APR 19 1974

Docket Nos. ~~50-280~~
and 50-281

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

Change No. 15
Licenses Nos. DPR-32
and DPR-37

Gentlemen:

By letter dated March 21, 1974, you submitted proposed changes to the Surry 1 and 2 Technical Specifications to Licenses Nos. DPR-32 and DPR-37. The proposed changes would establish new Safety Limits and Limiting Safety System Settings, limiting conditions of operation, and surveillance requirements that apply to the present core loadings allowing for fuel densification effects when analyzed using the new Westinghouse fuel densification model.

Your proposed changes to the Technical Specifications were submitted in response to our letter of February 27, 1974, which informed you that the Westinghouse Fuel Densification and Power Spike Model as described in WCAP-8218(P) and WCAP-8219(NP) has been reviewed by the staff and is acceptable for use.

We have completed our review of the proposed changes, and we conclude that these proposed changes do not present a significant hazards consideration and that there is reasonable assurance that the health and safety of the public will not be endangered. A copy of our Safety Evaluation supporting this action is enclosed.

We have designated our action as Change No. 15. Pursuant to 10 CFR Part 50, Section 50.59, the Technical Specifications appended to Licenses Nos. DPR-32 and DPR-37 are changed as shown in Attachment A.

Sincerely,

Karl R. Goller
Assistant Director for
Operating Reactors
Directorate of Licensing

C/P

Enclosures and cc on next page						
OFFICE ➤						RG
RNAME ➤						
DATE ➤						

APR 19 1974

Enclosures:
1. Attachment A - Change No. 15 to the Technical Specifications
2. Safety Evaluation

cc w/enclosures:
George D. Gibson, Esquire
Hunton, Williams, Gay & Gibson
P. O. Box 1535
Richmond, Virginia 23212

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Williamsburg, Virginia 23185

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MKarman, OGC
SATeets, L:OR-1
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OFFICE➤	L:ORB#1	L:ORB#1	L:OR			
SURNAME➤	VLRooney:dc	RAPurple	KRGoller			
DATE➤	4/18/74	4/ /74	4/ /74			

ATTACHMENT A

CHANGE NO. 15 TO THE TECHNICAL SPECIFICATIONS

VIRGINIA ELECTRIC & POWER COMPANY

DOCKET NOS. 50-280 AND 50-281

Delete the appropriate pages from the Technical Specifications and insert the attached replacement pages.

OFFICE ➤						
SURNAME ➤						
DATE ➤						

UNITED STATES ATOMIC ENERGY COMMISSION

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

VIRGINIA ELECTRIC & POWER COMPANY

DOCKET NOS. 50-280 AND 50-281

On February 27, 1974, we informed Virginia Electric & Power Company (VEPCO) that the Westinghouse Fuel Densification and Power Spike Model as described in WCAP-8218(P) and WCAP-8219(NP) has been reviewed by the staff. This new model is acceptable under certain conditions for the evaluation of densification effects in prepressurized PWR fuels that have been manufactured by Westinghouse. Reanalysis using the new model was not required. However, VEPCO was informed that they might elect to reanalyze the densification effects on Surry 1 and 2 core performance using the new model subject to the conditions provided. In that event, we would be prepared to receive and review their proposed Technical Specifications to relax existing limits together with the supporting reanalysis.

By letter dated March 21, 1974, VEPCO submitted proposed changes to the Surry 1 and 2 Technical Specifications. The Regulatory staff has reviewed the changes requested by VEPCO in Proposed Change No. 15, Surry Units 1 and 2 Technical Specifications, and finds them acceptable. Our basis for arriving at this conclusion is our review of their submittal and the applicability of the new Westinghouse fuel densification model to the Surry 1 and 2 fuel.

Based on the above considerations, we conclude that this action does not present a significant hazards consideration and that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner.

/s/

Vernon L. Rooney
Operating Reactors Branch #1
Directorate of Licensing

/s/

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

Date: APR 19 1974

OFFICE						
SURNAME						
DATE						

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

April 19, 1974

Docket Nos. 50-280
and 50-281

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

Change No. 15
Licenses Nos. DPR-32
and DPR-37

Gentlemen:

By letter dated March 21, 1974, you submitted proposed changes to the Surry 1 and 2 Technical Specifications to Licenses Nos. DPR-32 and DPR-37. The proposed changes would establish new Safety Limits and Limiting Safety System Settings, limiting conditions of operation, and surveillance requirements that apply to the present core loadings allowing for fuel densification effects when analyzed using the new Westinghouse fuel densification model.

Your proposed changes to the Technical Specifications were submitted in response to our letter of February 27, 1974, which informed you that the Westinghouse Fuel Densification and Power Spike Model as described in WCAP-8218(P) and WCAP-8219(NP) has been reviewed by the staff and is acceptable for use.

We have completed our review of the proposed changes, and we conclude that these proposed changes do not present a significant hazards consideration and that there is reasonable assurance that the health and safety of the public will not be endangered. A copy of our Safety Evaluation supporting this action is enclosed.

We have designated our action as Change No. 15. Pursuant to 10 CFR Part 50, Section 50.59, the Technical Specifications appended to Licenses Nos. DPR-32 and DPR-37 are changed as shown in Attachment A.

