December 2, 1977

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Dockets Nos.: 50-280
and 50-281

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 Nos. 35 and 34 to Facility Operating Licenses Nos. DPR-32 and DPR-37 for the Surry Power Station, Units Nos. 1 and 2, respectively. These amendments include changes to the Technical Specifications for each license in response to your application dated August 9, 1977, as supplemented August 26, 1977, October 14, 1977, and November 16, 1977.

These amendments establish peaking factors [$F_Q(Z)$] to be used when the steam generator tube plugging limit exceeds 19% for Surry Unit No. 1 and 20% for Surry Unit No. 2; in addition, a tube plugging limit of 25% is authorized for both units.

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

Sincerely,

Original signed by

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

Enclosures:
1. Amendments Nos. 35 and 34
2. Safety Evaluation
3. Notice

cc w/enclosures: See next page

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DATE	12/2/77	12/2/77	12/ /77	12/2/77	12/2/77	12/2/77

NRC FORM 318 (9-76) NRCM 0240

Virginia Electric & Power Company

cc w/enclosure(s):

Michael W. Maupin, Esq.
Hunton, Williams, Gay & Gibson
P. O. Box 1535
Richmond, Virginia 23213

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

Mr. James C. Dunstan
State Corporation Commission
Commonwealth of Virginia
Blandon Building
Richmond, Virginia 23209

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

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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. 35
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 August 9, 1977, as supplemented August 26, 1977, October 14, 1977, and November 16, 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, Facility Operating License No. DPR-32 is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and by the following additional changes:

   A. Change Paragraph 3.B. to read:

      "B. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 35 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications."

   B. Delete Paragraph 3.F. in its entirety.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 2, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 35

FACILITY OPERATING LICENSE NO. DPR-32

DOCKET NO. 50-280

Revise the Technical Specifications as follows:

Remove Pages	Insert Pages
2.1-1	2.1-1
2.1-3	2.1-3
Fig. 2.1-1	Fig. 2.1-1
2.3-2	2.3-2
2.3-3	2.3-3
2.3-5	2.3-5
3.12-2	3.12-2
3.12-4	3.12-4
3.12-4a	3.12-4a
3.12-12	3.12-12
3.12-13	3.12-13
3.12-14	3.12-14
-	3.12-14a
3.12-15	3.12-15
-	3.12-15a
3.12-17	3.12-17
-	Table 3.12-2
Fig. 3.12-7	Fig. 3.12-7
Fig. 3.12-8	Fig. 3.12-8
4.10-1	4.10-1
4.10-2	4.10-2
4.10-3	4.10-3
5.3-2	5.3-2

Changes on the revised pages are shown by marginal lines.

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 90% of design 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 90% 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,

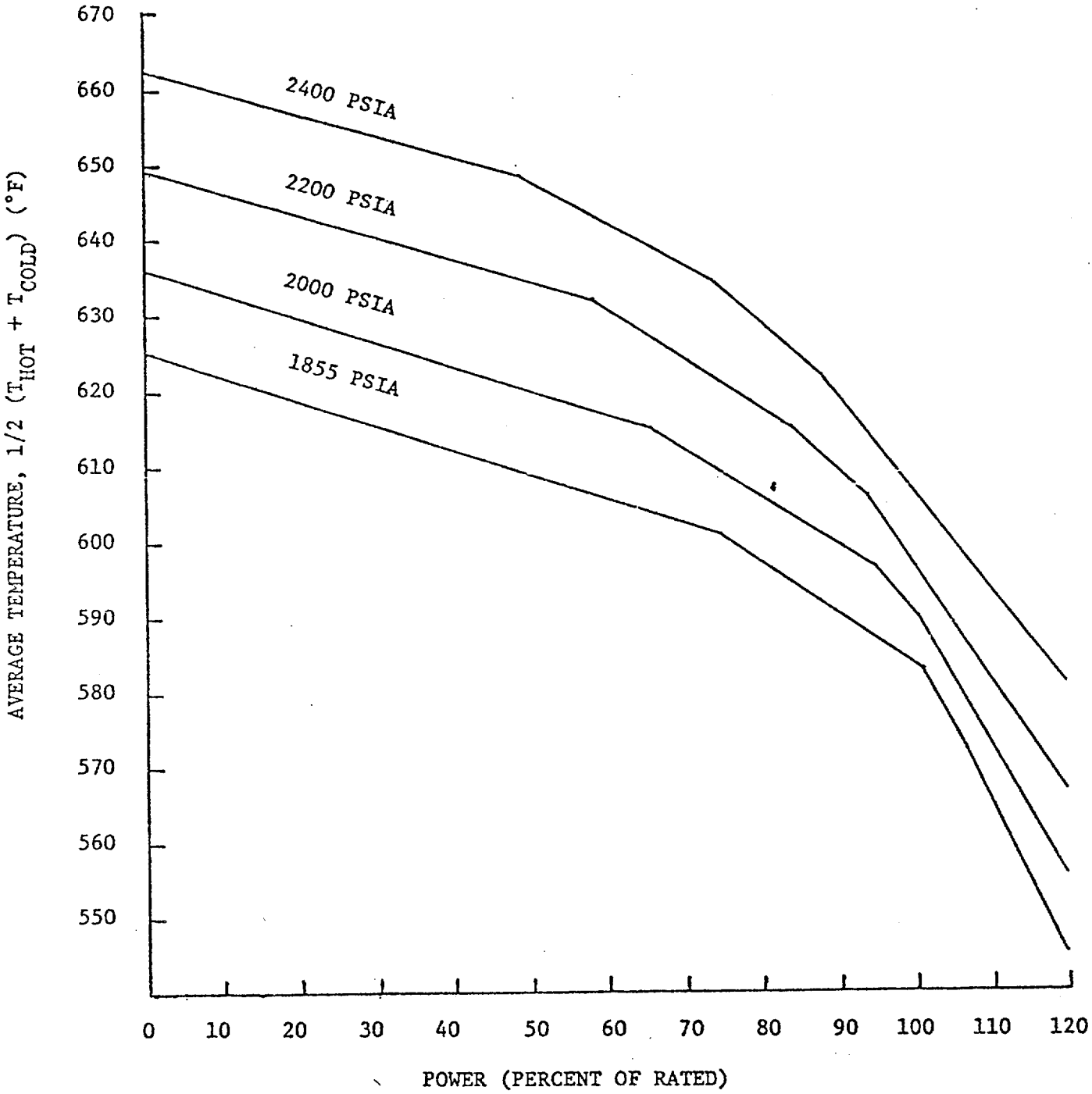


FIGURE 2.1-1 REACTOR CORE THERMAL & HYDRAULIC SAFETY LIMITS-
THREE LOOP OPERATION, 90% DESIGN FLOW

(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 = 1.07

K_2 = 0.0095

K_3 = 0.0005 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 = 30 seconds

τ_2 = 4 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 = 1.07

K_5 = 0 for decreasing average temperature

A constant, for increasing average temperature 0.02/°F

K_6 = 0 for $T \leq T'$

= 0.0011 for $T > T'$

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

τ_3 = 10 seconds

(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 settings

(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.

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 90% 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

culations and physics data obtained during unit startup and subsequent operation, will be permitted.

c. The shutdown margin with allowance for a stuck control rod assembly shall be greater than or equal to 1.77% reactivity under all steady-state operation conditions, except for physics tests, from zero to full power, including effects of axial power distribution. The shutdown margin as used here is defined as the amount by which the reactor core would be subcritical at hot shutdown conditions ($T_{avg} \geq 547^{o}F$) if all control rod assemblies were tripped, assuming that the highest worth control rod assembly remained fully withdrawn, and assuming no changes in xenon, boron, or part-length rod position.

