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OPA, C. Miles
R. Diggs
NSIC

September 22, 1983

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. 90 to Facility Operating License No. DPR-32 and Amendment No. 89 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 May 2, 1983.

These amendment would revise the Technical Specifications to change the fracitonal power limit to a 0.3 multiplier instead of a 0.2 multiplier for Units 1 and 2 and restore rod insertion limits to pre-Cycle 7 values for Unit 1.

A copy of the related Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's next regular Federal Register.

Sincerely,

/s/

Joseph D. Neighbors, Project Manager
Operating Reactors Branch No. 1
Division of Licensing

Enclosures:
1. Amendment No. 90 to DPR-32
2. Amendment No. 89 to DPR-37
3. Safety Evaluation

cc w/enclosures:
See next page

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PDR ADOCK 05000280
P PDR

OFFICE	ORB 1	ORB 1	ORB 1	AD:OR:DL	OELD	D:DL	
SURNAME	CParrish	DNeighbors/rs	SVarga	GLainas		DEisenhut	
DATE	9/14/83	9/14/83	9/ /83	9/14/83	9/15/83	9/ /83	

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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. 90
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 May 2, 1983, 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. 90 , 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: September 22, 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. 89
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 May 2, 1983, 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. 89 , 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: September 22, 1983

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 90 TO FACILITY OPERATING LICENSE NO. DPR-32

AMENDMENT NO. 89 TO FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NOS. 50-280 AND 50-281

Revise Appendix A as follows:

Remove Pages	Insert Pages
2.1-3	2.1-3
2.1-4	2.1-4
2.1-6	2.1-6
TS Figure 2.1-1	TS Figure 2.1-1
2.3-2	2.3-2
2.3-3	2.3-3
TS Figure 2.3-1	TS Figure 2.3-1
3.12-3	3.12-3
3.12-15	3.12-15
3.12-12	3.12-12
TS Figure 3.12-1A	TS Figure 3.12-1A

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 vessel 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, 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 procedure including the fuel densification power spiking [4] as part of the generic margin to accommodate rod bowing [5][6]. TS Figure 2.1-1 is also valid for the following limit of the enthalpy rise hot channel factor: $F^N_{\Delta H} = 1.55 (1 + 0.3 (1-P))$ where P is the fraction of rated power. TS Figures 2.1-2 and 2.1-3 include a 0.2 rather than 0.3 part power multiplier for the enthalpy rise hot channel factor.

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 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.

References

1) FSAR Section 3.4

2) FSAR Section 3.3.

3) FSAR Section 14.2

4) WCAP-8012, "Fuel Densification-Surry Power Station", December 1972 Section 4.3

5) Westinghouse (C. Eicheldinger) to NRC (V. Stello) letter dated August 13, 1976, Serial No. NS-CE-1163

