DEC 27 1974

Docket Nos. 50-280 ✓
and 50-281

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

Gentlemen:

The Commission has issued the enclosed Amendments No. 4 to Facility Licenses No. DPR-32 and DPR-37 for the Surry Power Station, Units 1 and 2. The amendments include Change No. 19 to your Technical Specifications for each license and are in response to your request dated October 17, 1974, as supplemented November 20, 1974.

The amendments revise the provisions in the Technical Specifications relating to the replacement of 84 of 157 fuel assemblies in the reactor core, constituting refueling of the core for second cycle operation of Unit 1.

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

Sincerely,

/S/ D. L. Rooney

for

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

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

cc: See next page

Const R

DEC 27 1974

cc w/enclosures:
Michael W. Maupin, Esquire
Hunton, Williams, Gay & Gibson
Post Office Box 1535
Richmond, Virginia 23261

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

Swem Library
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Williamsburg, Virginia 23185

cc w/enclosures & incoming:
Ms. Susan T. Wilburn
Commonwealth of Virginia
Council on the Environment
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Richmond, Virginia 23219

Mr. Robert Blanco
Environmental Protection Agency
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6th and Walnut Streets
Philadelphia, Pennsylvania 19106

bcc: H. J. McAlduff, OROO
J. R. Buchanan, ORNL
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VIRGINIA ELECTRIC & POWER COMPANY

DOCKET NO. 50-280

SURRY POWER STATION UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 4
License No. DPR-32

1. The Atomic Energy Commission (the Commission) having found that:

   A. The application for amendment by Virginia Electric & Power Company (the licensee) dated October 17, 1974, as supplemented November 20, 1974, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended, ~~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. No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

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

OFFICE ➤						
SURNAME ➤						
DATE ➤						

"3.B Technical Specifications

The Technical Specifications contained in Appendix A, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 19."

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

FOR THE ATOMIC ENERGY COMMISSION

/S/ Roger S. Boyd

for A. Giambusso, Deputy Director
for Reactor Proejcts
Directorate of Licensing

Attachment:
Change No. 19 to the
Technical Specifications

Date of Issuance: DEC 27 1974

OFFICE ➤						
SURNAME ➤						
DATE ➤						

ATTACHMENT TO LICENSE AMENDMENT NO. 4

CHANGE NO. 19 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-32

VIRGINIA ELECTRIC & POWER COMPANY

SURRY POWER STATION UNIT NO. 1

DOCKET NO. 50-280

Revise Appendix A as follows:

Remove Page	Insert New Page
2.1-1	2.1-1
2.1-2	2.1-2
2.1-3	2.1-3
2.1-4	2.1-4
2.1-5	2.1-5
2.1-6	2.1-6
Figure 2.1-1	Figure 2.1-1A
—	Figure 2.1-1B
Figure 2.1-2	Figure 2.1-2A
—	Figure 2.1-2B
Figure 2.1-3	Figure 2.1-3A
—	Figure 2.1-3B
Figure 2.3-1	Figure 2.3-1
Figure 2.3-2	Figure 2.3-2
Figure 2.3-3	Figure 2.3-3
Figure 2.3-4	Figure 2.3-4
Figure 2.3-5	Figure 2.3-5
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3.12-13	3.12-13
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Remove Page	Insert New Page
Figure 3.12-1	Figure 3.12-1A
—	Figure 3.12-1B
Figure 3.12-2	Figure 3.12-2
Figure 3.12-4	Figure 3.12-4A
—	Figure 3.12-4B
4.3-1	4.3-1
4.3-4	4.3-4
5.3-1	5.3-1
5.3-2	5.3-2

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SURNAME ▶						
DATE ▶						

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMIT, REACTOR CORE

Applicability

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

Objective

To maintain the integrity of the fuel cladding.

Specification

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

1. Exceed the limits shown in TS Figures 2.1-1A (Unit 1) and 2.1-1B (Unit 2) when full flow from three reactor coolant pumps exist. 19
2. Exceed the limits shown in TS Figures 2.1-2A (Unit 1) and 2.1-2B (Unit 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 Figures 2.1-3A (Unit 1) and 2.1-3B 19 (Unit 2) 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 2.1-1A, 2.1-1B, 19
2.1-2A, 2.1-2B, 2.1-3A, or 2.1-3B; or the core thermal power exceeds 118% of rated power.

C. The fuel residence time shall be limited to 15,500 effective full power hours (EFPH) for Cycles 1 and 2 of Unit 1 and to 10,000 EFPH 19
for Cycle 1 of Unit 2 provided the Unit 2 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

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.

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

1. Exceed the limits shown in TS Figures 2.1-1A (Unit 1) and 2.1-1B (Unit 2) when full flow from three reactor coolant pumps exist. 19
2. Exceed the limits shown in TS Figures 2.1-2A (Unit 1) and 2.1-2B (Unit 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 Figures 2.1-3A (Unit 1) and 2.1-3B (Unit 2) when full flow from two reactor coolant pumps exist and the reactor coolant loop stop valves in the non-operating loop are closed. 19

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 2.1-1A, 2.1-1B, 2.1-2A, 2.1-2B, 2.1-3A, or 2.1-3B; or the core thermal power exceeds 118% of rated power. 19

C. The fuel residence time shall be limited to 15,500 effective full power hours (EFPH) for Cycles 1 and 2 of Unit 1 and to 10,000 EFPH for Cycle 1 of Unit 2 provided the Unit 2 primary system pressure is reduced to 2000 psia by 3500 EFPH. 19

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 Figures 2.1-1A (Unit 1) and 2.1-1B (Unit 2) 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. 19

The curves of TS Figures 2.1-2A (Unit 1), 2.1-2B (Unit 2), 2.1-3A (Unit 1), and 2.1-3B (unit 2) which show the allowable power level decreasing with increasing temperature at selected pressures for constant flow (two loop operation), represent limits equal to, or more conservative, than the loci of 19

points of thermal power, coolant system average temperature, and coolant system
pressure for which either the DNB ratio is equal to 1.30 or the average enthalpy
at the exit of the core is equal to the saturation value. At low pressures or
high temperatures the average enthalpy at the exit of the core reaches saturation
before the DNB ratio reaches 1.30 and, thus, this arbitrary limit is conservative
with respect to maintaining clad integrity. The plant conditions required to 19
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. 19

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 2.1-1A through 2.1-3B are based on a $F^N_{\Delta H}$ of 1.55, a 1.55 cosine axial 19
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 Figures 3.12-1A (Unit 1) and 3.12-1B (Unit 2) ensure that the 19
DNBR is always greater at partial power than at full power.

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions for Reactor Coolant System temperature, pressure and thermal power level that would result in a DNB ratio of less than 1.30[3] based on steady state nominal operating power levels less than or equal to 100%, steady state nominal operating Reactor Coolant System average temperatures less than or equal to 574.4°F for Unit 1 and 563.5°F for Unit 2 and a steady state nominal operating pressure of 2235 psig for Unit 1 19
and 1985 psig for Unit 2. Allowances are made in initial conditions assumed for transient analyses for steady state errors of +2% in power, +4°F in Reactor Coolant System average temperature and ±30 psi in pressure. The combined steady state errors result in the DNB ratio at the start of a transient being 10 per cent less than the value at nominal full power operating conditions. The steady state nominal operating parameters and allowances for steady state errors given above are also applicable for two loop operation except that the steady state nominal operating power level is less than or equal to 60%.