4. Whenever the reactor is subcritical, except for physics tests, the critical rod position, i.e., the rod position at which criticality would be achieved if the control rod assemblies were withdrawn in normal sequence with no other reactivity changes, shall not be lower than the insertion limit for zero power.

5. Operation with part length rods shall be restricted such that except during physics tests, the part length rod banks are withdrawn from the core at all times.

6. Insertion limits do not apply during physics tests or during periodic excerise of individual rods. However, the shutdown margin indicated above must be maintained except for the low power physics test to measure control rod worth and shutdown margin. For this test the reactor may be critical with all but one full length control rod, expected to have the highest worth, inserted and part length rods fully withdrawn.

$F_Q(Z) \leq PF(S) \times K(Z)$ for $P > .5$

$F_Q(Z) \leq 2 \times PF(S) \times K(Z)$ for $P \leq .5$

$F^N_{\Delta H} \leq 1.55\ (1 + 0.2(1-P)) \times T(BU)$

$F^N_{\Delta H}\Big|^{LOCA}_{Assm.} \leq 1.38/P$ (See Note 1)

$F^N_{\Delta H}\Big|^{LOCA}_{Rod} \leq 1.45/P$ (See Note 1)

where P is the fraction of rated power at which the core is operating, PF(S) is the function given in TS Table 3.12-2, K(Z) is the function given in TS Figure 3.12-8, Z is the core height location of F_Q, and T(BU) is the interim thimble cell rod bow penalty on $F^N_{\Delta H}$ given in TS Figure 3.12-9.

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, and the hot assembly enthalpy rise factor, $F^N_{\Delta H}\Big|^{LOCA}_{Assm.}$, shall be increased by four percent to account for measurement error. If any measured hot channel factor exceeds its limit specified under 3.12.B.1, the reactor power and high neutron flux trip setpoint shall be reduced until the limits under 3.12.B.1 are met. If the hot channel factors cannot be brought to within the limits $F_Q \leq PF(S) \times K(Z)$, $F^N_{\Delta H} \leq 1.55 \times T(BU)$, $F^N_{\Delta H}\Big|^{LOCA}_{Rod} \leq 1.45$, and $F^N_{\Delta H}\Big|^{LOCA}_{Assm.} \leq 1.38$ within 24 hours, the Overpower ΔT and Overtemperature ΔT trip setpoints shall be similarly reduced.

NOTE 1: Only applicable for Unit 1 when steam generator tube plugging levels exceed 19% and for Unit 2 when steam generator tube plugging levels exceed 20%. $F_{\Delta H}\Big|^{LOCA}_{Assm.}$ and $F_{\Delta H}\Big|^{LOCA}_{Rod}$ are evaluated between the 1.5 and 10.5ft. elevations in the core.

b. $F_Q(Z)$ shall be evaluated for normal (Conditon I) operation of each unit by combining the measured values of $F_{xy}(Z)$ with the design Condition I axial peaking factor values, $F_Z(Z)$, as listed in TS Table 3.12-1A and TS Table 3.12-1B. For the purpose of this specification $F_{xy}(Z)$ shall be determined between 1.5 feet and 10.5 feet elevations of the core exclusive of grid plane regions located at 25.9 ±3.2 inches, 52.1 ±3.2 inches, 78.3 ±3.2 inches, and 104.5 ±3.2 inches. The measured values of $F_{xy}(Z)$ shall be increased by nine percent to account for manufacturing tolerances, measurement error, rod bow, xenon redistribution, and any burnup dependent peaking factor increases. If the results of this evaluation predict that $F_Q(Z)$ could potentially violate its limiting values as established in Specification 3.12.B.1, either:

(1) the thermal power and high neutron flux trip setpoint shall be reduced at least 1% for each 1% of the potential violation (for the purpose of this specification, this power level shall be called $P_{THRESHOLD}$), or

(2) movable detector surveillance shall be required for operation when the reactor thermal power exceeds $P_{THRESHOLD}$. This surveillance shall be performed in accordance with the following:

(a) The normalized power distribution, $F_Q(Z)\Big|^{j}_{APDM}$, from thimble j at core elevation Z shall be measured utilizing at least two thimbles of the movable incore flux system for

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 analyses 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 revised 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 and part-length rods) 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 (±7.5 inches) under steady state conditions. The relative accuracy of the linear position indicator is such that, with the most adverse errors, an alarm is actuated if any two assemblies within a bank deviate by more than 14 inches. In the event that the linear position indicator is not in service, the effects of

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 both units. 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, the nuclear enthalpy rise hot channel factor, and the hot assembly enthalpy rise factor must not exceed their limiting 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.

$F_Q(Z)$, Height Dependent Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

F_Q^E, Engineering Heat Flux Hot Channel Factor, is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

$F_{\Delta H}^N$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power for both LOCA and non-LOCA considerations.

$F_{\Delta H}^N \Big|_{\text{Assm.}}^{\text{LOCA}}$, Hot Assembly Nuclear Enthalpy Rise Factor, is defined as the ratio of the integral of linear power along the assembly with the highest integrated power to the average assembly power.

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

steam generator tube plugging levels for each unit. TS Table 3.12-2 lists the applicable values of PF(S) as determined by specific LOCA-ECCS analyses.

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 (≥40 thimbles 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 $F^N_{\Delta H} \leq 1.55(1 + 0.2(1-P)) \times T(BU)/1.08$ where T(BU) is the interim thimble cell rod bow penalty on $F^N_{\Delta H}$ given in TS Figure 3.12-9. 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 be compensated for by tighter axial control. Four percent is the appropriate allowance for measurement uncertainty for $F^N_{\Delta H}$ obtained from a full core map (≥40 thimbles monitored) taken with the movable incore detector flux mapping system.

The values specified for the limits of $F^N_{\Delta H}\Big|^{LOCA}_{Rod}$ and $F^N_{\Delta H}\Big|^{LOCA}_{Assm.}$ are the values used in the LOCA analysis. It has been determined that four percent is the appropriate allowance to be applied for measurement uncertainty for each of these parameters. Measurement of the hot channel factors are required as part of startup physics tests, during each effective full power month of operation,

and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following core loading provides confirmation of the basic nuclear design bases including proper fuel loading patterns. The periodic incore mapping provides additional assurance that the nuclear design bases remain inviolate and identify operational anomalies which would, otherwise, affect these bases.

For normal (Condition I) operation, it may be necessary to perform surveillance to insure that the heat flux hot channel factor, $F_Q(Z)$, limit is met. To determine whether and at what power level surveillance is required, the potential (Condition I) values of $F_Q(Z)$ shall be evaluated monthly by combining the measured values of $F_{xy}(Z)$ obtained from the analysis of the monthly incore flux map with the values of the design Condition I axial peaking factors, $F_Z(Z)$. The product of these shall be increased by nine percent to account for measurement uncertainty, manufacturing tolerances, rod bow, radial redistribution of xenon during normal (Condition I) operation, and any burnup dependent peaking factor increases. $P_{THRESHOLD}$ is defined as the value of rated power minus one percent power for each percent of potential $F_Q(Z)$ violation. If the potential values of $F_Q(Z)$ for normal (Condition I) operation are greater than the $F_Q(Z)$ limit, then surveillance shall be performed at all power levels above $P_{THRESHOLD}$.

