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

March 14, 1983

Posted
Amdt. 87
to DPR 37

Docket Nos. 50-280
and 50-281

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

Dear Mr. Stewart:

The Commission has issued the enclosed Amendment No. 86 to Facility Operating License No. DPR-32 and Amendment No. 87 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 November 22, 1982.

These amendments revise the Technical Specifications to restore the core thermal limits and overtemperature and overpower ΔT setpoints to values consistent with 100% of thermal design flow.

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

Sincerely,

Joseph D. Neighbors

Joseph D. Neighbors, Project Manager
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 86 to DPR-32
2. Amendment No. 87 to DPR-37
3. Safety Evaluation
4. Notice of Issuance

cc w/enclosures:
See next page

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

cc: Mr. Michael W. Maupin
Hunton and Williams
Post Office Box 1535
Richmond, Virginia 23213

Mr. J. L. Wilson, Manager
P. O. Box 315
Surry, Virginia 23883

Donald J. Burke, Resident Inspector
Surry Power Station
U. S. Nuclear Regulatory Commission
Post Office Box 166
Route 1
Surry, Virginia 23883

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

Attorney General
1101 East Broad Street
Richmond, Virginia 23219

Mr. James R. Wittine
Commonwealth of Virginia
State Corporation Commission
Post Office Box 1197
Richmond, Virginia 23209

Regional Radiation Representative
EPA Region III
Curtis Building - 6th Floor
6th and Walnut Streets
Philadelphia, Pennsylvania 19106

Mr. J. H. Ferguson
Executive Vice President - Power
Virginia Electric and Power Company
Post Office Box 26666
Richmond, Virginia 23261

James P. O'Reilly
Regional Administrator - Region II
U. S. Nuclear Regulatory Commission
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

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. 86
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 November 22, 1982, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 86 , 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

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 14, 1983

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. 87
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 November 22, 1982, 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. 87 , 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

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 14, 1983

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 86 TO FACILITY OPERATING LICENSE NO. DPR-32

AMENDMENT NO. 87 TO FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NOS. 50-280 AND 50-281

Revise Appendix A as follows:

Remove Pages	Insert Pages
2.1-1	2.1-1
2.1-3	2.1-3
Figure 2.1-1	Figure 2.1-1
2.3-2	2.3-2
2.3-3	2.3-3
2.3-5	2.3-5
6.5-3	6.5-3

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMIT, REACTOR CORE

Applicability

Applies to the limiting combinations of thermal power, Reactor Coolant System pressure, coolant temperature and coolant flow when a reactor is critical.

Objective

To maintain the integrity of the fuel cladding.

Specification

A. The combination of reactor thermal power level, coolant pressure, and coolant temperature shall not:

1. Exceed the limits shown in TS Figure 2.1-1 when full flow from three reactor coolant pumps exist.
2. Exceed the limits shown in TS Figure 2.1-2 when full flow from two reactor coolant pumps exist and the reactor coolant loop stop valves in the non-operating loop are open.
3. Exceed the limits shown in TS Figure 2.1-3 when full flow from two reactor coolant pumps exist and the reactor coolant loop stop valves in the non-operating loop are closed.

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

The curves of TS Figure 2.1-1 which show the allowable power level decreasing with increasing temperature at selected pressures for constant flow (three loop operation) represent limits equal to, or more conservative than, the loci of points of thermal power, coolant system average temperature, and coolant system pressure for which the DNB ratio is equal to 1.30 or the average enthalpy at the exit of the core is equal to the saturation value. The area where clad integrity is assured is below these lines. The temperature limits are considerably more conservative than would be required if they were based upon a minimum DNB ratio of 1.30 alone but are such that the plant conditions required to violate the limits are precluded by the self-actuated safety valves on the steam generators. The three loop operation safety limit curve has been revised to allow for heat flux peaking effects due to fuel densification and to apply to 100% of design flow. The effects of rod bowing are also considered in the DNBR analyses.