Sincerely,

Karl R. Goller

Karl R. Goller
Assistant Director for
Operating Reactors
Directorate of Licensing

Enclosures and cc on next page

Enclosures:
1. Attachment A - Change No. 15 to the Technical Specifications
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ATTACHMENT A

CHANGE NO. 15 TO THE TECHNICAL SPECIFICATIONS

VIRGINIA ELECTRIC & POWER COMPANY

DOCKET NOS. 50-280 AND 50-281

Delete the appropriate pages from the Technical Specifications and insert the attached replacement pages.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMIT, REACTOR CORE

Applicability

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

Objective

To maintain the integrity of the fuel cladding.

Specification

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

1. Exceed the limits shown in TS Figure 2.1-1 when full flow from three reactor coolant pumps exist.
2. Exceed the limits shown in TS Figure 2.1-2 when full flow from two reactor coolant pumps exist and the reactor coolant loop stop valves in the non-operating loop are open.
3. Exceed the limits shown in TS Figure 2.1-3 when full flow from two reactor coolant pumps exist and the reactor coolant loop stop valves in the non-operating loop are closed.
4. The reactor thermal power level shall not exceed 118% of rated power.

B. The safety limit is exceeded if the combination of Reactor Coolant System average temperature and thermal power level is at any time above the appropriate pressure line in TS Figures 2.1-1, 2.1-2 or 2.1-3; or the core thermal power exceeds 118% of rated power.

C. The fuel residence time shall be presently limited to 10,000 effective full power hours (EFPH) under design operating conditions provided the primary system pressure is reduced to 2000 psia by 3500 EFPH.

Basis

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the reactor coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed Departure From Nucleate Boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters; thermal power, reactor coolant temperature and pressure have been related to DNB through the W-3 correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNB ratio (DNBR) during steady state operation, normal operational transients and anticipated transients, is limited to 1.30. A DNBR

of 1.30 corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.[1]

The curves of TS Figure 2.1-1 which show the allowable power level decreasing with increasing temperature at selected pressures for constant flow (three loop operation) represent the loci of points of thermal power, coolant system average temperature, and coolant system pressure for which the DNB ratio is not less than 1.30. The area where clad integrity is assured is below these lines. In order to completely specify limits at all power levels, arbitrary constant upper limits of average temperature are shown for each pressure at powers lower than approximately 75% of rated power. The temperature limits at low power are considerably more conservative than would be required if they were based upon a minimum DNB ratio of 1.30 but are such that the plant conditions required to violate the limits are precluded by the self actuated safety valves on the steam generators. The three loop operation safety limit curve has been revised to allow for heat flux peaking effects due to fuel densification.

The curves of TS Figures 2.1-2 and 2.1-3, which show the allowable power level decreasing with increasing temperature at selected pressures for constant flow (two loop operation), represent 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. In order to completely specify limits at all

power levels, arbitrary constant upper limits of average temperatures are shown for each pressure at powers lower than approximately 45% of rated power. The limits at low power as well as the limits based on the average enthalpy at the exit of the core are considerably more conservative than would be required if they were based upon a minimum DNB ratio of 1.30. 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. TS Figures 2.1-2 and 2.1-3 have not been revised as these have been found to be adequate and conservative even including the heat flux peaking effects due to fuel densification.

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

These hot channel factors are higher than those calculated at full power over the range between that of all control rod assemblies fully withdrawn to maximum allowable control rod

assembly insertion. The control rod assembly insertion limits are covered by Specification 3.12. Adverse power distribution factors could occur at lower power levels because additional control rod assemblies are in the core; however, the control rod assembly insertion limits dictated by TS Figure 3.12-1 ensure that the DNBR is always greater at partial power than at full power.

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions for Reactor Coolant System temperature, pressure and thermal power level that would result in a DNB ratio of less than 1.30[3] based on steady state nominal operating power levels less than or equal to 100%, steady state nominal operating Reactor Coolant System average temperatures less than or equal to 563.5°F and a steady state nominal operating pressure of 1985 psig. Allowances are made in initial conditions assumed for transient analyses for steady state errors of +2% in power, +4°F in Reactor Coolant System average temperature and ±30 psi in pressure. The combined steady state errors result in the DNB ratio at the start of a transient being 10 percent less than the value at nominal full power operating conditions. The steady state nominal operating parameters and allowances for steady state errors given above are also applicable for two loop operation except that the steady state nominal operating power level is less than or equal to 60%.

The fuel overpower design limit is 118% of rated power. The overpower limit criterion is that core power be prevented from reaching a value at which fuel pellet melting would occur. The value of 118% power allows substantial margin to this limiting criterion. Additional peaking factors to account for local peaking due to fuel rod axial gaps and reduction in fuel pellet stack length have been included in the calculation of this limit.

The fuel residence time for Cycle 1 is limited to 10,000 EFPH to assure no fuel clad flattening without prior review by the Regulatory staff. If residence time of the present core will exceed 10,000 hours under design operating conditions, the assumption of clad flattening is presently required. Prior to 10,000 hours, the licensee may provide the additional analyses required for operation beyond 10,000 EFPH.