6) NRC (A. Schwencer) to Vepco (W. L. Proffitt) letter dated July 27, 1979

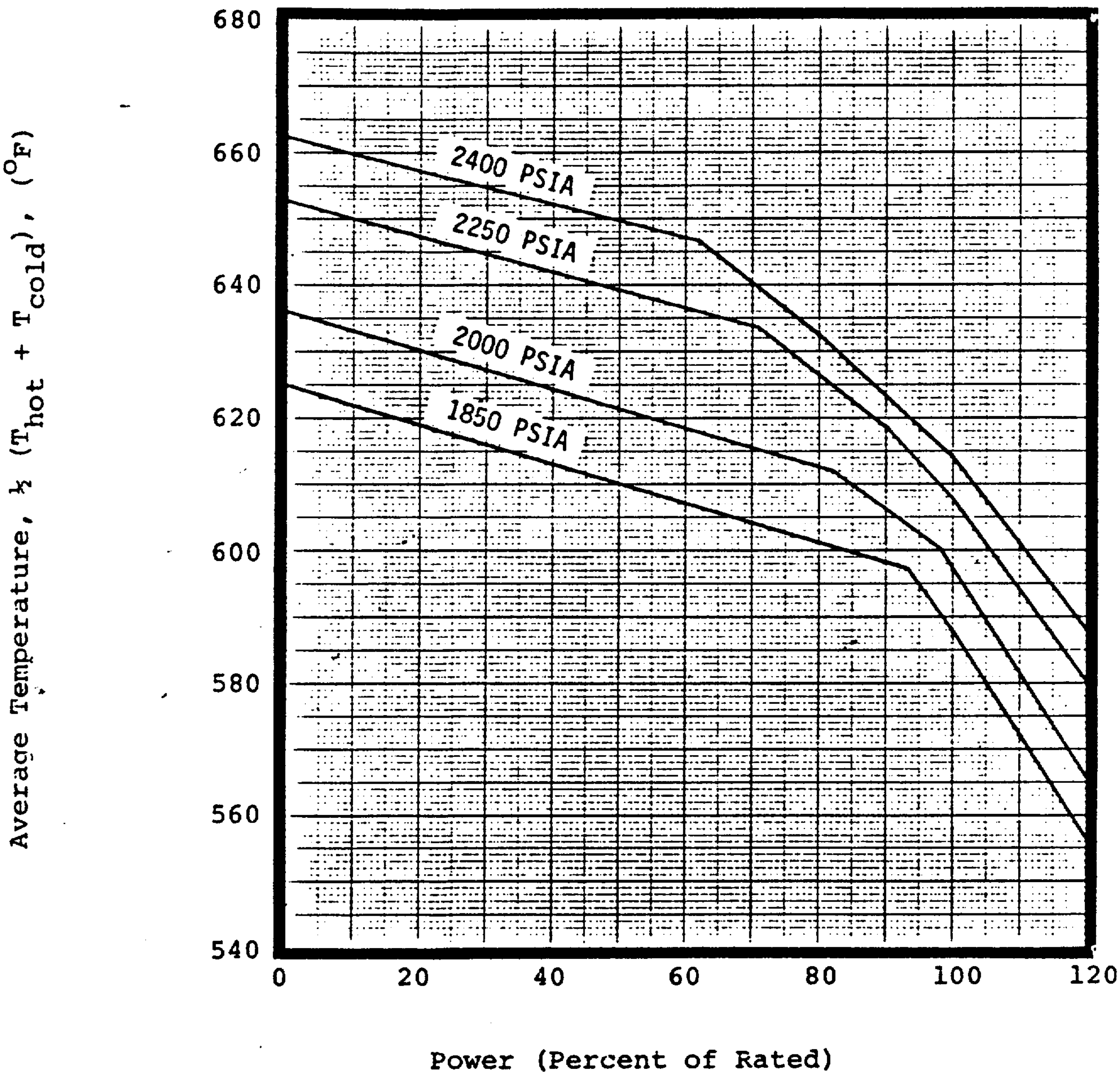

FIGURE 2.1-1 Reactor Core Thermal & Hydraulic Safety Limits Three Loop Operation, 100% Flow

(b) High pressurizer pressure - ≤ 2385 psig.
(c) Low pressurizer pressure - ≥ 1860 psig.
(d) Overtemperature T

$$\Delta T \leq \Delta T_o \left[K_1 - K_2 \left(\frac{1 + \tau_1 S}{1 + \tau_2 S}\right) (T - T') + {}_3K (P - P') - f(\Delta I)\right]$$

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 = 1.135	
K_2 = 0.01072	
K_3 = 0.000566	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 = 1.089

K_5 = 0 for decreasing average temperature

A constant, for increasing average temperature 0.02/°F

K_6 = 0 for $T \leq T'$

= 0.001086 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.

