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DESCRIPTION:
Ltr notarized 9-20-74, trans the following:

ENCLOSURES:
(1) ATTACHMENT 1-Proposed Tech Specs Changes
(2) Babcock & Wilcox Rpt BAW-1409, "Oconee 1 Cycle 2 Reload Report". see Rpt

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PLANT NAME: Oconee Unit # 1

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DUKE POWER COMPANY
POWER BUILDING
422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28201

A. C. THIES
SENIOR VICE PRESIDENT
PRODUCTION AND TRANSMISSION

P. O. Box 2178

September 20, 1974

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Angelo Giambusso
Deputy Director of Reactor Projects
Directorate of Licensing
Office of Regulation
U. S. Atomic Energy Commission
Washington, D. C. 20545

RE: Oconee Unit 1
Docket No. 50-269

Dear Mr. Giambusso:

Please find attached for your review proposed changes to the Oconee Nuclear Station Technical Specifications. The purpose of these revisions is to ensure operation of the Oconee Unit 1, Cycle 2 Core within appropriate fuel design criteria and LOCA limits. Proposed changes are shown in Attachment 1 as replacement pages for the Oconee Nuclear Station Technical Specifications:

2.1 SAFETY LIMITS, REACTOR CORE
2.3 LIMITING SAFETY SYSTEMS SETTINGS, PROTECTIVE INSTRUMENTATION
3.5.2 CONTROL ROD GROUP AND POWER DISTRIBUTION LIMIT

Also attached is the Babcock & Wilcox report BAW-1409, "Oconee 1 Cycle 2 Reload Report." This report includes a summary of Cycle 2 operating parameters, and contains safety analyses supporting operations of the Oconee 1 Cycle 2 Core at rated power.

Refueling operations for Oconee Unit 1 are presently scheduled to commence on October 20, 1974, and power operation should resume on November 20, 1974. Your prompt review of these proposed revisions is requested.

Very truly yours,

A C Thies

A. C. Thies

ACT:gje

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A. C. THIES, being duly sworn, states that he is Senior Vice President of Duke Power Company; that he is authorized on the part of said Company to sign and file with the Atomic Energy Commission this request for amendment of the Oconee Nuclear Station Technical Specifications, Appendix A to Facility Operating License DPR-38; and that all statements and matters set forth therein are true and correct to the best of his knowledge.

A C Thies

A. C. Thies, Senior Vice President

ATTEST:

Dorothea B. Stroupe

Dorothea B. Stroupe
Assistant Secretary

Subscribed and sworn to before me this 20th day of September, 1974.

Edna B. Farmer

Notary Public

My Commission Expires:

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ATTACHMENT 1

OCONEE NUCLEAR STATION

PROPOSED TECHNICAL SPECIFICATION REPLACEMENT PAGES

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2 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS, REACTOR CORE

Applicability

Applies to reactor thermal power, reactor power imbalance, reactor coolant system pressure, coolant temperature, and coolant flow during power operation of the plant.

Objective

To maintain the integrity of the fuel cladding.

Specification

The combination of the reactor system pressure and coolant temperature shall not exceed the safety limit as defined by the locus of points established in Figure 2.1-1A - Unit 1. If the actual pressure/temperature point is below
2.1-1B - Unit 2
2.1-1C - Unit 3
and to the right of the line, the safety limit is exceeded.

The combination of reactor thermal power and reactor power imbalance (power in the top half of the core minus the power in the bottom half of the core expressed as a percentage of the rated power) shall not exceed the safety limit as defined by the locus of points (solid line) for the specified flow set forth in Figure 2.1-2A - Unit 1. If the actual reactor-thermal-power/
2.1-2B - Unit 2
2.1-2C - Unit 3
power-imbalance point is above the line for the specified flow, the safety limit is exceeded.

Bases

To maintain the integrity of the fuel cladding and to prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB). At this point, there is a sharp reduction of the heat transfer coefficient, which would result in high cladding temperatures and the possibility of cladding failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature, and pressure can be related to DNB through the use of the W-3 correlation.(1) The W-3 correlation has been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), 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 DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.3. A DNBR of 1.3 corresponds to a 94.3% probability at a 99% confidence level that DNB will not occur; this is considered a conservative margin to DNB for all operating conditions. The difference between the actual core outlet pressure and the indicated reactor coolant system pressure has been considered in determining the core protection safety limits. The difference in these two pressures is nominally 45 psi; however, only a 30 psi drop was assumed in reducing the pressure trip set points to correspond to the elevated location where the pressure is actually measured.

The curve presented in Figure 2.1-1A
2.1-1B
2.1-1C
represents the conditions at which a minimum DNBR of 1.3 is predicted for the maximum possible thermal power (112%) when four reactor coolant pumps are operating(minimum reactor coolant flow is 131.3 x 10^6 lbs/hr.) This curve is based on the following nuclear power peaking factors(2) with potential fuel densification effects;

$$F_q^N = 2.67; \; F_{\Delta H}^N = 1.78; \; F_z^N = 1.50$$

The design peaking combination results in a more conservative DNBR than any other shape that exists during normal operation.

The curves of Figure 2.1-2A
2.1-2B
2.1-2C
are based on the more restrictive of two thermal limits and include the effects of potential fuel densification:

1. The 1.3 DNBR limit produced by a nuclear power peaking factor of $F_q^N = 2.67$ or the combination of the radial peak, axial peak and position of the axial peak that yields no less than a 1.3 DNBR.

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 20.15 kw/ft - Unit 1
19.8 kw/ft - Unit 2
19.8 kw/ft - Unit 3

Power peaking is not a directly observable quantity and therefore limits have been established on the bases of the reactor power imbalance produced by the power peaking.

The specified flow rates for Curves 1, 2, 3, and 4 of Figure 2.1-2A
2.1-2B
2.1-2C
correspond to the expected minimum flow rates with four pumps, three pumps, one pump in each loop and two pumps in one loop, respectively.

The curve of Figure 2.1-1A
2.1-1B
2.1-1C
is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in Figure 2.1-3A.
2.1-3B
2.1-3C

The curves of Figure 2.1-3B
2.1-3C
represent the conditions at which a minimum DNBR

of 1.3 is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation or the local quality at the point of minimum DNBR is equal to 15%, (3) whichever condition is more restrictive.

Using a local quality limit of 15% at the point of minimum DNBR as a basis for Curves 2 and 4 of Figure 2.1-3B is a conservative criterion even though the
2.1-3C
quality of the exit is higher than the quality at the point of minimum DNBR.

The DNBR as calculated by the W-3 correlation continually increases from point of minimum DNBR, so that the exit DNBR is 1.7 or higher, depending on the pressure. Extrapolation of the W-3 correlation beyond its published quality range of +15% is justified on the basis of experimental data. (4)

The maximum thermal power for three pump operation is 86% - Unit 2
86% - Unit 3
due to a power level trip produced by the flux-flow ratio 75% flow x 1.07 = 80%
1.07 = 80%
power
plus the maximum calibration and instrument error. The maximum thermal power for other coolant pump conditions are produced in a similar manner.

For each curve of Figure 2.1-3B, a pressure-temperature point above and to the
2.1-3C
left of the curve would result in a DNBR greater than 1.3 or a local quality at the point of minimum DNBR less then 15% for that particular reactor coolant pump situation. The 1.3 DNBR curve for four pump operation is more restrictive than any other reactor coolant pump situation because any pressure/temperature point above and to the left of the four pump curve will be above and to the left of the other curves.

The safety limits presented for Oconee Unit 1 have been generated using BAW-2 critical heat flux (CHF) correlation and the actual measured flow rate at Oconee Unit 1.(5) This development is discussed in the Oconee 1, Cycle 2-Reload Report, reference.(6) These limits have been developed using the procedures previously discussed in the Bases with the exception of the CHF correlation and the measured flow rate. The minimum DNBR for the BAW-2 correlation is 1.32 which corresponds to a 95% probability at a 99% confidence level that DNB will occur. The flow rate utilized is 107.6% of the design flow (131.32 X 10^6 lbs/HR) based on four pump operation.(6) Because the four pump pressure-temperature restriction is known to be more limiting than the 3 and 2 pump combinations, only the four pump limit has been shown on Figure 2.1-3A. The maximum thermal power for three pump operation for Unit 1 is 87% due to a power level trip produced by the flux-flow ratio 75% flow X 1.08 = 81% power plus the maximum calibration and instrument error.

