Distribution:
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Docket (2)
PWR-1 Rdg
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RCDeYoung
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DSkovholt
RKlecker
VSTello
MJinks (4)
DRoss
ADromerick
DVassallo
JLee
ACRS (16)
WMiller

AUG 0 9 1973

Docket Nos. 50-280
and 50-281

Mr. Stanley Ragone
Vice President
Virginia Electric and Power Company
P. O. Box 26666
Richmond, Virginia 23261

Licenses Nos. DPR-32
and DPR-37

Change No. 9

Dear Mr. Ragone:

Your letters dated June 4, 1973 and July 5, 1973 enclosed proposed changes in Section 2.1, "Safety Limit, Reactor Core"; Section 2.3, "Limiting Safety System Settings, Protection Instrumentation"; and Section 3.12, "Control Rod Assemblies and Power Distribution Limits" of the Technical Specifications for Facility Operating Licenses Nos. DPR-32 and DPR-37 for Surry Power Station Units 1 and 2. You also proposed to add Section 3.20, "100-Hour Full Power Test," to the Technical Specifications. The proposed changes include power escalation above 92% of rated power, a revised incore surveillance program, and a 100-Hour Full Power Performance Test at 2250 psia.

We have reviewed the reports "Fuel Densification - Surry Power Station Unit 1,""WCAP-8012, Addendum 1, dated April 1973, and "Fuel Densification - Surry Units 1 and 2 - Low Pressure Analysis," WCAP-8116, dated April 1973, which you submitted in support of your proposed changes for power escalation above 92% of rated power to 100% of rated power at a reduced primary coolant system pressure of 2000 psia.

On the basis of our review, we have determined that, for this power escalation, the three areas requiring assessment were minimum DNB ratio (DNBR), stored energy, and creep collapse. The DNBR analysis was performed using the methods as described in the FSAR for 100% of rated power. The minimum value of the DNB ratio during normal operation and anticipated transients is limited to 1.30. Since the calculated DNBR is above this limit for power operation up to 100% of rated power, including anticipated transients, we find this acceptable. We have reviewed the Westinghouse time dependent creep collapse and stored energy models and find them acceptable for a fuel residence time of 10,000 EFPH. We have determined that the effects of fuel densification and reduced primary coolant system pressure above 92% of rated power have been adequately analyzed for a fuel residence time of 10,000 EFPH and that the plant power capability, with respect to such effects, is acceptably defined in Figure 3.12-8 of

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DATE ▶						

the Technical Specifications and therefore, operation at 100% of rated power at a reduced primary system pressure of 2000 psia is permitted. We have also determined that the proposed incore surveillance program provides the necessary assurance that the reactor will be operated within the prescribed limits.

With respect to your proposed changes involving escalation above 92% of rated power and the revised incore surveillance program, we have concluded that the proposed changes do not involve significant hazards consideration and there is reasonable assurance that the health and safety of the public will not be endangered by operation of the reactor in the manner proposed. Accordingly, pursuant to Section 50.59 of 10 CFR Part 50, the Technical Specifications of Facility Operating Licenses Nos. DPR-32 and DPR-37 are hereby changed as set forth in Sections 2.1, 2.3, and 3.12, Change No. 9, copies of which are enclosed. Although every page of every section of Change No. 9 has not been changed, the entire sections are being replaced as separate entities.

In your letter of July 5, 1973, you also requested authorization to proceed with the 100-Hour Full Power Test at a primary coolant system pressure of 2250 psia. If you require that the 100-Hour Full Power Test be performed, it is our judgement that the test should be conducted at the primary system pressure of 2000 psia, which is the pressure the reactor will be operating in accordance with Technical Specification Change No. 8. Therefore, your request to perform the 100-Hour Full Power Test at 2250 psia cannot be granted.

Please contact us if you desire any discussion or clarification of these matters.

Sincerely,

Original signed by R. C. DeYoung

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

Enclosure:
As stated

cc: George D. Gibson, Esq.
Hunton, Williams, Gay, and Gibson
P. O. Box 1535
Richmond, Virginia 23213

bcc: J. R. Buchanan, ORNL
Thomas B. Abernathy, DTIE

OFFICE ▶	PWR-1	PWR-1	TR	OGC	RO	AD:PWRs
SURNAME ▶	ADromerick:ms	DBVassallo	DRoss		JHS for GK	RCDeYoung
DATE ▶	8/8/73	8/8/73	8/8/73	8/ /73	8/9/73	8/9/73

Docket Nos. 50-280
and 50-281

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Vice President
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Licenses Nos. DPR-32
and DPR-37

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We have reviewed the reports "Fuel Densification - Surry Power Station Unit 1," WCAP-8012, Addendum 1, dated April 1973, and "Fuel Densification - Surry Units 1 and 2 - Low Pressure Analysis," WCAP-8116, dated April 1973, which you submitted in support of your proposed changes for power escalation above 92% of rated power at a reduced primary coolant system pressure of 2000 psia. On the basis of our review, we have determined that, for this power escalation, the three areas requiring assessment were minimum DNB ratio (DNBR), stored energy, and creep collapse. The DNBR analysis was performed using the methods as described in the FSAR for 100% of rated power. The minimum value of the DNB ratio during normal operation and anticipated transients is limited to 1.30. Since the calculated DNBR is above this limit for power operation up to 100% of rated power, including anticipated transients, we find this acceptable. We have reviewed the Westinghouse time dependent creep collapse and stored energy models and find them acceptable for a fuel residence time of 9000 EFPH. We are presently reviewing the analysis which will permit the fuel residence time to be extended beyond the 9000 EFPH, as you requested. Upon completion of our review we will determine the acceptable fuel residence time. We have determined that the effects of fuel densification and reduced primary coolant system pressure above 92% of rated power have been adequately analyzed for a fuel residence time of 9000 EFPH and that the plant power capability, with respect to such effects, is acceptably defined in Figure 3.12-8 of the Technical

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DATE ▶						

Specifications. We have also determined that the proposed incore surveillance program provides the necessary assurance that the reactor will be operated within the prescribed limits.

With respect to your proposed changes involving escalation above 92% of rated power and the revised incore surveillance program, we have concluded that the proposed changes do not involve significant hazards consideration and there is reasonable assurance that the health and safety of the public will not be endangered by operation of the reactor in the manner proposed. Accordingly, pursuant to Section 50.59 of 10 CFR Part 50, the Technical Specifications of Facility Licenses Nos. DPR-32 and DPR-37 are changed as set forth in Sections 2.1, 2.3, and 3.12, Change No. 9, copies of which are enclosed, contingent upon the satisfactory resolution of the following matters. (Although every page of every section of Change No. 9 has not been changed, the entire sections are being replaced as separate entities.)

As discussed in your letters of June 22, 1973 and July 5, 1973 and in telephone conversations between the Regulatory staff and your representatives, approval to escalate above 92% of rated power is dependent upon the satisfactory resolution of our concerns with respect to a postulated high energy line rupture outside of containment. These concerns were discussed with your representatives at a meeting held on July 6, 1973. Therefore, Technical Specification Change No. 9 shall not be implemented until you receive written Commission approval of the plans and schedule for modification of the facilities to mitigate the consequences of a postulated high energy line rupture outside of containment.

In your letter of July 5, 1973, you also requested authorization to proceed with the 100-Hour Full Power Test at 2250 psia. If you require that the 100-Hour Full Power Test be performed, it is our judgement that the test should be conducted at 2000 psia conditions, which are the conditions the reactor will be operating at 100% of rated power. Therefore, your request to perform the 100-Hour Full Power Test at 2250 psia cannot be granted.

Please contact us if you desire any discussion or clarification of these matters.

Sincerely,

[redacted]

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

Enclosure:
As stated

cc: See page 3

OFFICE ▶	PWR-1	PWR-1	TR	OGC	RO	AD:PWRs
SURNAME ▶	[illegible]	DBVassallo	Per for DRoss			RCDeYoung
DATE ▶	7/19/73	7/20/73	7/19/73	7/ /73	7/ /73	7/ /73

Distribution:
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AUG 0 9 1973

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and 50-281

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Vice President
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Licenses Nos. DPR-32
and DPR-37

Change No. 9

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We have reviewed the reports "Fuel Densification - Surry Power Station Unit 1,""WCAP-8012, Addendum 1, dated April 1973, and "Fuel Densification - Surry Units 1 and 2 - Low Pressure Analysis," WCAP-8116, dated April 1973, which you submitted in support of your proposed changes for power escalation above 92% of rated power to 100% of rated power at a reduced primary coolant system pressure of 2000 psia.

On the basis of our review, we have determined that, for this power escalation, the three areas requiring assessment were minimum DNB ratio (DNBR), stored energy, and creep collapse. The DNBR analysis was performed using the methods as described in the FSAR for 100% of rated power. The minimum value of the DNB ratio during normal operation and anticipated transients is limited to 1.30. Since the calculated DNBR is above this limit for power operation up to 100% of rated power, including anticipated transients, we find this acceptable. We have reviewed the Westinghouse time dependent creep collapse and stored energy models and find them acceptable for a fuel residence time of 10,000 EFPH. We have determined that the effects of fuel densification and reduced primary coolant system pressure above 92% of rated power have been adequately analyzed for a fuel residence time of 10,000 EFPH and that the plant power capability, with respect to such effects, is acceptably defined in Figure 3.12-3 of

AUG 0 9 1973

the Technical Specifications and therefore, operation at 100% of rated power at a reduced primary system pressure of 2000 psia is permitted. We have also determined that the proposed incore surveillance program provides the necessary assurance that the reactor will be operated within the prescribed limits.

With respect to your proposed changes involving escalation above 92% of rated power and the revised incore surveillance program, we have concluded that the proposed changes do not involve significant hazards consideration and there is reasonable assurance that the health and safety of the public will not be endangered by operation of the reactor in the manner proposed. Accordingly, pursuant to Section 50.59 of 10 CFR Part 50, the Technical Specifications of Facility Operating Licenses Nos. DPR-32 and DPR-37 are hereby changed as set forth in Sections 2.1, 2.3, and 3.12, Change No. 9, copies of which are enclosed. Although every page of every section of Change No. 9 has not been changed, the entire sections are being replaced as separate entities.

