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Docket Nos. 50-280
and 50-281

Virginia Electric & Power Company
ATTN: Mr. Stanley Ragone
Senior Vice President
Post Office Box 26666
Richmond, Virginia 23261

Gentlemen:

The Commission has issued the enclosed Amendments No. 6 to Facility Licenses No. DPR-32 and DPR-37 for the Surry Power Station, Units 1 and 2. The amendments include Change No. 21 to your Technical Specifications for each license and are in response to your request dated March 12, 1975, as supplemented April 9, 1975.

The amendments revise the provisions in the Technical Specifications relating to the replacement of 84 of 157 fuel assemblies in the reactor core, constituting refueling of the core for second cycle operation of Unit 2.

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

Sincerely,

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

Enclosures:
1. Amendment No. 6 to DPR-32
2. Amendment No. 6 to DPR-37
3. Safety Evaluation
4. Federal Register Notice

cc w/enclosures:
See next page

OI&E(3)
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BJones(4)
JMcGough
JSaltzman
BScharf(15)
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Mr. Robert Blanco
Environmental Protection Agency
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6th and Walnut Street
Philadelphia, Pennsylvania 19106

VIRGINIA ELECTRIC & POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 6
License No. DPR-37

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

   A. The application for amendment by Virginia Electric & Power Company (the licensee) dated March 12, 1975, as supplemented April 9, 1975, 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; and

   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.

2. Accordingly, the license is amended by a change 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:

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"3.B Technical Specifications

The Technical Specifications contained in Appendix A, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 21."

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

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By
Roger S. Boyd

A. Giambusso, Director
Division of Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Change No. 21 to the
Technical Specifications

Date of Issuance: JUN 10 1975

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ATTACHMENT TO LICENSE AMENDMENT NO. 6

CHANGE NO. 21 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NO. 50-281

Revise Appendix A as follows:

Remove Page	Insert New Page
2.1-1	2.1-1
2.1-2	2.1-2
2.1-3	2.1-3
2.1-4	2.1-4
2.1-5	2.1-5
2.1-6	2.1-6
Figure 2.1-1A	Figure 2.1-1
Figure 2.1-1B	Blank Page - Figure 2.1-1B
Figure 2.1-2A	Figure 2.1-2
Figure 2.1-2B	Blank Page - Figure 2.1-2B
Figure 2.1-3A	Figure 2.1-3
Figure 2.1-3B	Blank Page - Figure 2.1-3B
2.3-2	2.3-2
2.3-3	2.3-3
2.3-4	2.3-4
2.3-5	2.3-5
2.3-6	2.3-6
2.3-7	2.3-7
2.3-8	2.3-8
Figure 3.12-1B	Figure 3.12-1B
Figure 3.12-4B	Figure 3.12-4B

OFFICE➤						
SURNAME➤						
DATE➤						

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.

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
2.1-3; or the core thermal power exceeds 118% of rated power.

C. The fuel residence time shall be limited to 26,000 effective full power hours (EFPH) for Cycles 1 and 2 of Unit 1 and to 17,000 EFPH for Cycles 1 and 2 of Unit 2.

D. Unit 2 shall not be operated at power levels exceeding those 21
required for low power physics tests until the Technical Specifications for Unit 2 are changed to incorporate the Emergency Core Cooling Final Acceptance Criteria.

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 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 21
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.

The curves of TS Figures 2.1-2 and 2.1-3 which show the allowable power level 21
decreasing with increasing temperature at selected pressures for constant flow (two 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 either the DNB ratio is equal to 1.30 or the average enthalpy at the exit of the core is equal to the saturation value. At low pressures or high temperatures the average enthalpy at the exit of the core reaches saturation before the DNB ratio reaches 1.30 and, thus, this arbitrary limit is conservative with respect to maintaining clad integrity. The plant conditions required to violate these limits are precluded by the protection system and the self-actuated safety valves on the steam generator. Upper limits of 70% power for loop stop valves open and 75% with loop stop valves closed are shown to completely bound the area where clad integrity is assured. These latter limits are arbitrary but cannot be reached due to the Permissive 8 protection system setpoint which will trip the reactor on high nuclear flux when only two reactor coolant pumps are in service.

Operation with natural circulation or with only one loop in service is not allowed since the plant is not designed for continuous operation with less than two loops in service.

TS Figures 2.1-1 through 2.1-3 are based on a $F^N_{\Delta H}$ of 1.55, a 1.55 cosine axial flux shape and a DNB analysis as described in Section 4.3 of the report Fuel Densification Surry Power Station, Unit 1 dated December 6, 1972 (including the effects of fuel densification). They also include an allowance for an increase in the enthalpy rise hot channel factor based on the expression: $F^N_{\Delta H} = 1.55\ (1 + 0.2\ (1-P))$ where P is fraction of rated power. 21

These hot channel factors are higher than those calculated at full power over the range between that of all control rod assemblies fully withdrawn to maximum allowable control rod assembly insertion. The control rod assembly insertion limits are covered by Specification 3.12. Adverse power distribution factors could occur at lower power levels because additional control rod assemblies are in the core; however, the control rod assembly insertion limits dictated by TS Figures 3.12-1A (Unit 1) and 3.12-1B (Unit 2) ensure that the DNBR is always greater at partial power than at full power.