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

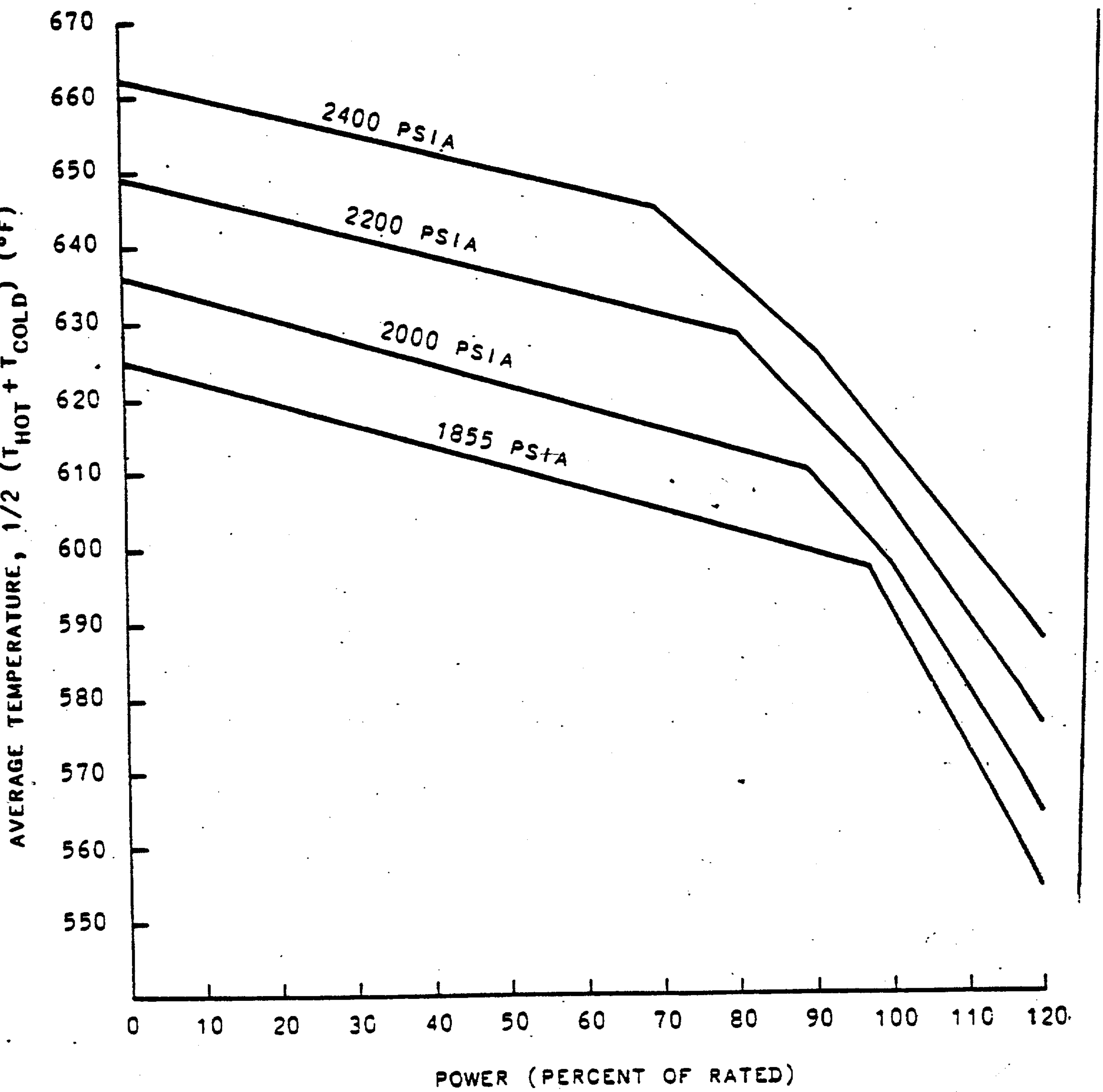

FIGURE 2.1-1 REACTOR CORE THERMAL & HYDRAULIC SAFETY LIMITS-
THREE LOOP OPERATION, 100% FLOW

(b) High pressurizer pressure - $\leq$ 2385 psig.

(c) Low pressurizer pressure - $\geq$ 1860 psig.