In your letter of July 5, 1973, you also requested authorization to proceed with the 100-Hour Full Power Test at a primary coolant system pressure of 2250 psia. If you require that the 100-Hour Full Power Test be performed, it is our judgement that the test should be conducted at the primary system pressure of 2000 psia, which is the pressure the reactor will be operating in accordance with Technical Specification Change No. 8. Therefore, your request to perform the 100-Hour Full Power Test at 2250 psia cannot be granted.

Please contact us if you desire any discussion or clarification of these matters.

Sincerely,

Original signed by R. C. DeYoung

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

Enclosure:
As stated

cc: George D. Gibson, Esq.
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Richmond, Virginia 23213

bcc: J. R. Buchanan, ORNL
Thomas B. Abernathy, DTIE

OFFICE	PWR-1	PWR-1	TR	OGC	RO	AD:PWRs
SURNAME	ADromerick:ms	DBVassallo	DRoss			RCDeYoung
DATE	8/8/73	8/8/73	8/8/73	8/ /73	8/9/73	8/9/73

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 the limit shown in TS Figure 2.1-4. 9

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 the limit on TS Figure 2.1-4 9

C. The reactor thermal power shall not exceed 1220 megawatts thermal until the results of the environmental qualification tests performed on the recirculation spray pump motors have been evaluated and approved in writing by the Atomic Energy Commission.

D. 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. 6 9

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. 6

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 a 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. 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 the curves shown in Figures 2.1-1, 2 and 3. 6

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

The curves 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). 6

These hot channel factors are higher than those calculated at full power over the range between that of all control rod assemblies full 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.

TS Figure 2.1-2 and 2.1-3 have not been revised as these have been found to be adequate and conservative even including the effects of densification. Figure 2.1-1, 2 and 3 also include an allowance for an increase in the enthalpy rise hot channel factor at reduced power based on the expression: 6

$F^N_{\Delta H} = 1.55\ [1 + 0.2\ (1-P)]$ where P is the fraction of rated power.

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions for Reactor Coolant System temperature, pressure and thermal power level that would result in a DNB ratio of less than 1.30(3) based on steady state nominal operating power levels less than or equal to 100%, steady state nominal operating Reactor Coolant System average temperatures less than or equal to 574.4°F and a steady state nominal operating pressure of 2235 psig. 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 curve of TS Figure 2.1-4 represents the fuel overpower design limit as a function of burnup. This limit is the fuel melting temperature or a linear heat rate of 21.1 kw/ft, whichever is more restrictive. 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 curve. 9

The Commission is presently evaluating the results of the post-loss-of-coolant accident environmental qualification tests performed to determine the acceptability of the inside containment recirculation spray pump motors. Two of the motors are located outside the containment and would not be subjected to the post-loss-of-coolant accident environment. These two motors and their associated pumps provide adequate redundancy up to 50 percent of rated power. Accordingly, operation up to 50 percent of rated power (1220 megawatts thermal) is permitted. However, until the Commission has determined that the recirculation spray pump motors located in the containment are adequate for their intended service, operation above 50 percent of rated power is not permitted.

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. 6 9

References

(1) FSAR Section 3.4

(2) FSAR Section 3.3

(3) FSAR Section 14.2

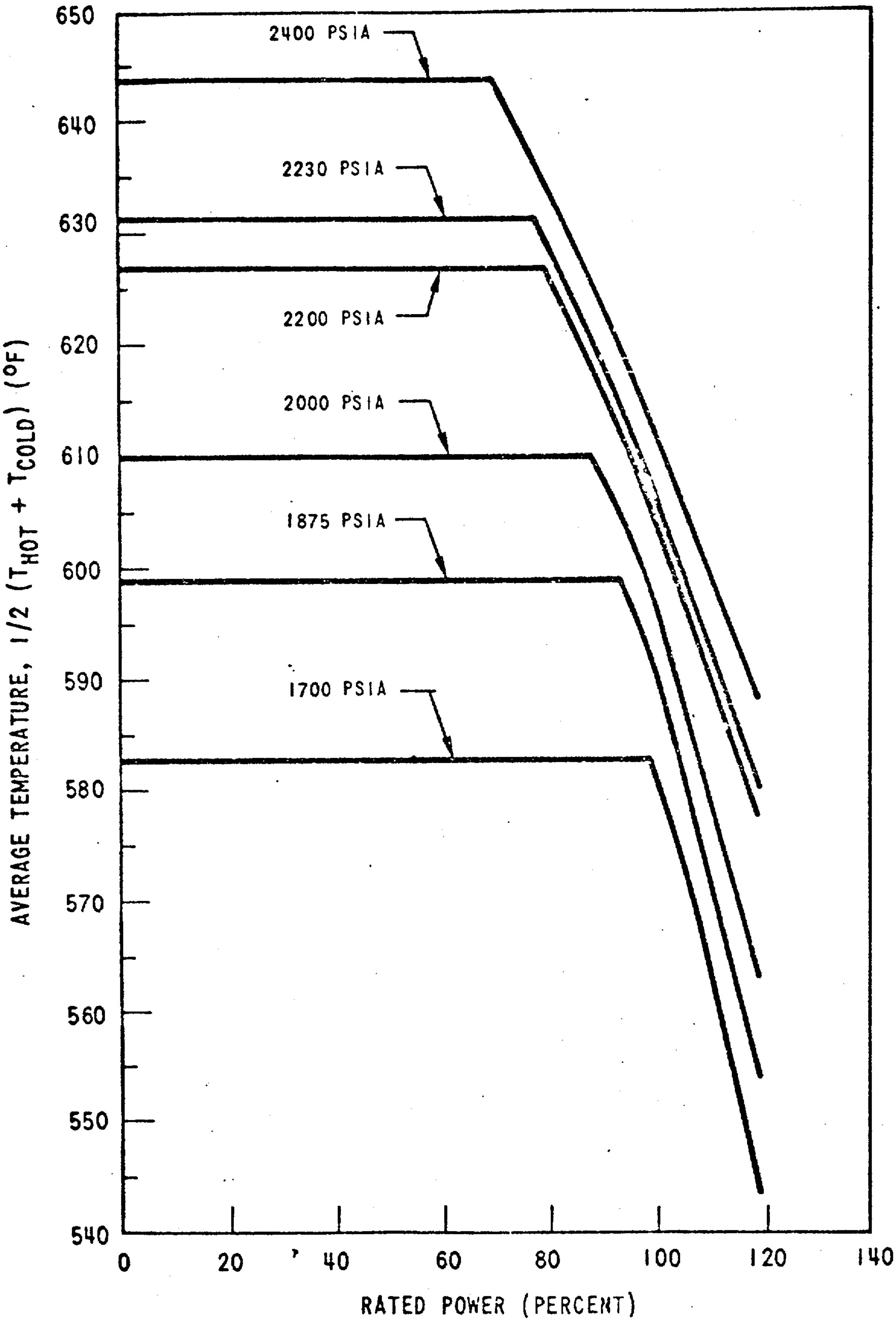

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Figure 2.1-1 Reactor Core Thermal & Hydraulic Safety Limits - Three Loop Operation, 100% Flow

CHANGE NO.9

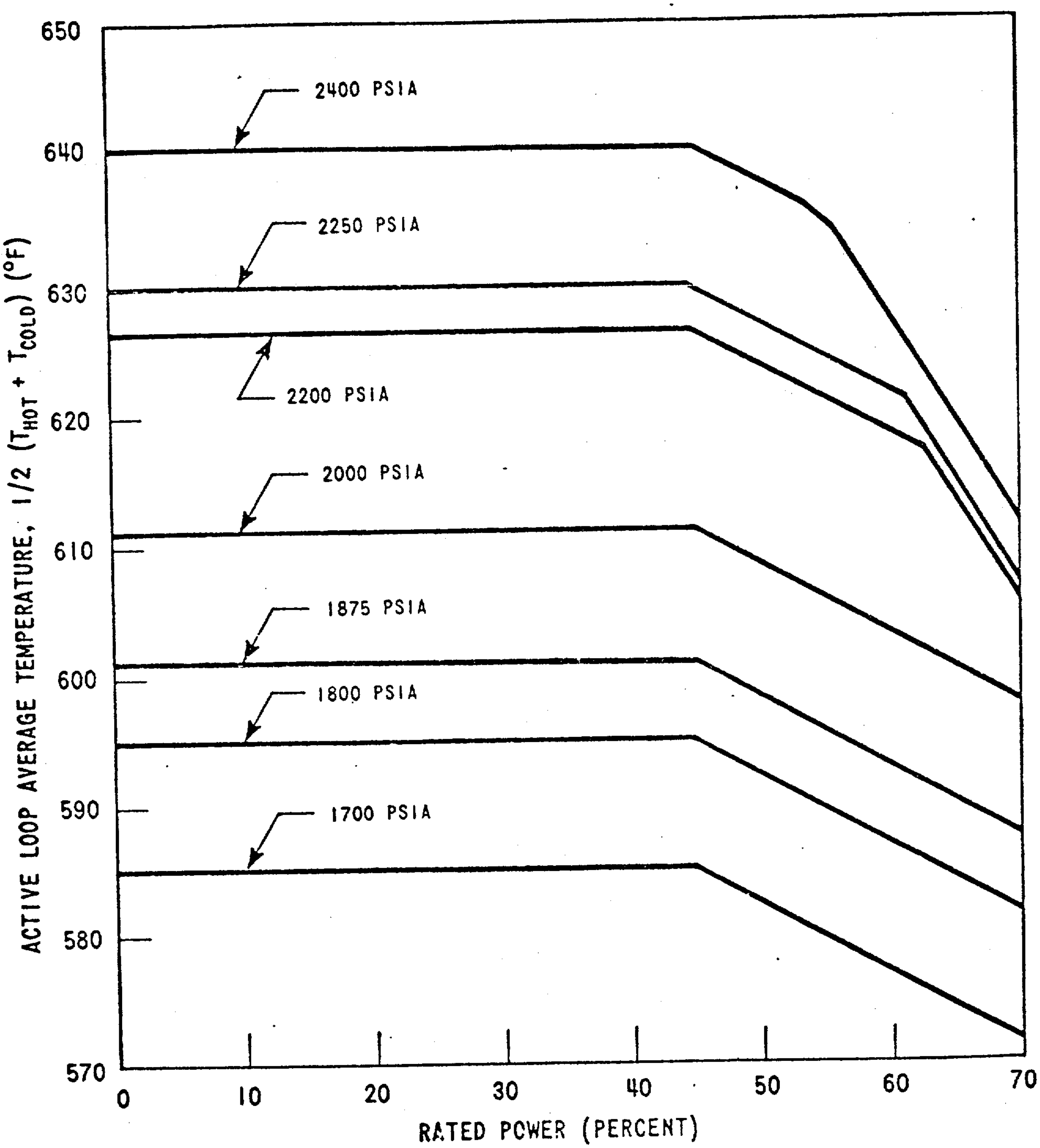

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Figure 2.1-2. Reactor Core Thermal and Hydraulic Safety Limits, Two Loop Operation, Loop Stop Valves Open