References

(1) FSAR Section 3.4

(2) FSAR Section 3.3

(3) FSAR Section 14.2

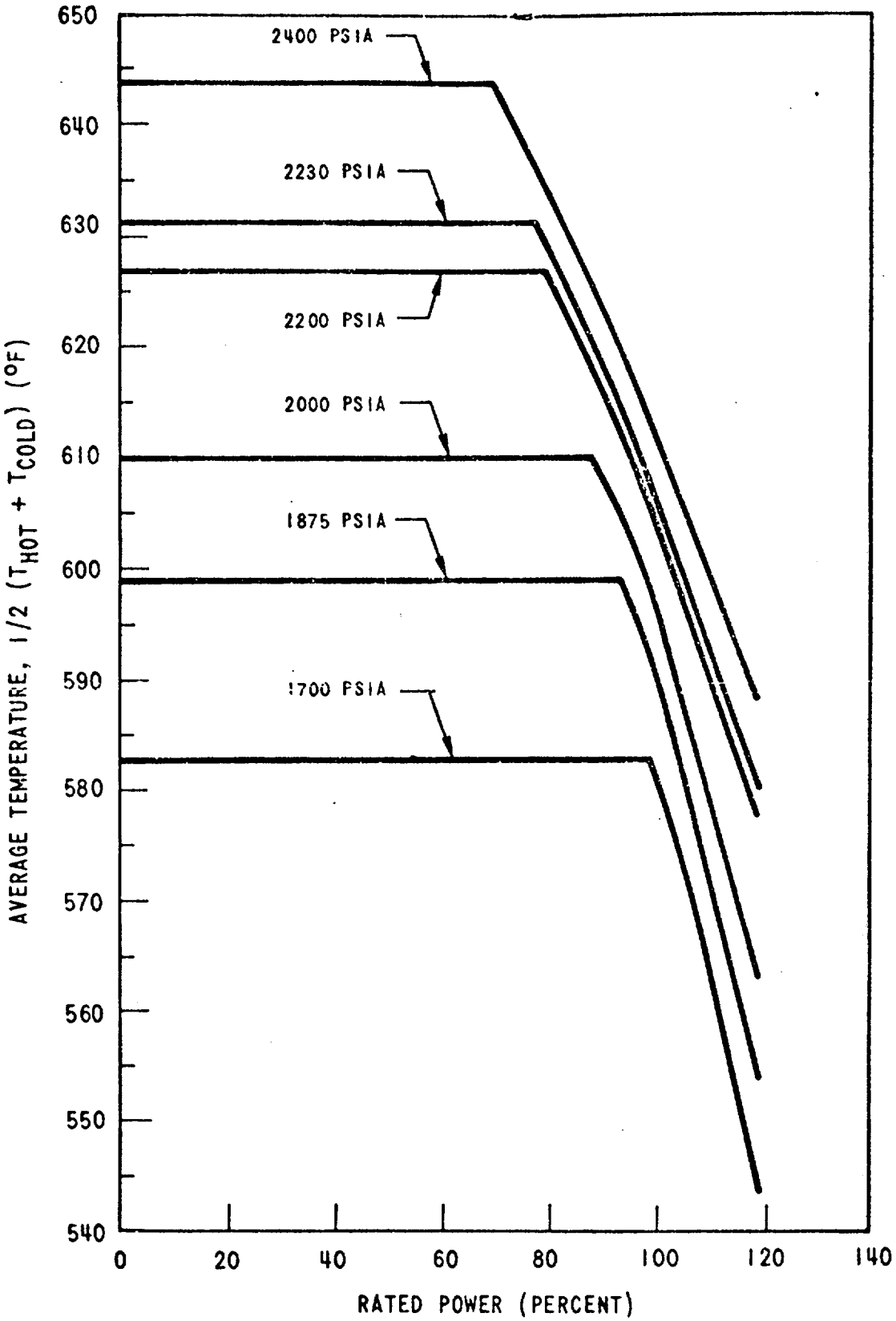


Figure 2.1-1 Reactor Core Thermal & Hydraulic Safety Limits - Three Loop Operation, 100% Flow