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions for Reactor Coolant System temperature, pressure and thermal power level that would result in a DNB ratio of less than 1.30[3] based on steady state nominal operating power levels less than or equal to 100%, steady state nominal operating Reactor Coolant System average temperatures less than or equal to 574.4°F and a steady state nominal
operating pressure of 2235 psig. 21
Allowances are made in initial conditions assumed for transient analyses for steady state errors of +2% in power, +4°F in Reactor Coolant System average temperature and ±30 psi in pressure. The combined steady state errors result in the DNB ratio at the start of a transient being 10 per cent less than the value at nominal full power operating conditions. The steady state nominal operating parameters and allowances for steady state errors given above are also applicable for two loop operation except that the steady state nominal operating power level is less than or equal to 60%.

The fuel overpower design limit is 118% of rated power. The overpower limit criterion is that core power be prevented from reaching a value at which fuel pellet melting would occur. The value of 118% power allows substantial margin 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 26,000 EFPH for Cycles 1 and 2 of Unit 1 and to 17,000 EFPH for Cycles 1 and 2 of Unit 2 to assure no fuel clad flattening will occur in the cores without prior review by the Regulatory Staff. 21

The limit on reactor operation to low power physics tests (paragraph 2.1.D) is temporarily required until the provisions of the Emergency Core Cooling System Final Acceptance Criteria can be incorporated into the Technical Specifications. 21