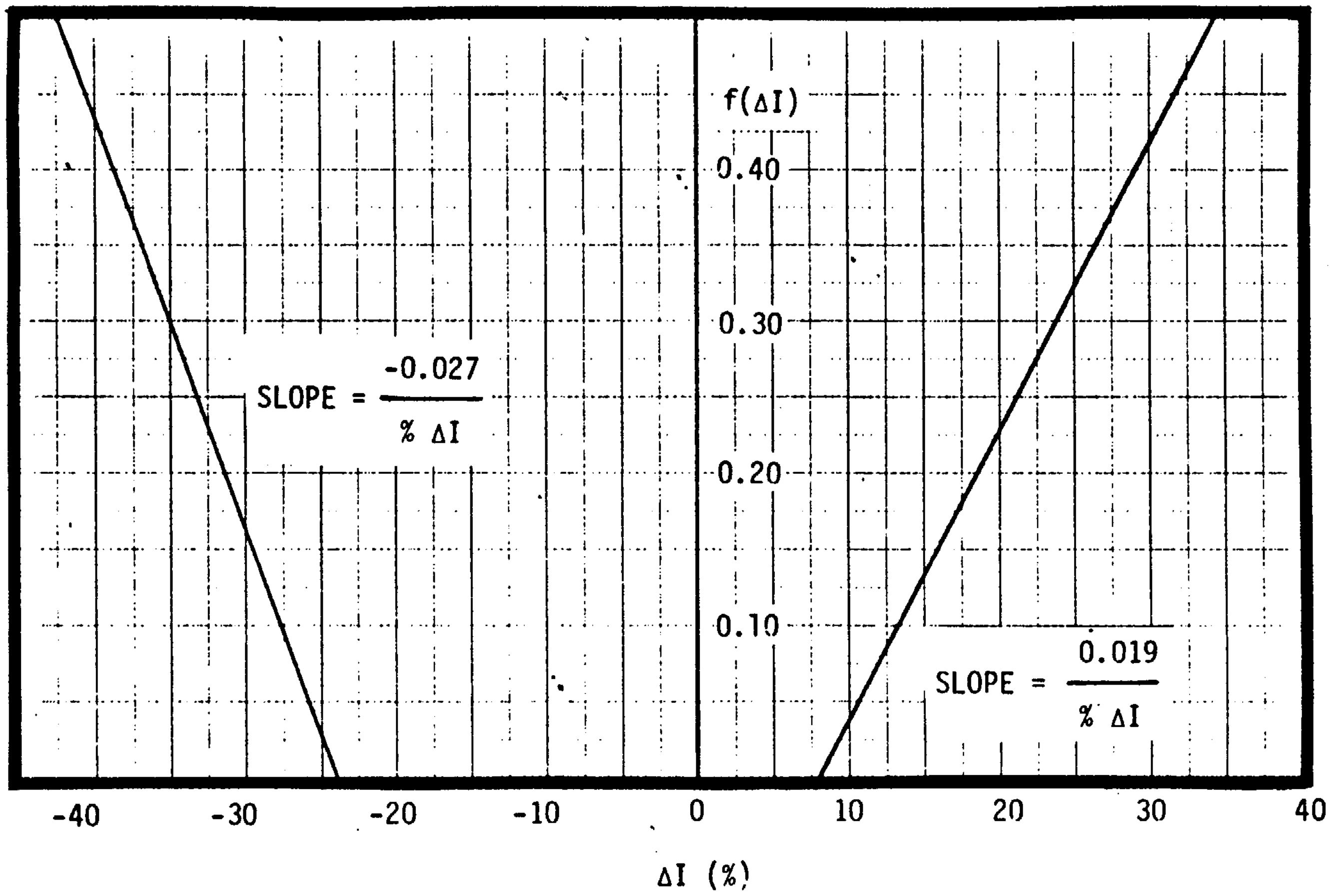

Figure 2.3-1 OPΔT and OTΔT f(ΔI) Function

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:

   $F_Q(Z) \leq 2.18/P \times K(Z)$ for $P > 0.5$

   $F_Q(Z) \leq 4.36 \times K(Z)$ 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 $F_Q(Z)$ $2.18 \times K(Z)$ and $F^N_{\Delta H} \leq 1.55$ within 24 hours, the Overpower ΔT and Overtemperature ΔT trip setpoints shall be similarly reduced.