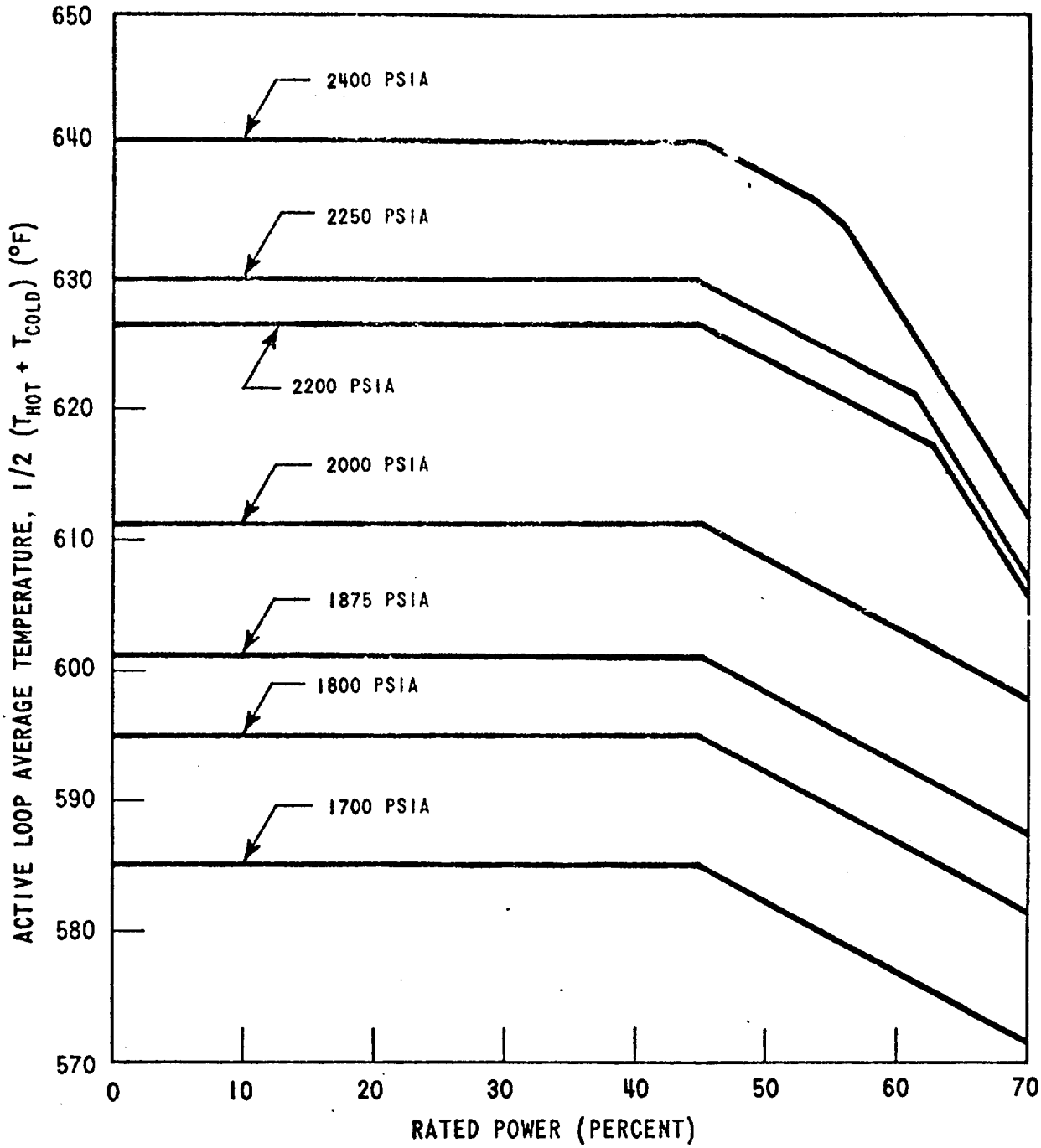


Figure 2.1-2. Reactor Core Thermal and Hydraulic Safety Limits, Two Loop Operation, Loop Stop Valves Open

