<u>PROPOSED TECHNICAL SPECIFICATION CHANGES</u>

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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 when the reactor is critical.

Objective

To maintain the integrity of the fuel cladding.

Specification

2.1.1 The maximum local fuel pin centerline temperature shall be $\leq 5080 - (6.5 \times 10^{-3} \times (\text{Burnup, MWD/MTU})^{\circ}F)$ for TACO2 applications and $\leq 4642 - (5.8 \times 10^{-3} \times (\text{Burnup, MWD/MTU})^{\circ}F$ for TACO3 applications. Operation within this limit is ensured by compliance with the Axial Power Imbalance protective limits preserved by Table 2.3-1 "Reactor Protection System Trip Setting Limits," as specified in the COLR.

2.1.2 The departure from nucleate boiling ratio shall be maintained greater than the limits of 1.3 for the BAW-2 correlation and 1.18 for the BWC correlation. Operation within this limit is ensured by compliance with Specification 2.1.3 and with the Axial Power Imbalance protective limits preserved by Table 2.3-1 "Reactor Protection System Trip Setting Limits," as specified in the COLR.

2.1.3 Reactor Coolant System (RCS) core outlet temperature and pressure shall be maintained above and to the left of the Variable Low RCS Pressure-Temperature Protective Limits as specified in the COLR.

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 could 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 a critical heat flux (CHF) correlation. The BAW-2(1) and BWC(2) correlations have been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The BAW-2 correlation applies to Mark-B fuel and the BWC correlation applies to Mark-BZ fuel. 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 1.30 (BAW-2) and 1.18 (BWC).

A DNBR of 1.30 (BAW-2) or 1.18 (BWC) corresponds to a 95 percent probability at a 95 percent 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 for the allowable RC pump combination has been considered in determining the Variable Low RCS Pressure-Temperature Protective Limits.

The Variable Low RCS Pressure-Temperature Protective Limits presented in the COLR represent the conditions at which the DNBR is greater than or equal to the minimum allowable DNBR for the limiting combination of thermal power and number of operating reactor coolant pumps which is based on the nuclear power peaking factors (3) as specified in the COLR with potential fuel densification effects.

The Axial Power Imbalance Protective Limits in the COLR are based on the more restrictive of two thermal limits and include the effects of potential fuel densification:

1. The DNBR limit produced by the limiting combination of the radial peak, axial peak, and position of the axial peak.

2. The combination of radial and axial peak that prevents central fuel melting at the hot spot as given in the COLR.

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

The flow rates for the Variable Low RCS Pressure-Temperature Protective Limits specified in the COLR correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop.

The Variable Low RCS Pressure-Temperature Protective Limit for four reactor coolant pumps operating is the most restrictive of all possible reactor coolant pump maximum thermal power combinations as specified in the COLR. The Variable Low RCS Pressure-Temperature Protective Limits in the COLR represent the conditions at which the DNBR limit is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation. If the actual pressure/temperature point is below and to the right of the pressure/temperature line, the Variable Low RCS Pressure-Temperature Protective Limit is exceeded. The local quality at the point of minimum DNBR is less than 22 percent (BAW-2)(1) or 26 percent (BWC)(2).

Using a local quality limit of 22 percent (BAW-2) or 26 percent (BWC) at the point of minimum DNBR as a basis for less than four reactor coolant pumps operating of the Variable Low RCS Pressure-Temperature Protective Limits specified in the COLR is a conservative criterion even though the quality at the exit is higher than the quality at the point of minimum DNBR.

The DNBR as calculated by the BAW-2 or the BWC correlation continually increases from point of minimum DNBR, so that the exit DNBR is always higher and is a function of the pressure.

The maximum thermal power, as a function of reactor coolant pump operation is limited by the power level trip produced by the flux-flow ratio (percent flow x flux-flow ratio), plus the appropriate calibration and instrumentation errors.