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

The fuel residence time is limited to 15,500 EFPH for Cycles 1 and 2 of Unit 1
and to 10,000 EFPH for Cycle 1 of Unit 2 to assure no fuel clad flattening will 19
occur in the cores without prior review by the Regulatory Staff. Prior to 10,000 EFPH for Cycle 1 of Unit 2 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

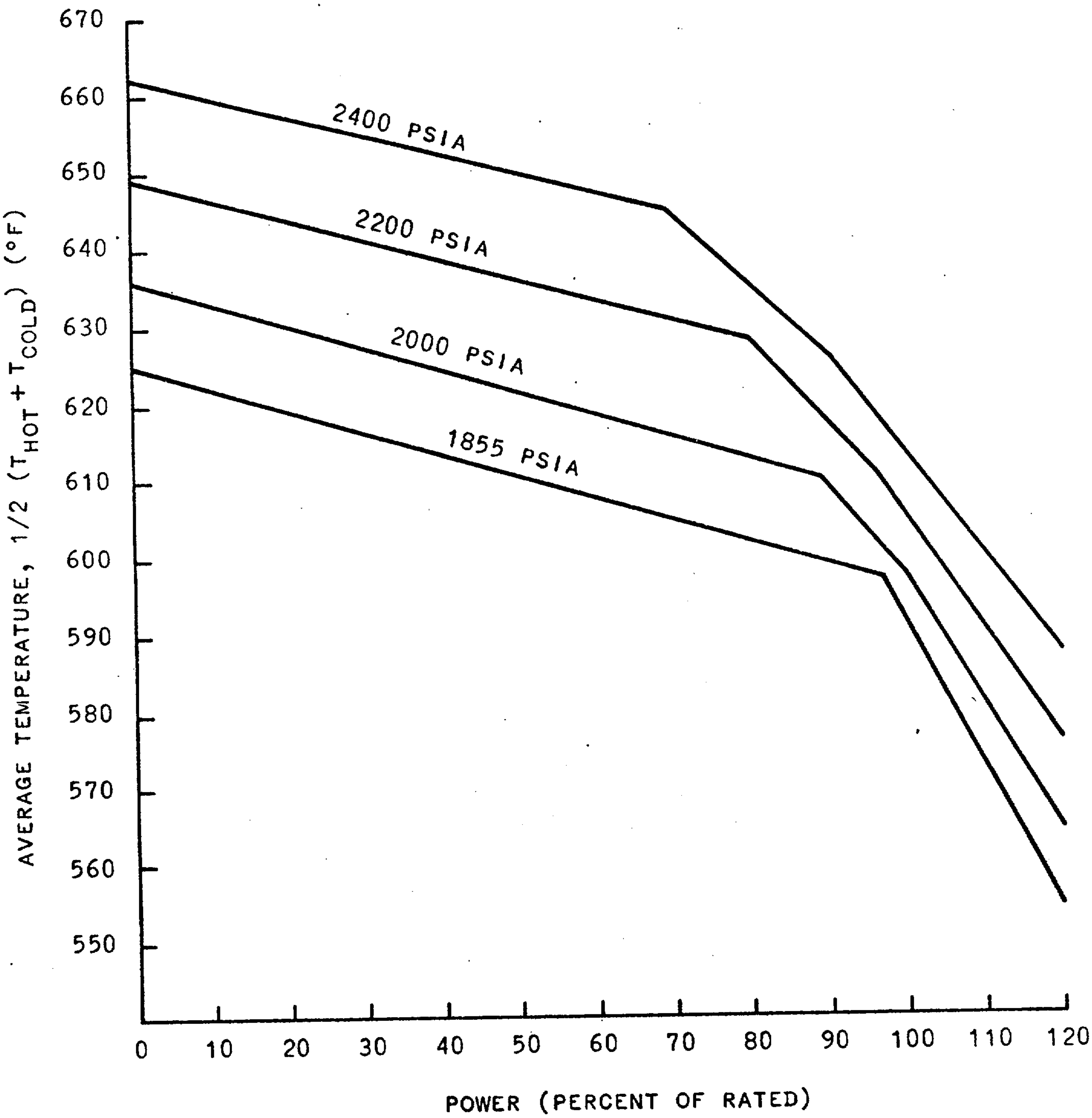


FIGURE 2.1-1A REACTOR CORE THERMAL & HYDRAULIC SAFETY LIMITS-
THREE LOOP OPERATION, 100% FLOW-UNIT NO. 1

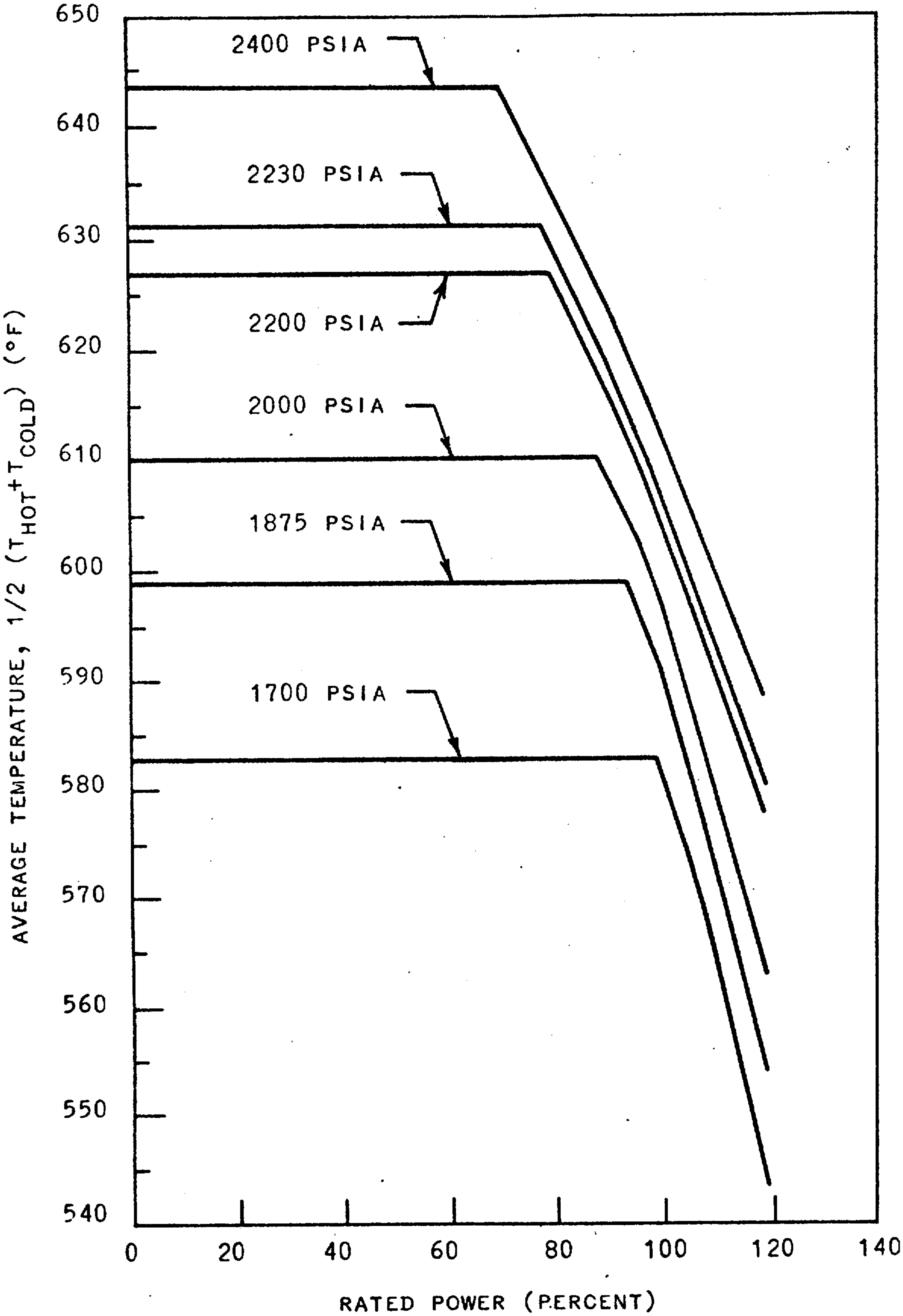


FIGURE 2.1-1B REACTOR CORE THERMAL & HYDRAULIC SAFETY LIMITS-THREE LOOP OPERATION, 100% FLOW-UNIT NO. 2

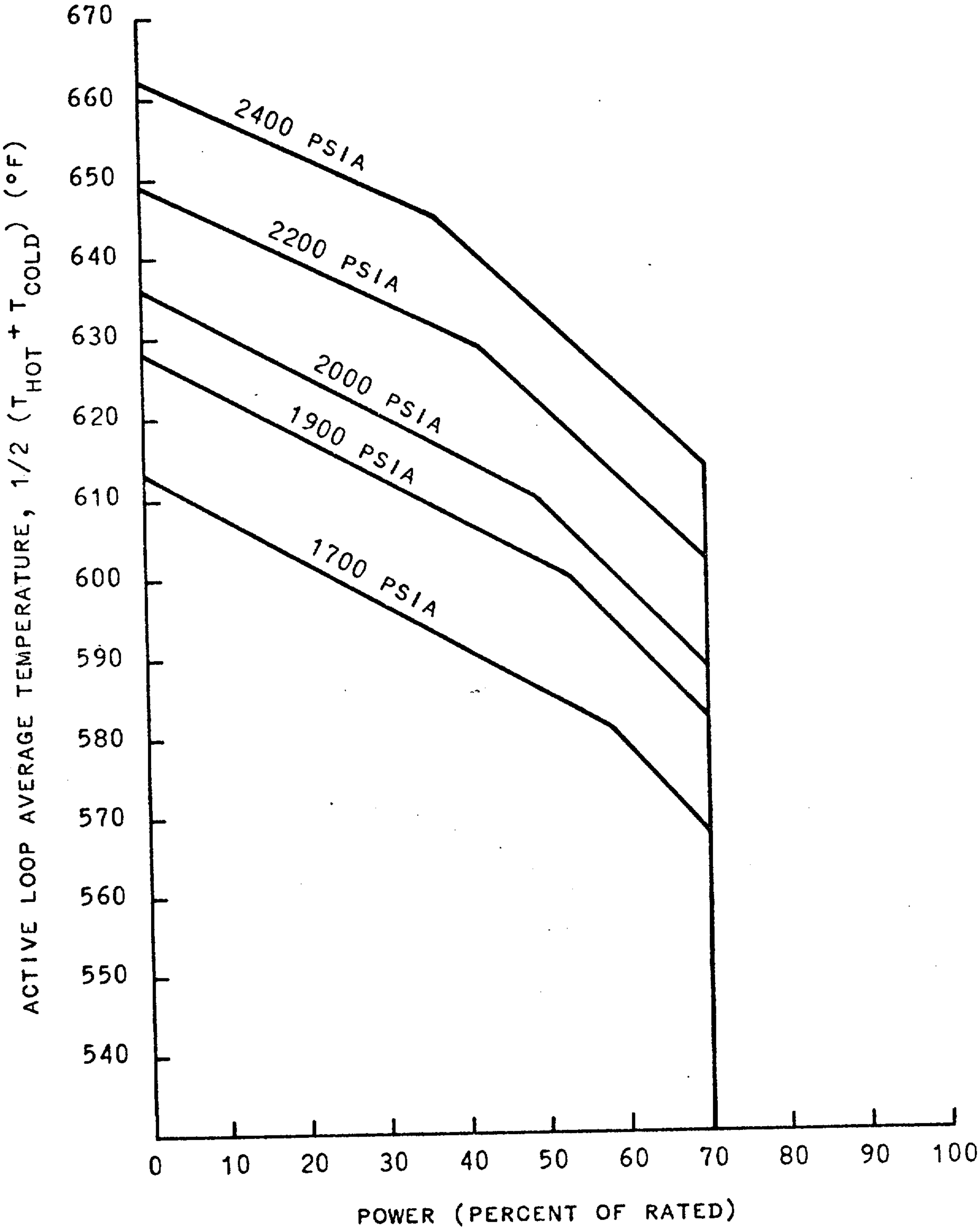


FIGURE 2.1-2A REACTOR CORE THERMAL AND HYDRAULIC SAFETY LIMITS, TWO LOOP OPERATION, LOOP STOP VALVES OPEN-UNIT NO. 1

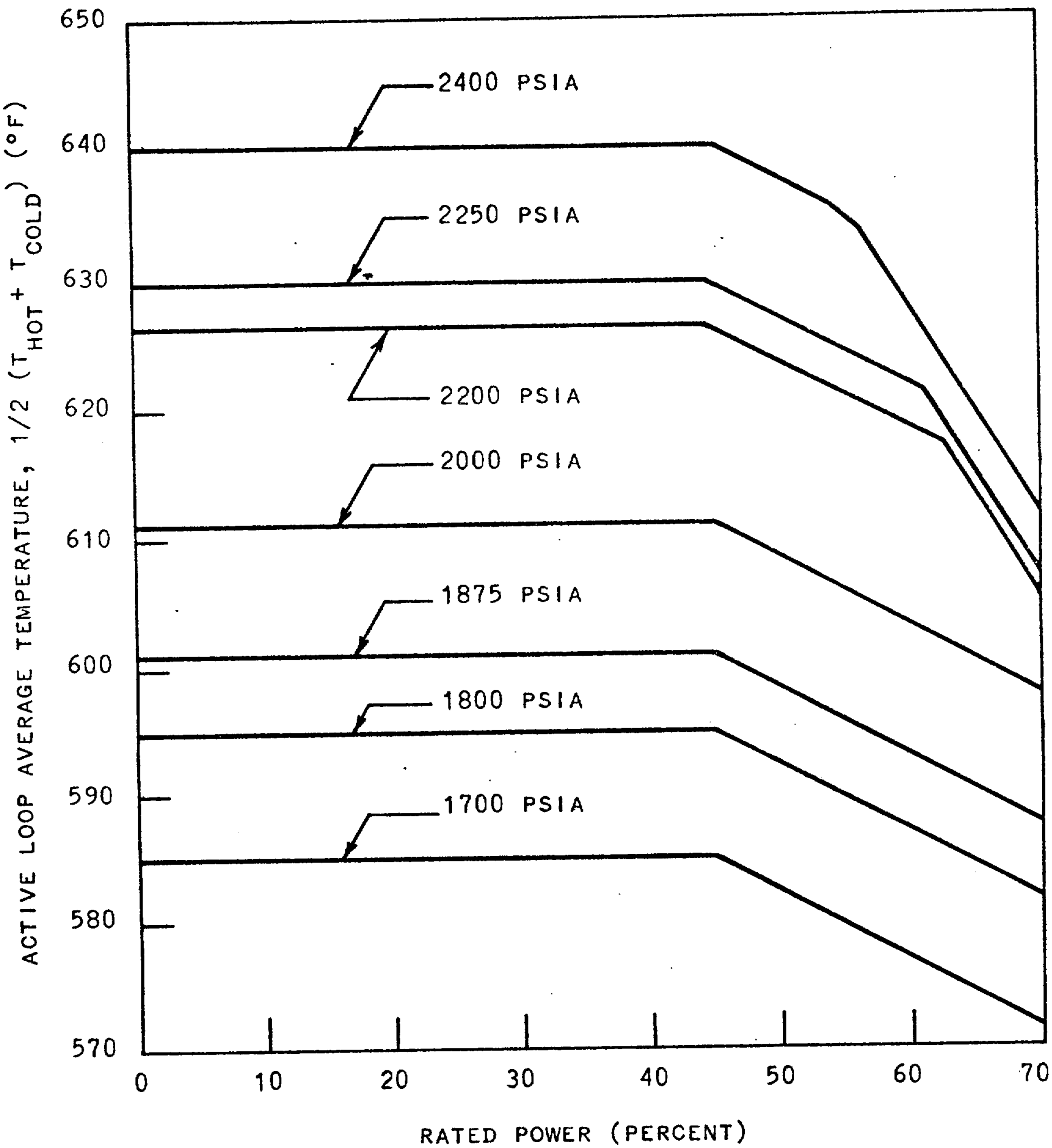


FIGURE 2.1-2B REACTOR CORE THERMAL AND HYDRAULIC SAFETY LIMITS, TWO LOOP OPERATION, LOOP STOP VALVES OPEN-UNIT NO. 2

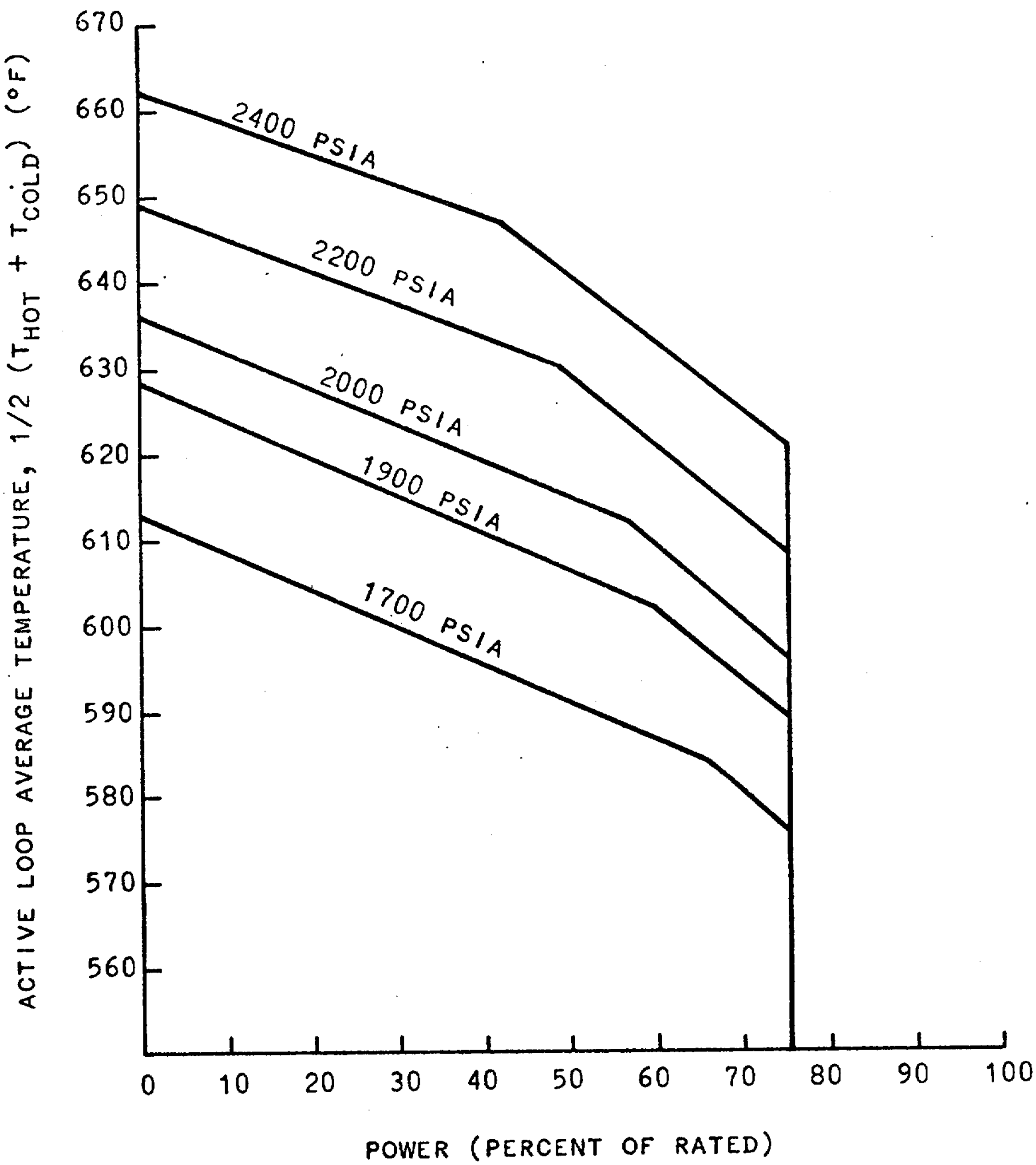


FIGURE 2.1-3A REACTOR CORE THERMAL AND HYDRAULIC SAFETY LIMITS, TWO LOOP OPERATION, LOOP STOP VALVES CLOSED-UNIT NO. 1

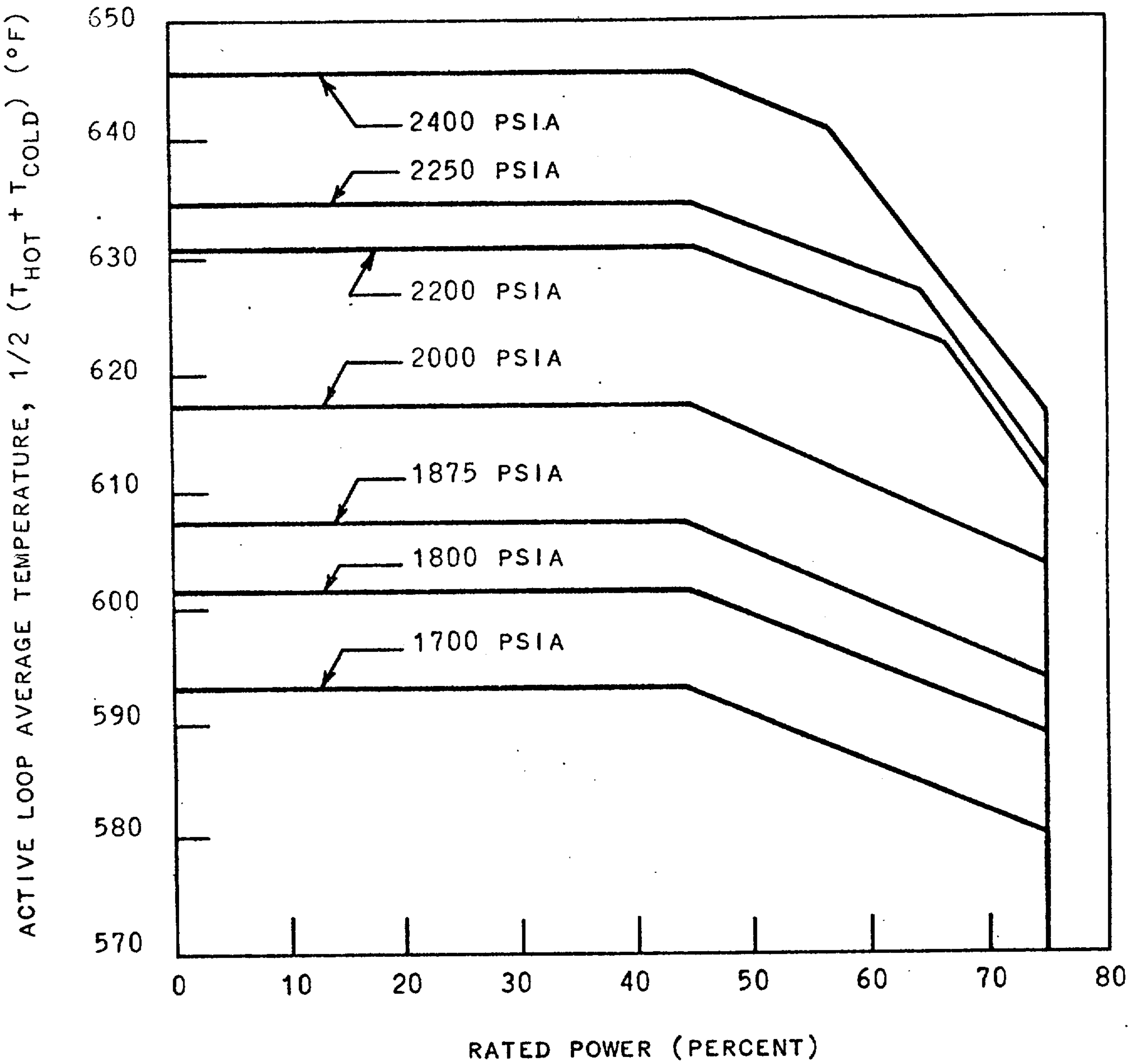


FIGURE 2.1-3B REACTOR CORE THERMAL AND HYDRAULIC SAFETY LIMITS, TWO LOOP OPERATION, LOOP STOP VALVES CLOSED-UNIT NO. 2

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 - ≤ 2385 psig.

(c) Low pressurizer pressure - ≥ 1860 psig - Unit 1 19

≥ 1715 psig - Unit 2

(d) Overtemperature ΔT

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

where

ΔT_o = Indicated ΔT at rated thermal power, ^oF