Movable incore instrumentation thimbles for surveillance are selected so that the measurements are representative of the peak core power density. By limiting the core average axial power distribution, the total power peaking factor $F_Q(Z)$ can be limited since all other components remain relatively fixed. The remaining part of the total power peaking factor can be derived based on incore measurements, i.e., an effective radial peaking factor, $\overline{R}$, can be determined as the ratio of the total peaking

STEAM GENERATOR TUBE PLUGGING LEVEL, S (%)	PF(S)
UNIT 1	
$S \leq 19$	2.00
$19 < S \leq 25$	1.85
UNIT 2	
$S \leq 20$	2.00
$20 < S \leq 25$	1.85

TABLE 3.12-2: MAXIMUM HEAT FLUX HOT CHANNEL FACTOR, PF vs. % STEAM GENERATOR TUBE PLUGGING, S.

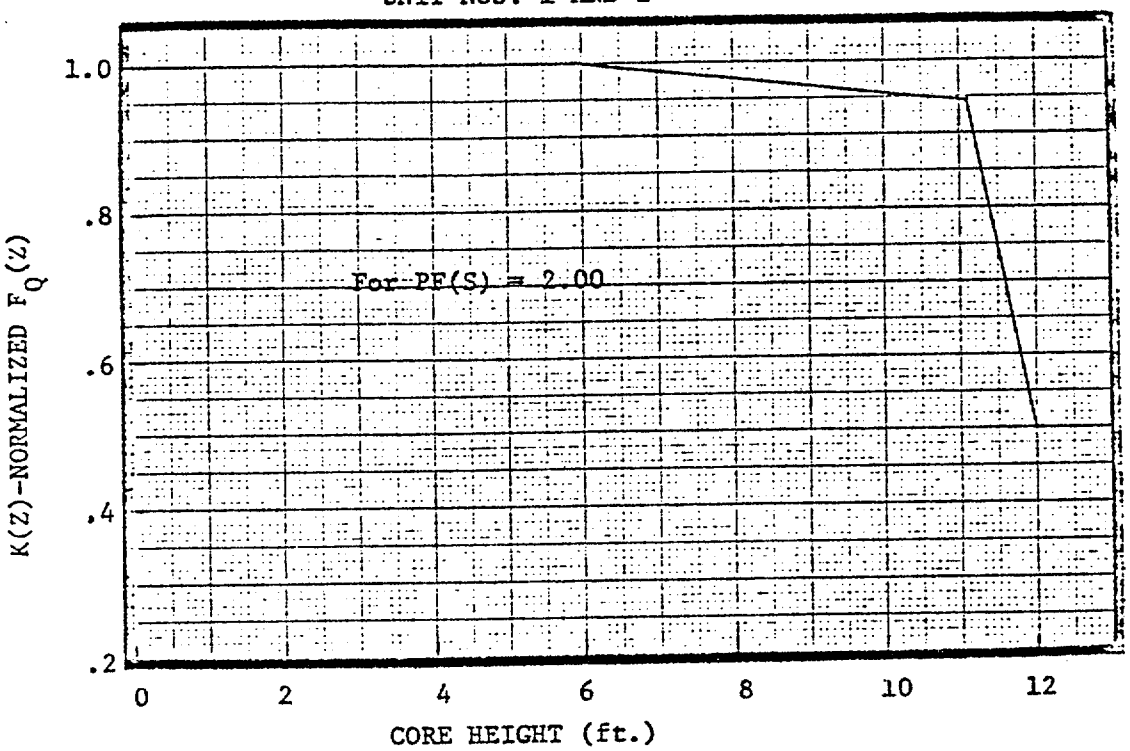
DELETE

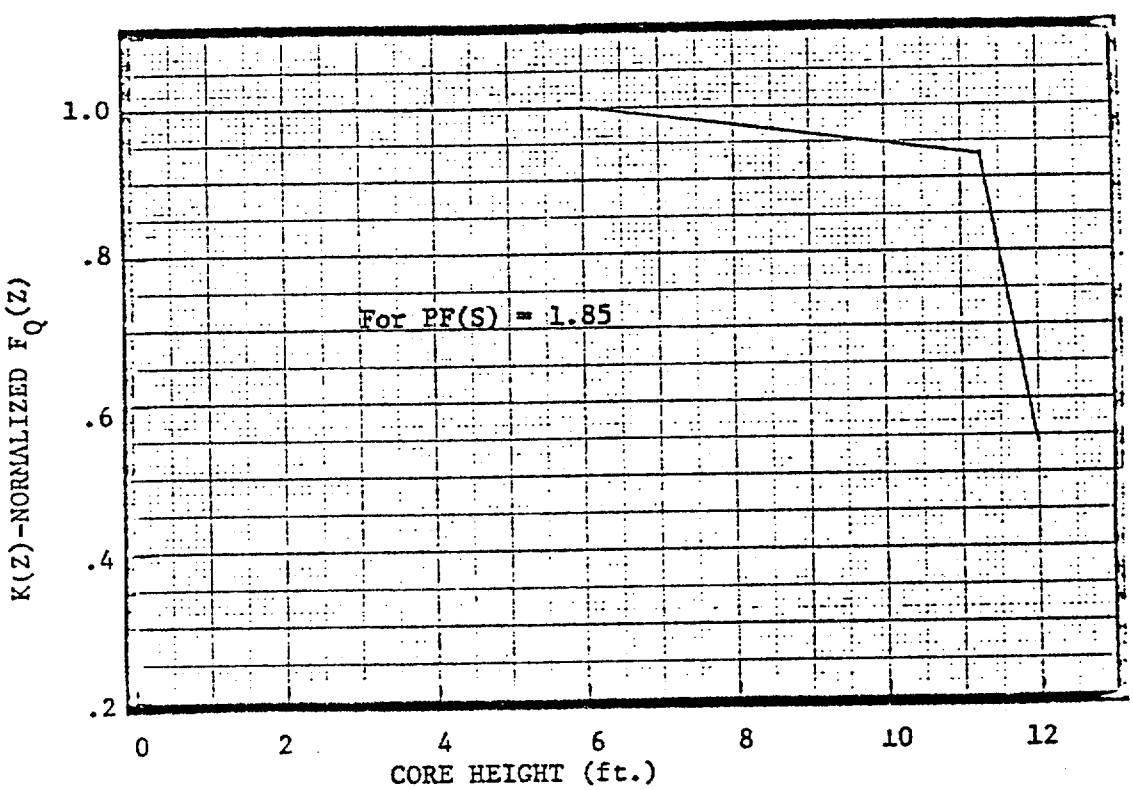
HOT CHANNEL FACTOR NORMALIZED

OPERATING ENVELOPE

SURRY POWER STATION

UNIT NOS. 1 AND 2





4.10 REACTIVITY ANOMALIES

Applicability

Applies to potential reactivity anomalies.

Objective

To require evaluation of applicable reactivity anomalies within the reactor.

Specification

A. Following a normalization of the computed boron concentration as a function of burnup, the actual boron concentration of the coolant shall be compared monthly with the predicted value. If the difference between the observed and predicted steady-state concentrations reaches the equivalent of one percent in reactivity, an evaluation as to the cause of the discrepancy shall be made and reported to the Nuclear Regulatory Commission per Section 6.6 of these Specifications.