REFERENCES

(1) FSAR, Section 3.2.3.1.1
(2) FSAR, Section 3.2.3.1.1.c
(3) FSAR, Section 3.2.3.1.1.k

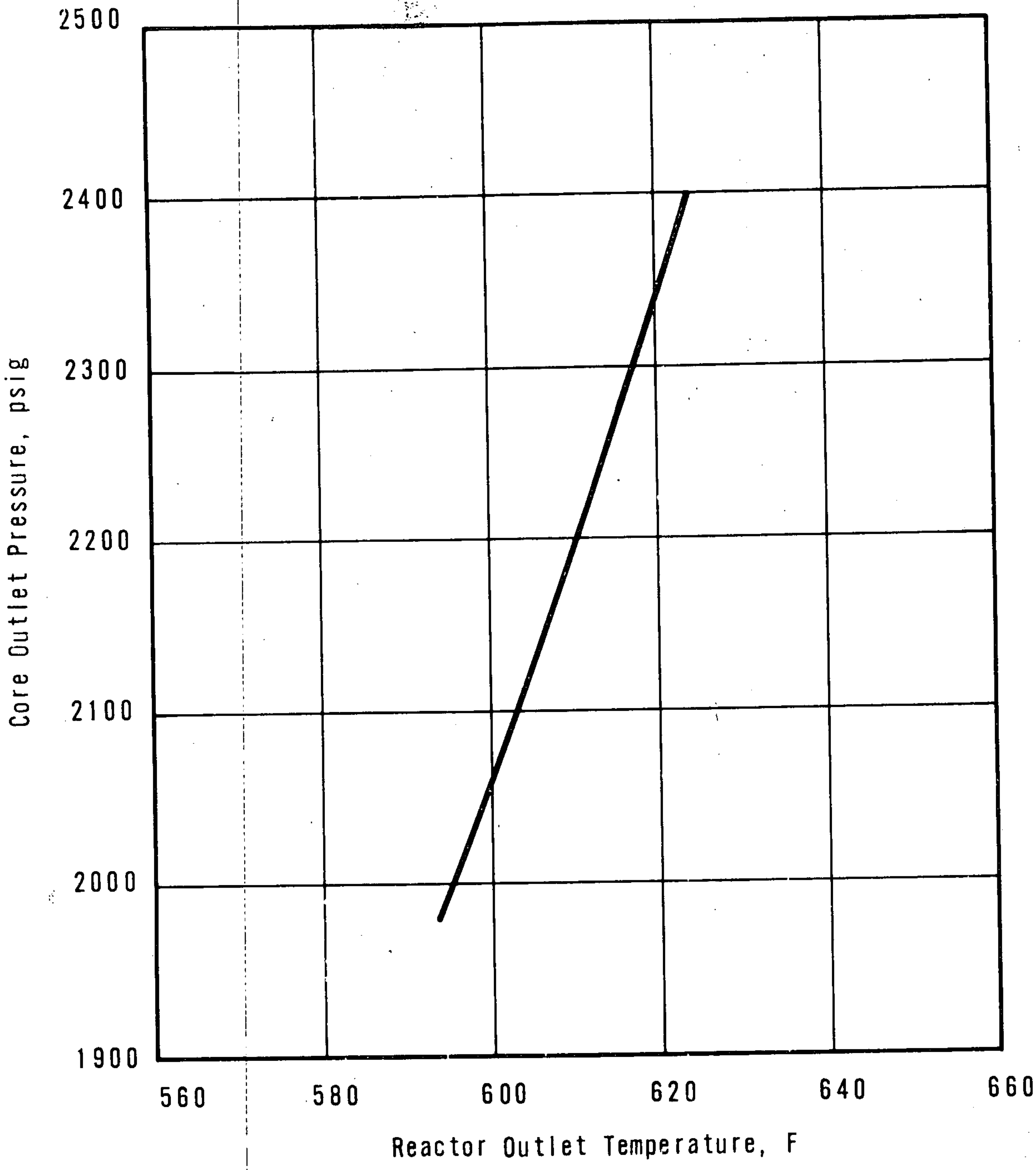
(4) The following papers which were presented at the Winter Annual Meeting, ASME, November 18, 1969, during the "Two-phase Flow and Heat Transfer in Rod Bundles Symposium:"

(a) Wilson, et. al.
"Critical Heat Flux in Non-Uniform Heater Rod Bundles".

(b) Gellerstedt, et. al.
"Correlation of a Critical Heat Flux in a Bundle Cooled by Pressurized Water".

(5) Corrélation of Critical Heat Flux in a Bundle Cooled by Pressurized Water, BAW-10000, Mar. 1970.

(6) Oconee 1, Cycle 2 - Reload Report - BAW-1409, September, 1974.