T = Average coolant temperature, ^oF

T' = 574.4^oF - Unit 1

= 563.5^oF - Unit 2

P = Pressurizer pressure, psig

P' = 2235 psig - Unit 1

= 1985 psig - Unit 2

$$\left.\begin{array}{ll} K_1 & = 1.12 \text{ - Unit 1} \\ & = 1.095 \text{ - Unit 2} \\ K_2 & = 0.01012 \text{ - Unit 1} \\ & = 0.0139 \text{ - Unit 2} \\ K_3 & = 0.000554 \text{ - Unit 1} \\ & = 0.000751 \text{ - Unit 2} \end{array}\right\} \text{for 3-loop operation}$$

19

$$\left.\begin{array}{ll} K_1 & = 0.951 \text{ - Unit 1} \\ & = 1.036 \text{ - Unit 2} \\ K_2 & = 0.01012 \text{ - Unit 1} \\ & = 0.0139 \text{ - Unit 2} \\ K_3 & = 0.000554 \text{ - Unit 1} \\ & = 0.000944 \text{ - Unit 2} \end{array}\right\} \text{for 2-loop operation with loop stop valves open in operable loop}$$

K_1 = 1.026 - Unit 1
= 1.095 - Unit 2
K_2 = 0.01012 - Unit 1
= 0.0139 - Unit 2
K_3 = 0.000554 - Unit 1
= 0.000751 - Unit 2

for 2-loop operation with loop stop valves closed in inoperable loop 19

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

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

(e) Overpower ΔT

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

where

ΔT_o = Indicated ΔT at rated thermal power, ^oF

T = Average coolant temperature, ^oF

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

K_4 = A constant = 1.09

K_5 = 0 for decreasing average temperature

A constant, for increasing average temperature, 0.02/^oF 19

K_6 = 0 for $T \leq T'$

= 0.00108 for $T > T'$

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

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

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

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

3. Other reactor trip setting

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

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

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

   (d) Turbine trip

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

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

1. The reactor trip on low pressurizer pressure, high pressurizer level, turbine trip, and low reactor coolant flow for two or more loops shall be unblocked when power $\geq$10% of rated power.
2. The single loop loss of flow reactor trip shall be unblocked when the power range nuclear flux $\geq$50% of rated power. During two loop operation with the loop stop valves in the inactive loop open, this blocking setpoint, established by Permissive 8, may be increased to 60% of rated power only after the overtemperature ΔT setpoint is adjusted to the mandatory two loop value. For two loop operation with the loop stop valves of the inactive loop closed, Permissive 8 may be increased to 65% of rated power only after the overtemperature ΔT setpoint is adjusted to the mandatory value for this condition. 19

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

Basis

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

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

The overtemperature ΔT reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided only that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 3 seconds), and pressure is within the range between high and low pressure reactor trips.