References

(1) FSAR Section 3.4

(2) FSAR Section 3.3

(3) FSAR Section 14.2

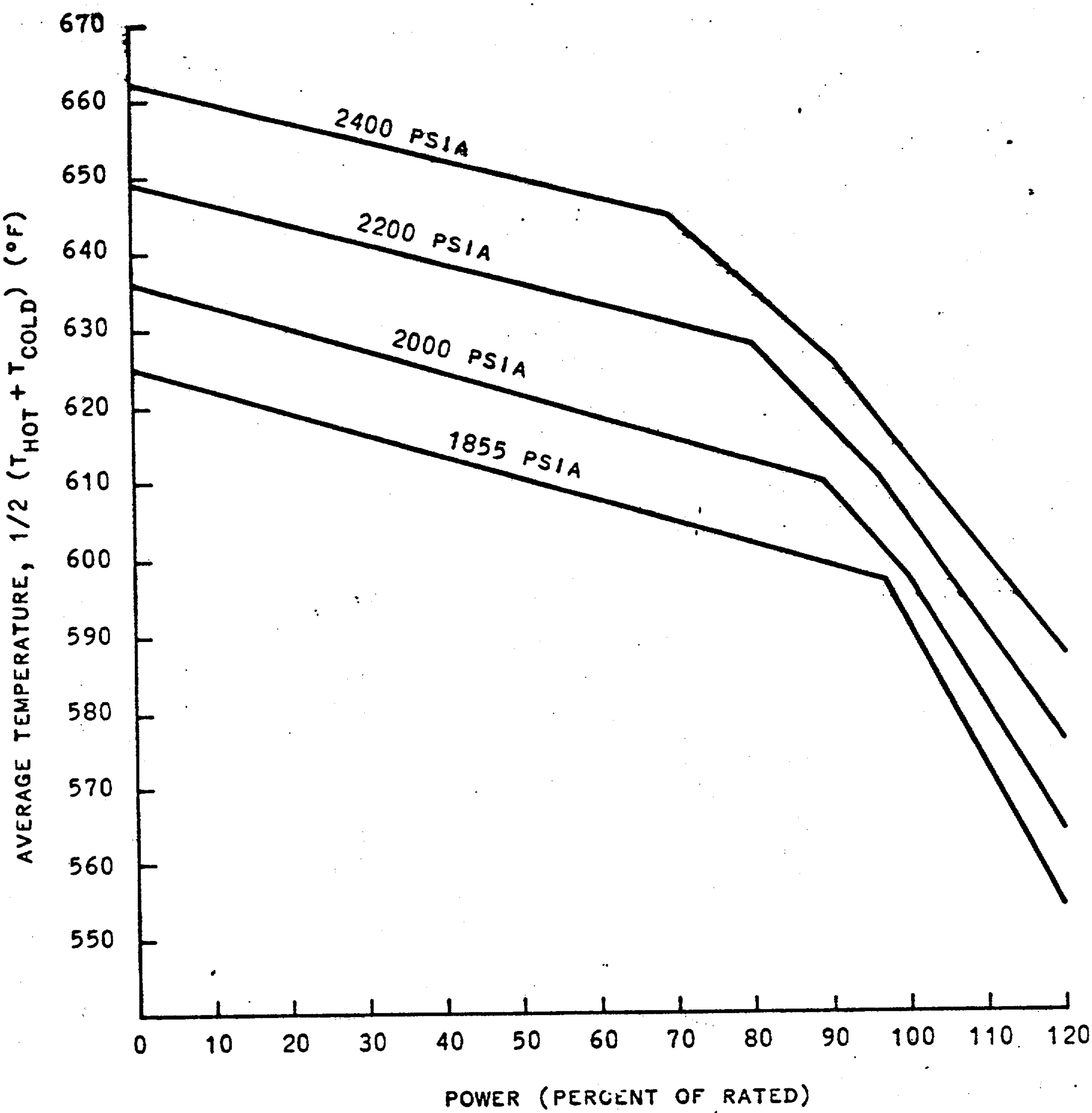

FIGURE 2.1-1 REACTOR CORE THERMAL & HYDRAULIC SAFETY LIMITS-
THREE LOOP OPERATION, 100% FLOW

TS Figure 2.1-1B

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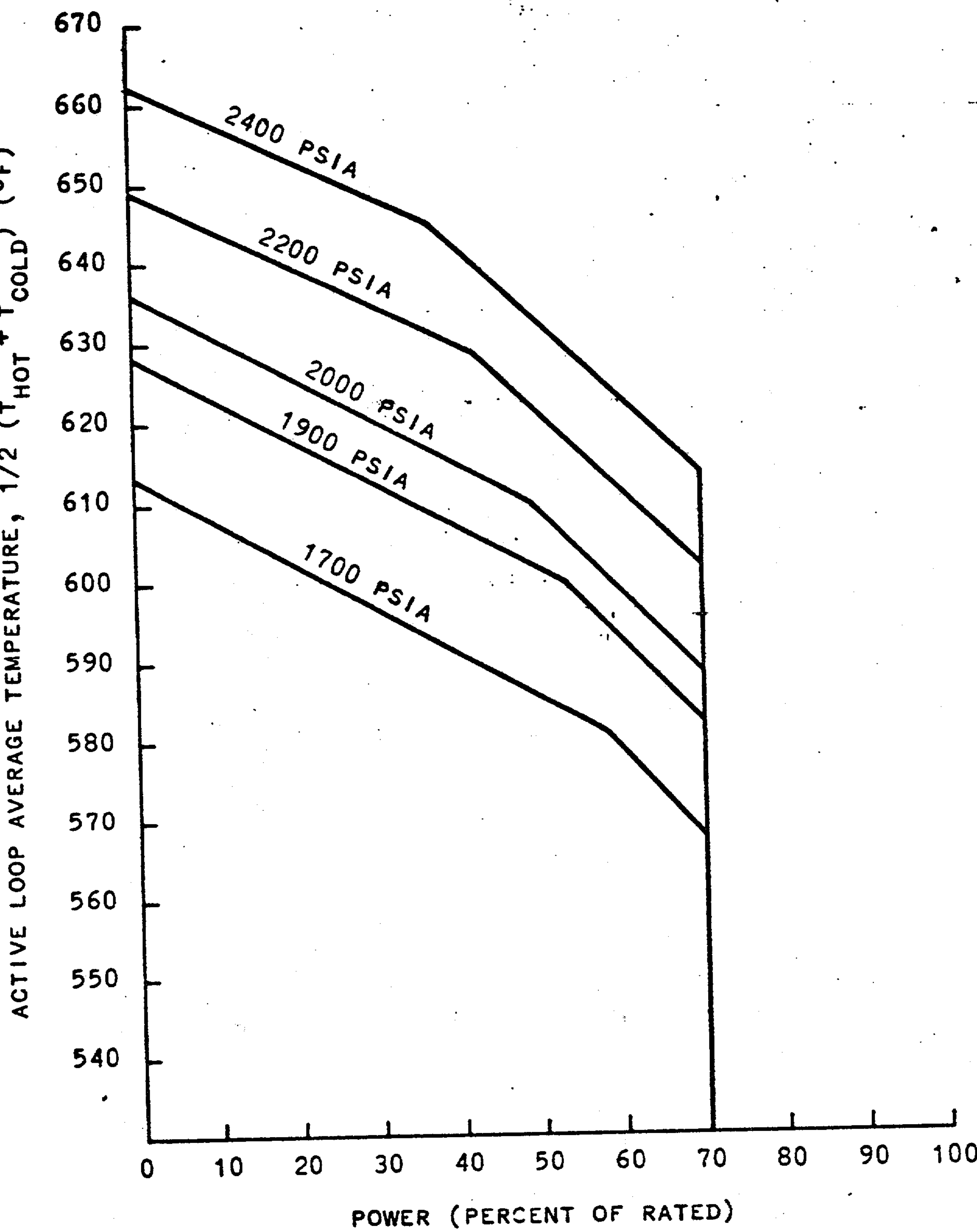