B. During periods of power operation at greater than 10% of rated power, the hot channel factors identified in Section 3.12. shall be determined during each effective full power month of operation using data from limited core maps. If these factors exceed their limits, an evaluation as to the cause of the anomaly shall be made.

DELETED

Basis

BORON CONCENTRATION

To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burnup and the boron concentration necessary to maintain adequate control characteristics must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the control rod assembly groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration, and the slope of the curve relating burnup and reactivity is compared with that predicted. This process of normalization should be completed after about 10% of the total core burnup. Thereafter, actual boron concentration can be compared with prediction, and the reactivity status of the core can be continuously evaluated. Any reactivity anomaly greater than 1% would be unexpected, and its occurrence would be thoroughly investigated and evaluated.

The value of 1% is considered a safe limit since a shutdown margin of at least 1% with the most reactive control rod assembly in the fully withdrawn position is always maintained.

PEAKING FACTORS

A thermal criterion in the reactor core design specified that "no fuel melting during any anticipated normal operating condition" should occur. To meet the above criterion during a thermal overpower of 118% with additional margin for design uncertainties, a steady state maximum linear power is selected. This then is an upper linear power limit determined by the maximum central temperature of the hot pellet.

The peaking factor is a ratio taken between the maximum allowed linear power density in the reactor to the average value over the whole reactor. It is of course the average value that determines the operating power level. The peaking factor is a constraint which must be met to assure that the peak linear power density does not exceed the maximum allowed value.

During normal reactor operation, measured peaking factors should be significantly lower than design limits. As core burnup progresses, measured designed peaking factors typically decrease. A determination of these peaking factors during each effective full power month of operation is adequate to ensure that core reactivity changes with burnup have not significantly altered peaking factors in an adverse direction.

3. Reload fuel will be similar in design to the initial core. The enrichment of reload fuel will not exceed 3.60 weight percent of U-235.

4. Burnable poison rods are incorporated in the initial core. There are 816 poison rods in the form of 12 rod clusters, which are located in vacant control rod assembly guide thimbles. The burnable poison rods consist of pyrex clad with stainless steel.

5. There are 48 full-length control rod assemblies and 5 part-length control rod assemblies in the reactor core. The full-length control rod assemblies contain a 144-inch length of silver-indium-cadmium alloy clad with stainless steel. The part-length control rod assemblies contain a 36-inch length of silver-indium-cadmium alloy with the remainder of the stainless steel sheath filled with Al_2O_3.

6. Surry Unit 1, Cycle 4, Surry Unit 2, Cycle 3, and subsequent cores will meet the following criteria at all times during the operation lifetime.

   a. Hot channel factor limits as specified in Section 3.12 shall be met.

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. 34
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 August 9, 1977, as supplemented August 26, 1977, October 14, 1977, and November 16, 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, Facility Operating License No. DPR-37 is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and by the following additional changes:

   A. Change Paragraph 3.B. to read:

      "B. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 34, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications."

   B. Delete Paragraph 3.F. in its entirety.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 2, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 34

FACILITY OPERATING LICENSE NO. DPR- 37

DOCKET NO. 50- 281

Revise the Technical Specifications as follows:

Remove Pages	Insert Pages
2.1-1	2.1-1
2.1-3	2.1-3
Fig. 2.1-1	Fig. 2.1-1
2.3-2	2.3-2
2.3-3	2.3-3
2.3-5	2.3-5
3.12-2	3.12-2
3.12-4	3.12-4
3.12-4a	3.12-4a
3.12-12	3.12-12
3.12-13	3.12-13
3.12-14	3.12-14
-	3.12-14a
3.12-15	3.12-15
-	3.12-15a
3.12-17	3.12-17
-	Table 3.12-2
Fig. 3.12-7	Fig. 3.12-7
Fig. 3.12-8	Fig. 3.12-8
4.10-1	4.10-1
4.10-2	4.10-2
4.10-3	4.10-3
5.3-2	5.3-2

Changes on the revised pages are shown by marginal lines.

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 90% of design 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 90% 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,

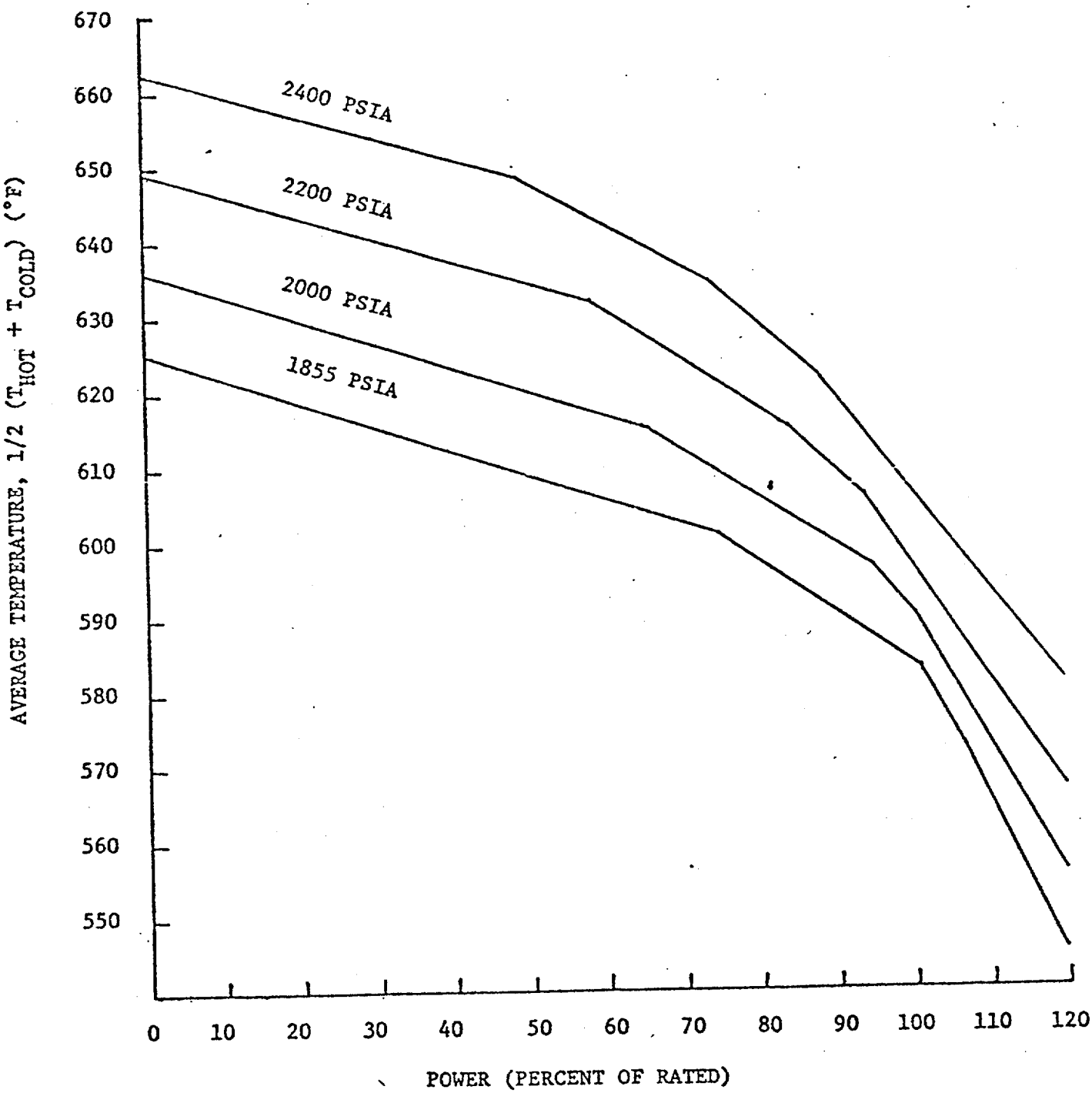


FIGURE 2.1-1 REACTOR CORE THERMAL & HYDRAULIC SAFETY LIMITS-
THREE LOOP OPERATION, 90% DESIGN FLOW

(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 = 1.07

K_2 = 0.0095

K_3 = 0.0005 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 = 30 seconds

τ_2 = 4 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 = 1.07

K_5 = 0 for decreasing average temperature

A constant, for increasing average temperature 0.02/°F

K_6 = 0 for $T \leq T'$

= 0.0011 for $T > T'$

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

τ_3 = 10 seconds

(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 settings

(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.