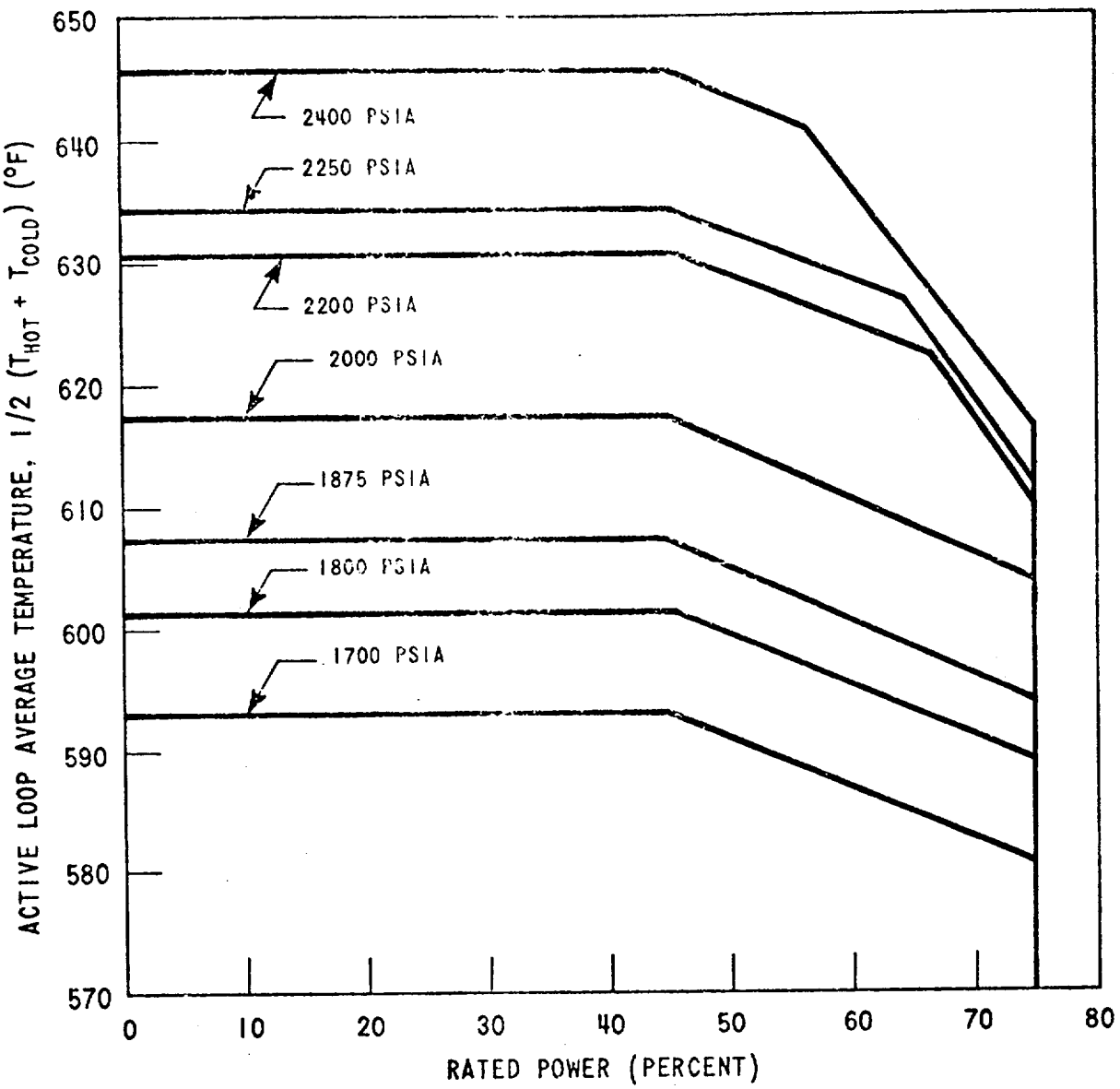


Figure 2.1-3. Reactor Core Thermal and Hydraulic Safety Limits, Two Loop Operation, Loop Stop Valves Closed

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Applicability

Applies to trip and permissive settings for instruments monitoring reactor power; and reactor coolant pressure, temperature, and flow; and pressurizer level.

Objective

To provide for automatic protective action in the event that the principal process variables approach a safety limit.

Specification

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

1. Startup protection

   (a) High flux, power range (low set point) - $\leq$ 25% of rated power.

   (b) High flux, intermediate range (high set point) - current equivalent to $\leq$ 25% of full power.

   (c) High flux, source range (high set point) - Neutron flux $\leq 10^6$ counts/sec.

2. Core Protection

   (a) High flux, power range (high set point) - $\leq$ 109% of rated power.

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

   (c) Low pressurizer pressure - $\geq$1715 psig.

(d) Overtemperature ΔT

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

where

ΔT_o = Indicated ΔT at rated thermal power, ^oF

T = Average coolant temperature, ^oF

T' = 563.5 ^oF

P = Pressurizer pressure, psig

P' = 1985 psig

K_1 = 1.095 (for 3 loop operation and 2 loop operation with the loop stop valves closed in the inoperable loop)

= 1.036 (for 2 loop operation with the loop stop valves open in the inoperable loop)

K_2 = 0.0139 (for 3 loop operation and 2 loop operation with the loop stop valves closed in the inoperable loop)

= 0.0139 (for 2 loop operation with the loop stop valves open in the inoperable loop)

K_3 = 0.000751 (for 3 loop operation and 2 loop operation with the loop stop valves closed in the inoperable loop)

= 0.000944 (for 2 loop operation with the loop stop valves open in the inoperable loop)

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

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

(e) Overpower ΔT

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

where

ΔT_o = Indicated ΔT at rated thermal power, °F

T = Average coolant temperature, °F

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

K_4 = A constant = 1.09

K_5 = 0 for decreasing average temperature

A constant, for increasing average temperature, 0.2 sec/°F

K_6 = 0 for $T \leq T'$

0.00108 for $T > T'$

f(ΔI) as defined in (d) above,

(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 setting

(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 - ≤ 1.0×10^6 lbs/hr

(d) Turbine trip

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

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

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

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

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

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

Basis

The power range reactor trip low setpoint provides protection in the power

range for a power excursion beginning from low power. This trip value was used in the safety analysis. (1) The intermediate range high flux, low setpoint and source range high flux, high setpoint trips provide additional protection against uncontrolled startup excursions. As power level increases, during startup, these trips are blocked to prevent unnecessary plant trips.

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

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

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

core safety limits as shown in Figures 2.1-1, 2, and 3. The reactor is prevented from reaching the overpower limit condition by action of the nuclear overpower and overpower ΔT trips. The overpower limit criteria is that core power be prevented from reaching a value at which fuel pellet centerline melting would occur. The overpower protection system set points include the effects of fuel densification.

In order to operate with a reactor coolant loop out of service (two-loop operation) and with the stop valves of the inactive loop open, the overtemperature ΔT trip setpoint calculation has to be modified by the adjustment of the variables K_1, K_2, and K_3. This adjustment, based on limits for two-loop operation, provides sufficient margin to DNB for the aforementioned transients during two loop operation. The required adjustment and subsequent mandatory calibrations are made in the protective system racks by qualified technicians* in the same manner as adjustments before initial startup and normal calibrations for three-loop operation. For two-loop operation with the inactive loop stop valves closed, the overtemperature ΔT trip setpoints used for three-loop operation

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

are adequate to protect against DNB for all combinations of pressure, power, coolant temperature, and axial power distribution provided only that the transient is slow with respect to transit delays from the core to the temperature detectors.

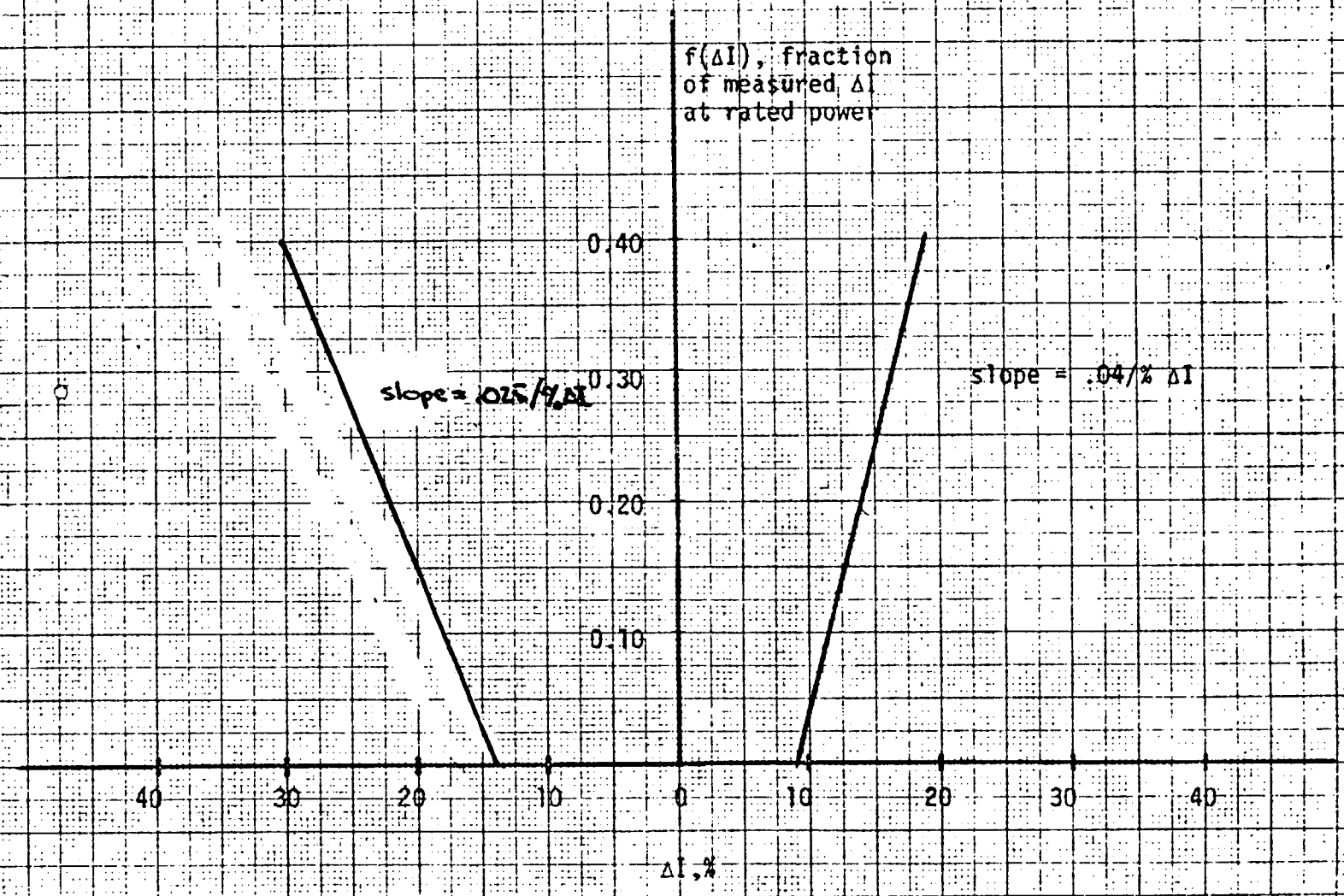
The overpower ΔT reactor trip prevents power density anywhere in the core from exceeding 118% of design power density as discussed Section 7 and specified in Section 14.2.2 of the FSAR and includes corrections for axial power distribution, change in density and heat capacity of water with temperature,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

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

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

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

FIGURE 2.3-1


f(ΔI), fraction
of measured ΔI
at rated power
0.40
0.30
0.20
0.10
slope = .0256/% ΔI
slope = .04/% ΔI
40
30
20
10
0
10
20
30
40
ΔI,%

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. The part length control rods will not be inserted. They will remain in the fully withdrawn position except for physics tests and for axial offset calibration which will be performed at 75% of rated power or less.

6. Insertion limits do not apply during physics tests or during periodic exercise of individual rods. However, the shutdown margin indicated in TS Figure 3.12-7 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.