FIGURE 2.1-2 REACTOR CORE THERMAL AND HYDRAULIC SAFETY LIMITS, TWO LOOP OPERATION, LOOP STOP VALVES OPEN

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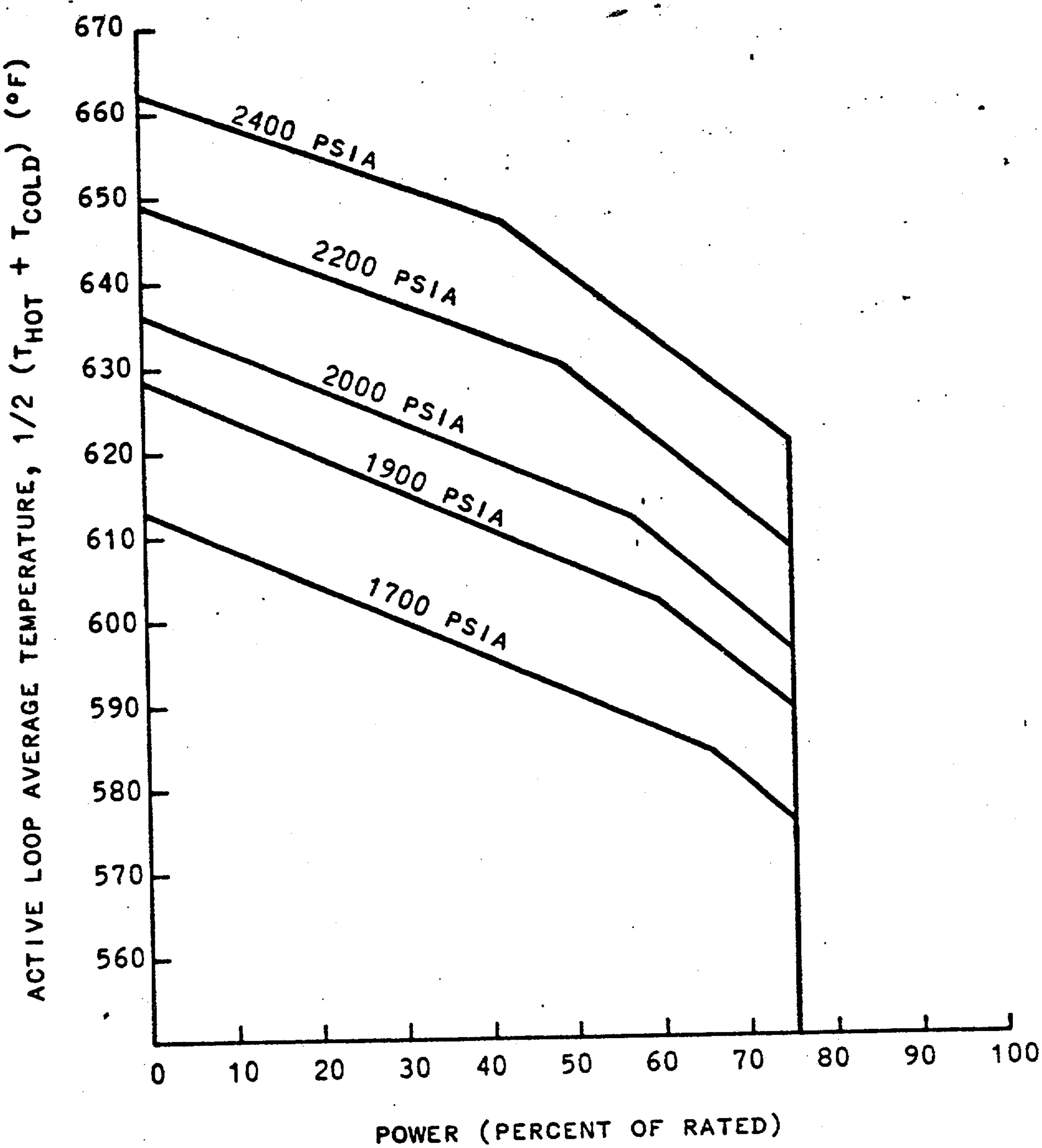

FIGURE 2.1-3 REACTOR CORE THERMAL AND HYDRAULIC SAFETY LIMITS, TWO LOOP OPERATION, LOOP STOP VALVES CLOSED

TS Figure 2.1-3B

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

(c) Low pressurizer pressure - $\geq$1860 psig 21

(d) Overtemperature ΔT

$$\Delta T \leq \Delta T_o \left(K_1 - K_2 (T - T') + K_3 (P - P') - f(\Delta I)\right)$$

where

ΔT_o = Indicated ΔT at rated thermal power, ^oF

T = Average coolant temperature, ^oF

T' = 574.4^oF 21

P = Pressurizer pressure, psig

P' = 2235 psig

K_1 = 1.12

K_2 = 0.01012 for 3-loop operation

K_3 = 0.000554

K_1 = 0.951

K_2 = 0.01012 for 2-loop operation with loop stop 21

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

(e) Overpower ΔT

$$\Delta T \leq \Delta T_o \left(K_4 - K_5 \frac{dT}{dt} - K_6 (T - T') - f(\Delta I)\right)$$

where

ΔT_o = Indicated ΔT at rated thermal power, ^oF

T = Average coolant temperature, ^oF

T' = Average coolant temperature measured at nominal conditions and rated power, ^oF

K_4 = A constant = 1.09

K_5 = 0 for decreasing average temperature

A constant, for increasing average temperature, 0.2 sec/^oF | 21

K_6 = 0 for $T \leq T'$

= 0.00108 for $T > T'$

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

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

(g) Low reactor coolant pump motor frequency - $\geq$57.5 Hz

(h) Reactor coolant pump under voltage - $\geq$70% of normal voltage

3. Other reactor trip setting

(a) High pressurizer water level - $\leq$92% of span

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

(c) Low steam generator water level - $\geq$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.