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 90% 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

culations and physics data obtained during unit startup and subsequent operatic will be permitted.

c. The shutdown margin with allowance for a stuck control rod assembly shall be greater than or equal to 1.77% reactivity under all steady-state operation conditions, except for physics tests, from zero to full power, including effects of axial power distribution. The shutdown margin as used here is defined as the amount by which the reactor core would be subcritical at hot shutdown conditions ($T_{avg} \geq 547^{\circ}F$) if all control rod assemblies were tripped, assuming that the highest worth control rod assembly remained fully withdrawn, and assuming no changes in xenon, boron, or part-length rod position.

4. Whenever the reactor is subcritical, except for physics tests, the critical rod position, i.e., the rod position at which criticality would be achieved if the control rod assemblies were withdrawn in normal sequence with no other reactivity changes, shall not be lower than the insertion limit for zero power.

5. Operation with part length rods shall be restricted such that except during physics tests, the part length rod banks are withdrawn from the core at all times.

6. Insertion limits do not apply during physics tests or during periodic excerise of individual rods. However, the shutdown margin indicated above must be maintained except for the low power physics test to measure control rod worth and shutdown margin. For this test the reactor may be critical with all but one full length control rod, expected to have the highest worth, inserted and part length rods fully withdrawn.

$F_Q(Z) \leq PF(S) \times K(Z)$ for $P > .5$

$F_Q(Z) \leq 2 \times PF(S) \times K(Z)$ for $P \leq .5$

$F^N_{\Delta H} \leq 1.55\ (1 + 0.2(1-P)) \times T(BU)$

$F^N_{\Delta H}\Big|^{LOCA}_{Assm.} \leq 1.38/P$ (See Note 1)

$F^N_{\Delta H}\Big|^{LOCA}_{Rod} \leq 1.45/P$ (See Note 1)

where P is the fraction of rated power at which the core is operating, PF(S) is the function given in TS Table 3.12-2, K(Z) is the function given in TS Figure 3.12-8, Z is the core height location of F_Q, and T(BU) is the interim thimble cell rod bow penalty on $F^N_{\Delta H}$ given in TS Figure 3.12-9.

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, and the hot assembly enthalpy rise factor, $F^N_{\Delta H}\Big|^{LOCA}_{Assm.}$, shall be increased by four percent to account for measurement error. If any measured hot channel factor exceeds its limit specified under 3.12.B.1, the reactor power and high neutron flux trip setpoint shall be reduced until the limits under 3.12.B.1 are met. If the hot channel factors cannot be brought to within the limits $F_Q \leq PF(S) \times K(Z)$, $F^N_{\Delta H} \leq 1.55 \times T(BU)$, $F^N_{\Delta H}\Big|^{LOCA}_{Rod} \leq 1.45$, and $F^N_{\Delta H}\Big|^{LOCA}_{Assm.} \leq 1.38$ within 24 hours, the Overpower ΔT and Overtemperature ΔT trip setpoints shall be similarly reduced.

NOTE 1: Only applicable for Unit 1 when steam generator tube plugging levels exceed 19% and for Unit 2 when steam generator tube plugging levels exceed 20%. $F_{\Delta H}\Big|^{LOCA}_{Assm.}$ and $F_{\Delta H}\Big|^{LOCA}_{Rod}$ are evaluated between the 1.5 and 10.5ft. elevations in the core.

b. $F_Q(Z)$ shall be evaluated for normal (Conditon I) operation of each unit by combining the measured values of $F_{xy}(Z)$ with the design Condition I axial peaking factor values, $F_Z(Z)$, as listed in TS Table 3.12-1A and TS Table 3.12-1B. For the purpose of this specification $F_{xy}(Z)$ shall be determined between 1.5 feet and 10.5 feet elevations of the core exclusive of grid plane regions located at 25.9 ±3.2 inches, 52.1 ±3.2 inches, 78.3 ±3.2 inches, and 104.5 ±3.2 inches. The measured values of $F_{xy}(Z)$ shall be increased by nine percent to account for manufacturing tolerances, measurement error, rod bow, xenon redistribution, and any burnup dependent peaking factor increases. If the results of this evaluation predict that $F_Q(Z)$ could potentially violate its limiting values as established in Specification 3.12.B.1, either:

(1) the thermal power and high neutron flux trip setpoint shall be reduced at least 1% for each 1% of the potential violation (for the purpose of this specification, this power level shall be called $P_{THRESHOLD}$), or

(2) movable detector surveillance shall be required for operation when the reactor thermal power exceeds $P_{THRESHOLD}$. This surveillance shall be performed in accordance with the following:

(a) The normalized power distribution, $F_Q(Z)\Big|^{j}_{APDM}$, from thimble j at core elevation Z shall be measured utilizing at least two thimbles of the movable incore flux system for

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 analyses 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 revised 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 and part-length rods) 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 (±7.5 inches) under steady state conditions. The relative accuracy of the linear position indicator is such that, with the most adverse errors, an alarm is actuated if any two assemblies within a bank deviate by more than 14 inches. In the event that the linear position indicator is not in service, the effects of

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 both units. 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, the nuclear enthalpy rise hot channel factor, and the hot assembly enthalpy rise factor must not exceed their limiting 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.

$F_Q(Z)$, Height Dependent Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

F_Q^E, Engineering Heat Flux Hot Channel Factor, is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

$F_{\Delta H}^N$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power for both LOCA and non-LOCA considerations.

$F_{\Delta H}^N \Big|_{\text{Assm.}}^{\text{LOCA}}$, Hot Assembly Nuclear Enthalpy Rise Factor, is defined as the ratio of the integral of linear power along the assembly with the highest integrated power to the average assembly power.

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

steam generator tube plugging levels for each unit. TS Table 3.12-2 lists the applicable values of PF(S) as determined by specific LOCA-ECCS analyses.

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 (≥40 thimbles 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 $F^N_{\Delta H} \leq 1.55(1 + 0.2(1-P)) \times T(BU)/1.08$ where T(BU) is the interim thimble cell rod bow penalty on $F^N_{\Delta H}$ given in TS Figure 3.12-9. 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 be compensated for by tighter axial control. Four percent is the appropriate allowance for measurement uncertainty for $F^N_{\Delta H}$ obtained from a full core map (≥40 thimbles monitored) taken with the movable incore detector flux mapping system.