B. Power Distribution Limits

1. At all times the hot channel factors defined in the basis must meet the following limits:

a. $F_Q^N \leq 2.39\ (1 + 0.2\ (1-P))$ in the flux difference range -14 to + 9 percent

$$F_{\Delta H}^N \leq 1.55\ (1 + 0.2\ (1-P))$$

where P is the fraction of rated power at which the core is operating.

b. If peaking factors exceed the limits of Section B.1.a, the reactor power and high neutron flux trip setpoint shall be reduced by 1 percent for every percent excess over $F^N_{\Delta H}$ or F^N_Q, whichever is limiting. If the peaking factors cannot be corrected within 1 day, the overpower ΔT and overtemperature ΔT trip setpoints shall be similarly reduced.

c. The allowable quadrant to average power tilt is

$$T = 2.0 + 50\ [1.42/F_{xy} - 1] \leq 10\%$$

where F_{xy} is 1.42, or the value of the unrodded horizontal plane peaking factor appropriate to F_Q as determined by a movable in-core detector map taken on at least a monthly basis; and T is the percentage operating quadrant tilt limit, having a value of 2% if F_{xy} is 1.42 or a value up to 10% as selected by the operator if the option to measure F_{xy} is in effect.

d. At rated power, the indicated axial flux difference must be maintained within the range +9 percent to -14 percent.

e. For every 4 percent below rated power, the permissible positive flux difference range is extended by 1 percent. For every 5 percent below rated power, the permissible negative flux difference is extended by 2 percent.

f. Following initial loading and each subsequent reloading, a power distribution map, using the Movable Detector System, shall be made to confirm that power distribution limits are met, in the full power configuration, before the plant is operated above 75 percent of rating.

g. For operation of the reactor above 75% of rated power a full movable incore detector map shall be taken monthly. A full map is defined as surveillance of a minimum of 40 fuel assembly detector thimbles with at least 8 per quadrant.

2. If the quadrant to average power tilt exceeds a value T% as selected in specification B.1.c., except for physics and rod exercise testing, then:

   a. The hot channel factors shall be determined within 2 hours and the power level adjusted to meet the specification of B.1.b., or

   b. If the hot channel factors are not determined within two hours, the power and high neutron flux trip setpoint shall be reduced from rated power, 2% for each percent of quadrant tilt.

   c. If the quadrant to average power tilt exceeds ± 10%, except for physics tests, the power level and high neutron flux trip setpoint will be reduced from rated power, 2% for each percent of quadrant tilt.

3. If after a further period of 24 hours, the power tilt in 2 above is not corrected to less than ± T%:

   a. If design hot channel factors for rated power are not exceeded, an evaluation as to the cause of the discrepancy shall be made and reported as an abnormal occurrence to the Atomic Energy Commission.

   b. If the design hot channel factors for rated power are exceeded and the power is greater than 10% - the Atomic Energy Commission shall be notified and the nuclear overpower,

D. If the reactor is operating above 75% of rated power with one excore nuclear channel out of service, the core quadrant power balance shall be determined.

1. Once per day, and
2. After a change in power level greater than 10% or more than 30 inches of control rod motion.

The core quadrant power balance shall be determined by one of the following methods:

1. Movable detectors (at least two per quadrant)
2. Core exit thermocouples (at least four per quadrant).

E. Inoperable Rod Position Indicator Channels

1. If a rod position indicator channel is out of service then:
   a) For operation between 50% and 100% of rated power, the position of the RCC shall be checked indirectly by core instrumentation (excore detector and/or thermocouples and/or movable incore detectors) every shift or subsequent to motion, of the non-indicating rod, exceeding 24 steps, whichever occurs first.
   b) During operation below 50% of rated power no special monitoring is required.

2. Not more than one rod position indicator (RPI) channel per group nor two RPI channels per bank shall be permitted to be inoperable at any time.

F. Misaligned or Dropped Control Rod

1. If the Rod Position Indicator Channel is functional and the associated part length or full length control rod is more than 15 inches out of alignment with its bank and cannot be realigned, then unless the hot channel factors are shown to be within design limits as specified in Section 3.12.B-1 within 8 hours, power shall be reduced so as not to exceed 75% of permitted power.

2. To increase power above 75% of rated power with a part-length or full length control rod more than 15 inches out of alignment with its bank an anlysis shall first be made to determine the hot channel factors and the resulting allowable power level based on Section 3.12.B.

Basis

The reactivity control concept assumed for operation is that reactivity changes accompanying changes in reactor power are compensated by control rod assembly motion. Reactivity changes associated with xenon, samarium, fuel depletion, and large changes in reactor coolant temperature (operating temperature to cold shutdown) are compensated for by changes in the soluble boron concentration. During power operation, the shutdown groups are fully

malpositioned control rod assemblies with inoperable rod position indicators because, even with an unnoticed complete assembly misalignment (part-length of 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. (2)

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 20.4 kW/ft. 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 initial steady state conditions for the peak linear power for a loss-of-coolant accident must not exceed the values assumed in the accident evaluation. This limit is required in order for the maximum clad temperature to remain below that established by the Interim Policy Statement for LOCA. To aid in specifying the

limits on power distribution the following hot channel factors are defined. F_Q, Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod divided by the average fuel rod heat flux, allowing for manufacturing tolerance on fuel pellets and rods.

F_Q^N, Nuclear Heat Flux Hot Channel Factor, is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and rod dimensions.

F_Q^E, Engineering Heat Flux Hot Channel Factor, is defined as the ratio between F_Q and F_Q^N and is the allowance on heat flux required for manufacturing tolerances.