(d) Overtemperature ΔT

$$\Delta T \leq \Delta T_o \left[K_1 - K_2\left(\frac{1 + \tau_1 S}{1 + \tau_2 S}\right)(T - T') + K_3(P - P') - f(\Delta I)\right]$$

where

ΔT_o = Indicated ΔT at rated thermal power, °F

T = Average coolant temperature, °F

T' = 574.4°F

P = Pressurizer pressure, psig

P' = 2235 psig

K_1 = 1.12

K_2 = 0.01012

K_3 = 0.000554 for 3-loop operation

K_1 = 0.951

K_2 = 0.01012 for 2-loop operation with loop stop

K_3 = 0.000554 valves open in inoperable loop

K_1 = 1.026

K_2 = 0.01012 for 2-loop operation with loop stop

K_3 = 0.000554 valves closed in inoperable loop

$\Delta I = q_t - q_b$, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of rated power

$f(\Delta I)$ = function of ΔI, percent of rated core power as shown in Figure 2.3-1

τ_1 = 25 seconds

τ_2 = 3 seconds

(e) Overpower ΔT

$$\Delta T \leq \Delta T_o \left[K_4 - K_5\left(\frac{\tau_3 S}{1 + \tau_3 S}\right) T - K_6 (T - T') - f(\Delta I)\right]$$

where

ΔT_o = Indicated ΔT at rated thermal power, °F

T = Average coolant temperature, °F

T' = Average coolant temperature measured at nominal conditions and rated power, °F

K_4 = A constant = 1.09

K_5 = 0 for decreasing average temperature

A constant, for increasing average temperature 0.02/°F

K_6 = 0 for $T \leq T'$

= 0.00108 for $T > T'$

$f(\Delta I)$ as defined in (d) above,

τ_3 = 10 seconds

(f) Low reactor coolant loop flow - ≥90% of normal indicated loop flow as measured at elbow taps in each loop

(g) Low reactor coolant pump motor frequency - ≥57.5 Hz

(h) Reactor coolant pump under voltage - ≥70% of normal voltage

3. Other reactor trip settings

(a) High pressurizer water level - ≤92% of span

(b) Low-low steam generator water level - ≥5% of narrow range instrument span

(c) Low steam generator water level - ≥15% of narrow range instrument span in coincidence with steam/feedwater mismatch flow - $\leq 1.0 \times 10^6$ lbs/hr

(d) Turbine trip

(e) Safety injection - Trip settings for Safety Injection are detailed in TS Section 3.7.

and source range high flux, high setpoint trips provide additional protection against uncontrolled startup excursions. As power level increases, during startup, these trips are blocked to prevent unnecessary plant trips.

The high and low pressurizer pressure reactor trips limit the pressure range in which reactor operation is permitted. The high pressurizer pressure reactor trip is also a backup to the pressurizer code safety valves for overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The low pressurizer pressure reactor trip also trips the reactor in the unlikely event of a loss-of-coolant accident.(3)

The overtemperature ΔT reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided only that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 3 seconds), and pressure is within the range between high and low pressure reactor trips. With normal axial power distribution, the reactor trip limit, with allowance for errors, (2) is always below the core safety limit as shown on TS Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor limit is automatically reduced.(4)(5)

The overpower and overtemperature protection system setpoints have been revised to include effects of fuel densification on core safety limits and to apply to 100% of design flow. The revised setpoints in the Technical Specifications will ensure that the combination of power, temperature, and pressure will not exceed the revised

9. Records of the service lives of all hydraulic and mechanical snubbers listed on Tables 4.17-1, 4.17-2, including the date at which the service life commences and associated installation and maintenance records.

10. Records of the annual audit of the Station Emergency Plan and implementing procedures.

11. Records of the annual audit of the Station Security Plan and implementing procedures.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 86 TO FACILITY OPERATING LICENSE NO. DPR-32

AND AMENDMENT NO. 87 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 November 22, 1982, the Virginia Electric and Power Company (the licensee) requested an amendment to Operating License Nos. DPR-32 and DPR-37 for the Surry Power Station, Unit Nos. 1 and 2. The amendment would revise the Technical Specifications to allow the core thermal limits and overtemperature and overpower ΔT setpoints to be restored to values corresponding to 100% thermal design flow.