For each combination of operating reactor coolant pumps of the Variable Low RCS Pressure-Temperature Protective Limits specified in the COLR, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 (BAW-2) or 1.18 (BWC) or a local quality at the point of minimum DNBR less than 22 percent (BAW-2) or 26 percent (BWC) for that particular reactor coolant pump combination. The Variable Low RCS Pressure-Temperature Protective Limit for four reactor coolant pumps operating is the most restrictive because any pressure-temperature point above and to the left of this curve will be above and to the left of the other curves.

REFERENCES

(1) Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water, BAW-10000A, May, 1976.

(2) BWC Correlation of Critical Heat Flux, BAW-10143P-A, April, 1985.

(3) FSAR, Section 3.2.3.1.1.c.

pumps(s). The pump monitors also restrict the power level for the number of pumps in operation.

C. RCS Pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure trip is reached before the nuclear overpower trip setpoint. The trip setting limit shown in Table 2.3-1 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.[2]

The low pressure (1800 psig) and variable low pressure (COLR) trip setpoint shown in Table 2.3-1 have been established to maintain the DNB ratio greater than or equal to the minimum allowable DNB ratio for those design accidents that result in a pressure reduction.[2,3]

To account for the calibration and instrumentation errors, the accident analysis used the protective limit specified in the COLR.

D. Coolant Outlet Temperature

The high reactor coolant outlet temperature trip setting limit (618F) shown in Table 2.3-1 has been established to prevent excessive core coolant temperatures in the operating range. Due to calibration and instrumentation errors, the safety analysis used a trip setpoint of 620 F.

E. 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 steam line failure in the reactor building or a loss-of-coolant accident, even in the absence of a low reactor coolant system pressure trip.

F. 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 2.3-1. Two conditions are imposed when the bypass is used:

1. A nuclear overpower trip set point of ≤5.0 percent of rated power is automatically imposed during reactor shutdown.

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

Table 2.3-1

Reactor Protection System Trip Setting Limits

	Four Reactor Coolant Pumps Operating (Nominal Operating Power - 100%)	Three Reactor Coolant Pumps Operating (Nominal Operating Power, 75%)	One Reactor Coolant Pump Operating in Each Loop[d] (Nominal Operating Power, 49%)	Shutdown Bypass
Nuclear power, % of rated, max	104.9	104.9	104.9	5.0[a]
Nuclear Power based on flow[b] and imbalance, % of rated, max	Protection System Maximum Allowable Setpoints for Axial Power Imbalance envelope in COLR	Protection System Maximum Allowable Setpoints for Axial Power Imbalance envelope in COLR	Protection System Maximum Allowable Setpoints for Axial Power Imbalance envelope in COLR	Bypassed
Nuclear Power based on pump monitors, % of rated, max[c]	NA	NA	55	Bypassed
High RC system pressure, psig, max	2355	2355	2355	1720[a]
Low RC system pressure, psig. min	1800	1800	1800	Bypassed
Variable low RC system pressure, psig, min	Specified in RCS Pressure-Temperature Protective Maximum Allowable Setpoints figure in COLR	Specified in RCS Pressure-Temperature Protective Maximum Allowable Setpoints figure in COLR	Specified in RCS Pressure-Temperature Protective Maximum Allowable Setpoints figure in COLR	Bypassed
RC temp, F, max	618	618	618	618
High reactor building pressure, psig, max	4(18.7 psia)	4(18.7 psia)	4(18.7 psia)	4(18.7 psia)

[a]Automatically set when other segments of the RPS (as specified) are bypassed.

[b]Reactor coolant system flow, %

[c]The pump monitors also produce a trip on (a) loss of two RC pumps in one RC loop, and (b) loss of one or two RC pumps during two-pump operation.

[d]Operation with one Reactor Coolant Pump operating in each loop is limited to 24 hrs. with the reactor critical.

6.12.3 CORE OPERATING LIMITS REPORT

6.12.3.1 The core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT prior to each reload cycle or prior to any remaining part of a reload cycle for the following Specifications:

2.1	Safety Limits, Reactor Core - Axial Power Imbalance protective limits and Variable Low RCS Pressure-Temperature Protective Limits
2.3.1	Reactor Protection System trip setting limits - Protection System Maximum Allowable Setpoints for Axial Power Imbalance and Variable low RC system pressure
3.1.8.3	Minimum Shutdown Margin for Low Power Physics Testing
3.5.2.1	Allowable Shutdown Margin limit during Power Operation
3.5.2.2	Allowable Shutdown Margin limit during Power Operation with inoperable control rods
3.5.2.4	Quadrant power Tilt limit
3.5.2.5	Control Rod and APSR position limits
3.5.2.6	Reactor Power Imbalance limits

6.12.3.2 The analytical methods used to determine the core operating limits addressed by the individual Technical Specification shall be those previously reviewed and approved by the NRC in Babcock & Wilcox Topical Report BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses" (the approved revision at the time the reload analyses are performed). The approved revision number shall be identified in the CORE OPERATING LIMITS REPORT.

6.12.3.3 The core operating limits shall be determined so that all applicable limits (e.g. fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.12.3.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance for each reload cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

MARKUP OF CURRENT ANO-1 TECHNICAL SPECIFICATIONS

(FOR INFO ONLY)

The pages would be revised as follows:

Remove	Insert
iv	iv
7	7
8	8
9	9
9a	--
9b	--
9c	--
13	13
14a	--
14b	--
15	15
142	142

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~~2.3-1~~	~~PROTECTIVE SYSTEM MAXIMUM ALLOWABLE SETPOINT~~	~~14a~~
~~2.3-2~~	~~PROTECTIVE SYSTEM MAXIMUM ALLOWABLE SETPOINTS~~	~~14b~~
3.1.2-1	REACTOR COOLANT SYSTEM HEATUP AND COOLDOWN LIMITATIONS	20a
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4.4.2-3	NORMALIZED LIFTOFF FORCE - VERTICAL TENDONS	85d
4.18.1	UPPER TUBE SHEET VIEW OF SPECIAL GROUPS PER SPECIFICATION 4.18.3.a.3	110o2
5.1-1	MAXIMUM AREA BOUNDARY FOR RADIOACTIVE RELEASE CALCULATION (EXCLUSION AREA)	111a
6.2-1	MANAGEMENT ORGANIZATION CHART	119
6.2-2	FUNCTIONAL ORGANIZATION FOR PLANT OPERATIONS	120

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 when the reactor is critical.

Objective

To maintain the integrity of the fuel cladding.

Specification

2.1.1 The maximum local fuel pin centerline temperature shall be $\leq 5080 - (6.5 \times 10^{-3} \times$ (Burnup, MWD/MTU)°F) for TACO2 applications and $\leq 4642 - (5.8 \times 10^{-3} \times$ (Burnup, MWD/MTU)°F for TACO3 applications. Operation within this limit is ensured by compliance with the Axial Power Imbalance protective limits preserved by Table 2.3-1 "Reactor Protection System Trip Setting Limits," as specified in the COLR.

2.1.2 The departure from nucleate boiling ratio shall be maintained greater than the limits of 1.3 for the BAW-2 correlation and 1.18 for the BWC correlation. Operation within this limit is ensured by compliance with Specification 2.1.3 and with the Axial Power Imbalance protective limits preserved by Table 2.3-1 "Reactor Protection System Trip Setting Limits," as specified in the COLR.

2.1.3 Reactor Coolant System (RCS) core outlet temperature and pressure shall be maintained above and to the left of the ~~Safety~~ Variable Low RCS Pressure-Temperature Protective Limits ~~shown~~ as specified in ~~Figure 2.1-1~~ the COLR.

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 could 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 a critical heat flux (CHF) correlation. The BAW-2(1) and BWC(2) correlations have been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The BAW-2 correlation applies to Mark-B fuel and the BWC correlation applies to Mark-BZ fuel. 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 1.30 (BAW-2) and 1.18 (BWC).