$F_{\Delta H}^N$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod on which minimum DNBR occurs to the average rod power.

It has been determined by analysis that the design limits on peak local power density on minimum DNBR at full power and LOCA are met, provided:

$$F_Q^N \leq 2.39 \text{ and } F_{\Delta H}^N \leq 1.55$$

These quantities are measurable although there is not normally a requirement to do so. Instead it has been determined that, provided certain conditions are observed, the above hot channel factor limits

will be met at rated power; these conditions are as follows:

1. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the bank demand position.

2. Control rod banks are sequenced with overlapping banks as shown in Figure 3.12-1 and 3.12-2.

3. The control bank insertion limits are not violated.

4. Axial power distribution guidelines, which are given in terms of flux difference control are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of axial offset which is defined as the difference in power between the top and bottom halves of the core. Calculation of core peaking factors under a variety of operation conditions have been correlated with axial offset. The correlation shows that an F_Q^N of 2.39 and allowed DNB shapes, including the effects of fuel densification, are not exceeded if the axial offset (flux difference) is maintained between -17 and +12%. The specified limits of -14 and +9% allow for a 3% error in the axial offset.

For operation at rated power, design limits are met, provided,

$F_Q^N \leq 2.39\ (1 + 0.2(1-P))$ in the indicated flux difference range of +9 to -14%

and $F_{\Delta H}^N \leq 1.55\ (1 + 0.2\ (1-P))$

The permitted relaxation allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, these hot channel factors limits are met.

For normal operation and anticipated transients the core is protected from exceeding 20.4 kW/ft locally, and from going below a minimum DNBR of 1.30, by automatic protection on power, flux difference, pressure and temperature. Only conditions 1 through 3, above, are mandatory since the flux difference is an explicit input to the protection system.

Measurements of the hot channel factors are required as part of startup physics tests and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors.

In the specified limit of F_Q^N there is a 5% allowance for uncertainties (1) which means that normal operation of the core within the defined conditions and procedures is expected to result in $F_Q^N \leq 2.39/1.05$ even on a

worst case basis. When a measurement is taken experimental error must be allowed for and 5% is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system.

In the specified limit of $F^N_{\Delta H}$ there is a 8% allowance for uncertainties (1) which means that normal operation of the core is expected to result in $F^N_{\Delta H} \leq 1.55/1.08$. The logic behind the larger uncertainty in this case is that (a) abnormal perturbations in the radial power shape (e.g., rod misalignment) affect $F^N_{\Delta H}$, in most cases without necessarily affecting F^N_Q, through movement of part length rods, and can limit it to the desired value, (b) while the operator has some control over F^N_Q through F^N_z by motion of control rods, 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 can be compensated for in F^N_Q by tighter axial control, but compensation for $F^N_{\Delta H}$ is less readily available.

At the option of the operator, credit may be taken for measured decreases in the unrodded horizontal plane peaking factor, F_{xy}. This credit may take the form of an expansion of permissible quadrant

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 Atomic Energy Commission per Section 6.6 of these Specifications.

B. During periods of power operation at greater than 10% of power, design peaking factors, F_Q^N and $F_{\Delta H}^N$, shall be determined monthly using data from limited core maps. If these factors exceed values of

$F_Q^N \leq 2.39\ (1 + 0.2\ (1-P))$ in the flux difference range -14 to +9 percent

$F_{\Delta H}^N \leq 1.55\ (1 + 0.2\ (1-P))$

(where P is the fraction of rated power at which the core is operating) an evaluation as to the cause of the anomaly shall be made.

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 specifies 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 are expected to decrease. A monthly determination of F_q^N and $F_{\Delta H}^N$ is adequate to ensure that core reactivity changes with burnup have not significantly altered peaking factors in an adverse direction.

UNITED STATES ATOMIC ENERGY COMMISSION

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

VIRGINIA ELECTRIC & POWER COMPANY

DOCKET NOS. 50-280 AND 50-281

On February 27, 1974, we informed Virginia Electric & Power Company (VEPCO) that the Westinghouse Fuel Densification and Power Spike Model as described in WCAP-8218(P) and WCAP-8219(NP) has been reviewed by the staff. This new model is acceptable under certain conditions for the evaluation of densification effects in prepressurized PWR fuels that have been manufactured by Westinghouse. Reanalysis using the new model was not required. However, VEPCO was informed that they might elect to reanalyze the densification effects on Surry 1 and 2 core performance using the new model subject to the conditions provided. In that event, we would be prepared to receive and review their proposed Technical Specifications to relax existing limits together with the supporting reanalysis.

By letter dated March 21, 1974, VEPCO submitted proposed changes to the Surry 1 and 2 Technical Specifications. The Regulatory staff has reviewed the changes requested by VEPCO in Proposed Change No. 15, Surry Units 1 and 2 Technical Specifications, and finds them acceptable. Our basis for arriving at this conclusion is our review of their submittal and the applicability of the new Westinghouse fuel densification model to the Surry 1 and 2 fuel.

Based on the above considerations, we conclude that this action does not present a significant hazards consideration and that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner.

Vernon L. Rooney
Operating Reactors Branch #1
Directorate of Licensing

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

Date: April 19, 1974