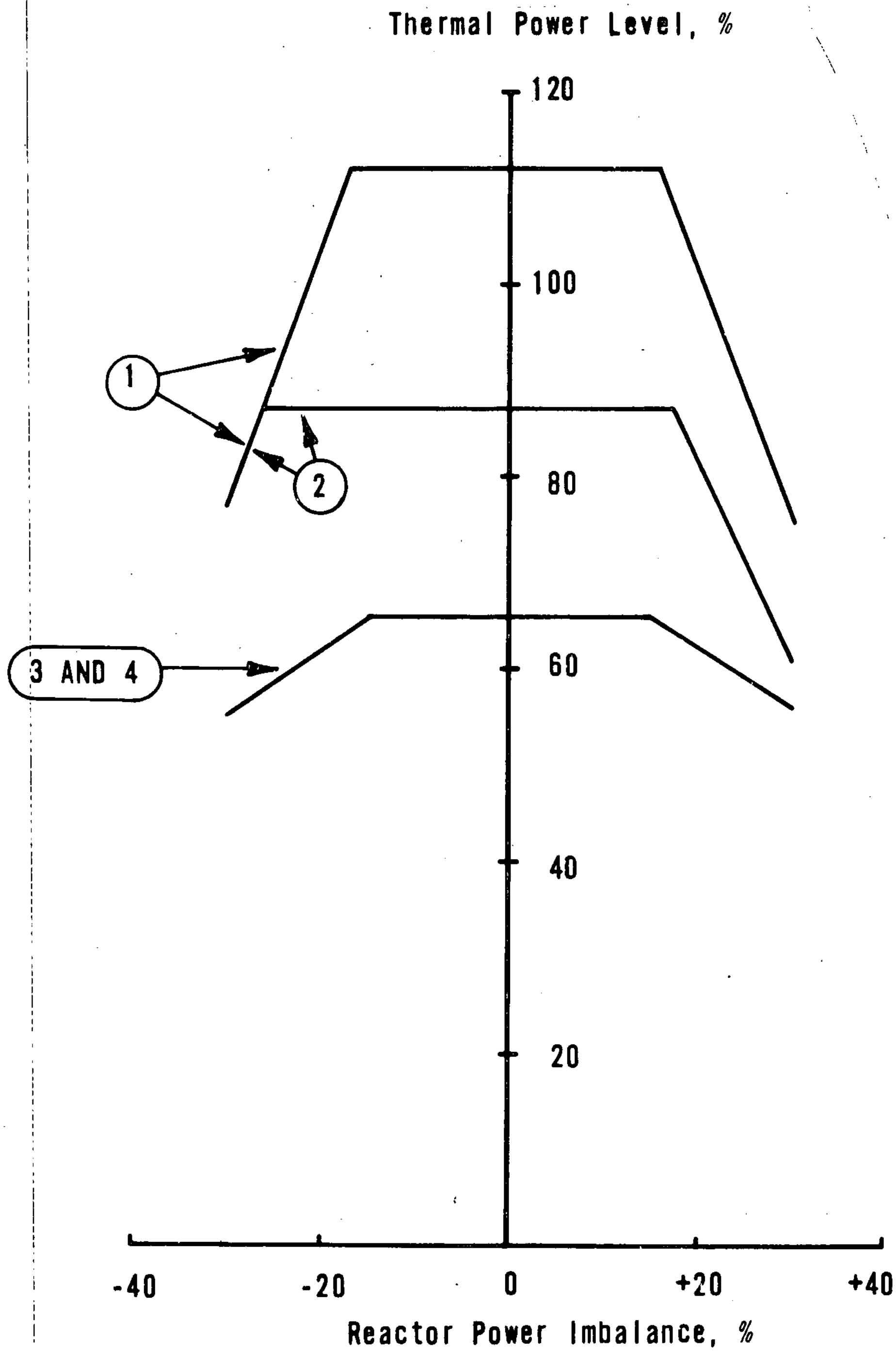
CORE PROTECTION SAFETY LIMITS

UNIT 1

2.1-4a

OCONEE NUCLEAR STATION

Figure 2.1-1A



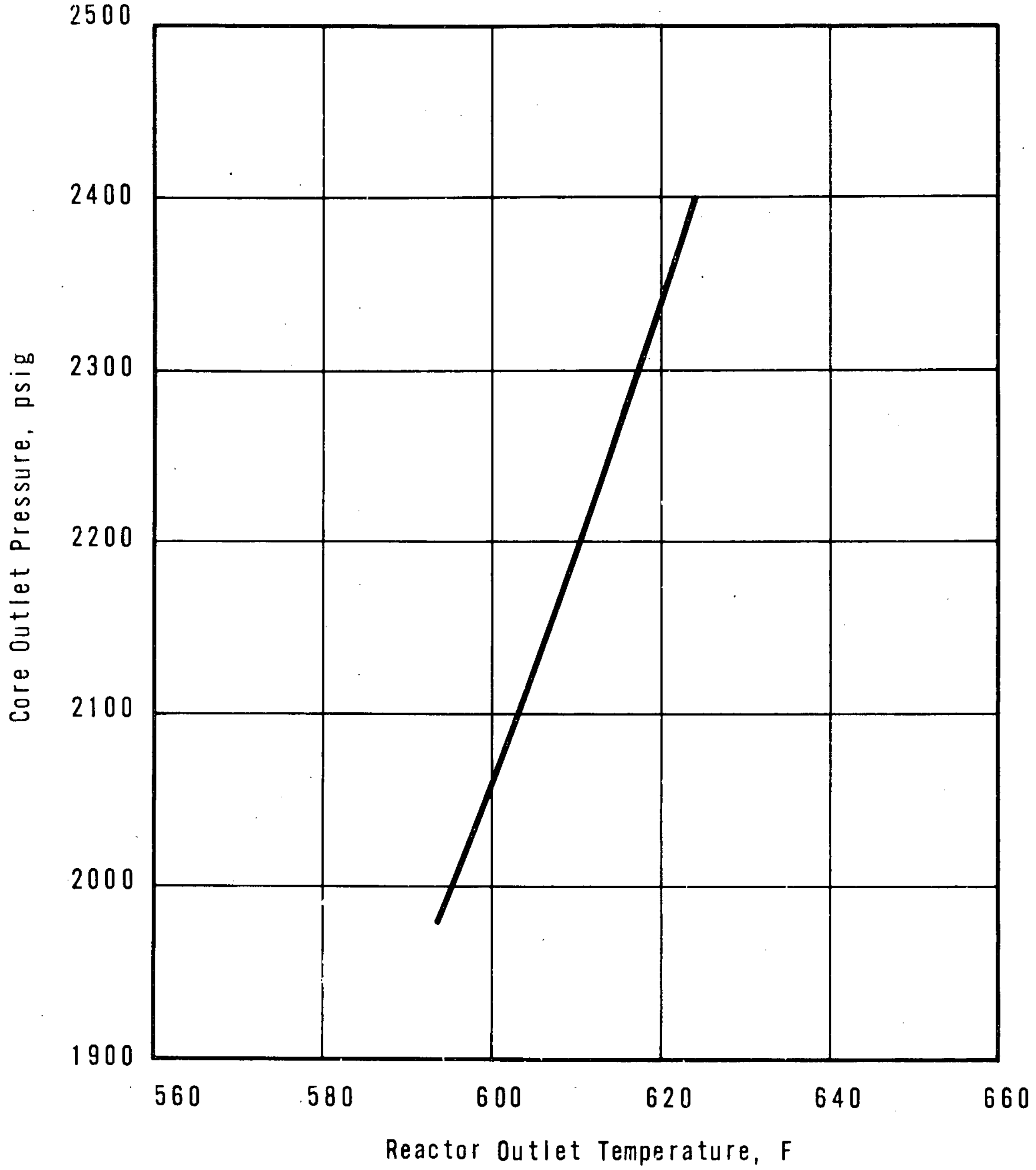
CURVE	REACTOR COOLANT FLOW (LB/HR)
1	131.3×10^6
2	98.1×10^6
3	64.4×10^6
4	60.1×10^6

CORE PROTECTION SAFETY LIMITS

UNIT 1

OCONEE NUCLEAR STATION

Figure 2.1-2A



CORE PROTECTION SAFETY LIMITS

UNIT 1

OCONEE NUCLEAR STATION

Figure 2.1-3A

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Applicability

Applies to instruments monitoring reactor power, reactor power imbalance, reactor coolant system pressure, reactor coolant outlet temperature, flow, number of pumps in operation, and high reactor building pressure.

Objective

To provide automatic protective action to prevent any combination of process variables from exceeding a safety limit.

Specification

The reactor protective system trip setting limits and the permissible bypasses for the instrument channels shall be as stated in Table 2.3-1A - Unit 1 and
2.3-1B - Unit 2
2.3-1C - Unit 3

Figure 2.3-2A1 } Unit 1
2.3-2A2
2.3-2B - Unit 2
2.3-2C - Unit 3

The pump monitors shall produce a reactor trip for the following conditions:

a. Loss of two pumps and reactor power level is greater than 55% (0.0% for Unit 1) of rated power.

b. Loss of two pumps in one reactor coolant loop and reactor power level is greater than 0.0% of rated power. (Reactor power level trip setpoint is reset to 55% of rated power for single loop operation. Reactor power level trip setpoint is reset to 55% for all modes of 2 pump operation for Unit 1.)

c. Loss of one or two pumps during two-pump operation.

Bases

The reactor protective system consists of four instrument channels to monitor each of several selected plant conditions which will cause a reactor trip if any one of these conditions deviates from a pre-selected operating range to the degree that a safety limit may be reached.

The trip setting limits for protective system instrumentation are listed in Table 2.3-1A - Unit 1. The safety analysis has been based upon these protective
2.3-1B - Unit 2
2.3-1C - Unit 3
system instrumentation trip set points plus calibration and instrumentation errors.

Nuclear Overpower

A reactor trip at high power level (neutron flux) is provided to prevent damage to the fuel cladding from reactivity excursions too rapid to be detected by pressure and temperature measurements.

During normal plant operation with all reactor coolant pumps operating, reactor trip is initiated when the reactor power level reaches 105.5% of rated power. Adding to this the possible variation in trip setpoints due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which is more conservative than the value used in the safety analysis.(4)

Overpower Trip Based on Flow and Imbalance

The power level trip set point produced by the reactor coolant system flow is based on a power-to-flow ratio which has been established to accommodate the most severe thermal transient considered in the design, the loss-of-coolant flow accident from high power. Analysis has demonstrated that the specified power-to-flow ratio is adequate to prevent a DNBR of less than 1.3 should a low flow condition exist due to any electrical malfunction.

The power level trip set point produced by the power-to-flow ratio provides both high power level and low flow protection in the event the reactor power level increases or the reactor coolant flow rate decreases. The power level trip set point produced by the power-to-flow ratio provides overpower DNB protection for all modes of pump operation. For every flow rate there is a maximum permissible power level, and for every power level there is a minimum permissible low flow rate. Typical power level and low flow rate combinations for the pump situations of Table 2.3-1A are as follows:

1. Trip would occur when four reactor coolant pumps are operating if power is 108% and reactor flow rate is 100%, or flow rate is 93% and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is 81.0% and reactor flow rate is 74.7% or flow rate is 69% and power level is 75%.

3. Trip would occur when two reactor coolant pumps are operating in a single loop if power is 59% and the operating loop flow rate is 54.5% or flow rate is 43% and power level is 46%.

4. Trip would occur when one reactor coolant pump is operating in each loop (total of two pumps operating) if the power is 53% and reactor flow rate is 49.0% or flow rate is 45% and the power level is 49%.

For safety calculations the maximum calibration and instrumentation errors for the power level trip were used.

The power-imbalance boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kw/ft limits or DNBR limits. The reactor power imbalance (power in the top half of core minus power in the bottom half of core) reduces the power level trip produced by the power-to-flow ratio such that the boundaries of Figure 2.3-2A1, 2.3-2A2 }Unit 1; 2.3-2B - Unit 2; 2.3-2C - Unit 3 are produced. The power-to-flow ratio reduces the power

Shutdown Bypass

In order to provide for control rod drive tests, zero power physics testing, and startup procedures, there is provision for bypassing certain segments of the reactor protection system. The reactor protection system segments which can be bypassed are shown in Table 2.3-1A. Two conditions are imposed when
2.3-1B
2.3-1C
the bypass is used:

1. By administrative control the nuclear overpower trip set point must be reduced to a value $\leq$ 5.0% of rated power during reactor shutdown.

2. A high reactor coolant system pressure trip setpoint of 1720 psig is automatically imposed.

The purpose of the 1720 psig high pressure trip set point is to prevent normal operation with part of the reactor protection system bypassed. This high pressure trip set point is lower than the normal low pressure trip set point so that the reactor must be tripped before the bypass is initiated. The over power trip set point of $\leq$ 5.0% prevents any significant reactor power from being produced when performing the physics tests. Sufficient natural circulation (5) would be available to remove 5.0% of rated power if none of the reactor coolant pumps were operating.

Two Pump Operation

A. Two Loop Operation

Operation with one pump in each loop will be allowed only following reactor shutdown. After shutdown has occurred, the following actions will permit operation with one pump in each loop:
1. Reset the pump contact monitor power level trip setpoint to 55.0%.
2. (Unit 1) Reset the protective system maximum allowable setpoint as shown in Figure 2.3-2A2.

B. Single Loop Operation

Single loop operation is permitted only after the reactor has been tripped. After the pump contact monitor trip has occurred, the following actions will permit single loop operation:
1. Reset the pump contact monitor power level trip setpoint to 55.0%.
2. Trip one of the two protective channels receiving outlet temperature information from sensors in the Idle Loop.
3. (Unit 1) Reset the protective system maximum allowable setpoints as shown in Figure 2.3-2A2. Tripping one of the two protective channels receiving outlet temperature information from the idle loop assures a protective system trip logic of one out of two.

REFERENCES

(1) FSAR, Section 14.1.2.2
(2) FSAR, Section 14.1.2.7
(3) FSAR, Section 14.1.2.8
(4) FSAR, Section 14.1.2.3
(5) FSAR, Section 14.1.2.6

level trip and associated reactor power/reactor power-imbalance boundaries by
1.08% - Unit 1 for a 1% flow reduction.
1.07% - Unit 2
1.07% - Unit 3

Pump Monitors

The pump monitors prevent the minimum core DNBR from decreasing below 1.3 by tripping the reactor due to the loss of reactor coolant pump(s). The circuitry monitoring pump operational status provides redundant trip protection for DNB by tripping the reactor on a signal diverse from that of the power-to-flow ratio. The pump monitors also restrict the power level for the number of pumps in operation.

Reactor Coolant System Pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure set point is reached before the nuclear overpower trip set point. The trip setting limit shown in Figure 2.3-1A - Unit 1
2.3-1B - Unit 2
2.3-1C - Unit 3
for high reactor coolant system pressure (2355 psig) has been established to maintain the system pressure below the safety limit (2750 psig) for any design transient.(1)

The low pressure (1985) psig and variable low pressure ($13.77\ T_{out} - 6181$) trip
(1800) psig ($16.25\ T_{out} - 7756$)
(1800) psig ($16.25\ T_{out} - 7756$)
setpoints shown in Figure 2.3-1A have been established to maintain the DNB
2.3-1B
2.3-1C
ratio greater than or equal to 1.3 for those design accidents that result in a pressure reduction.(2,3)

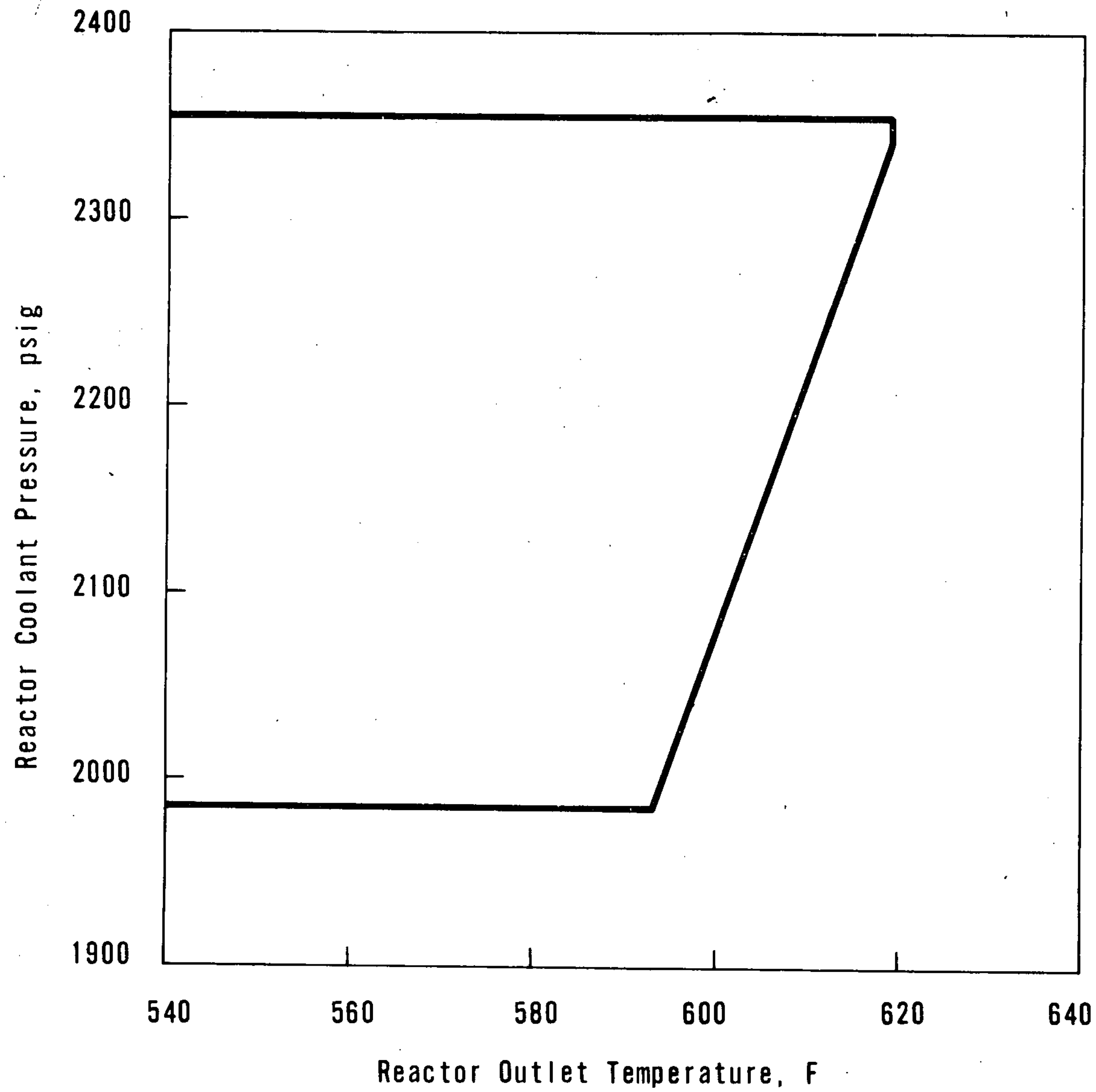
Due to the calibration and instrumentation errors the safety analysis used a variable low reactor coolant system pressure trip value of ($13.77\ T_{out} - 6221$)
($16.25\ T_{out} - 7796$)
($16.25\ T_{out} - 7796$)

Coolant Outlet Temperature

The high reactor coolant outlet temperature trip setting limit (619 F) shown in Figure 2.3-1A has been established to prevent excessive core coolant
2.3-1B
2.3-1C
temperatures in the operating range. Due to calibration and instrumentation errors, the safety analysis used a trip set point of 620°F.

Reactor Building Pressure

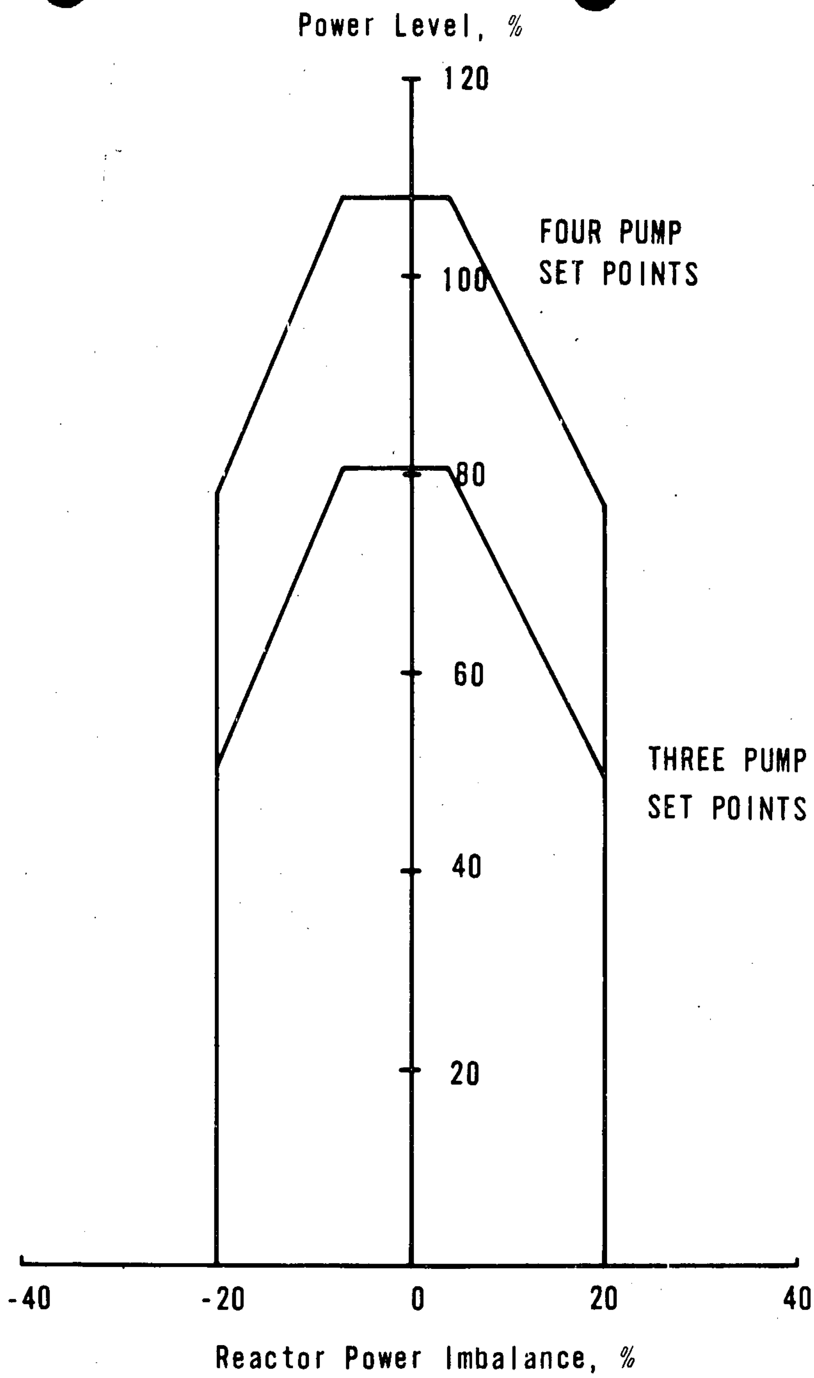
The high reactor building pressure trip setting limit (4 psig) provides positive assurance that a reactor trip will occur in the unlikely event of a loss-of-coolant accident, even in the absence of a low reactor coolant system pressure trip.



PROTECTIVE SYSTEM MAXIMUM
ALLOWABLE SET POINTS

UNIT 1

OCONEE NUCLEAR STATION
Figure 2.3-1A

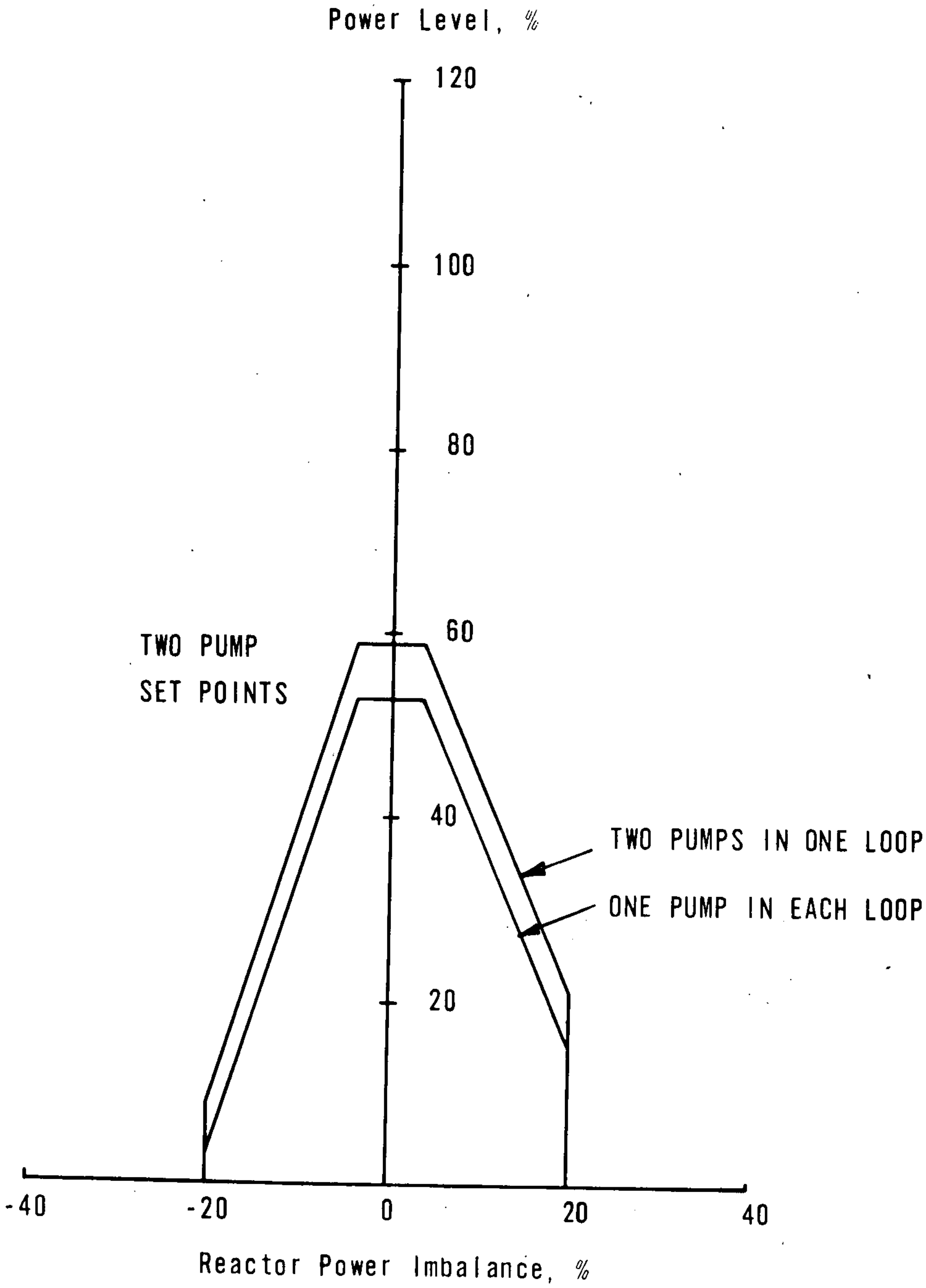


PROTECTIVE SYSTEM MAXIMUM
ALLOWABLE SET POINTS

UNIT 1

OCONEE NUCLEAR STATION

Figure 2.3-2A1



PROTECTIVE SYSTEM MAXIMUM
ALLOWABLE SET POINTS

UNIT 1

OCONEE NUCLEAR STATION
Figure 2.3-2A2

Table 2.3-1A
Unit 1

Reactor Protective System Trip Setting Limits

	RPS Segment	Four Reactor Coolant Pumps Operating (Operating Power -100% Rated)	Three Reactor Coolant Pumps Operating (Operating Power -75% Rated)	Two Reactor Coolant Pumps Operating in A Single Loop (Operating Power -46% Rated)	One Reactor Coolant Pump Operating in Each Loop (Operating Power -49% Rated)	Shutdown Bypass
1.	Nuclear Power Max. (% Rated)	105.5	105.5	105.5	105.5	5.0(3)
2.	Nuclear Power Max. Based on Flow (2) and Imbalance, (% Rated)	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	Bypassed
3.	Nuclear Power Max. Based on Pump Monitors, (%, Rated)	NA	NA	55% (5)(6)	55% (5)	Bypassed
4.	High Reactor Coolant System Pressure, psig, Max.	2355	2355	2355	2355	1720(4)
5.	Low Reactor Coolant System Pressure, psig, Min.	1985	1985	1985	1985	Bypassed
6.	Variable Low Reactor Coolant System Pressure psig, Min.	$(13.77\ T_{out} - 6181)$(1)	$(13.77\ T_{out} - 6181)$(1)	$(13.77\ T_{out} - 6181)$(1)	$(13.77\ T_{out} - 6181)$(1)	Bypassed
7.	Reactor Coolant Temp. F., Max.	619	619	619 (6)	619	619
8.	High Reactor Building Pressure, psig, Max.	4	4	4	4	4

(1) T_{out} is in degrees Fahrenheit (^{o}F).

(2) Reactor Coolant System Flow, %.

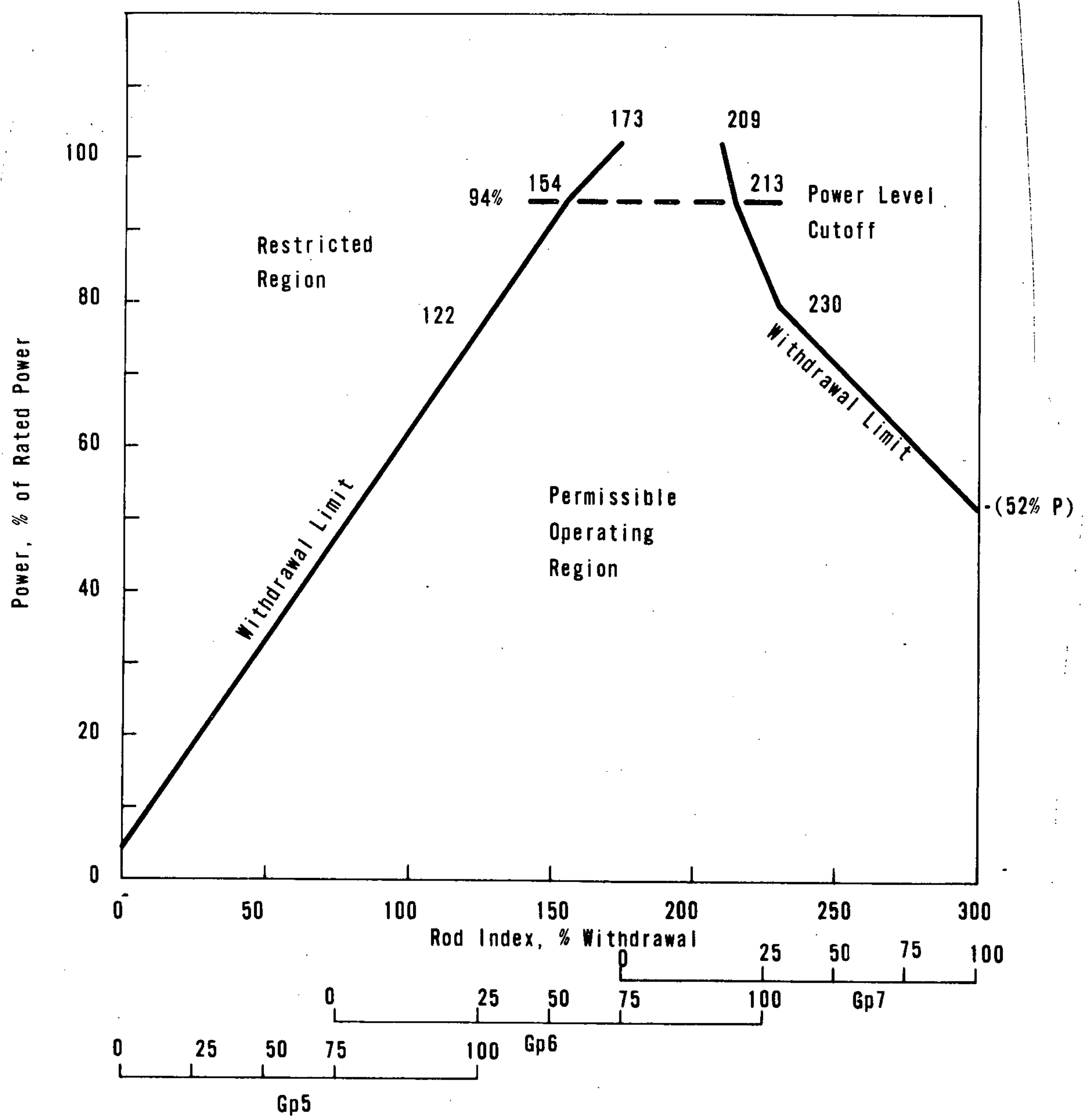
(3) Administratively controlled reduction set only during reactor shutdown.

(4) Automatically set when other segments of the RPS are bypassed.

(5) Reactor power level trip set point produced by pump contact monitor reset to 55.0%.

(6) Specification 3.1.8 applies. Trip one of the two protection channels receiving outlet temperature information from sensors in the idle loop.

1. Rod index is the percentage sum of the withdrawal of the operating groups.
2. The withdrawal limits are modified after 250 ± 5 full power days of operation.



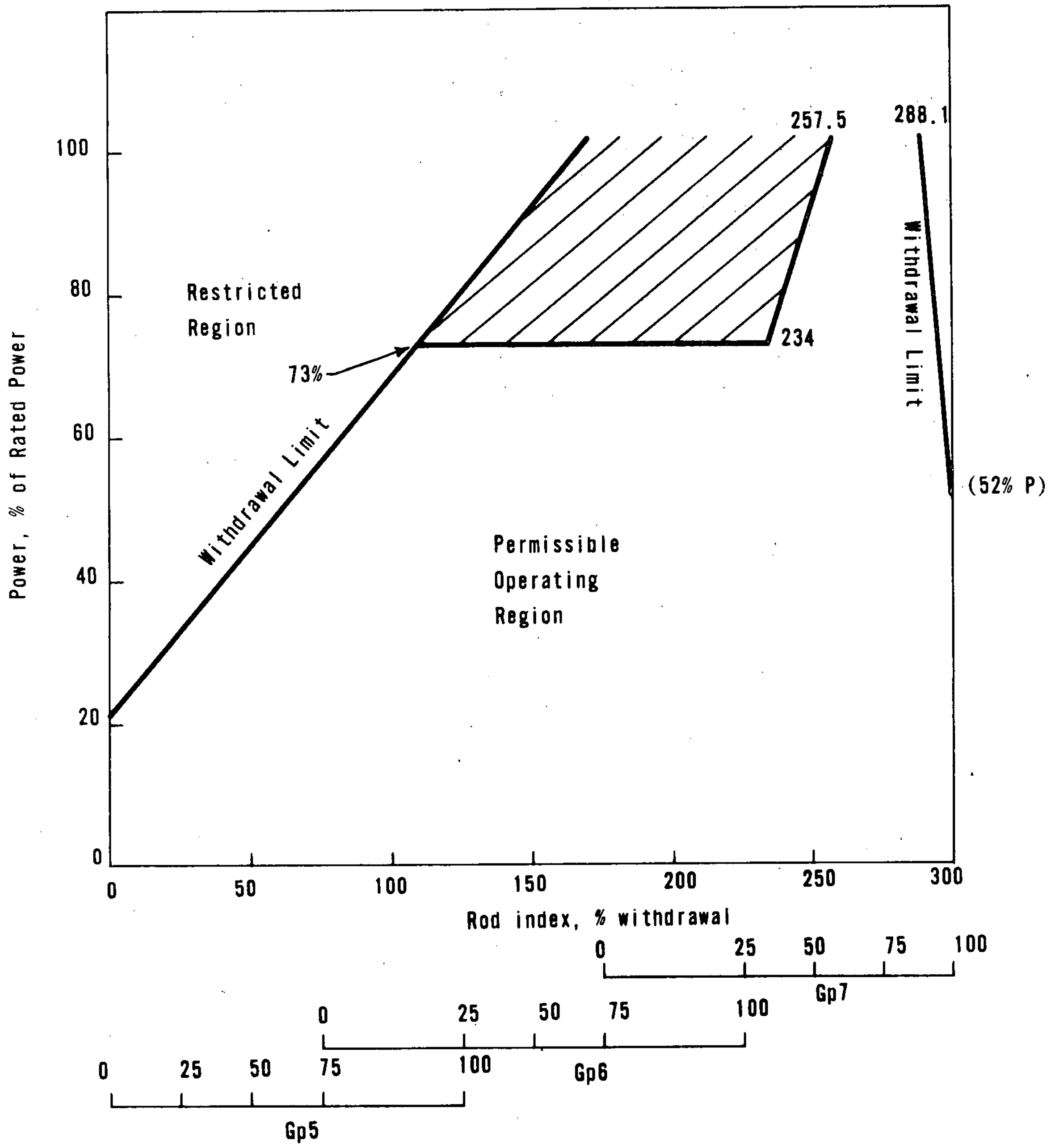
CONTROL ROD GROUP WITHDRAWAL LIMITS FOR 4 PUMP OPERATION

UNIT 1

OCONEE NUCLEAR STATION

Figure 3.5.2-1A1

1. Rod Index is the percentage sum of the withdrawal of the operating groups.
2. The withdrawal limits are in effect after 250 ± 5 full power days of operation. (The applicable power level cutoff is 100% power)