Technical Specification page 6.5-3 is added to this amendment because it was inadvertently omitted from Amendment Nos. 82 and 83 dated March 1, 1983. This page pertains to retention of records and was submitted February 18, 1983, as a supplement to the licensee's application dated December 1, 1982.

Discussion

In a letter dated August 9, 1977, (Ref. 1), the licensee provided the justification for operation of Surry Units 1 and 2 with substantial steam generator tube plugging. It addressed the impact on non-LOCA accident analyses of steam generator tube plugging of up to 40% of the tubes with consequent Reactor Coolant System (RCS) flow reductions to as low as 90% of the thermal design flow rate considered in the Final Safety Analysis Report (FSAR). A revised set of core thermal operating limits and corresponding overtemperature and overpower ΔT setpoints consistent with the assumption of 90% of RCS design flow were also submitted. The safety analyses presented in Reference 1 were based on these low flow setpoints and were approved by the NRC on December 2, 1977.

From 1979 to 1980, the licensee undertook an extensive steam generator repair program resulting in total replacement of the steam generator tube bundle for both Units 1 and 2. The licensee indicates that the startup measurements for subsequent cycles have confirmed that RCS flow rates for both units are well in excess of the 100% design value. Steam generator performance following the tube bundle replacement program has been good. No tube plugging has been required to date. However, since the completion of the tube bundle repair program, the licensee has retained the conservative setpoint limits as provided in Reference 1.

Evaluation

The revised limits and setpoints contained in the proposed Technical Specification changes (Ref. 2) are consistent with the FSAR assumption of 100% of thermal design flow, and are identical to those provided in Reference 3. These were the previously applicable limits and setpoints in effect prior to the Reference 1 submittal. Since the proposed limits and setpoints are identical to those previously analyzed and have been checked for current fuel cycle conditions, and since the thermal design flows have been experimentally verified, operation with the proposed Technical Specification changes will not invalidate any existing safety analyses for Surry Units 1 and 2.

We have reviewed the core thermal limits and overtemperature and overpower ΔT setpoints provided in References 2 and 3 and concur with the licensee's assessment that these limits and setpoints are identical and therefore no accident reanalysis is required for the proposed changes in the Technical Specifications.

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 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 an accident previously evaluated, do not create the possibility of an accident of a type different from any evaluated previously, and do not involve a significant reduction in a margin of safety, 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 the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: March 14, 1983

Principal Contributors:
V. Leung
G. Schwenk

REFERENCE

1. Letter from L. M. Stallings (VEPCO) to E. G. Case (NRC), "Amendment to the Operating License, Technical Specifications Change No. 57," dated August 9, 1977.

2. Letter from W. L. Stewart (VEPCO) to H. R. Denton (NRC), "Amendment to Operating License DPR-32 and DPR-37, Proposed Technical Specification Changes," dated November 22, 1982.

3. Letter from L. M. Stallings (VEPCO) to K. R. Goller (NRC), "Amendment to Operating License DPR-32 and DPR-37, Technical Specifications Changes No. 27," dated March 12, 1975.

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

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 86 to Facility Operating License No. DPR-32 and Amendment No. 87 to Facility Operating License No. DPR-37 issued to Virginia Electric and Power Company (the licensee), which revised Technical Specifications for operation of the Surry Power Station, Unit Nos. 1 and 2, respectively, (the facilities), located in Surry County, Virginia. The amendments are effective as of the date of issuance.

The amendments revise the Technical Specifications to restore the core thermal limits and overtemperature and overpower ΔT setpoints to values consistent with 100% of thermal design flow.

The application for 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 these amendments do not involve a significant hazards consideration.

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

For further details with respect to this action, see (1) the application for amendments dated November 22, 1982, (2) Amendment Nos. 86 and 87 to License 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 23185. 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 Licensing.

Dated at Bethesda, Maryland, this 14th day of March, 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing