR-37 for the Surry Power Station, Units Nos. 1 and 2. These amendments consist of changes to the Technical Specifications for each license in response to your application dated September 27, 1976, as supplemented October 19 and 29, November 26, December 15, 1976, and January 3 and 11, 1977.

These amendments and evaluation relate to the replacement of 97 of 157 fuel assemblies in the reactor core of Surry Unit No. 1 constituting refueling of the core for fourth cycle operation, and the emergency core cooling system analysis for an average of 15% of the steam generator tubes plugged in Surry Units Nos. 1 and 2.

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

Sincerely,

Original Signed By

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

Enclosures and cc: See next page

1 const. 1

*SEE PREVIOUS YELLOW FOR CONCURRENCES

OFFICE ➤	ORB#4:DOR	ORB#4:DOR	AD-OT:DOR	OELD	C-ORB#4:DOR
SURNAME ➤	RIngram*	MFairtile	DEisenhut*	*	RReid
DATE ➤	1/ /77	1/ /77	1/ /77	1/ /77	1/ /77

Enclosures:
1. Amendment No. 28 to DPR-32
2. Amendment No. 28 to DPR-37
3. Safety Evaluation
4. Federal Register Notice

cc w/enclosures: See next page

OFFICE ➤						
SURNAME ➤						
DATE ➤						

Virginia Electric & Power Company

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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

cc w/enclosures and incoming
dtd.: 11/26, 12/15, 1/3 & 1/11
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. 28
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 27, 1976, as supplemented October 29, November 26, December 15, 1976, January 3, and January 11, 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.

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: January 19, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 28

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
3.12-13	3.12-13

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 exceed 118% of rated power.

C. The fuel residence time shall be limited to 21,348 effective full power hours(EFPH) for Cycle 4 of Unit 1 and to 6699 EFPH for Cycle 3 of Unit 2.

Basis

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

The fuel residence time is limited to 21,348 EFPH for Cycle 4 of Unit 1 and to 6699 EFPH for Cycle 3 of Unit 2 to assure no fuel clad flattening will occur in the cores without prior review by the NRC staff.

References

1) FSAR Section 3.4
2) FSAR Section 3.3
3) FSAR Section 14.2

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 Unit 1 and 20.4 kw/ft for Unit 2. 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.

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. 28
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 27, 1976, as supplemented October 29, November 26, December 15, 1976, January 3, and January 11, 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.

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: January 19, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 28

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
3.12-13	3.12-13

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 exceed 118% of rated power.

C. The fuel residence time shall be limited to 21,348 effective full power hours(EFPH) for Cycle 4 of Unit 1 and to 6699 EFPH for Cycle 3 of Unit 2.

Basis

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

The fuel residence time is limited to 21,348 EFPH for Cycle 4 of Unit 1 and to 6699 EFPH for Cycle 3 of Unit 2 to assure no fuel clad flattening will occur in the cores without prior review by the NRC staff.

References

1) FSAR Section 3.4

2) FSAR Section 3.3

3) FSAR Section 14.2

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 Unit 1 and 20.4 kw/ft for Unit 2. 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENTS NO. 28 TO LICENSES NOS. DPR-32 AND DPR-37

VIRGINIA ELECTRIC & POWER COMPANY

SURRY POWER STATION UNITS NOS. 1 AND 2

DOCKETS NOS. 50-280 AND 50-281

Introduction

By letter dated September 27, 1976, as supplemented October 19 and 29, November 26, December 15, 1976, January 3 and 11, 1977, Virginia Electric and Power Company (VEPCO) requested amendments to Facility Operating Licenses Nos. DPR-32 and DPR-37. The purpose of the request is to permit the Cycle 4 reload of Surry Unit No. 1, and to provide a reanalysis of the emergency core cooling system (ECCS) with 15% of the steam generator tubes plugged in Surry Units Nos. 1 and 2. There are two safety related sections of this evaluation.

1. Reload Safety Evaluation and

2. Evaluation of ECCS with 15% Steam Generator Tube Plugging.

1. Reload Safety Evaluation

Discussion

On September 27, 1976, VEPCO submitted proposed Technical Specification Change No. 47 to Facility Operating Licenses DPR-32 and DPR-37 for Surry Units Nos. 1 and 2. The September 27, 1976, proposal analyzed the Surry Unit No. 1 Cycle 4 reload core but did not include the emergency core cooling system (ECCS) reanalysis required by the August 27, 1976 Order for Modification of License. The ECCS analysis was submitted by VEPCO on October 29, 1976, as an amendment to proposed Change No. 47. The ECCS analysis, the proposed axial power distribution monitoring system (APDMS) and thermal and hydraulic design were previously reviewed by the NRC staff and License Amendments No. 26 were issued on November 26, 1976, to authorize the return to power of Surry Unit No. 2. This license amendment addresses the remaining items in proposed Change No. 47 which apply to the Surry Unit No. 1 Cycle 4 reload core.

During the Cycle 4 refueling, 97 fuel assemblies of 157, including two region 4 17x17 demonstration assemblies will be replaced with 84 fresh assemblies and 13 previously burned assemblies. Four of the previously burned assemblies were irradiated in Cycle 2 of Surry Unit No. 2. Cycle 4 is planned to be a nominal 18 month cycle and is designed to produce approximately 13,200 MWD/MTU (9,299 EFPH) of energy.

Evaluation

Our evaluation of the Cycle 4 design as it affects safety-related plant parameters and postulated accident analyses is described below, in addition to the items covered in License Amendments No. 26.

Reactor Design

The basic design parameters for Cycle 4 are core average power of 2441 Mwt, system pressure of 2250 psia, coolant mass flow rate of 2.31×10^6 lb/hr-ft^2, average temperature T_{AVG} of 574.4^oF and core average linear power density of 6.2 Kw/ft. These parameters are identical to the Cycle 3 core parameters. The Cycle 4 core will contain 720 fresh borosilicate burnable poison rods and 48 depleted burnable poison rods. Also two unirradiated secondary neutron sources will be activated during Cycle 4. All fuel in Cycle 4 will be in either its first or second cycle of operation.

Mechanical design of the fresh fuel is the same as the original fuel.

Clad flattening time is predicted to be 30,730 EFPH for the most limiting assemblies. These assemblies have now accumulated 9,382 EFPH and Cycle 4 will nominally operate for 9,299 EFPH providing sufficient margin for the duration of Cycle 4. See Table 1 for summary of fuel assembly design parameters.

Thermal and Hydraulic Design

The thermal and hydraulic design was addressed in our license amendments dated November 26, 1976 and is acceptable.

Nuclear Design

A comparison of Cycle 4 core kinetics characteristics with reference analysis limits is shown in Table 2.

All kinetics characteristics except the delayed neutron fraction β and the prompt neutron lifetime are bounded by the current limit which reflects the values used in reference analyses. These parameters are used in the rod ejection accident analysis which is discussed below and in the ECCS analysis which we reanalyzed for Cycle 4.

Table 3 presents control rod worths and shutdown margins for the Cycle 4 core. Sufficient shutdown margins exist for the Cycle 4 core and are conservative with respect to previously approved analyses.

Accident Analysis

For the beginning of cycle rod ejection accident cases, the delayed neutron fraction, β, is below the current limit value. However, the ejected rod worth is also lower than for previous cycles so that the ejected rod reactivity, which is defined as the ratio of rod worth to delayed neutron fraction, is lower than the current limit. The net effect is a conservative ejected rod worth so that it was not necessary to reanalyze the rod ejection accident for beginning-of-cycle. The end-of-cycle rod ejection parameters are within the bounds analyzed for Surry Unit No. 2 Cycle 3, and therefore the end-of-cycle cases are acceptable.

The steamline break accidents were reanalyzed due to a decrease in the integral return to power coefficient. The integral return to power coefficient is the change in the reactivity as a function of power during the cooldown due to steamline breaks and is represented as a combination of the fuel temperature and moderator temperature coefficients integrated over the temperature change.

Four steamline break cases were analyzed:

Case 1 - Complete severance of a pipe outside the containment, downstream of the steam flow measuring nozzle, with the plant initially at end of life (EOL) no load condition, full reactor coolant flow with offsite power available.

Case 2 - Complete severance of a pipe inside the containment at the outlet of the steam generator with the other conditions being the same as Case 1.

Case 3 - Same conditions as Case 1 with the loss of offsite power.

Case 4 - Same conditions as Case 2 with the loss of offsite power.

The same analysis methods were used as were previously used in the current accepted analysis. The analysis included the assumption that upper head temperature was equal to hot leg temperature.

Table 4 shows results of the two most limiting breaks which were Case 2 and Case 4. For all cases analyzed the minimum DNBR was greater than 1.3.

Conclusions

We conclude that the Surry Unit No. 1 Cycle 4 reload core can be operated safely under the provisions of the proposed Change No. 47 to the plant technical specifications and that these changes do not represent a decrease in safety margins.

2. Evaluation of ECCS with 15% Steam Generator Tube Plugging

Discussion

License Amendments No. 26 issued November 26, 1976, for the Surry Power Station Units Nos. 1 and 2, covered the loss of coolant accident (LOCA) reanalysis submitted to us by VEPCO on October 29, 1976, a proposed axial power distribution monitoring system (APDMS), and core thermal and hydraulic data. Our License Amendments No. 26 considered up to 12% of steam generator tubes plugged in the ECCS reanalysis. Since that time the licensee has decided to plug more than 12% of the steam generator tubes. On January 3, 1977, an additional LOCA anlaysis was submitted by VEPCO with 15% of the steam generator tubes assumed plugged. This Safety Evaluation presents our evaluation of the January 3 LOCA analysis with 15% steam generator tubes plugged.

Evaluation

We have determined that all analysis parameters are identical to the ECCS analysis of November 26, 1976, with 12% tubes plugged except that the tube plugging was increased to 15%. VEPCO's analysis results indicate a peak clad temperature of 2120°F, a maximum local metal-water reaction of 6.7 percent, and a total metal-water reaction of less than 0.3 percent at the 15% plugged condition.

Only one large LOCA was considered by VEPCO, namely the previously determined worst case, $C_D = 0.4$ for a double-ended cold leg guillotine break. Table 5 presents the time sequence of events for the previously analyzed case with 12% tubes plugged and the new case with 15% of tubes plugged. Table 6 compares the results of the two cases for the 10 CFR 50.46 parameters.

Conclusions

We conclude that the analysis with 15% steam generator tubes plugged is acceptable and in conformance with paragraph 50.46 of 10 CFR 50 and will permit operation of Surry Units Nos. 1 and 2 within the constraints of the technical specifications.

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

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: January 19, 1977

Table 1

SURRY UNIT 1 - CYCLE 4

FUEL ASSEMBLY DESIGN PARAMETERS*

Region	1	4B	4C	5	6A	6B	6C	S2/4A**
Enrichment (w/o U 235)*	1.87	2.61	3.33	2.11	2.62	2.60	2.90	2.61
Density (% Theoretical)*	93.6	94.6	94.4	94.6	94.5	95.0	95.0	94.4
Number of Assemblies	1	8	52	8	24	8	52	4
Approximate Burnup at Beginning of Cycle 4 (MWD/MTU)	15,240	8,690	13,560	10,380	0	0	0	9,890
Conservative Estimate of Burnup at End of Cycle 4 (MWD/MTU)	28,430	23,170	29,930	24,980	16,950	15,220	11,280	26,380

*All regions except Regions 6B and 6C are as-built values; Regions 6B and 6C reflect the nominal values; however, an average density of 94.5% theoretical was used in thermal evaluations. The conservative estimate of end-of-cycle burnup is based on Cycle 3 end-of-cycle burnup of 9,500 MWD/MTU and on Cycle 4 end-of-cycle burnup of 14,200 MWD/MTU.

** Fuel transferred from Region 4A of Surry Unit No. 2.

Table 2

SURRY UNIT 1 - CYCLE 4

KINETICS CHARACTERISTICS

	Current Limit	Cycle 4
Moderator Temperature Coefficient (pcm/^{o}F)**	+3.0* to -35	-0 to -35
Least Negative Doppler - Only Power Coefficient, Zero to Full Power (pcm/percent power)	-11.5 to -6.0	-12.2 to -8.4
Delayed Neutron Fraction (percent)	.60 to .50	.59 to .50
Prompt Neutron Lifetime (μ sec)	26	19
Maximum Positive Reactivity Insertion Rate from Subcritical (pcm/sec)	65	63

*The moderator temperature coefficient may be positive up to full power according to the following program:

+3.0 pcm/^{o}F from 0 to 50% power and is linearly ramped down to 0.0 pcm/^{o}F from 50 to 100% power (see Reference 1).

**pcm = 10^{-5} $\Delta K/K$

Table 3

SURRY UNIT 1 - CYCLE 4

SHUTDOWN REQUIREMENTS AND MARGINS

	Cycle 4 BOC	Cycle 4 EOC
Control Rod Worth (% $\Delta\rho$)		
All Rods Inserted Less Most Reactive Stuck Rod	7.51	7.99
(1) Less 10% Uncertainty	6.76	7.19
Control Rod Requirements (% $\Delta\rho$)		
Reactivity Defects (i.e., Doppler, Moderator, Void, and Redistribution)	1.83	2.83
Rod Insertion Allowance	0.90	0.90
(2) Total Requirements	2.73	3.73
Shutdown Margin {(1)-(2)} (% $\Delta\rho$)	4.03	3.46
Required Shutdown Margin (% $\Delta\rho$)	1.00	1.77

Table 4

STEAMLINE BREAK REANALYSIS RESULTS

	Cycle 3		Cycle 4	
Peak Core Average Power, %	15.8*	8.5**	28.6*	9.2**
Reactor Inlet Temp., Failed Loop, °F	370	332	373	347
Reactor Inlet Temp., Intact Loops, °F	497	519	502	521
Reactor Coolant Pressure, psia	794	875	1167	1215
Reactor Coolant Flow, % of Nominal	100	24.	100	23.
Min. DNBR	>1.3	>1.3	>1.3	>1.3

*Inside Break (Case 2) with Power.

**Inside Break (Case 4) without Power.

TABLE 5

TIME SEQUENCE OF EVENTS

	12% Plugging	15% Plugging
START	0.0 (sec.)	0.0 (sec.)
Reactor Trip Signal	0.648	0.648
S. I. Signal	2.23	2.23
Acc. Injection	16.2	16.1
End of Bypass	24.26	24.05
End of Blowdown	27.81	26.97
Bottom of Core Recovery	37.88	37.66
Acc. Empty	55.99	55.71
Pump Injection	27.23	27.23

TABLE 6

ANALYSIS RESULTS

	12% Plugging	15% Plugging
Peak Clad Temp., ^{o}F	2107	2120
Peak Clad Location, ft.	9.0	9.0
Local Zr/H_2O Rxn (max), %	6.234	6.704
Local Zr/H_2O Location ft.	9.0	9.0
Total Zr/H_2O Rxn, %	<0.3	<0.3
Hot Rod Burst Time, sec.	28.2	28.6
Hot Rod Burst Location, Ft.	6.0	6.25

Initial Conditions	
Core Power, Mwt, 102% of	2441
Peak Linear Power, kw/ft 102% of	12.74
Peaking Factor	2.00
Accumulator Water Volume (ft^3)	1075 (per accumulator)

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS NOS. 50-280 AND 50-281

VIRGINIA ELECTRIC & POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments No. 28 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 Station 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 fourth cycle operation for Surry Unit No. 1 and modify clad flattening limitations and consider the emergency core cooling system analysis for an average of 15% of the steam generator tubes plugged in Surry Units Nos. 1 and 2.

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 licensee's filings dated September 27, 1976, as supplemented October 19 and 29, November 26, December 15, 1976, January 3, and January 11, 1977, (2) Amendments No. 28 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 19th day of January 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

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