B. Protective instrumentation settings for reactor trip interlocks shall be as follows:

1. The reactor trip on low pressurizer pressure, high pressurizer level, turbine trip, and low reactor coolant flow for two or more loops shall be unblocked when power $\geq$10% of rated power.

2. The single loop loss of flow reactor trip shall be unblocked when the power range nuclear flux $\geq$50% of rated power. During two loop operation with the loop stop valves in the inactive loop open, this blocking setpoint, established by Permissive 8, may be increased to 60% of rated power only after the overtemperature ΔT setpoint is adjusted to the mandatory two loop value. For two loop operation with the loop stop valves of the inactive loop closed, Permissive 8 may be increased to 65% of rated power only after the overtemperature ΔT setpoint is adjusted to the mandatory value for this condition.

3. The power range high flux, low setpoint trip and the intermediate range high flux, high setpoint trip shall be unblocked when power $\leq$10% of rated power.

4. The source range high flux, high setpoint trip shall be unblocked when the intermediate range nuclear flux is $\leq 5 \times 10^{-11}$ amperes.

Basis

The power range reactor trip low setpoint provides protection in the power range for a power excursion beginning from low power. This trip value was used in the safety analysis.[1] The intermediate range high flux, low setpoint 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 trip 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. The revised setpoints in the Technical Specifications will ensure that the combination of power, temperature, and pressure will not exceed the revised core safety limits as shown in Figures 2.1-1 through 2.1-3. The reactor is prevented from reaching the overpower limit condition by action of the nuclear overpower and overpower ΔT trips. The overpower limit criteria is that core power be prevented from reaching a value at which fuel pellet centerline melting would occur. The overpower protection system set points include the effects of fuel densification. 21

In order to operate with a reactor coolant loop out of service (two-loop operation) and with the stop valves of the inactive loop either open or closed, the overtemperature ΔT trip setpoint calculation has to be modified by the adjustment of the variable K_1. This adjustment, based on limits of two-loop operation, provides sufficient margin to DNB for the aforementioned transients during two loop operation. The required adjustment and subsequent mandatory calibrations are made in the protective system racks by qualified technicians* in the same manner as adjustments before initial startup and normal calibrations for three-loop operation. 21

The overpower ΔT reactor trip prevents power density anywhere in the core from exceeding 118% of design power density as discussed Section 7 and specified in Section 14.2.2 of the FSAR and includes corrections for axial power distribution, change in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The specified setpoints meet this requirement and include allowance for instrument errors.[2]

*As used here, a qualified technician means a technician who meets the requirements of ANS-3. He shall have a minimum of two years of working experience in his speciality and at least one year of related technical training.

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The low flow reactor trip protects the core against DNB in the event of a sudden loss of power to one or more reactor coolant pumps. The setpoint specified is consistent with the value used in the accident analysis.[6] The underfrequency reactor coolant pump trip protects against a decrease in flow caused by low electrical frequency. The specified setpoint assures a reactor trip signal before the low flow trip point is reached.

The high pressurizer water level reactor trip protects the pressurizer safety valves against water relief. Approximately 1154 ft^3 of water corresponds to 92% of span. The specified setpoint allows margin for instrument error[7] and transient level overshoot beyond this trip setting so that the trip function prevents the water level from reaching the safety valves.

The low-low steam generator water level reactor trip protects against loss of feedwater flow accidents. The specified setpoint assures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays for the Auxiliary Feedwater System.[7]

The specified reactor trips are blocked at low power where they are not required for protection and would otherwise interfere with normal unit operations. The prescribed setpoint above which these trips are unblocked assures their availability in the power range where needed.

Above 10% power, an automatic reactor trip will occur if two or more reactor coolant pumps are lost. Above 50% power during three-loop operation, an automatic reactor trip will occur if any pump is lost or de-energized. This latter trip will prevent the minimum value of the DNBR from going below 1.30 during normal operational transients and anticipated transients when only two loops are in operation and the overtemperature ΔT trip setpoint is adjusted to the valve specified for three-loop operation. During two-loop operation with the loop stop valves in the inactive loop open, and the overtemperature ΔT trip setpoint is adjusted to the value specified for this condition, a reactor trip at 60% power will prevent the minimum value of DNBR from going below 1.30 during normal operational transients and anticipated transients when only two loops are in operation. During two-loop operation with the inactive loop stop valves closed and the overtemperature ΔT trip setpoint is adjusted to the value specified for this condition, a reactor trip at 65% power will prevent the minimum DNBR from going below 1.30 during normal operational transients and anticipated transients.

Although not necessary for core protection other reactor trips provide additional protection. The steam/feedwater flow mismatch is coincidence with a low steam generator water level is designed for protection from a sudden loss of the reactor's heat sink. Upon the actuation of the safety injection circuitry, the reactor is tripped to decrease the severity of the accident condition. Upon turbine trip, at greater than 10% power, the reactor is tripped to reduce the severity of the ensuing transient.