on the maximum inserted rod worth in the unlikely event of a hypothetical assembly ejection and provide for acceptable nuclear peaking factors. The limit may be determined on the basis of unit startup and operating data to provide a more realistic limit which will allow for more flexibility in unit operation and still assure compliance with the shutdown requirement. The maximum shutdown margin requirement occurs at end of core life and is based on the value used in the analysis of the hypothetical steam break accident. The rod insertion limits are based on end of core life conditions. The shutdown margin for the entire cycle length is established at 1.77% reactivity. All other accident analysis with the exception of the chemical and volume control system malfunction analysis are based on 1% reactivity shutdown margin. Relative positions of control rod banks are determined by a specified control rod bank overlap. This overlap is based on the consideration of axial power shape control. The specified control rod insertion limits have been established to limit the potential ejected rod worth in order to account for the effects of fuel densification. The various control rod assemblies (shutdown banks, control banks A, B, C, and D) are each to be moved as a bank; that is, with all assemblies in the bank within one step (5/8 inch) of the bank position. Position indication is provided by two methods: a digital count of actuating pulses which shows the demand position of the banks, and a linear position indicator, Linear Variable Differential Transformer, which indicates the actual assembly position. The position indication accuracy of the Linear Differential Transformer is approximately ±5% of span (±12 steps) under steady state conditions. The relative accuracy of the linear position indicator has been considered in establishing the maximum allowable deviation of a control rod assembly from its indicated group step demand position. In the event that the linear position indicator is not

It should be noted that the enthalpy rise factors are based on intergrals 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 an upper bound envelope of 2.18 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 (≥38 thimbles, including a minimum of 2 thimbles per core quadrant, monitored) taken with the movable incore detector flux mapping system, three percent is the appropriate allowance for manufacturing tolerances, and five percent is the 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