A DNBR of 1.30 (BAW-2) or 1.18 (BWC) corresponds to a 95 percent probability at a 95 percent 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 for the allowable RC pump combination has been considered in determining the ~~core protection safety~~ Variable Low RCS Pressure-Temperature Protective ~~l~~Limits.

The ~~curve~~ Variable Low RCS Pressure-Temperature Protective Limits presented in ~~Figure 2.1-1~~ the COLR represents the conditions at which the DNBR is greater than or equal to the minimum allowable DNBR for the limiting combination of thermal power and number of operating reactor coolant pumps~~. This curve~~ which is based on the ~~following~~ nuclear power peaking factors (3) as specified in the COLR with potential fuel densification effects~~:~~.

$$\cancel{F_q^N = 2.83; \quad F_{\Delta H}^N = 1.71; \quad F_z^N = 1.65.}$$

The Axial Power Imbalance Protective Limits in the COLR are based on the more restrictive of two thermal limits and include the effects of potential fuel densification:

1. The DNBR limit produced by ~~a nuclear power peaking factor of~~ $\cancel{F_q^N = 2.83}$ ~~or~~ the limiting combination of the radial peak, axial peak, and position of the axial peak. ~~that yields no less than the DNBR limit.~~

2. The combination of radial and axial peak that prevents central fuel melting at the hot spot as given in the COLR.

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

The flow rates for ~~curves 1, 2, and 3 of Figure 2.1-3~~ the Variable Low RCS Pressure-Temperature Protective Limits specified in the COLR correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop~~, respectively~~.

The ~~curve of Figure 2.1-1~~ Variable Low RCS Pressure-Temperature Protective Limit for four reactor coolant pumps operating is the most restrictive of all possible reactor coolant pump maximum thermal power combinations as specified ~~shown~~ in ~~Figure 2.1-3~~the COLR. The ~~curves of Figure 2.1-3~~ Variable Low RCS Pressure-Temperature Protective Limits in the COLR represent the conditions at which the DNBR limit is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation. If the actual pressure/temperature point is below and to the right of the pressure/temperature line, the ~~Safety~~ Variable Low RCS Pressure-Temperature Protective Limit is exceeded. The local quality at the point of minimum DNBR is less than 22 percent (BAW-2)(1) or 26 percent (BWC)(2).

Using a local quality limit of 22 percent (BAW-2) or 26 percent (BWC) at the point of minimum DNBR as a basis for ~~curves 2 and 3~~ less than four reactor coolant pumps operating of ~~Figure 2.1-3~~ the Variable Low RCS Pressure-Temperature Protective Limits specified in the COLR is a conservative criterion even though the quality at the exit is higher than the quality at the point of minimum DNBR.

The DNBR as calculated by the BAW-2 or the BWC correlation continually increases from point of minimum DNBR, so that the exit DNBR is always higher and is a function of the pressure.

The maximum thermal power, as a function of reactor coolant pump operation is limited by the power level trip produced by the flux-flow ratio (percent flow x flux-flow ratio), plus the appropriate calibration and instrumentation errors.