References

(1) FSAR Section 14.2.1

(2) FSAR Section 14.2

(3) FSAR Section 14.5

(4) FSAR Section 7.2

(5) FSAR Section 3.2.2

(6) FSAR Section 14.2.9

(7) FSAR Section 7.2

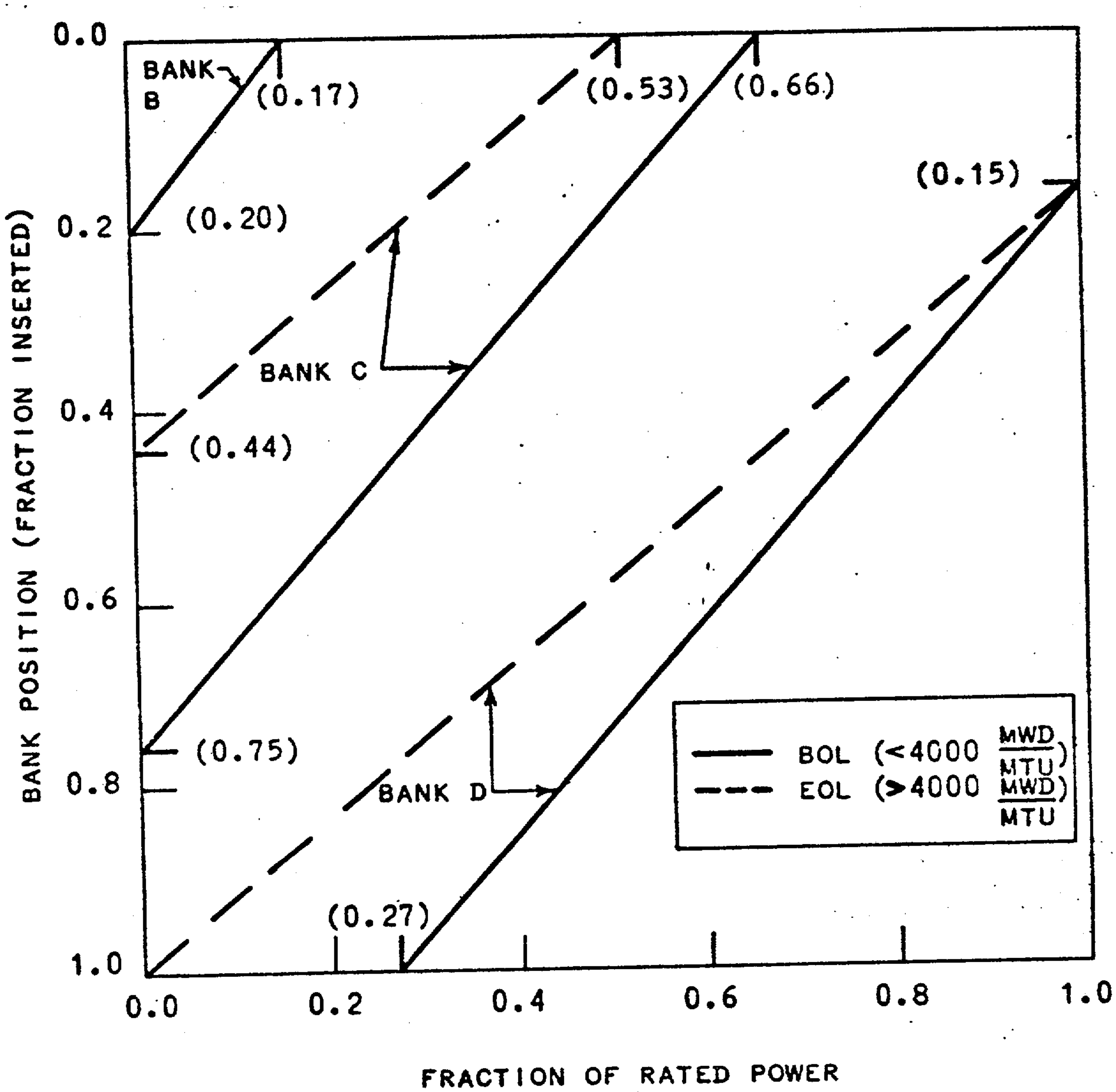
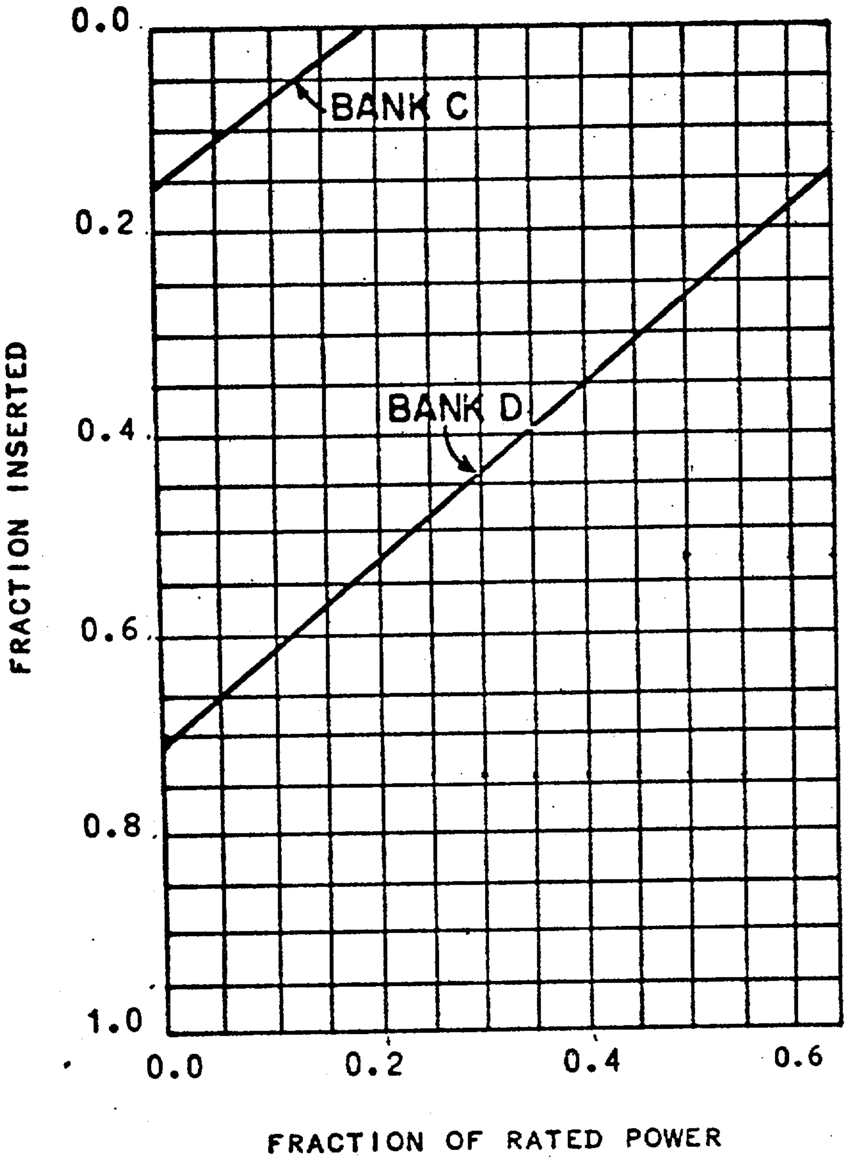