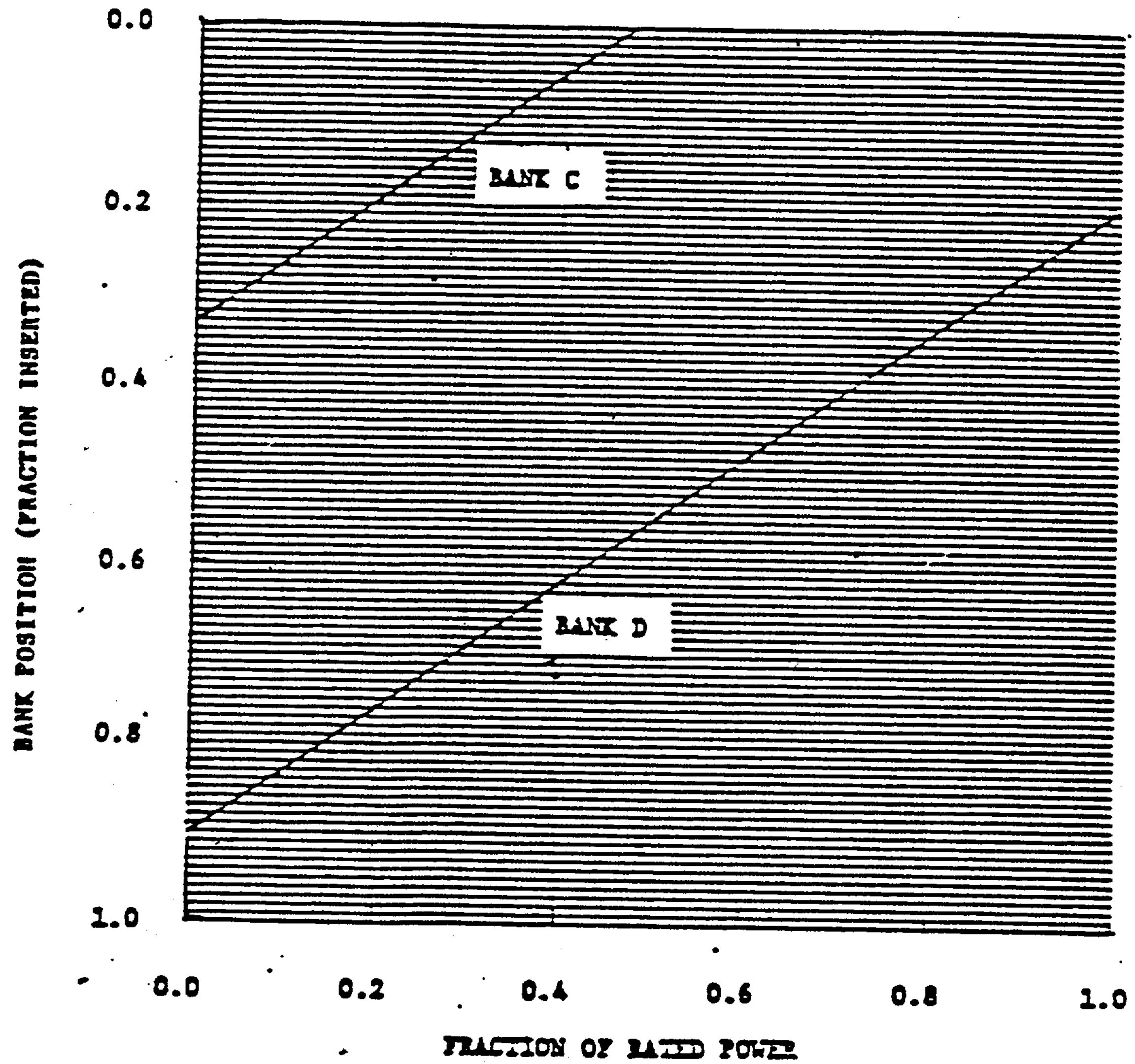

FIGURE 3.12-1A CONTROL BANK INSERTION LIMITS FOR 3-LOOP NORMAL OPERATION-UNIT 1

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 90 TO FACILITY OPERATING LICENSE NO. DPR-32

AND AMENDMENT NO. 89 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 May 2, 1983, the Virginia Electric and Power Company (the licensee) requested amendments to the facility operating licenses for Surry Power Station, Unit Nos. 1 and 2. The licensee proposed increasing the partial power multiplier for $F^N_{\Delta H}$ from 0.2 to 0.3 for both units and changing the rod insertion limits for Unit No. 1.

Discussion and Evaluation

$F^N_{\Delta H}$ Technical Specification Change

Historically, increasing the allowable $F^N_{\Delta H}$ with decreasing power has been permitted for all previously approved Westinghouse designs. The increase is permitted by the DNB protection setpoints and allows for radial power distribution changes with rod insertion to the insertion limit. The change to a larger (0.2 to 0.3) partial power multiplier is requested for Surry Units 1 and 2 to allow optimization of the core loading pattern by minimizing restrictions on $F^N_{\Delta H}$ at low power. The change will also minimize the probability of making rod insertion limit changes (such as was made for Unit 1 prior to Cycle 7) to satisfy peaking factor criteria at low power with the control rod banks at the insertion limit.

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The Surry core thermal limits and axial offset limits for an increased allowable $F^N_{\Delta H}$ at reduced power levels were determined using VEPCo's version of the COBRA code and standard Westinghouse methodology.

As a result of the multiplier change, small changes to the core thermal limits, overtemperature and overpower ΔT setpoints and the $F(\Delta I)$ function were necessary. The required changes were made to the Technical Specifications.

We have approved the 0.3 partial power multiplier for $F^N_{\Delta H}$ for WCAP-9500 and several other plants. The licensee's request for the Surry Units 1 and 2 changes is similar. Based on our review we find this change acceptable.

Rod Insertion Technical Specification Change for Unit 1

For Cycle 7, the rod insertion limits for Unit 1 were raised from the previously established limits in order to maintain the radial power peaking factors ($F^N_{\Delta H}$) below the Technical Specification limits (i.e., $F^N_{\Delta H} \leq 1.55\ (1.0 + 0.2(1-P))$ when P = fraction of rated thermal power). The reload safety evaluation of Unit 1 Cycle 7 established that after 1000 MWD/MTU of Cycle 7 burnup, $F^N_{\Delta H}$ would stay within the limits defined by $F^N_{\Delta H} \leq 1.55\ (1.0 + 0.3(1-P))$ with the previously acceptable rod insertion limits. Since the 0.3 partial power multiplier is established, the licensee has requested to change the rod insertion limits back to the previously established limits after 1000 MWD/MTU of Cycle 7 operation. We were aware that the licensee would be proposing this change as soon as the 0.3 partial power multiplier was established and we agree that the change is appropriate. Based on our review we find the rod insertion limit change for Unit 1 to be acceptable after 1000 MWD/MTU of Cycle 7 operation.

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

Date: September 22, 1983

Principal Contributor:
Margaret Chatterton