The values specified for the limits of $F^N_{\Delta H}\Big|^{LOCA}_{Rod}$ and $F^N_{\Delta H}\Big|^{LOCA}_{Assm.}$ are the values used in the LOCA analysis. It has been determined that four percent is the appropriate allowance to be applied for measurement uncertainty for each of these parameters. Measurement of the hot channel factors are required as part of startup physics tests, during each effective full power month of operation,

and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following core loading provides confirmation of the basic nuclear design bases including proper fuel loading patterns. The periodic incore mapping provides additional assurance that the nuclear design bases remain inviolate and identify operational anomalies which would, otherwise, affect these bases.

For normal (Condition I) operation, it may be necessary to perform surveillance to insure that the heat flux hot channel factor, $F_Q(Z)$, limit is met. To determine whether and at what power level surveillance is required, the potential (Condition I) values of $F_Q(Z)$ shall be evaluated monthly by combining the measured values of $F_{xy}(Z)$ obtained from the analysis of the monthly incore flux map with the values of the design Condition I axial peaking factors, $F_Z(Z)$. The product of these shall be increased by nine percent to account for measurement uncertainty, manufacturing tolerances, rod bow, radial redistribution of xenon during normal (Condition I) operation, and any burnup dependent peaking factor increases. $P_{THRESHOLD}$ is defined as the value of rated power minus one percent power for each percent of potential $F_Q(Z)$ violation. If the potential values of $F_Q(Z)$ for normal (Condition I) operation are greater than the $F_Q(Z)$ limit, then surveillance shall be performed at all power levels above $P_{THRESHOLD}$.

Movable incore instrumentation thimbles for surveillance are selected so that the measurements are representative of the peak core power density. By limiting the core average axial power distribution, the total power peaking factor $F_Q(Z)$ can be limited since all other components remain relatively fixed. The remaining part of the total power peaking factor can be derived based on incore measurements, i.e., an effective radial peaking factor, $\overline{R}$, can be determined as the ratio of the total peaking

STEAM GENERATOR TUBE PLUGGING LEVEL, S (%)	PF(S)
UNIT 1	
$S \leq 19$	2.00
$19 < S \leq 25$	1.85
UNIT 2	
$S \leq 20$	2.00
$20 < S \leq 25$	1.85

TABLE 3.12-2: MAXIMUM HEAT FLUX HOT CHANNEL FACTOR, PF vs. % STEAM GENERATOR TUBE PLUGGING, S.

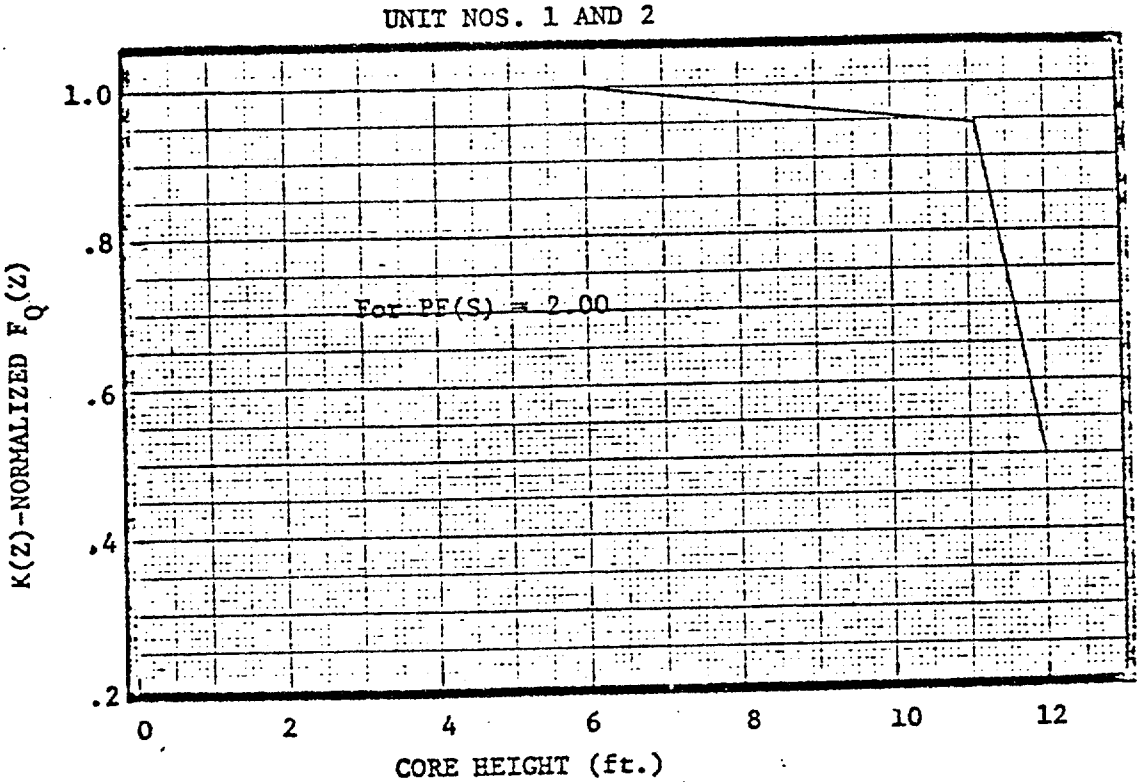
DELETE

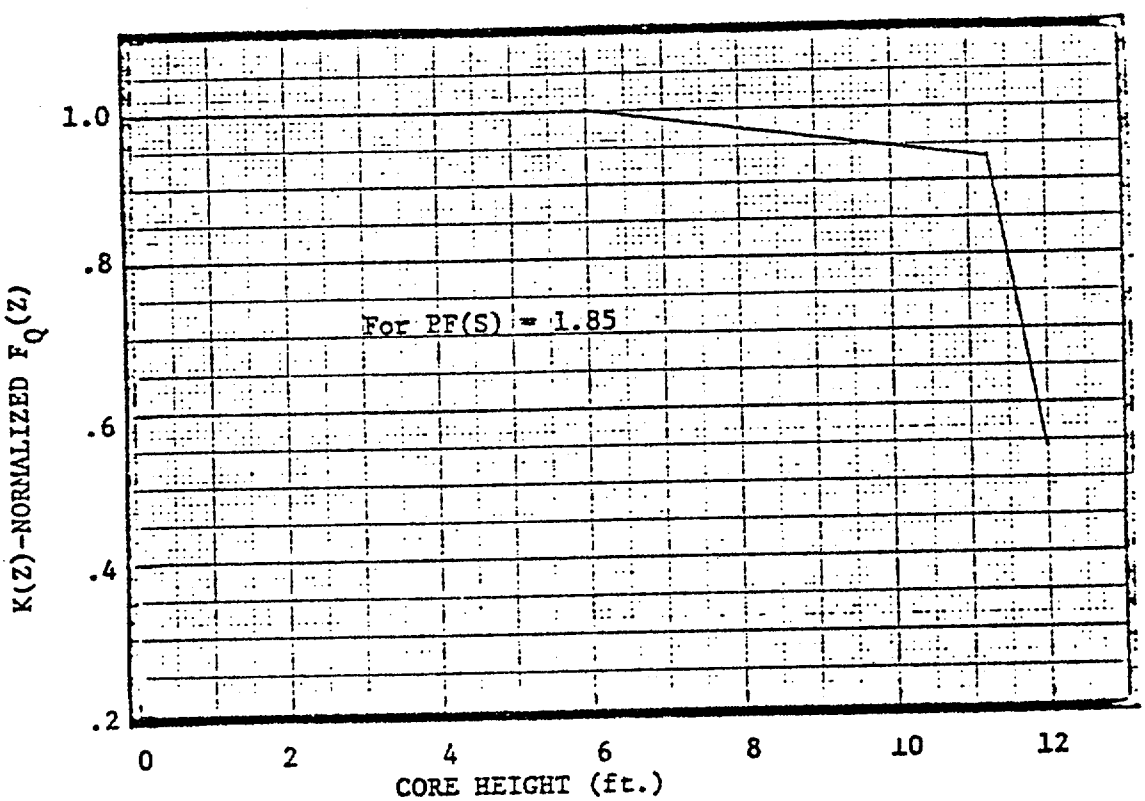
HOT CHANNEL FACTOR NORMALIZED

OPERATING ENVELOPE

SURRY POWER STATION

UNIT NOS. 1 AND 2





4.10 REACTIVITY ANOMALIES

Applicability

Applies to potential reactivity anomalies.