With normal axial power distribution, the reactor trip limit, with allowance for errors,[2] is always below the core safety limit as shown on TS Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip limit is automatically reduced.[4][5]

The overpower and overtemperature protection system setpoints have been revised to include effects of fuel densification on core safety limits. The revised setpoints in the Technical Specifications will ensure that the combination of power, temperature, and pressure will not exceed the revised
core safety limits as shown in Figures 2.1-1A through 2.1-3B. The reactor 19
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 either open or 19
closed, 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 of two-loop operation, provides sufficient margin to DNB for the aforementioned transients during two loop operation. The required adjustment and subsequent mandatory calibrations are made in the protective system racks by qualified technicians* in the same manner as adjustments before initial
startup and normal calibrations for three-loop operation. 19

*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.

19

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)

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

Above 10% power, an automatic reactor trip will occur if two or more reactor coolant pumps are lost. Above 50% power during three-loop operation, an automatic reactor trip will occur if any pump is lost or de-energized. This latter trip will prevent the minimum value of the DNBR from going below 1.30 during normal operational transients and anticipated transients when only two loops are in operation and the overtemperature ΔT trip setpoint is adjusted to the valve specified for three-loop operation. During two-loop operation with the loop stop valves in the inactive loop open, and the overtemperature ΔT trip setpoint is adjusted to the value specified for this condition, a reactor trip at 60% power will prevent the minimum value of DNBR from going below 1.30 during normal operational transients and anticipated transients when only two loops are in operation. During two-loop operation with the inactive loop stop valves closed and the overtemperature ΔT trip setpoint is adjusted to the value specified for this condition, a reactor trip at 65% power will prevent the minimum DNBR from 19
going below 1.30 during normal operational transients and anticipated transients.

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

References

(1) FSAR Section 14.2.1

(2) FSAR Section 14.2

(3) FSAR Section 14.5

(4) FSAR Section 7.2

(5) FSAR Section 3.2.2

(6) FSAR Section 14.2.9

(7) FSAR Section 7.2

plant shall be shutdown and the reactor made subcritical by inserting all control banks into the core. The shutdown rods may remain withdrawn.

c. A minimum of one pump in a non-isolated loop, or one residual heat removal pump and its associated flow path, shall be in operation during reactor coolant boron concentration reduction.

d. Reactor power shall not exceed 50% of rated power with only two pumps in operation unless the overtemperature ΔT trip setpoints have been changed in accordance with Section 2.3, after which power shall not exceed 60% with the inactive loop stop valves open and 65% with the inactive loop stop valves closed. 19

2. Steam Generator

A minimum of two steam generators in non-isolated loops shall be operable when the average reactor coolant temperature is greater than 350°F.

3. Pressurizer Safety Valves

a. One valve shall be operable whenever the head is on the reactor vessel, except during hydrostatic tests.

In addition to the above safeguards, interlocks are used during refueling to assure safe handling of the fuel assemblies. An excess weight interlock is provided on the lifting hoist to prevent movement of more than one fuel assembly at a time. The spent fuel transfer mechanism can accommodate only one fuel assembly at a time.

Upon each completion of core loading and installation of the reactor vessel head, specific mechanical and electrical tests will be performed prior to initial criticality.

The fuel handling accident has been analyzed based on the activity that could
be released from fuel rod gaps of 204 rods of the highest power assembly* with 19
a 100 hour decay period following power operation at 2550 MWt for 23,000 hours. The requirements detailed in Specification 3.10 provide assurance that refueling unit conditions conform to the operating conditions assumed in the accident analysis.

Detailed procedures and checks insure that fuel assemblies are loaded in the proper locations in the core. As an additional check, the moveable incore detector system will be used to verify proper power distribution. This system is capable of revealing any assembly enrichment error or loading error which could cause power shapes to be peaked in excess of design value.

*Fuel rod gap activity from 204 rods of the highest power 15x15 assembly is 19
greater than fuel rod gap activity from 264 rods of the highest power 17x17 demonstration assembly.

References

FSAR Section 5.2 Containment Isolation

3.12 CONTROL ROD ASSEMBLIES AND POWER DISTRIBUTION LIMITS

Applicability

Applies to the operation of the control rod assemblies and power distribution limits.

Objective

To ensure core subcriticality after a reactor trip, a limit on potential reactivity insertions from hypothetical control rod assembly ejection, and an acceptable core power distribution during power operation.

Specification

A. Control Bank Insertion Limits

1. Whenever the reactor is critical, except for physics tests and control rod assembly exercises, the shutdown control rods shall be fully withdrawn.

2. Whenever the reactor is critical, except for physics tests and control rod assembly exercises, the full length control rod banks shall be inserted no further than the appropriate limit determined by core burnup shown on TS Figures 3.12-1A, 3.12-1B,
3.12-2, or 3.12-3 for three-loop operation and TS Figures 3.12-4A, 19
3.12-4B, 3.12-5, or 3.12-6 for two-loop operation.

3. The limits shown on TS Figures 3.12-1A through 3.12-6 may be revised on the basis of physics calculations and physics data obtained during unit startup and subsequent operation, in accordance with the following: 19

a. The sequence of withdrawal of the controlling banks, when going from zero to 100% power, is A, B, C, D.

b. An overlap of control banks, consistent with physics calculations 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 exceed the applicable value shown on TS Figure 3.12-7 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.

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 per cent for every per cent 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

$$T = 2.0 + 50\ (1.435/F_{xy} - 1) \leq 10\%$$ 19

where F_{xy} is 1.435 or the value of the unrodded horizontal 19
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 1.435 or a value 19
up to 10% 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 per cent to -14 per cent.

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 21.1 kw/ft for Unit 1 and 20.4 kw/ft for Unit 2. Second, 19
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 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 Figures 3.12-1A, 3.12-1B, and 3.12-2. 19

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 factor limits are met.

For normal operation and anticipated transients the core is protected from exceeding 21.1 kw/ft for Unit 1 and 20.4 kw/ft for Unit 2 locally, and 19
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 test 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 the 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