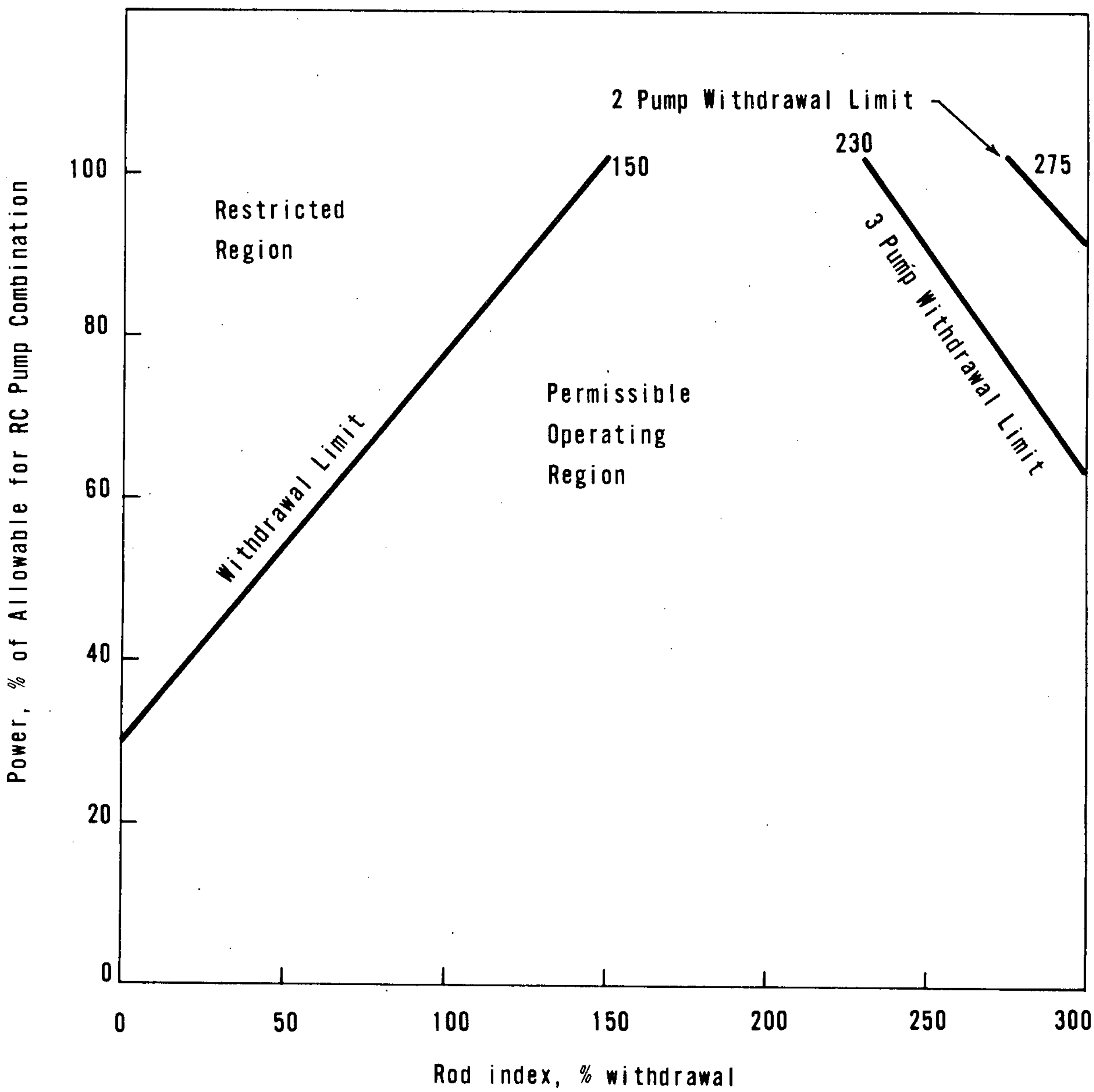


CONTROL ROD GROUP WITHDRAWAL LIMIT FOR
4 PUMP OPERATION

UNIT 1

OCONEE NUCLEAR STATION
Figure 3.5.2-1A2

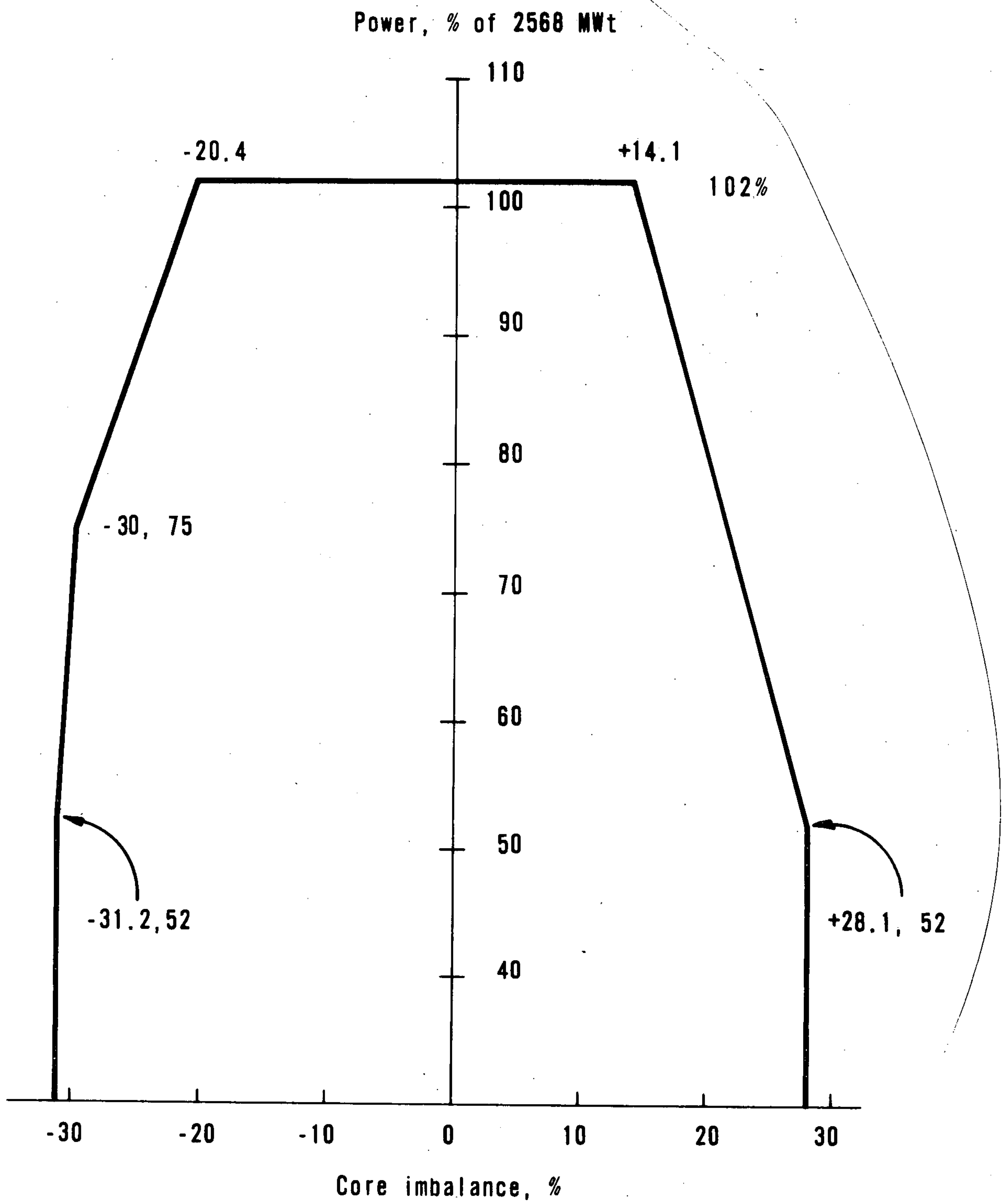
1. Rod index is the percentage sum of the withdrawal of the operating groups. (The applicable power level cutoff is 100% power)



CONTROL ROD GROUP WITHDRAWAL LIMITS FOR 3 AND 2 PUMP OPERATION

UNIT 1

Figure 3.5.2-2A

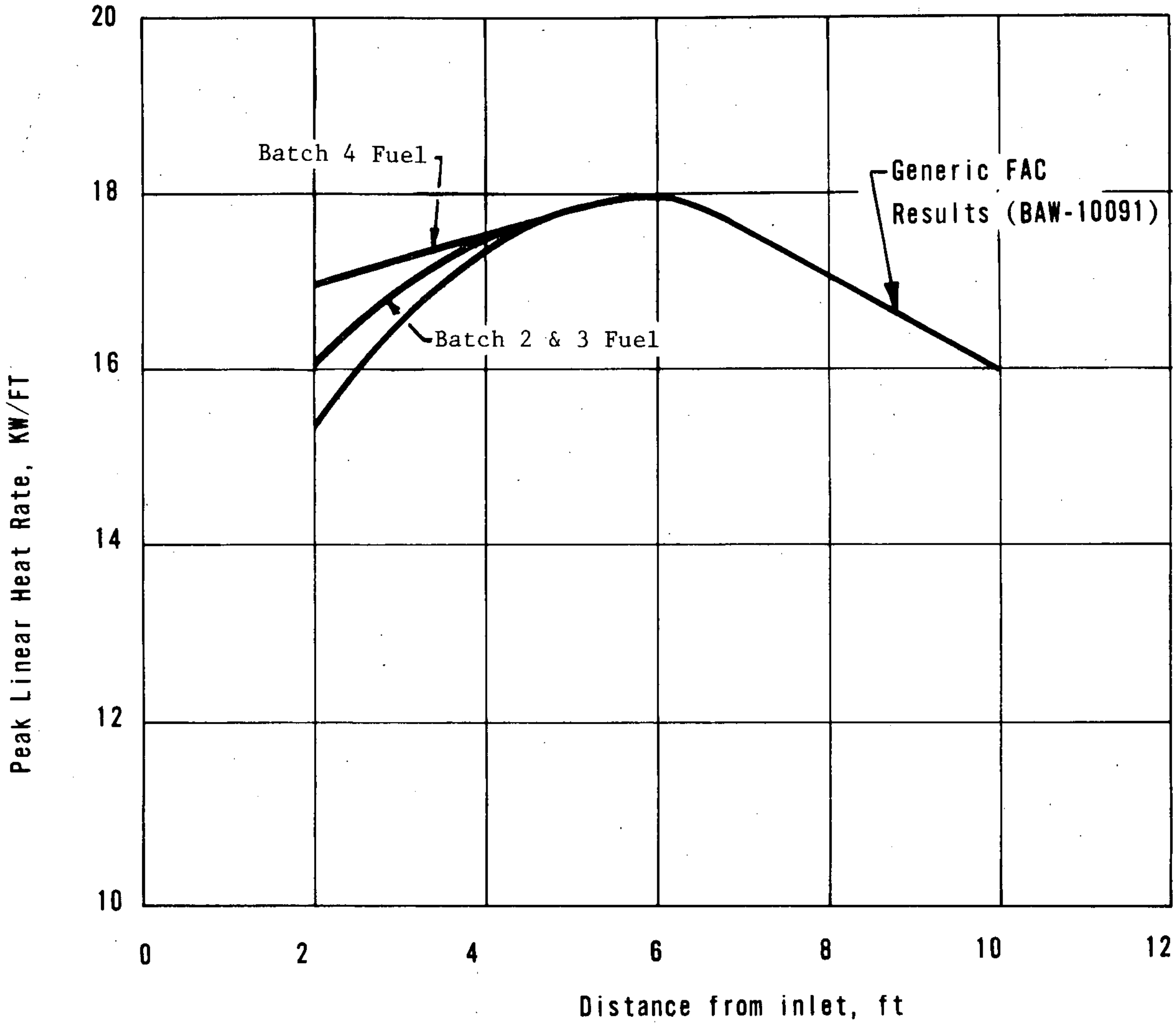


OPERATIONAL POWER IMBALANCE ENVELOPE

UNIT 1

OCONEE NUCLEAR STATION

Figure 3.5.2-3A



LOCA LIMITED MAXIMUM ALLOWABLE LINEAR HEAT RATE

OCONEE NUCLEAR STATION

Figure 3.5.2-4