Objective

To require evaluation of applicable reactivity anomalies within the reactor.

Specification

A. Following a normalization of the computed boron concentration as a function of burnup, the actual boron concentration of the coolant shall be compared monthly with the predicted value. If the difference between the observed and predicted steady-state concentrations reaches the equivalent of one percent in reactivity, an evaluation as to the cause of the discrepancy shall be made and reported to the Nuclear Regulatory Commission per Section 6.6 of these Specifications.

B. During periods of power operation at greater than 10% of rated power, the hot channel factors identified in Section 3.12. shall be determined during each effective full power month of operation using data from limited core maps. If these factors exceed their limits, an evaluation as to the cause of the anomaly shall be made.

DELETED

Basis

BORON CONCENTRATION

To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burnup and the boron concentration necessary to maintain adequate control characteristics must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the control rod assembly groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration, and the slope of the curve relating burnup and reactivity is compared with that predicted. This process of normalization should be completed after about 10% of the total core burnup. Thereafter, actual boron concentration can be compared with prediction, and the reactivity status of the core can be continuously evaluated. Any reactivity anomaly greater than 1% would be unexpected, and its occurrence would be thoroughly investigated and evaluated.

The value of 1% is considered a safe limit since a shutdown margin of at least 1% with the most reactive control rod assembly in the fully withdrawn position is always maintained.

PEAKING FACTORS

A thermal criterion in the reactor core design specified that "no fuel melting during any anticipated normal operating condition" should occur. To meet the above criterion during a thermal overpower of 118% with additional margin for design uncertainties, a steady state maximum linear power is selected. This then is an upper linear power limit determined by the maximum central temperature of the hot pellet.

The peaking factor is a ratio taken between the maximum allowed linear power density in the reactor to the average value over the whole reactor. It is of course the average value that determines the operating power level. The peaking factor is a constraint which must be met to assure that the peak linear power density does not exceed the maximum allowed value.

During normal reactor operation, measured peaking factors should be significantly lower than design limits. As core burnup progresses, measured designed peaking factors typically decrease. A determination of these peaking factors during each effective full power month of operation is adequate to ensure that core reactivity changes with burnup have not significantly altered peaking factors in an adverse direction.

3. Reload fuel will be similar in design to the initial core. The enrichment of reload fuel will not exceed 3.60 weight percent of U-235.

4. Burnable poison rods are incorporated in the initial core. There are 816 poison rods in the form of 12 rod clusters, which are located in vacant control rod assembly guide thimbles. The burnable poison rods consist of pyrex clad with stainless steel.

5. There are 48 full-length control rod assemblies and 5 part-length control rod assemblies in the reactor core. The full-length control rod assemblies contain a 144-inch length of silver-indium-cadmium alloy clad with stainless steel. The part-length control rod assemblies contain a 36-inch length of silver-indium-cadmium alloy with the remainder of the stainless steel sheath filled with Al_2O_3.

6. Surry Unit 1, Cycle 4, Surry Unit 2, Cycle 3, and subsequent cores will meet the following criteria at all times during the operation lifetime.

   a. Hot channel factor limits as specified in Section 3.12 shall be met.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENTS NOS. 35 AND 34 TO LICENSES NOS. DPR-32 AND DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION UNITS NOS. 1 AND 2

DOCKETS NOS. 50-280 AND 50-281

INTRODUCTION

By letter dated August 9, 1977, as supplemented August 26, 1977, October 14, 1977, and November 16, 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 establish peaking factors [$F_Q(Z)$] to be used when the steam generator tube plugging limit exceeds 19% for Surry Unit No. 1 and 20% for Surry Unit No. 2; in addition, a tube plugging limit of 25% is requested for both units.

EVALUATION

Through references 1, 2, and 3 VEPCO has requested amendment to the operating licenses for Surry Units 1 and 2 to permit an increase in the plugging level of the steam generator tubes. The current analyses are valid for steam generator tube plugging levels of up to 19% on Unit 1 and up to 20% on Unit 2. The submittals which contain accident analyses and Technical Specification changes are in support of a request to raise the steam generator tube plugging level limit to 25% for both units.

Reactor Coolant System (RCS) Flow Rate

As the level of steam generator tube plugging increases, several factors affect the assumptions used for the analyses of anticipated and design basis accidents. Among the affected parameters are the pump coast down rate, heat transfer area to the secondary side, and loop flow rate. To assess the effect of steam generator tube plugging on RCS loop flow VEPCO has taken measurements to obtain the loop flow rate at several levels of steam generator tube plugging. These data points are presented in Table 1.

TABLE 1

Unit	% S. G. Tubes Plugged	Ave. RCS Loop Flow (gpm)
1	5.7	94,500
1	14.4	92,400
1	18.6	90,700
2	3.4	100,000
2	16.8	94,000

The average loop flow rates are obtained from measurements used to perform a plant heat balance. The data include primary temperatures and pressure, secondary system calorimetric data, reactor coolant pump heat, and estimates of heat losses from both the primary and secondary side.

The data points were compared with the flow rate predictions obtained with with the Westinghouse analytical model. The maximum deviation between the measured and predicted curves was used as a constant bias to reduce the predicted curve of flow rate versus percent steam generator tubes plugged. This curve was then further reduced by 2% to account for measurement uncertainty, which VEPCO has shown(5) to be greater than the 2σ confidence limit on the measured flow rate.

Further conservatism has been used in the following analyses by extrapolating the flow rate versus steam generator tube plugging level curve to 40% plugging level and using the corresponding flow rate in the analyses. This flow rate, 79,560 gpm per loop, is 10% below the thermal design flow rate of 88,500 gpm per loop. This assumption is conservative since the proposed authorization is for operation to 25% of the steam generator tubes plugged.

ACCIDENT ANALYSES

A. Non-Loss-of-Coolant Accident (LOCA)

As a result of the proposed 40% plugging level three factors made the current analyses invalid:

1) The RCS flow rate is lower than the thermal design value,

2) The RCS volume is less than that assumed in the analyses, and

3) The pump coast down characteristics are more severe than assumed in the analyses.

VEPCO has submitted[1] an assessment of the impact on the non-LOCA of steam generator tube plugging up to a level of 40%. The individual accidents were examined to determine which of the parameters affected by increased steam generator tube plugging were important for each accident. These affected accidents which are limiting or very sensitive to the effects of higher steam generator tube plugging levels were reanalyzed. The following assumptions were used in the analyses:

	Proposed	Current
Thermal design flow, gpm/loop	79,650	88,500
S. G. tube plugging, %	40	19 Unit 1 20 Unit 2
Power, Mwt (102% of)	2441	2441
T_{AVE}, °F (+ 4°F)	574.4	574.4
ΔT at 100% power, °F	69.1	62.8
$F^N_{\Delta H}$	1.55	1.55
F_Q	2.0	2.0

Three accidents are limiting or most sensitive to the 40% steam generator tube plugging level.