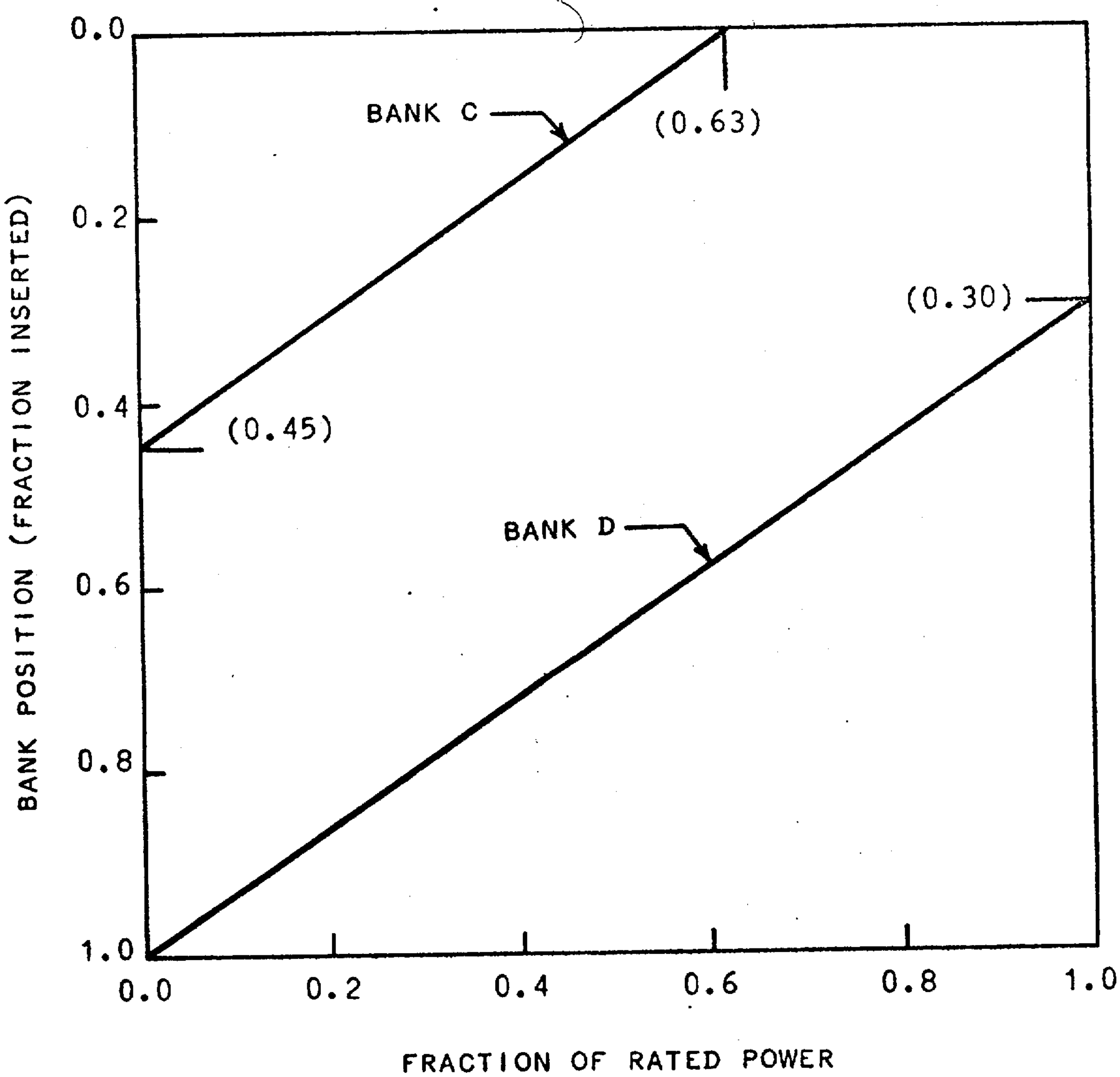


FIGURE 3.12-1A CONTROL BANK INSERTION LIMITS FOR 3-LOOP NORMAL OPERATION-UNIT 1

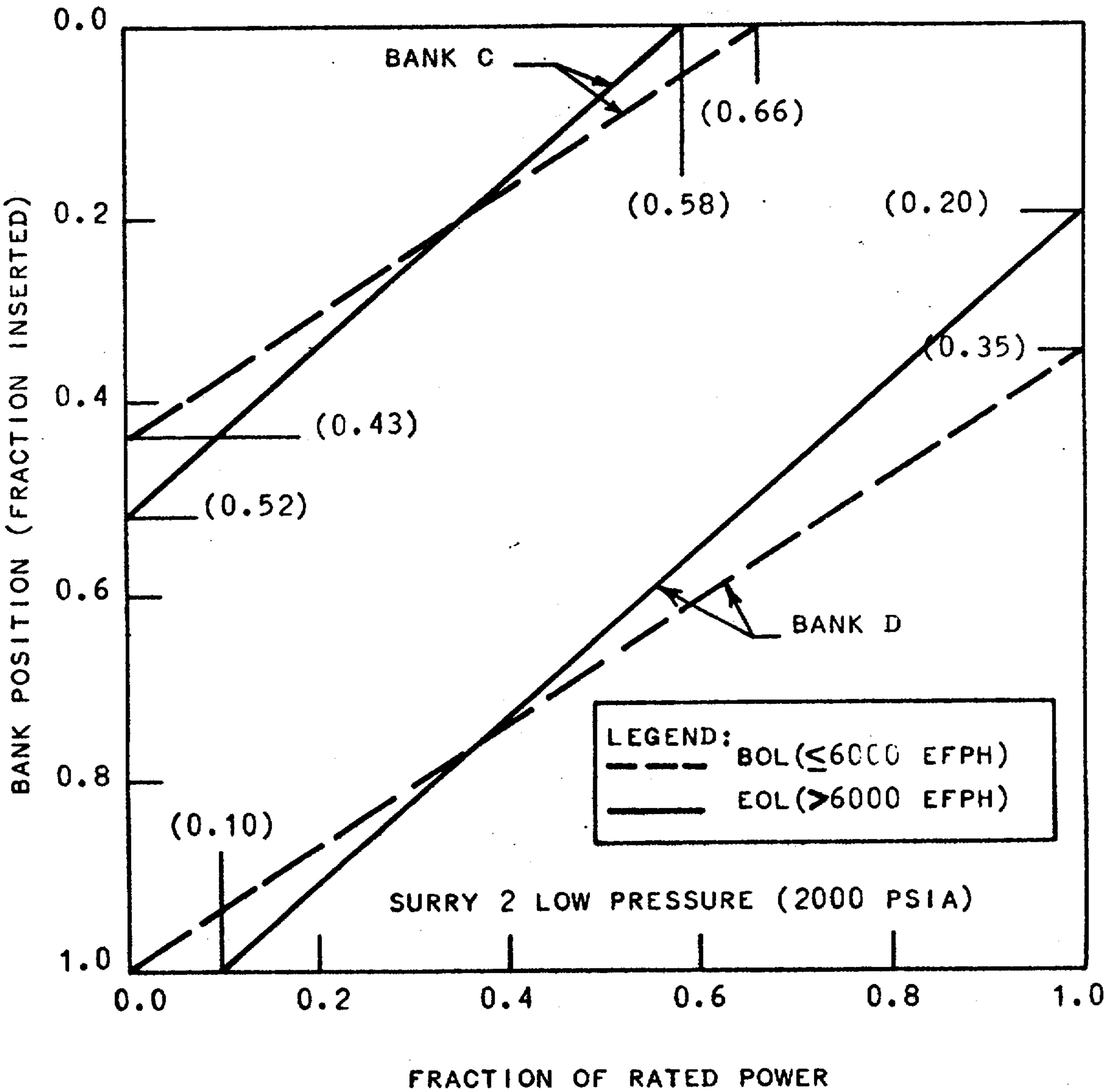


FIGURE 3.12-1B CONTROL BANK INSERTION LIMITS FOR NORMAL 3 LOOP OPERATION-UNIT NO. 2

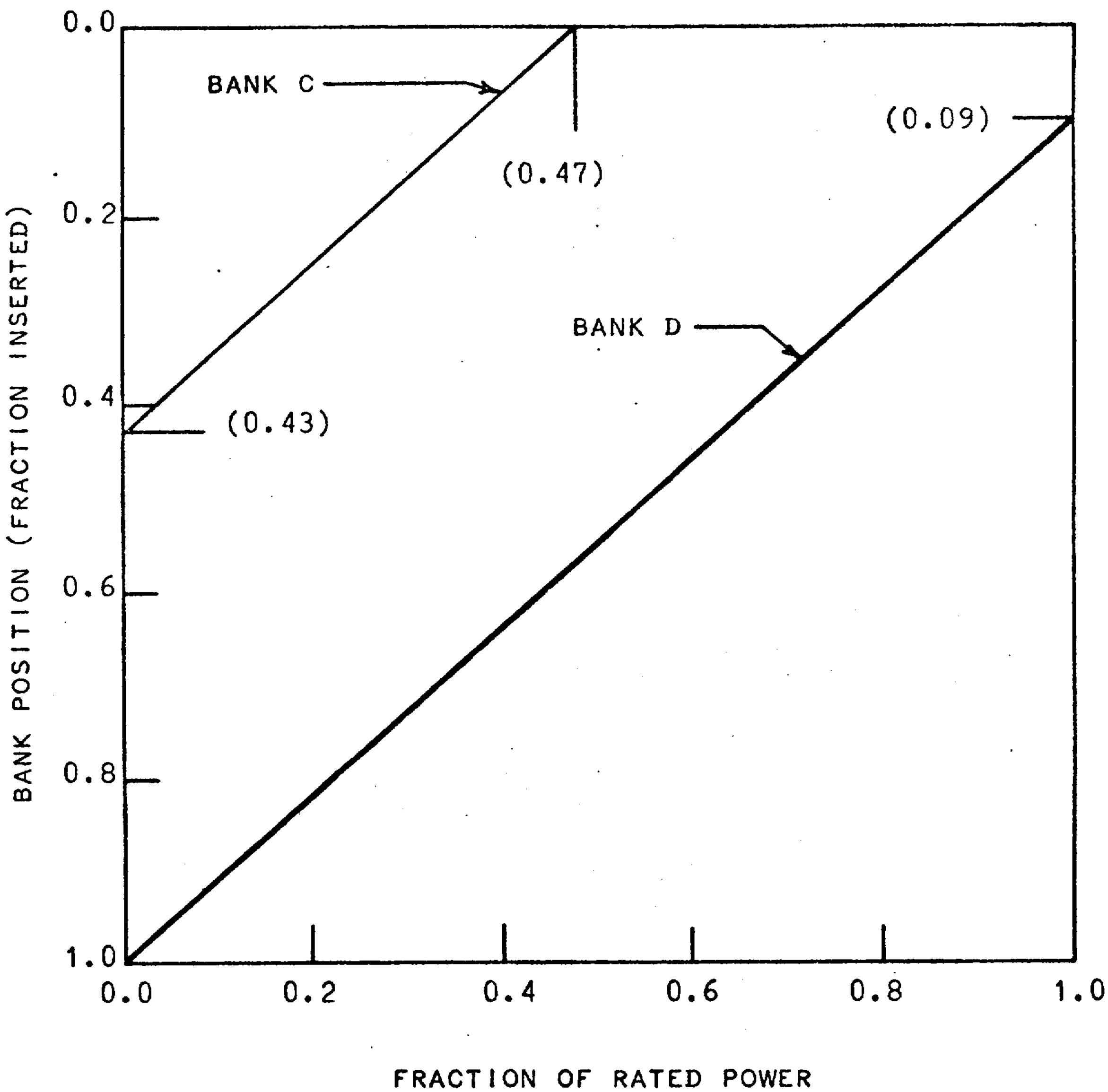


FIGURE 3.12-2 CONTROL BANK INSERTION LIMITS FOR 3 LOOP OPERATION WITH ONE BOTTOMED ROD

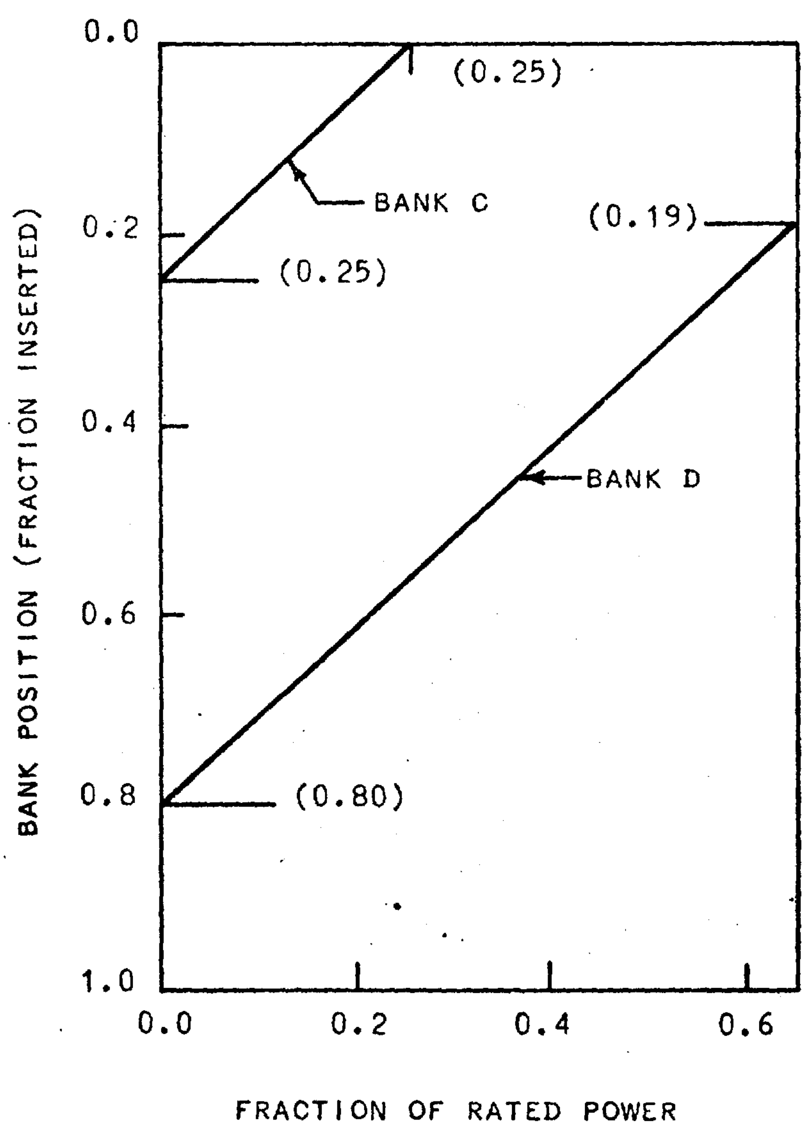
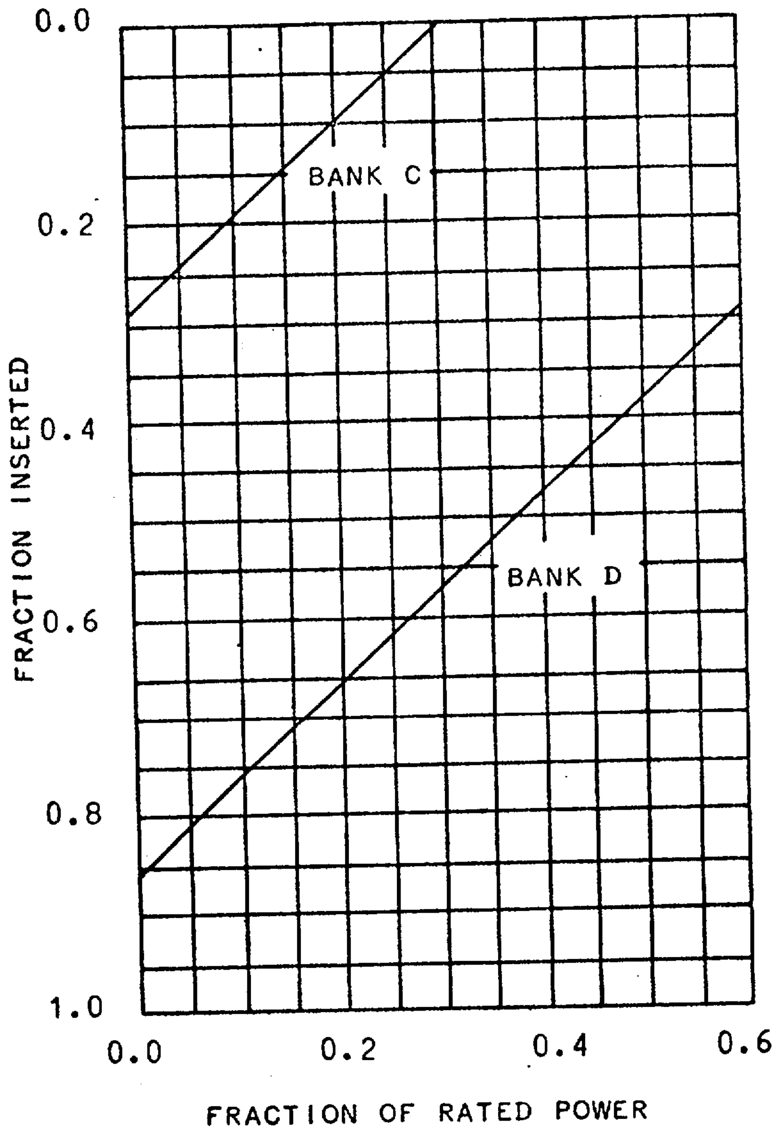


FIGURE 3.12-4A CONTROL BANK INSERTION LIMITS FOR 2-LOOP NORMAL OPERATION UNIT NO. 1

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



4.3 REACTOR COOLANT SYSTEM INTEGRITY TESTING FOLLOWING OPENING

Applicability

Applies to test requirements for Reactor Coolant System integrity. In this context, closed is defined as that state of system integrity which permits pressurization and subsequent normal operation after the system has been opened.

Objective

To specify tests for Reactor Coolant System integrity after the system is closed following normal opening, modification or repair.

Specification

A. Each time the Reactor Coolant System is closed, the system will be leak tested at not less than the nominal operating pressure +100 psi in conformance with NDT requirements. 19

B. When Reactor Coolant System modifications or repairs have been made which involved new strength welds on piping and components greater than 2 in. diameter, the new welds will receive both a surface and 100% volumetric non-destructive examination and meet applicable code requirements.

structural integrity of the component shall be performed in accordance with the provisions of Article IS-311 of Section XI Code.*

(2) Repairs of corroded areas, if necessary, shall be performed in accordance with the procedures of Article IS-400 of Section XI Code.*

Basis

For normal opening the integrity of the system, in terms of strength, is unchanged. If the system does not leak at the nominal operating pressure plus 100 psi, it will be leaktight during normal operation. 19

For repairs on piping and components greater than 2 in. diameter, the thorough nondestructive testing gives a very high degree of confidence in the integrity of the system, and will detect any significant defects in and near the new welds.