FIGURE 3.12-1B CONTROL BANK INSERTION LIMITS FOR NORMAL 3 LOOP OPERATION-UNIT NO. 2

FIGURE 3.12-4B
CONTROL BANK INSERTION LIMITS
FOR 2 LOOP NORMAL OPERATION-UNIT NO.2



JUN 10 1975

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENTS NO. 6 TO LICENSES NO. DPR-32 AND DPR-37

CHANGE NO. 21 TO TECHNICAL SPECIFICATIONS

VIRGINIA ELECTRIC & POWER COMPANY

SURRY POWER STATION UNITS 1 & 2

DOCKET NOS. 50-280 AND 50-281

Introduction

By a letter dated March 12, 1975, and supplemented by a letter dated April 9, 1975, Virginia Electric & Power Company (the licensee) requested changes to the Technical Specifications appended to Facility Operating Licenses No. DPR-32 and DPR-37 for the Surry Power Station Units 1 and 2. The purpose of the request is to revise the Surry 2 Technical Specifications as required to operate within the apporpriate fuel and core design limits during the second fuel cycle.

Discussion

The reloading of the core for fuel cycle 2 will involve the replacement of 84 of the 157 fuel assemblies in the core. The second cycle core will consist of five regions of fuel: three that are carried over from the first cycle, Regions 1, 2, and 3 and two that are fresh, Regions 4 and 4A. The fuel to be added to the core, with the exception of two 17x17 rod array demonstration assemblies, is not significantly different in design or in operating characteristics from the original fuel it replaces. The two 17x17 fuel assemblies are part of Region 4 and do not affect reactor performance adversely relative to an all 15x15 fuel assembly core (cycle 1 design). The rearrangement of fuel assemblies in the reloaded core does affect core physics and thermal-hydraulic calculations and, as a result, changes to the Technical Specifications are required.

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EVALUATION

Nuclear Design

The second cycle core for Surry Unit 2 will involve the replacement of 84 of the 157 fuel assemblies in the core. Two of the 84 assemblies will be 17x17 assemblies. These are designed to have the same nuclear characteristics as the 15x15 fuel.

The licensee's evaluation of the second cycle for Unit 2 compares values of kinetics parameters, fuel temperatures, and core limits with those that were used in the Final Safety Analysis Report (FSAR). This evaluation showed that the second cycle core kinetics and accident analysis parameters are within the bounds or are conservative when compared to those of the FSAR, with the exception of the maximum reactivity insertion rate achievable during a startup from subcritical, the minimum boron concentration for criticality for refueling, and the end of life full power control rod ejection hot channel factor. These conditions were reanalyzed by the licensee and are discussed under accident analysis.

Control rod insertion limits were established for three-loop and two-loop operation to maintain the shutdown margin, power distribution limits, and control rod ejection worth consistent with or more conservative than the applicable accident analysis, and are acceptable.

We agree with the licensee and his fuel supplier, Westinghouse, in their predictions of the nuclear characteristics of the second cycle for Unit 2, and concur with the conclusion that the parameters used for accident analyses are conservative.

Accident Analysis

As mentioned above, the Unit 2 second cycle nuclear design evaluation indicated changes in three areas that required reevaluation of accidents.