1. An uncontrolled control rod assembly withdrawal at power produces a mismatch in reactor power and steam flow. The result is an increase in reactor coolant temperature. Reanalysis of the uncontrolled control rod withdrawal at power indicates that adequate margin to DNB exists at the higher steam generator tube plugging level. The minimum Departure from Nucleate Boiling Ratio (DNBR) calculated remains above 1.30, thus indicating that the core and reactor coolant system are not adversely affected.

2. Boron dilution accidents have been reanalyzed primarily due to their sensitivity to the reduction in the reactor coolant system volume. The case of boron dilution during refueling

was not reanalyzed since the steam generator tube volume is not considered in the analysis. The case of boron dilution during startup was reanalyzed due to the significant change in system volume from the exclusion of 40% of the steam generator tube volume. The reanalysis indicates a return to criticality in 82 minutes. This is longer than our required minimum time of 30 minutes. Thus adequate time exists for operator action.

Two cases were analyzed for boron dilution while at power: automatic and manual control of the reactor. Boron dilution at power with the reactor in automatic control results in an operator action time of 24 minutes. Boron dilution of power with the reactor in manual control results in an operator action time of 22 minutes. Both cases exceed our required minimum of 15 minutes for operator action and thus are acceptable.

3. The loss of flow accident was reanalyzed assuming loss of all three reactor coolant pumps. The significant factor affecting the loss of flow accident is the increased loop resistances, due to the 40% level of steam generator tube plugging, resulting in a more rapid coastdown. Using analysis methods and assumptions consistent with those employed in the Final Safety Analysis Report (FSAR), the loss of flow accident for the higher level of steam generator tube plugging results in a minimum DNBR of 1.33. Thus, adequate margin exists for the loss of flow accident with a higher level of steam generator tube plugging. The result of the evaluations and reanalyses performed indicate that with the reduced flow rate and changed pump coastdown, the anticipated transients presented in the FSAR will meet NRC requirements for safety margin.

B. Loss-of-Coolant Accident (LOCA)

The emergency core cooling system (ECCS) performance has been reanalyzed[2,3] for a postulated large break LOCA with the assumed flow rate in the RCS flow rate section above. The reanalysis was performed using the October, 1975 version of the Westinghouse Evaluation Model. That model is in compliance with Appendix K to 10 CFR 50 and the August 27, 1976 Order for Modification of License.

The assumptions and initial operating conditions used in the reanalysis are in accord with those of the current approved LOGA-ECCS analysis with the exception of:

1) The heat flux hot channel factor changed from 2.0 to 1.85,

2) RCS flow rate changed from 88,500 gpm/loop to 79,650 gpm/loop,

3) Number of steam generator tubes plugged changed from 20% to 25%,

4) Core inlet temperature uncertainty of +4°F removed,

5) Containment spray initiation time changed from 20 sec to 46 sec,

6) Inside and outside recirculation spray system initiation times of 120 sec and 300 sec, respectively, and

7) Hot assembly enthalpy rise peaking factor change from 1.435 to 1.38.

Results have been submitted for the double ended cold leg guillotine break (DECLG) with a discharge coefficient CD = 0.4, 0.6, and 1.0. As with all previous analyses for the Surry Units, the break with CD=0.4 is the limiting case. The results of the reanalysis indicate a peak clad temperature of 2177°F, a maximum local clad oxidation rate of 7.4%, and a total core metal-water reaction of less than 0.3% taking into account the seven above changes in the original analysis. The results given above were obtained using a core power shape axially peaked at the core centerline. The Westinghouse ECCS sensitivity studies [6] indicate that the center peaked shape is the limiting power shape for peaking factors (F_Q) greater than 2.32. The staff has discussed with the licensee the applicability of the Westinghouse sensitivity study's result that the center peaked shape is limiting when the peaking factor is less than 2.32. In response the licensee has referenced Amendment 66[7,8] to the D.C. Cook FSAR which supports a peaking factor of 2.05. Although D. C. Cook is a different plant, the results of calculations performed with two power shapes, center and top peaked, demonstrate that at a peaking factor of 2.05 the center peaked shape remains limiting. The staff has concluded that the peaking of the Surry plants (F_Q= 1.85) is sufficiently lower than that of the D. C. Cook plant (F_Q= 2.05) to warrant additional analytical studies to support the conclusion that the center peaked power shape is the limiting shape. The licensee has committed to provide within approximately ninety days sufficient analytical studies to justify use of the center peaked power shape as the limiting shape. We conclude that the ECCS meets the acceptance criteria as presented in 10 CFR 50.46 with up to 25% of the steam generator tubes plugged.

TECHNICAL SPECIFICATIONS

The proposed Technical Specification changes have been reviewed to ensure that the assumptions and limitations imposed due to the accident analyses and LOCA reanalyses will be met. The proposed Technical Specifications with existing operating surveillance procedures ensure maintenance of the hot channel factor normalized operating envelope, [K(7) versus core height], and will provide for safe operation of Surry Units 1 and 2 with up to 25% of the steam generator tubes plugged. The changes in Technical Specifications 4.10 and 5.3 merely removed data repeated from Technical Specification 3.12 and added references to T.S. 3.12 for the data; thus, these changes are administrative in nature. Therefore, we find the proposed Technical Specification changes acceptable.

SUMMARY

The licensee has shown by the use of conservative assumptions and acceptable analyses that Surry Units 1 and 2 can be safely operated with up to 25% of the steam generator tubes plugged. The limiting transients were reanalyzed and the results are within required safety margins. The ECCS performance has been reanalyzed for the large break LOCA with the result that compliance with the requirements of 10 CFR 50.46 is assured. For the limiting case the peak clad temperature does not exceed 2200°F.

We conclude that operation of Surry Units 1 and 2 with up to 25% of the steam generator tubes plugged will not result in undue risk to the health and safety of the public.

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

Date: December 2, 1977

REFERENCES

1. Letter C. M. Stallings (Vepco) to E. G. Case (NRC) dated August 9, 1977.

2. Letter C. M. Stallings (Vepco) to E. G. Case (NRC) dated August 26, 1977.

3. Letter C. M. Stallings (Vepco) to E. G. Case (NRC) dated October 14, 1977.

4. Letter R. W. Reid (NRC) to C. M. Stallings (Vepco) dated November 10, 1977.

5. Letter C. M. Stallings (Vepco) to E. G. Case (NRC) dated November 16, 1977.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS 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 Amendments Nos. 35 and 34 to Facility Operating Licenses Nos. DPR-32 and DPR-37, issued to Virginia Electric & Power Company (the licensee), which revised the license and 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 establish peaking factors to be used when the steam generator tube plugging limit exceeds 19% for Surry Unit No. 1 and 20% for Surry Unit No. 2; in addition, a tube plugging limit of 25% is authorized for both units.

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.

Notice of Proposed Issuance of Amendments to Facility Operating Licenses in connection with this action was published in the FEDERAL REGISTER on September 15, 1977 (42 F.R. 46431) and corrected on September 29, 1977 (42 F.R. 51680). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

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, negative declaration or 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 August 9, 1977, as supplemented August 26, 1977, October 14, 1977, and November 16, 1977, (2) Amendments Nos. 35 and 34 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 2nd day of December 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

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