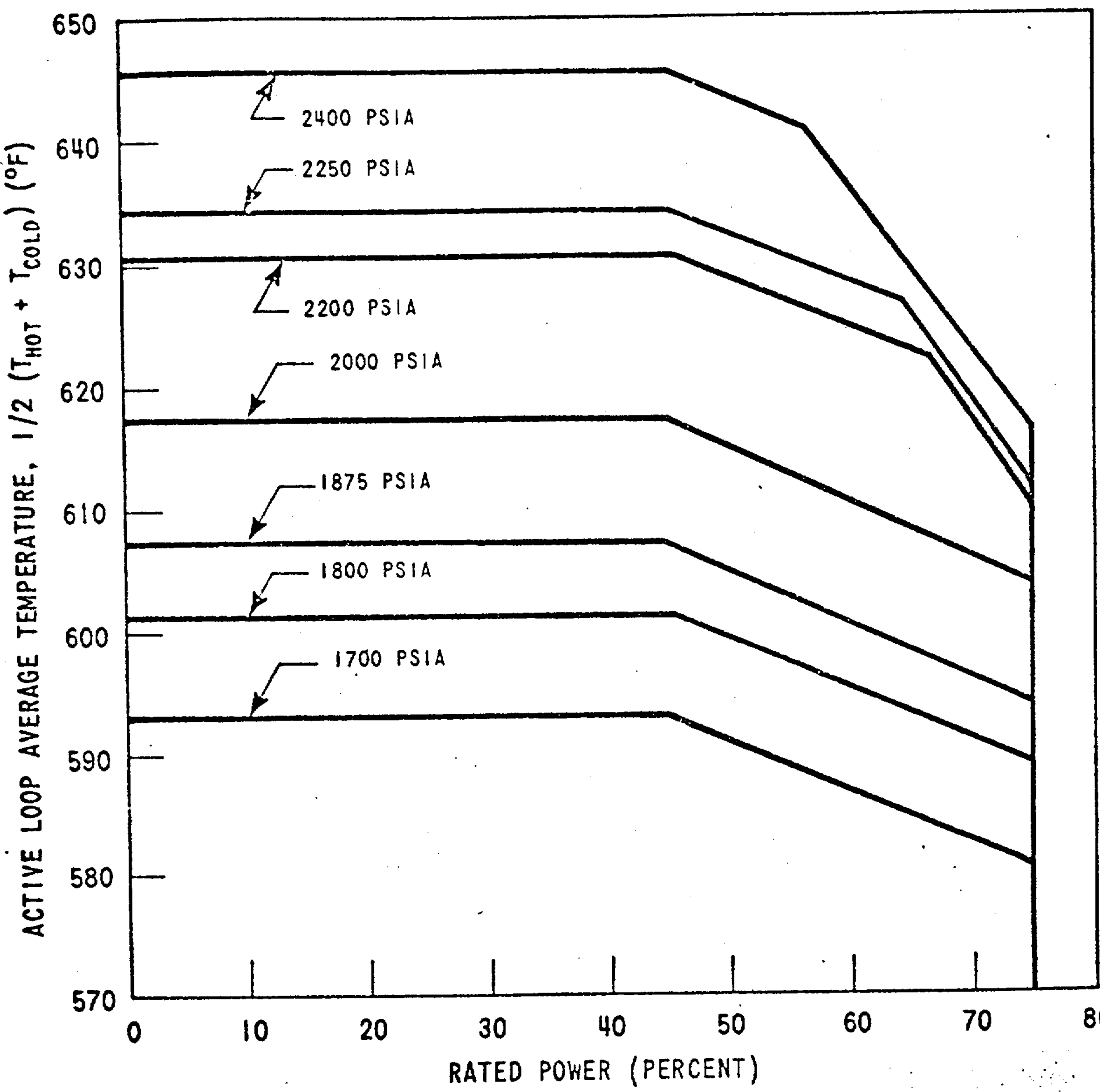

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Figure 2.1-3. Reactor Core Thermal and Hydraulic Safety Limits, Two Loop Operation, Loop Stop Valves Closed

CHANGE NO. 6

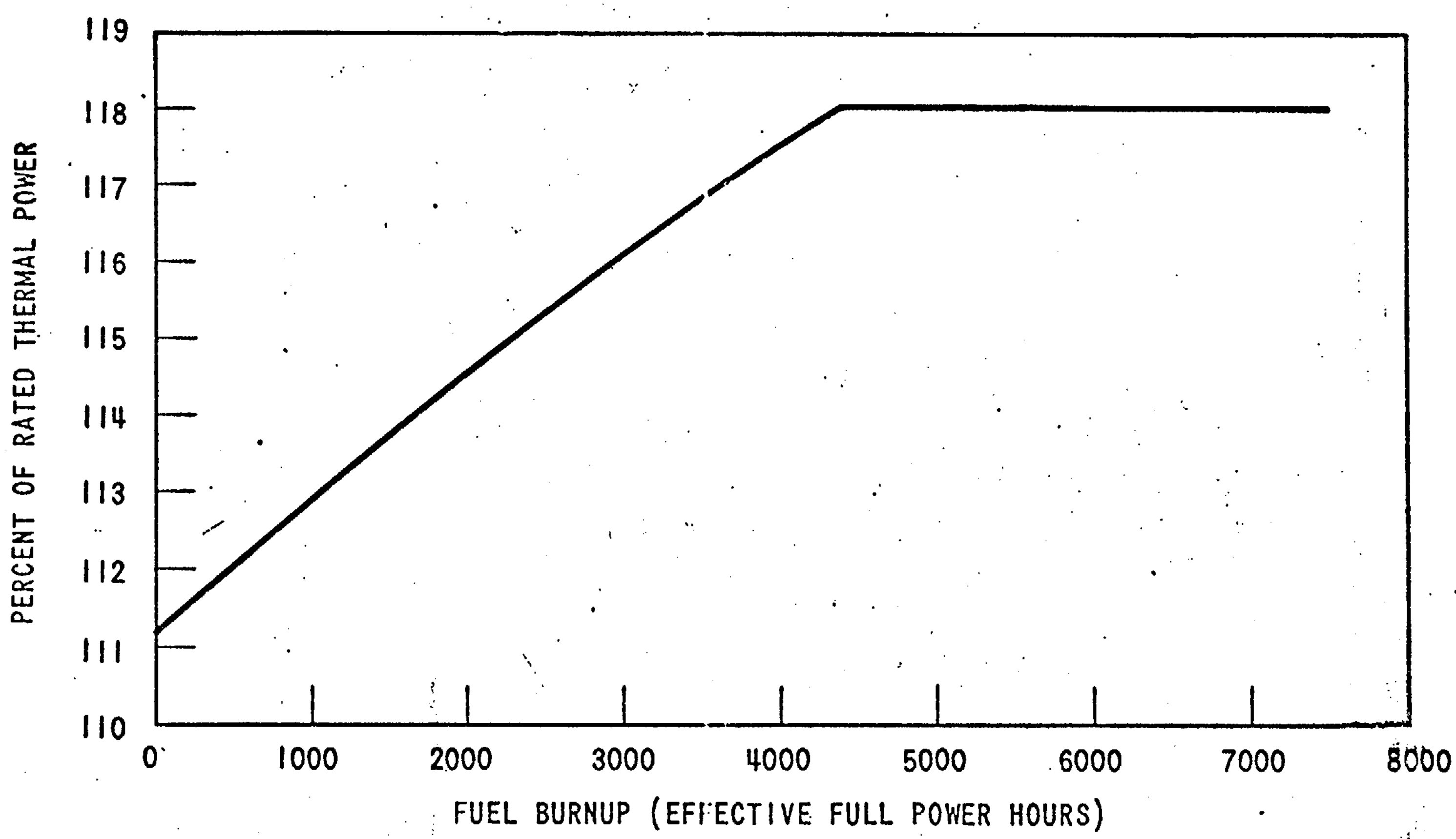

Figure 2.1-4. Thermal Overpower Limit

CHANGE NO. 9

TS Figure 2.1-4
AUG 0 9 1973

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) - $\leq$ value shown in TS Figure 2.3-1 (as a fraction of rated thermal power) at intervals no more frequent than 750 EFPH. 9
   - (b) High pressurizer pressure - $\leq$ 2385 psig.
   - (c) Low pressurizer pressure - $\geq$ 1715 psig. 8

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(d) Overtemperature T

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

where

ΔT_o = Indicated ΔT at rated thermal power, ^oF

T = Average coolant temperature, ^oF 8-1

T' = 563.5 ^oF

P = Pressurizer pressure, psig

P' = 1985 psig 8-1

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) 8-1

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 6

$f(\Delta I)$ = function of ΔI, percent of rated core power as shown in Figure 2.3-2 9

(e) Overpower T

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

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 = 1.02 at beginning of core life 6

= value shown in TS Figure 2.3-1 at intervals subsequent to beginning of core life determined at intervals no more frequent than 750 EFPH. 9

K_5 = 0 for decreasing average temperature; A constant, for increasing average temperature, 0.2 sec/ ^oF

K_6 = 0 for $T \leq T'$
0.00108 for $T > T'$ 6

$f(\Delta I)$ 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 trips 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 (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 6 core safety limits as shown in Figures 2.1-1, 2, 3, and 4.

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. Fuel temperature decreases due to cladding creepdown with burnup and consequential reduction of pellet-cladding gap. Thus overpower limits become less restrictive as fuel burnup proceeds and the safety system setpoints for these trips can 9
be increased accordingly. The overpower protection system set points include the effects of fuel densification on core safety limits.

Increase in the limiting safety system settings for the nuclear overpower and overpower ΔT trips shall be done in a series of discrete steps at intervals no more frequent than 750 EFPH which assures a limited number of trip resets.

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 8
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 112% 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 is 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 over-temperature ΔT trip setpoint is adjusted to the value specified for two-loop operation, 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, a reactor trip at 65% power will prevent the minimum DNBR from going below 1.30 during normal operational transients and anticipated transients. For this latter case the overtemperature ΔT trip setpoints may remain at the values used for three-loop operation.

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.

(1) FSAR Section 14.2.1

(2) FSAR Section 14.2

(3) FSAR Section 14.5

(4) FSAR Section 7.2

(5) FSAR Section 3.2.2

(6) FSAR Section 14.2.9

(7) FSAR Section 7.2

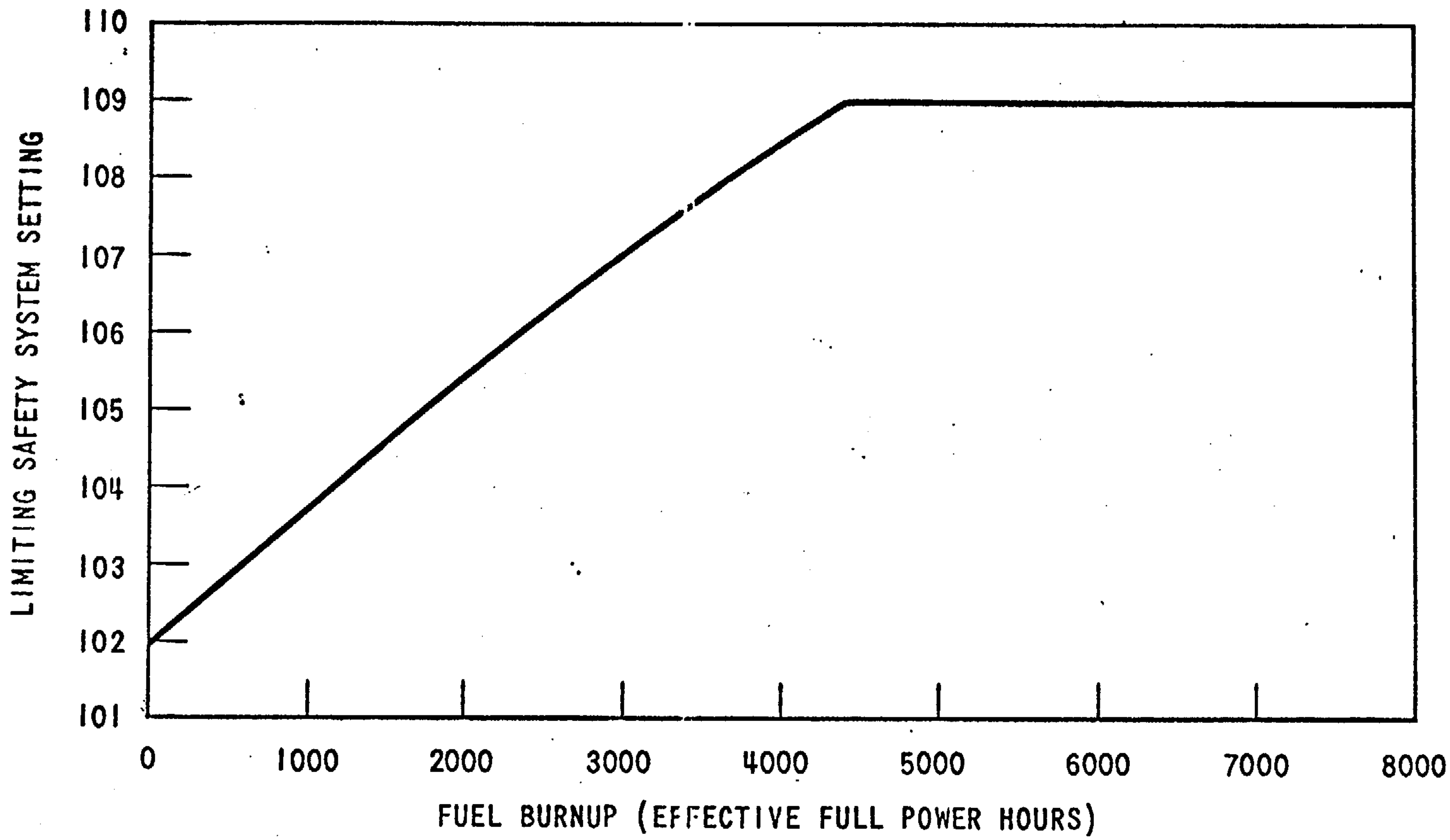

Figure 2.3-1. Limiting Safety System Setting for High Flux and Overpower ΔT Trips versus Fuel Burnup

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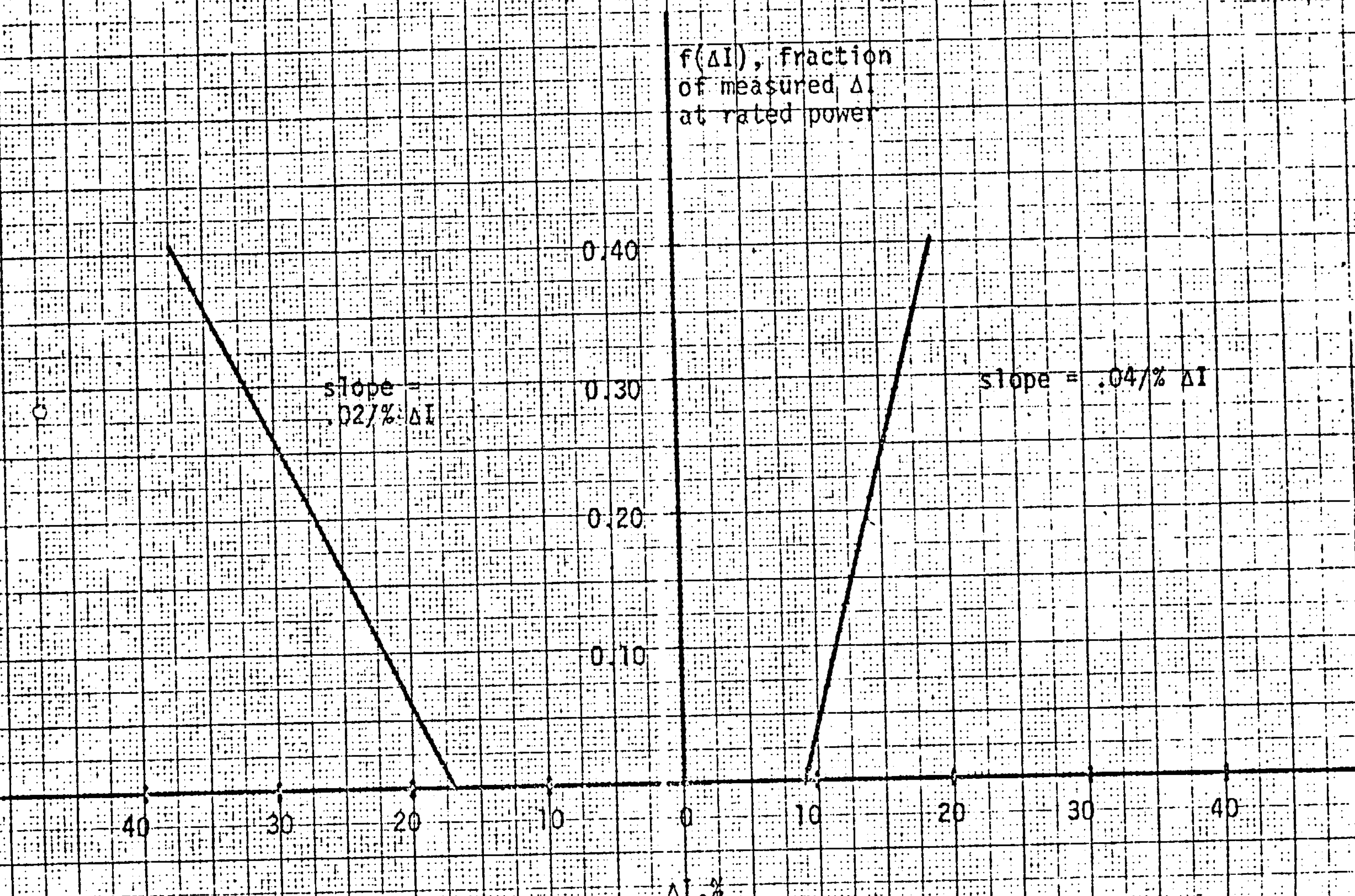
FIGURE 2.3-2



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TS Figure 2.3-2

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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 a 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 Fig. 3.12-1, 3.12-2, or 3.12-3 for three-loop operation and TS Fig. 3.12-4, 3.12-5 or 3.12-6 for two-loop operation. 6

3. The limits shown on TS Figures 3.12-1 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:

   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 operating 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 ($T_{avg} \geq 547°F$) if all control rod assemblies were tripped, assuming that the highest worth control rod assembly remained fully withdrawn, and assuming no changes in xenon, boron, or part-length rod position.

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

5. 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 permitted power or less. 6

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: 6

a. $F_Q^N \leq 2.48\ [1 + 0.2\ (1-P)]$ in the flux difference range − 17 to + 9 percent

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

where P is the quotient of the actual power (as fraction of 2441 Mwt) at which the core is operating to the permitted power. The permitted power is given in TS Figure 3.12-8. 9

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 fraction of 2441 Mwt at which the core can be operated, N, not to exceed the permitted power given in TS Figure 3.12-8, shall be determined by 9 6

$$N = \frac{Q}{6.2 \times 1.02 \times 1.017 \times 1.007 \times M}$$ 9

where $M = 2.55 \times \frac{F_{xy}}{1.42} [1+2(T/100 - 0.02)]$;

where Q is given in Figure 3.12-9; F_{xy} is 1.42, or the value of the unrodded horizontal plane peaking factor appropriate to Fq 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 permitted power, the indicated axial flux difference must be maintained within the range +9 percent to -17 percent. 9

e. For every 4 percent below permitted power, the permissible positive flux difference range is extended by +1 percent and the permissible negative flux difference is extended by 2 percent. 9

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:

(1) 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. 6

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(2) A partial movable incore detector map must be taken 10 to 17 days after the full map. A partial map is defined as surveillance of a minimum of 20 fuel assembly detector thimbles with at least 4 per quadrant.

(3) Two traverses with the movable incore detectors in appropriate alternate thimbles shall be taken during each calendar week.

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 permitted 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 permitted power, 2% for each percent of quadrant tilt. 6

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 permitted 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 permitted power are exceeded and the power is greater than 10% - the Atomic Energy Commission shall be notified and the nuclear overpower,

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overpower ΔT and overtemperature ΔT trips shall be reduced one percent for each percent the hot channel factor exceeds the rated power design values.

c. If the hot channel factors are not determined, the Atomic Energy 6
Commission shall be notified and the overpower ΔT and overtemperature ΔT trip settings shall be reduced by the equivalent of 2% power for every 1% quadrant to average power tilt.

C. Inoperable Control Rods

1. A control rod assembly shall be considered inoperable if the assembly cannot be moved by the drive mechanism, or the assembly remains misaligned from its bank by more than 15 inches. A full-length control rod shall be considered inoperable if its rod drop time is greater than 1.8 seconds to dashpot entry.

2. No more than one inoperable control rod assembly shall be permitted when the reactor is critical.

3. If more than one control rod assembly in a given bank is out of service because of a single failure external to the individual rod drive mechanisms, i. e. programming circuitry, the provisions of Specification C1 and 2 shall not apply and the reactor may remain critical for a period not to exceed two hours provided immediate attention is directed toward making the necessary repairs. In the event the affected assemblies cannot be returned

to service within this specified period the reactor will be brought to hot shutdown conditions.

4. The provisions of Specifications C1 and 2 shall not apply during physics test in which the assemblies are intentionally misaligned.

5. If an inoperable full-length rod is located below the 200 step level and is capable of being tripped, or if the full-length rod is located below the 30 step level whether or not it is capable of being tripped, then the insertion limits in TS Figure 3.12-2 apply.

6. If an inoperable full-length rod cannot be located, or if the inoperable full-length rod is located above the 30 step level and cannot be tripped, then the insertion limits in TS Figure 3.12-3 apply.

7. No insertion limit changes are required by an inoperable part-length rod.

8. If a full-length rod becomes inoperable and reactor operation is continued the potential ejected rod worth and associated transient power distribution peaking factors shall be determined by analysis within 30 days. The analysis shall include due allowance for nonuniform fuel depletion in the neighborhood of the inoperable rod. If the analysis results in a more limiting hypothetical transient than the cases reported in the safety analysis, the plant power level shall be reduced to an analytically determined part power level which is consistent with the safety analysis.

D. If the reactor is operating above 75% of permitted power with one excore nuclear channel out of service, the core quadrant power balance shall be determined. 6

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. 6

2. To increase power above 75% of permitted power with a part-length or full length control rod more than 15 inches out of alignment with its bank an analysis 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 withdrawn and control of power is by the control groups. A reactor trip occurring during power operation will place the reactor into the hot shutdown condition.

The control rod assembly insertion limits provide for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod assembly remains fully withdrawn, with sufficient margins to meet the assumptions used in the accident analysis. In addition, they provide a limit on the maximum inserted rod worth in the unlikely event of a hypothetical assembly ejection, and provide for acceptable nuclear peaking factors. The limit may be determined on the basis of unit startup and operating data to provide a more realistic limit which will allow for more flexibility in unit operation and still assure compliance with the shutdown requirement. The maximum shutdown margin requirement occurs at end of core life and is based on the value used in analysis of the hypothetical steam break accident. The rod insertion limits are based on end of core life conditions. Early in core life, less shutdown margin is required, and TS Figure 3.12-7 shows the shutdown margin equivalent to 1.77% reactivity at end-of-life with respect to an uncontrolled cooldown. All other accident analyses are based on 1% reactivity shutdown margin.

Relative positions of control rod banks are determined by a specified control rod bank overlap. This overlap is based on the considerations of axial power shape control.

The specified control rod insertion limits have been revised to limit the potential ejected rod worth in order to account for the effects of fuel densification. 6

Part length rod insertion has been limited to eliminate adverse power shapes.

The various control rod assemblies (shutdown banks, control banks A, B, C, D and part-lenth rods) are each to be moved as a bank, that is, with all assemblies in the bank within one step (5/8 inch) of the bank position. Position indication is provided by two methods: a digital count of actuating pulses which shows the demand position of the banks and a linear position indicator, Linear Variable Differential Transformer, which indicates the actual assembly position. The position indication accuracy of the pulse count is within one step (5/8 inch). The accuracy of the Linear Differential Transformer is approximately ±5% of span (±7.5 inches) under steady state conditions.(1) The relative accuracy of the linear position indicator is such that, with the most adverse errors, an alarm is actuated if any two assemblies within a bank deviate by more than 14 inches. In the event that the linear position indicator is not in service, the effects of malpositioned control rod assemblies are observable from nuclear and process information displayed in the Main Control Room and by core thermocouples and in-core movable detectors. Below 50% power, no special monitoring is required for malpositioned control rod assemblies with inoperable rod position indicators because, even with an unnoticed complete assembly misalignment (part-length 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 18.1 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 6 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 tolerances 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. 6

$F_{\Delta H}^N$, 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:

$$F_Q^N \leq 2.48 \text{ and } F_{\Delta H}^N \leq 1.55$$

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 permitted power; these conditions are as follows: 9

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 6
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 operating conditions have been correlated with axial offset. The correlation shows that an F_Q^N of 2.48 and allowed DNB shapes, including the effects of fuel densification, are not exceeded if the axial offset (flux difference) is maintained between -20 and +12%. The specified limits of -17 and +9% allow for a 3% error in the axial offset. In order to gain more information on the margin of safety in the correlation, a temporary movable incore detector surveillance 9
program, which consists of taking two traces weekly and a partial

map monthly, has been specified. 9

For operation at permitted power, design limits

are met, provided,

$F_Q^N \leq 2.48\ [1 + 0.2(1-P)]$ in the indicated flux difference range of +9 to -17%

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 18.1 kW/ft locally, and from going below a minimum DNBR of 1.30, by automatic protection on power, flux difference, pressure 6
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 $F_Q^N \leq 2.48/1.05$ 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.

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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. 6

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 a reduction in F_Q or expansion of permissible quadrant tilt limits over the 2% value, up to a value of 10%, at which point specified power reductions are prudent. Monthly surveillance of F_{xy} bounds the quantity because it decreases with burnup. (WCAP-7912 L).

A 2% quadrant tilt allows that a 5% tilt might actually be present in the core because of insensitivity of the excore detectors for disturbances near the core center such as misaligned inner control rods and an error allowance. No increase in F_Q occurs with tilts up to 5% because misaligned control rods producing such tilts do not extend to the unrodded plane, where the maximum F_Q occurs. 6

References

(1) FSAR Section 7.2

(2) FSAR Section 14.

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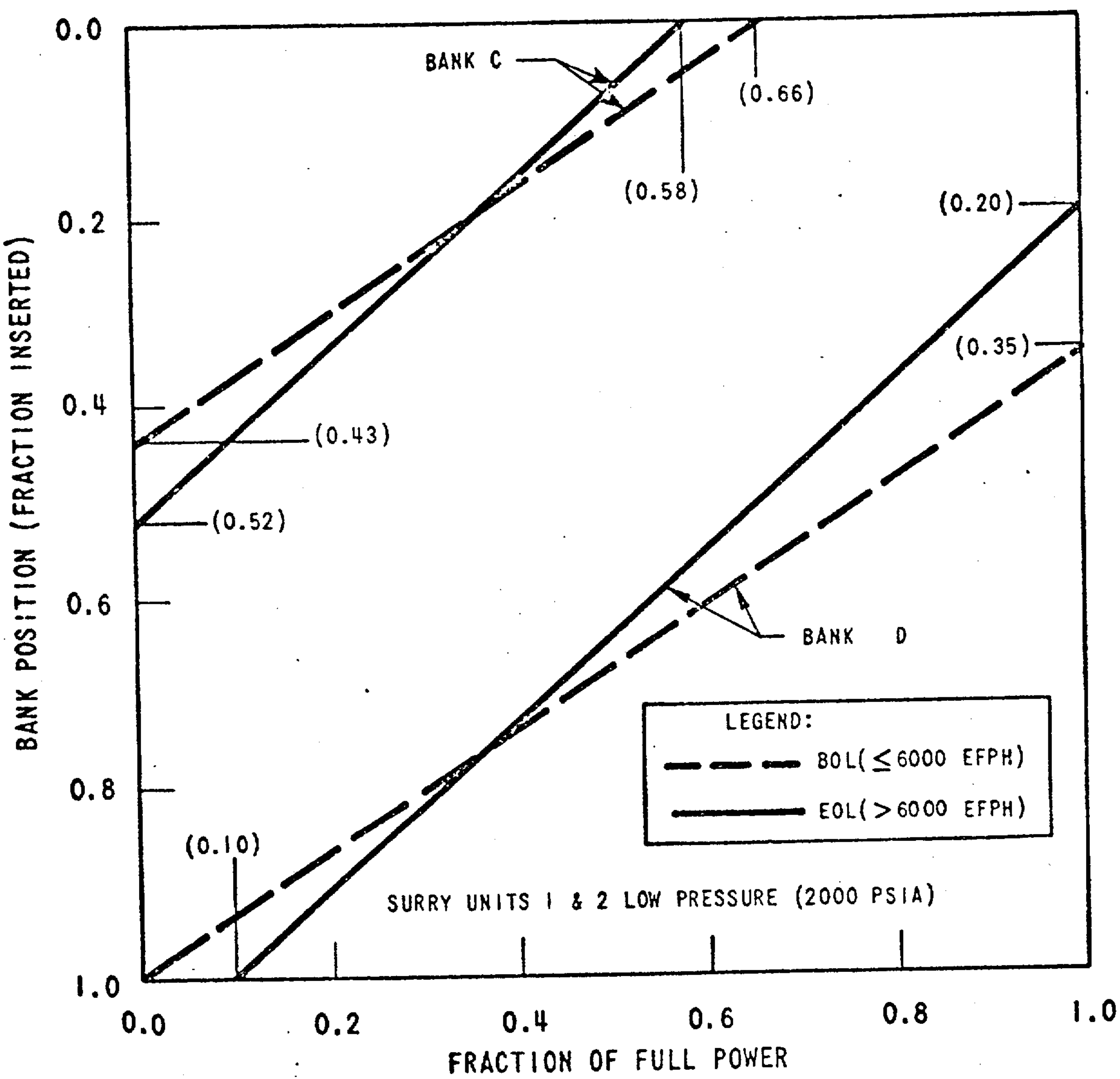

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Figure 3.12-1 Control Bank Insertion Limits for Normal 3 Loop Operation

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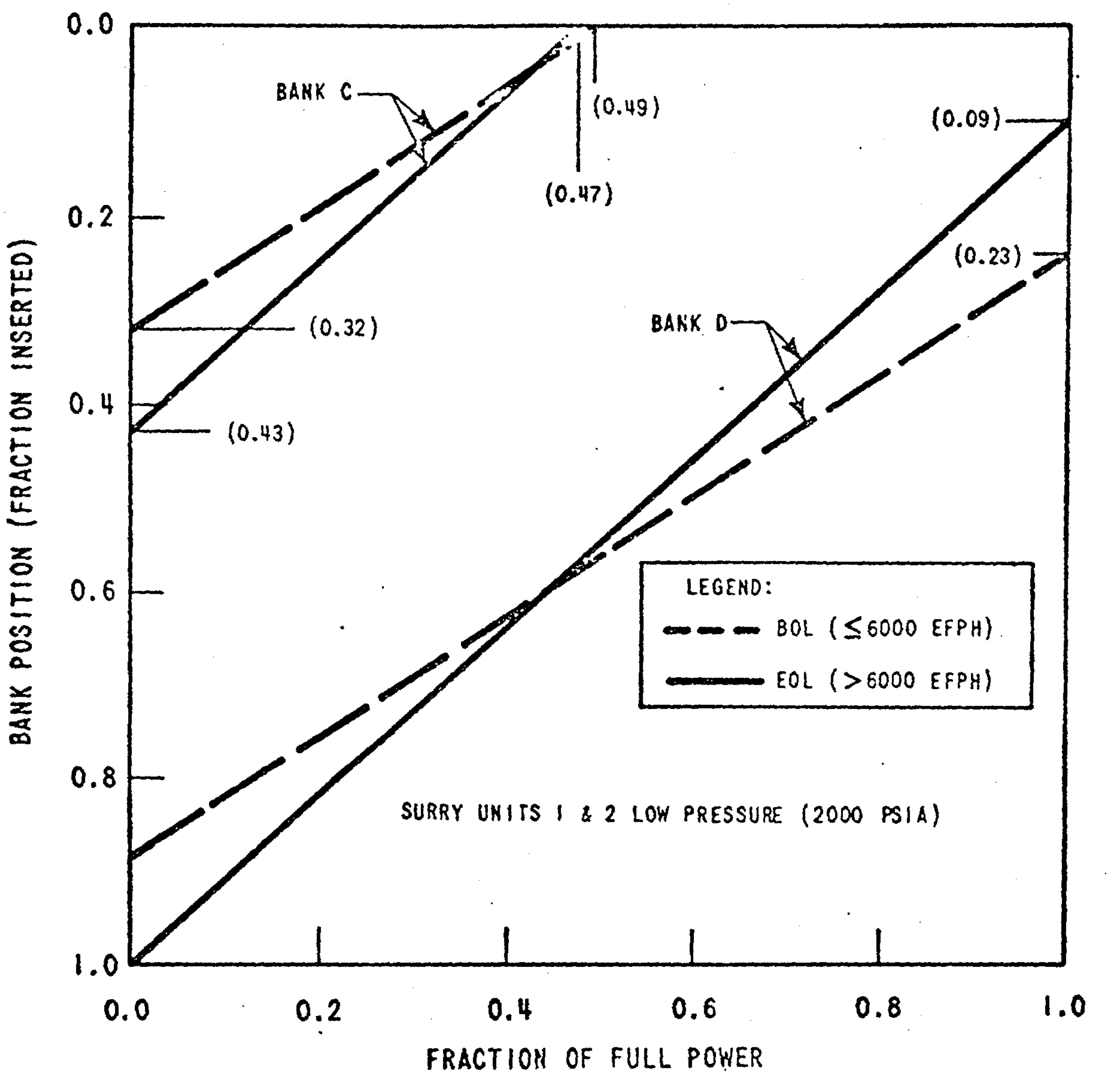

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Figure 3.12-2 Control Bank Insertion Limits for 3 Loop Operation with One Bottomed Rod

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

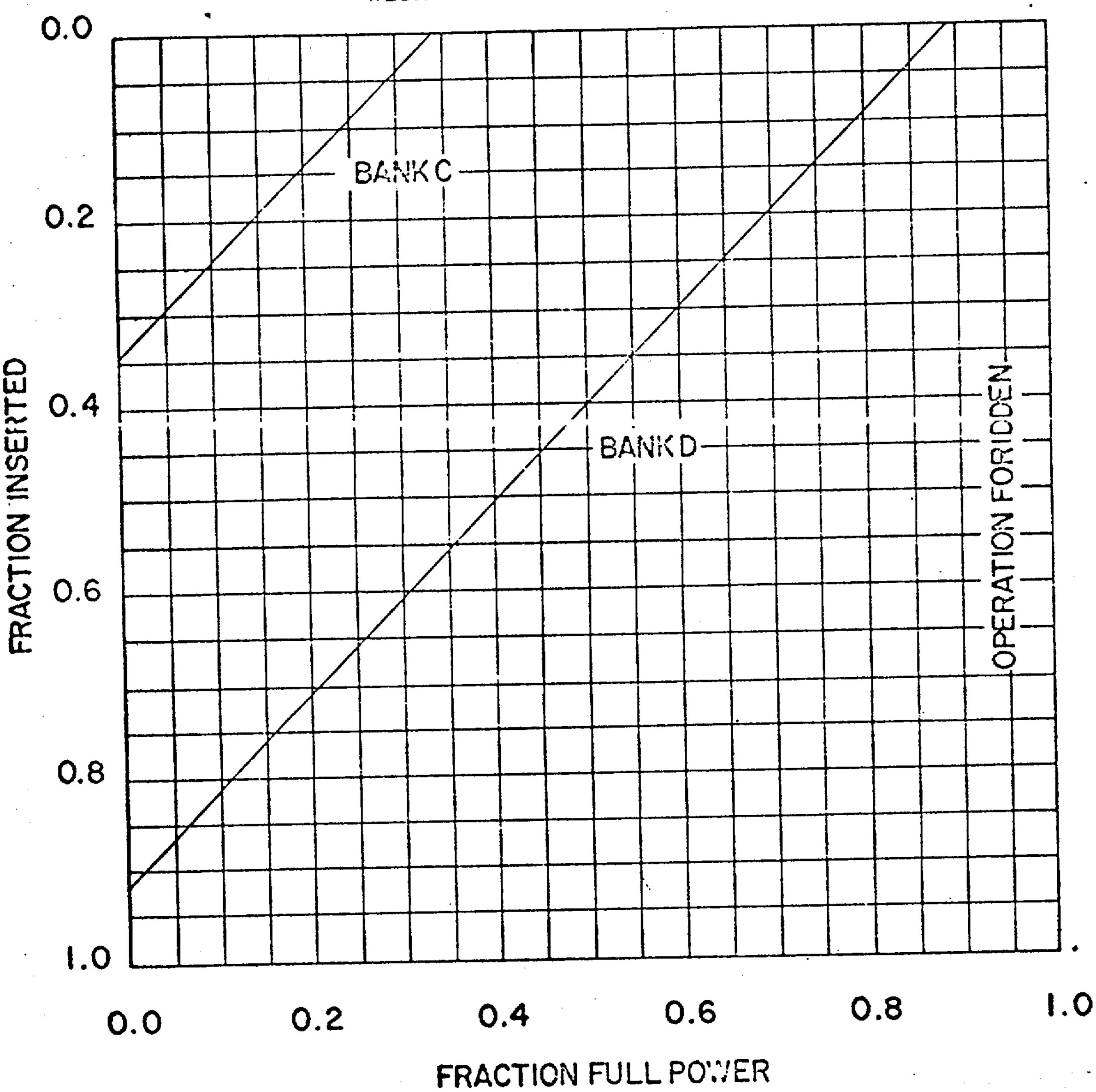
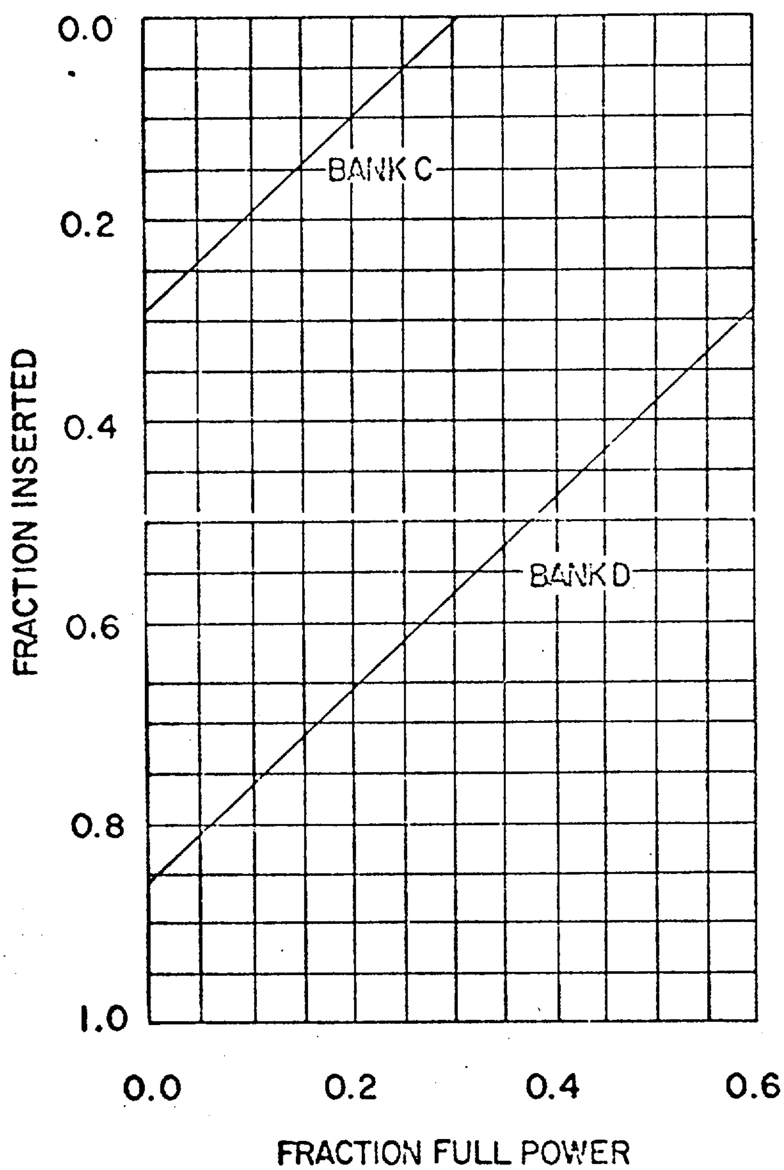
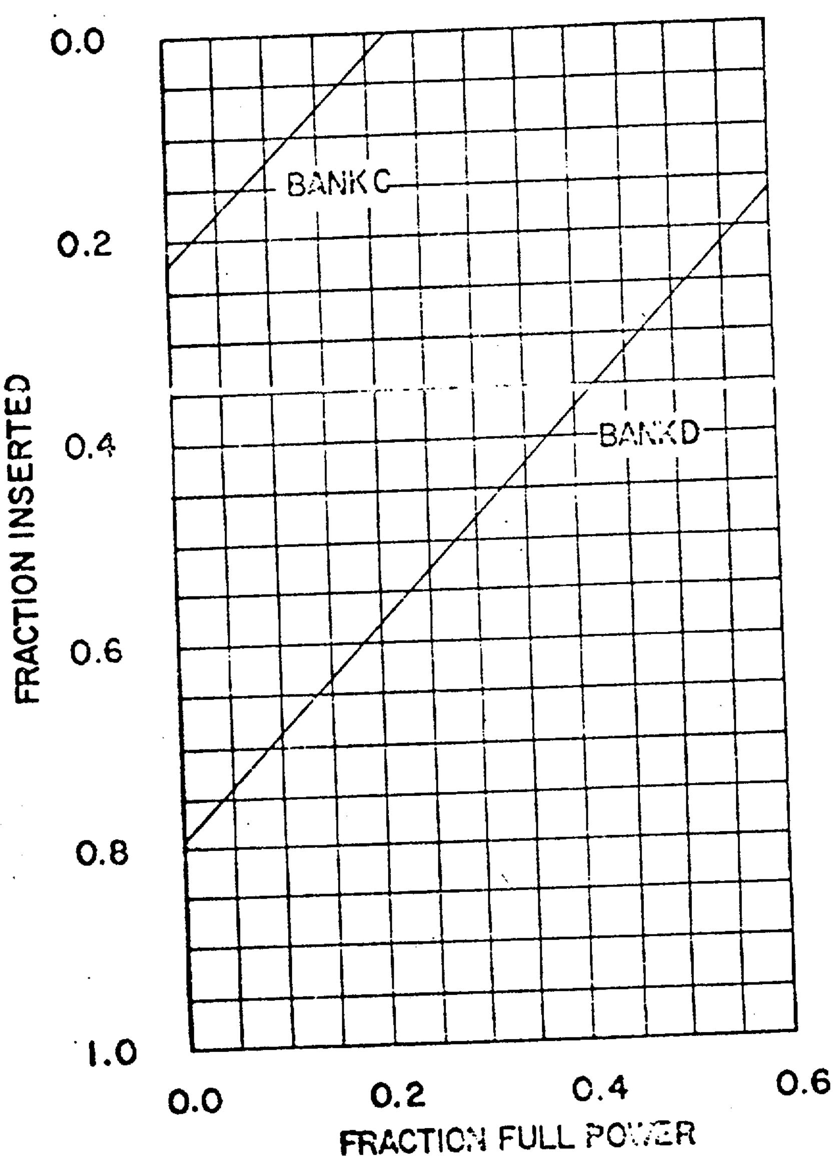