1. The calculated maximum reactivity insertion rate for the RCCA control banks moving together in the highest worth region was 65 pcm/sec. Since this exceeded the value of 60 pcm/sec used in the analysis of the RCCA withdrawal from subcritical presented in the FSAR, the accident was reanalyzed using 65 pcm/sec. The results show that the effect of the higher insertion rate is to increase the peak heat flux only 4%. Since the peak heat flux for the analysis presented in the FSAR reached only 67% of the nominal value, the effects of the higher reactivity insertion rate does not affect the conclusions in the FSAR.

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2. At the beginning of life for the second cycle core, the value of 1100 ppm boron concentration for criticality at refueling conditions exceeds the value of 1000 ppm assumed in the Chemical and Volume Control System Malfunction incident presented in Section 14.2.5 of the FSAR. Therefore, reanalysis of the incident was performed using the same initial conditions and method of analysis presented in the FSAR. It was found that the time to dilute from 2000 ppm, corresponding to a shutdown of approximately 10% Δk with all RCCAs inserted, to 1100 ppm would take about 50 minutes, only about 7 minutes shorter than the dilution time to 1000 ppm. In our view, 50 minutes is ample time for the operator to recognize a high count rate signal and isolate the reactor make-up water source. The time for operator action during startup conditions where the Reactor Coolant System is filled with water is even longer, about 1.4 hours.

3. The end of life RCCA ejection accident from the hot full power condition was reanalyzed due to the higher transient hot channel factor for this core. The results showed a net reduction in peak fuel and clad temperatures due to the lower initial fuel temperature of the limiting region at the end of the second cycle.

We conclude that the results of these reanalyses do not affect the acceptability of the three areas discussed above.

Technical Specifications Section 3.12

The changes to Section 3.12 proposed by the licensee in Technical Specification Change No. 27 involve replacement of Figures 3.12-1B and 4B, control rod insertion limits for three- and two-loop operation consistent with the Unit 2 cycle 2 reload analysis. The insertion limits maintain the shutdown margin throughout cycle life as well as maintaining potential ejected rod worth and power distribution within the limits used in the accident analysis and are acceptable as proposed.

Cladding Collapse

The requested return to 2250 psia coolant pressure is consistent with the FSAR original safety analysis for Surry Unit 2. However, the return to a higher pressure required a reanalysis of a predicted time to cladding collapse. This reanalysis employed the Westinghouse revised cladding flattening model (WCAP-8381) which has been reviewed and accepted by the NRC staff (letter from V. Stello, TR, to R. DeYoung, RL, dated January 14, 1975, TAR-1112). We have concluded that the return to 2250 psia is acceptable and that no cladding collapse is predicted until beyond 17,000 EFPH(effective full power hours) of operation. The technical specifications will limit cycle 2 operation to 17,000 EFPH.

Demonstration Assemblies

VEPCO plans to insert two demonstration assemblies in Surry Unit 2. The demonstration assemblies are 17x17 arrays of fuel rods for which the safety analysis has been performed by Westinghouse (WCAP-8185) and approved by the staff. We conclude that the substitution of these two demonstration assemblies does not adversely affect reactor performance nor the course of any accident.

Reactor Fuel

The licensee examined the parameters for the same set of transient analyses reported in the Final Safety Analysis Report. Input parameters to most of the transients changed in a manner which increased the margin of safety or remained the same. The remaining transients were reanalyzed and found to be acceptable.

We have reviewed the proposed changes in the technical specifications to identify and evaluate any changes in margin to DNB. We find that the proposed technical specification changes are acceptable.

ECCS

Pursuant to the Commission's Order dated December 27, 1974, VEPCO has submitted a reanalysis of the ECCS performance for Surry Units 1 and 2. We are presently evaluating this submittal to determine conformance with 10 CFR Part 50, Section 50.46. Since our evaluation is still in progress, power operation of Surry Unit 2 for cycle 2 will not be authorized until our evaluation is completed. This will be complete in the near future. In the interim, the licensee may perform low power physics tests and other tests that do not entail power operations.

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

Date: **JUN 10 1975**

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-280 AND 50-281

VIRGINIA ELECTRIC & POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

Notice is hereby given that the U.S. Nuclear Regulatory Commission (the Commission) has issued Amendments No. 6 to Facility Operating Licenses No. DPR-32 and DPR-37 issued to Virginia Electric & Power Company which revised Technical Specifications for operation of the Surry Power Station, Units 1 and 2, located in Surry County, Virginia. The amendments are effective as of the date of issuance.

The amendments revise the provisions in the Technical Specifications relating to the replacement of 84 of 157 fuel assemblies in the reactor core, constituting refueling of the core for second cycle operation of Unit 2.

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. Notice of Proposed Issuance of Amendments to Facility Operating Licenses in connection with this action was published in the FEDERAL REGISTER on May 1, 1975 (40 FR 19043). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

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DATE ➤						

For further details with respect to this action, see (1) the application for amendments dated March 12, 1975, as supplemented April 9, 1975, (2) Amendments No. 6 to Licenses No. DPR-32 and DPR-37, with Change No. 21, 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, NW., Washington, D.C. and at the Swem Library, College of William & 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 Reactor Licensing.

Dated at Bethesda, Maryland, this JUN 10 1975

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
R. A. Purple

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

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