Significance of repairs on piping and components 2 in. diameter or smaller are relatively minor in comparison. The surface examination assures an adequate standard of integrity. In all cases, the leak test will ensure leaktightness during normal operation.

Experience has shown that corrosion potential might exist under certain conditions when borated fluid has prolonged contact with carbon steel. Undetected leakage of broated reactor coolant may cause such conditions to exist. To detect such leakage or its effects at an early stage the inspection program described in Specification D provides a means of detecting reactor coolant leakage and/or its effects at an early stage.

*1970 Edition with Winter 1970 Addenda

5.3 REACTOR

Applicability

Applies to the reactor core, Reactor Coolant System, and Safety Injection System.

Objective

To define those design features which are essential in providing for safe system operations.

Specifications

A. Reactor Core

1. The reactor core contains approximately 176,200 lbs of uranium dioxide in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 tubing to form fuel rods. All fuel rods are pressurized with helium during fabrication. The reactor core is made up of 157 fuel assemblies. Each fuel assembly contains 204 fuel rods except for two demonstration fuel assemblies which are part of Region 4 fuel. The demonstration assemblies each contain 264 fuel rods. 19

2. The average enrichment of the initial core is 2.51 weight per cent of U-235. Three fuel enrichments are used in the initial core. The highest enrichment is 3.12 weight per cent of U-235.

3. Reload fuel will be similar in design to the initial core. The enrichment of reload fuel will not exceed 3.60 weight per cent 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 glass 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. The initial core and subsequent cores will meet the following performance criteria at all times during the operating lifetime.

   a. Nuclear hot channel factors:

      $F^N_Q \leq 2.39\ (1 + 0.2\ (1-P))$ in the flux difference range -14 to + 9 percent 19

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

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

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

SUPPORTING AMENDMENTS NO. 4 TO LICENSES NO. DPR-32 AND DPR-37

CHANGE NO. 19 TO TECHNICAL SPECIFICATIONS

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-280 AND 50-281

Introduction

By two letters dated October 17, 1974 and supplemented by letters dated November 15, 1974 and November 20, 1974, Virginia Electric and Power Company (the licensee) requested changes to the Technical Specifications appended to Facility Operating Licenses DPR-32 and DPR-37 for the Surry Power Station Units 1 and 2. The purpose of the request is to revise the Surry 1 Technical Specifications as required to operate within the appropriate fuel and core design limits during the second fuel cycle.

Discussion

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

Evaluation

The submittal was reviewed with particular attention to the areas of revised safety analyses and safety margins, adherence to both the

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interim and final acceptance criteria, changes in the Technical Specifications, and generic considerations (e.g., fuel densification and cladding creep collapse).

All accidents analyzed in the Surry Unit 1 and Unit 2 Final Safety Analysis Report were reviewed and it was determined that these analyses remain valid except for the rod ejection accident and the probability or consequence of these accidents will not be increased. The rod ejection accident was reanalyzed for the cycle 2 core and it was determined that the consequences of this accident are less severe with the cycle 2.

We also determined that no safety margin or design limit will be exceeded as a result of this change and that the licensee's submittal appropriately accounts for the effect of fuel densification and fuel cladding creep collapse.

We have reviewed the licensee's request to remove the current operational restriction on Region 2 fuel. Our approval of operation to 15,500 EFPH is based on (1) specific confirmatory information, (2) BUCKLE collapse calculations and (3) vendor's revised model calculations. These items are discussed in more detail in the paragraphs that follow.

The fuel residence time in the Technical Specifications for the cycle 1 core with Region 2 most limiting, was conservatively restricted to 10,000 EFPH due to the lack of an adequate clad collapse model which in turn could not be generated without the observed performance of similar fuel in other operating reactors. The licensee requested in the October 17, 1974 letter a fuel residence time of 26,000 EFPH based on an as yet unapproved model (WCAP-8377) discussed below. The licensee subsequently amended his request, in the November 15, 1974 letter, from 26,000 EFPH to 14,200 EFPH based on a previously approved staff model. As discussed, the staff's evaluation indicates that the 14,200 EFPH request is overly conservative. We prefer to approve in the Cycle 2 Technical Specifications a longer, more realistic fuel residence time, so that Surry Unit 1 operations will not be unnecessarily restricted before the review of the revised model (WCAP-8377) is completed.

Specific confirmatory information is available from surveillance of other reactors, e.g., R. E. Ginna, H. B. Robinson and Point Beach Unit 1. Flattened rods have not been observed in H. B. Robinson (Regions 2 and 3) for exposures of 19,000 EFPH. Region 4A of Ginna has also been exposed for 19,400 EFPH with no observed flattened rods. A few (0.7% in Region 2 and 0.05% in Region 3) flattened rods have been observed in Point Beach for exposures of 21,000 EFPH. These reactors all have similar fuel. The initial rod internal pressures are higher in Surry than in the above mentioned reactors.

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Independent calculations to predict time for cladding creep collapse have been performed using the computer code BUCKLE which is conservatively biased and uses a general descriptive creep rate equation. BUCKLE predicts cladding collapse at 16,200 EFPH for Region 2 in Surry. These calculations take into account the planned increase in primary system pressure to 2250 psia for the second cycle.

The fuel vendor's revised clad flattening model predicts the initial flattening time and the flattened rod frequency as a function of time for a given fuel region. The analysis includes the effect of gap length in contrast to BUCKLE which conservatively assumes an "infinite" gap length. The new model predicts a cladding collapse time of 26,000 EFPH for Region 2 in Surry. It should be noted that the revised clad flattening model is still under our review.

We conclude that an interim fuel residence time should be specified until the revised clad flattening model (WCAP-8377) obtains our approval. Until WCAP-8377 is approved, we do not have sufficient basis to approve the initially proposed fuel residence time of 26,000 EFPH. In view of the calculations and available confirmatory information from in-reactor tests, we approve operation to 15,500 EFPH for Region 2 of Surry Power Station, Unit No. 1.

10 CFR 50.46 requires that the operation of the facility be within the limits of both the proposed Appendix K Technical Specifications and the existing Technical Specifications based on the Interim Policy Statement until the proposed Appendix K Technical Specifications have been approved. The licensee has stated that the proposed Technical Specifications are in conformance with both the interim acceptance criteria and Appendix K to 10 CFR Part 50. (TAKEN FROM OCONEE RECENTLY ISSUED)

The nuclear, mechanical, and thermal-hydraulic analyses that were performed by the licensee to establish the appropriate operating limits and set-points for cycle 2 operation were reviewed and found to be methods previously used and found acceptable by the AEC for Surry Unit 1. The proposed Technical Specification changes which incorporates these limits and set-points were reviewed and found to be consistent with the reanalyses, and therefore acceptable. None of the proposed Technical Specification changes would increase the probability or consequence of postulated accidents previously analyzed. The bases of the Technical Specifications have been revised to show the result of this reanalysis. However, the method and procedures described in these bases remain unchanged.

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Conclusion

We conclude that there is reasonable assurance (i) that the activities authorized by these amendments can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

/S/

Morton B. Fairtile
Operating Reactors Branch #1
Directorate of Licensing

/S/ O. L. Rooney

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

Date: DEC 27 1974

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UNITED STATES ATOMIC ENERGY COMMISSION

DOCKET NOS. 50-280 AND 50-281

VIRGINIA ELECTRIC & POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

No request for a hearing or petition for leave to intervene having been filed following publication of the notice of proposed action in the FEDERAL REGISTER on November 20, 1974 (39 F.R. 40810), the Atomic Energy Commission (the Commission) has issued Amendments No. 4 to Facility Operating Licenses No. DPR-32 and DPR-37 issued to Virginia Electric & Power Company (VEPCO) which revised Technical Specifications for operation of the Surry Power Station, Units 1 and 2, located in Surry County, Virginia. The amendments are effective as of the date of issuance.

The amendments revise the provisions in the Technical Specifications relating to the replacement of 84 of 157 fuel assemblies in the reactor core, constituting refueling of the core for second cycle operation of Unit 1.

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.

For further details with respect to this action, see (1) the application for amendments dated October 17, 1974, as supplemented November 20, 1974, (2) Amendments No. 4 to Licenses No. DPR-32 and DPR-37, with any attachments, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, D.C., and at the Swem Library, College of William & Mary, Williamsburg, Virginia 23185.

A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Atomic Energy Commission, Washington, D.C. 20545, Attention: Deputy Director for Reactor Projects, Directorate of Licensing - Regulation.

Dated at Bethesda, Maryland, this 27th day of December 1974.

FOR THE ATOMIC ENERGY COMMISSION

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