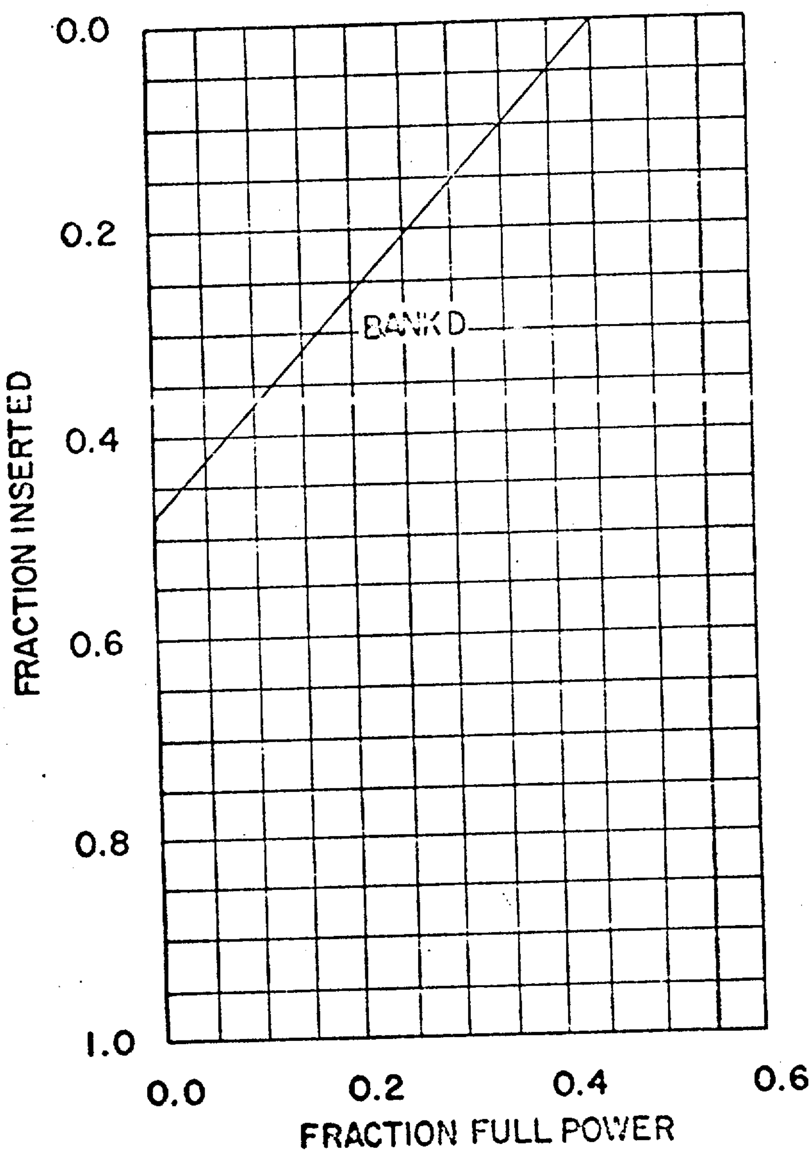
FIGURE 3.12-4
CONTROL BANK INSERTION LIMITS
FOR 2 LOOP NORMAL OPERATION



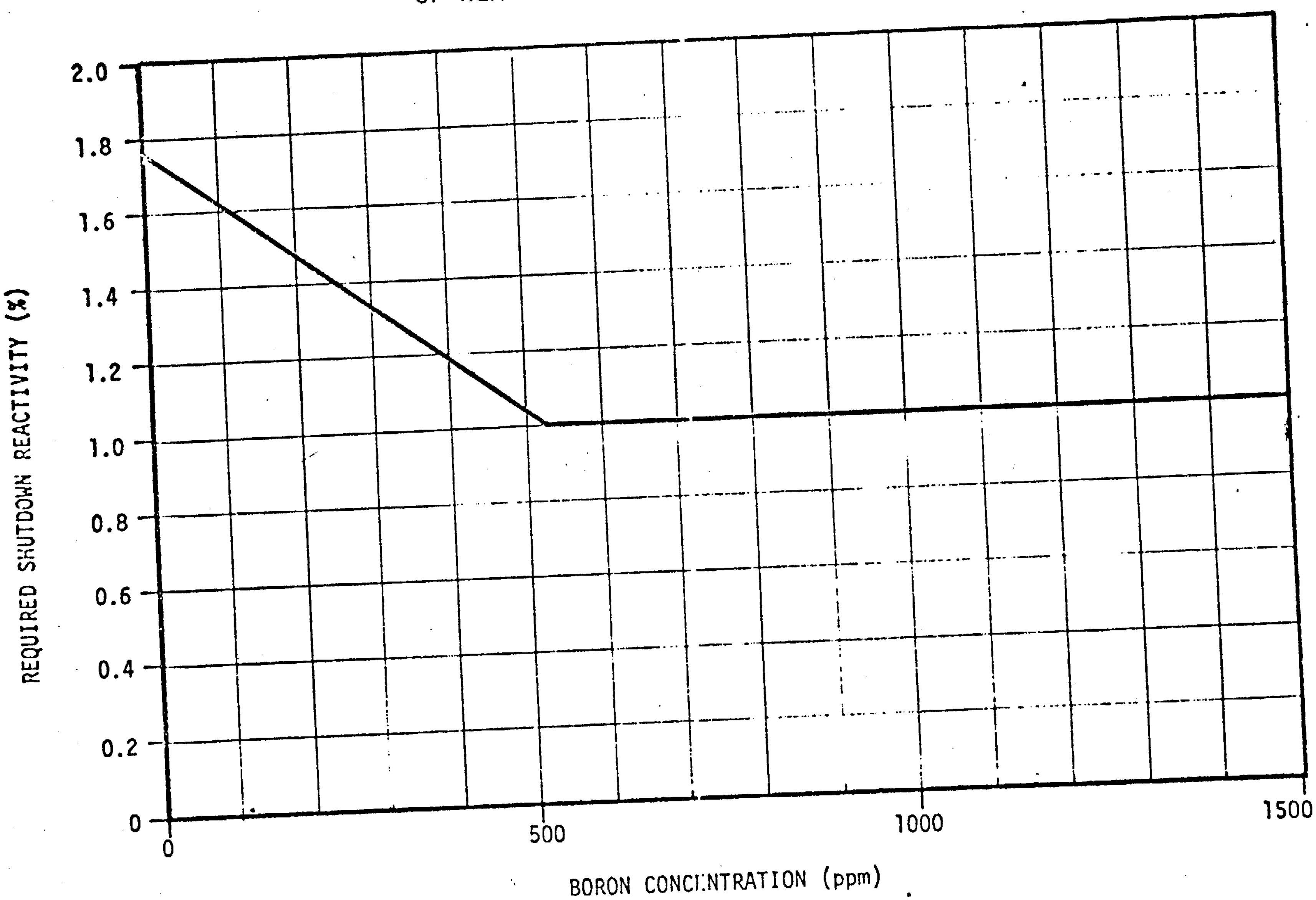
CONTROL BANK INSERTION LIMITS
FOR 2 LOOP OPERATION
WITH ONE BOTTOMED ROD



CONTROL BANK INSERTION LIMITS
FOR 2 LOOP OPERATION
WITH ONE INOPERABLE ROD



REQUIRED SHUTDOWN REACTIVITY AS A FUNCTION OF REACTOR COOLANT BORON CONCENTRATION



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TS FIGURE 3.12-7
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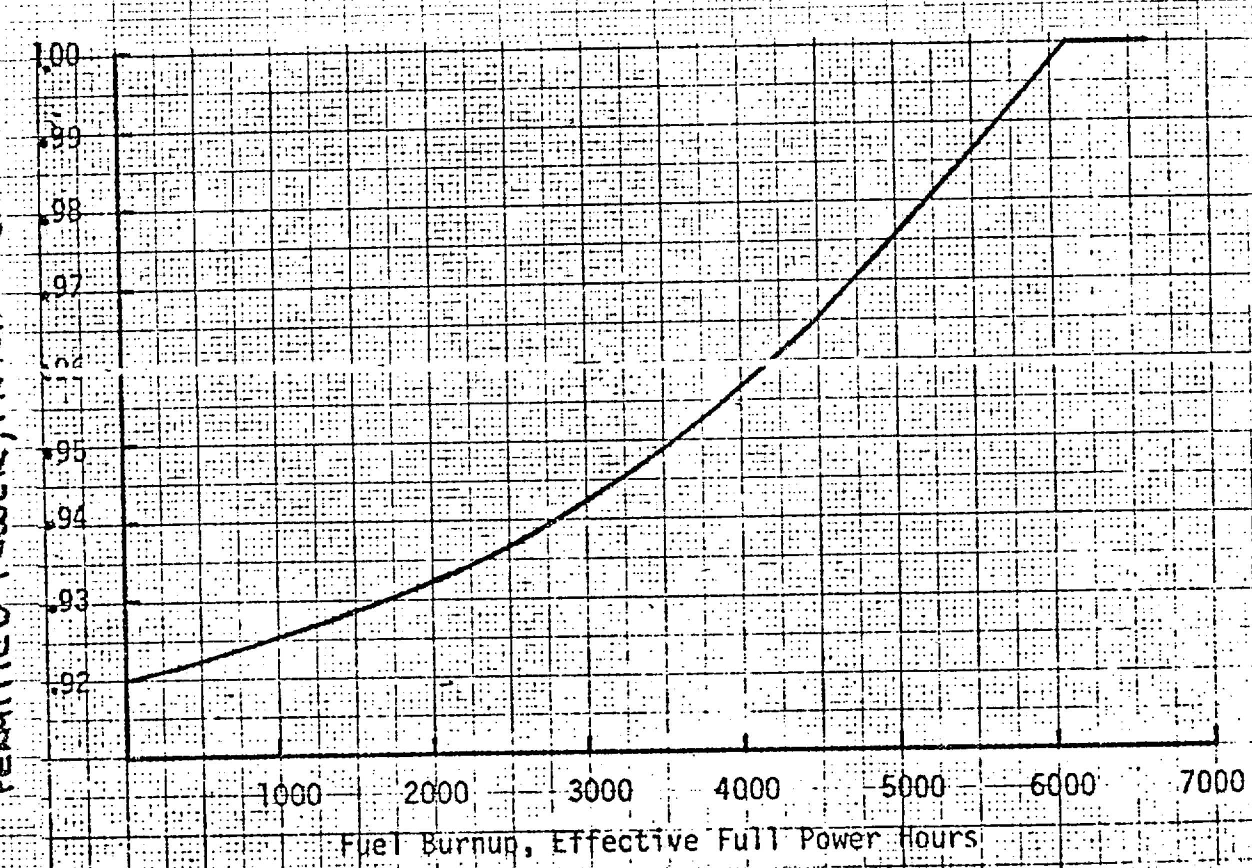
TS FIGURE 3.12-8

PERMITTED POWER TO MEET LOCA CRITERIA VS. BURNUP



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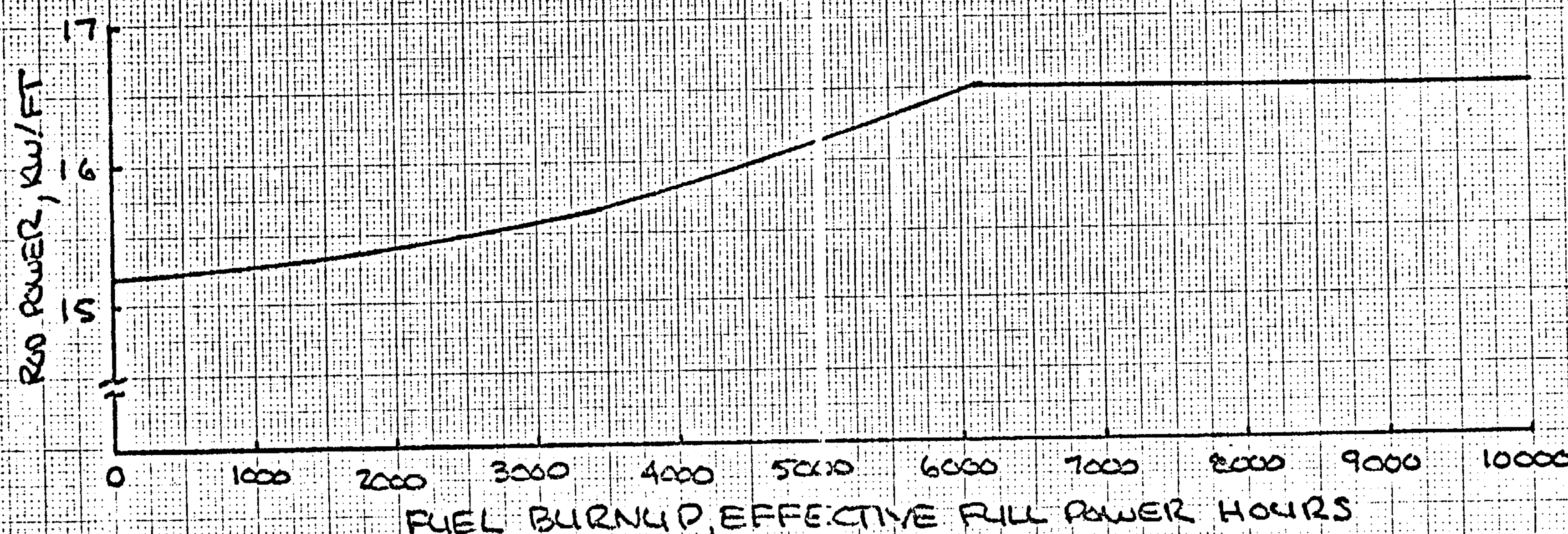
TS FIGURE 3.12-9

φ, LIMITING VALUES OF KW/FT
TO MEET LOCA CRITERIA VS. BURNUP



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TS Figure 3.12-9
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