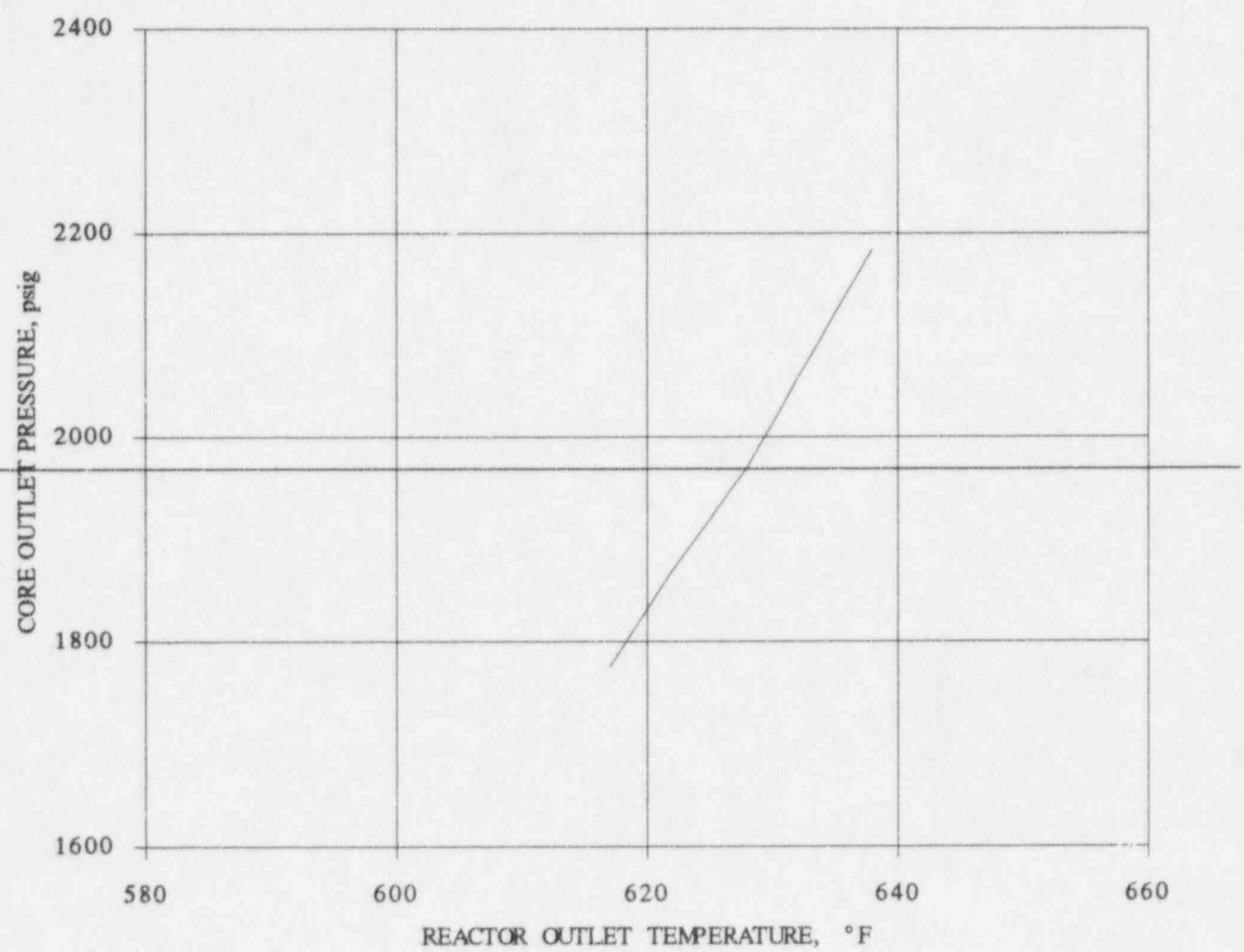
For each ~~curve~~ combination of operating reactor coolant pumps of ~~Figure 2.1-3~~ the Variable Low RCS Pressure-Temperature Protective Limits specified in the COLR, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 (BAW-2) or 1.18 (BWC) or a local quality at the point of minimum DNBR less than 22 percent (BAW-2) or 26 percent (BWC) for that particular reactor coolant pump ~~situation~~ combination. ~~Curve 1 of Figure 2.1-3~~ The Variable Low RCS Pressure-Temperature Protective Limit for four reactor coolant pumps operating is the most restrictive because any pressure-temperature point above and to the left of this curve will be above and to the left of the other curves.

REFERENCES

(1) Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water, BAW-10000A, May, 1976.

(2) BWC Correlation of Critical Heat Flux, BAW-10143P-A, April, 1985.

(3) FSAR, Section 3.2.3.1.1.c.



~~CORE PROTECTION SAFETY LIMIT~~ ~~FIG. NO. 2.1-1~~

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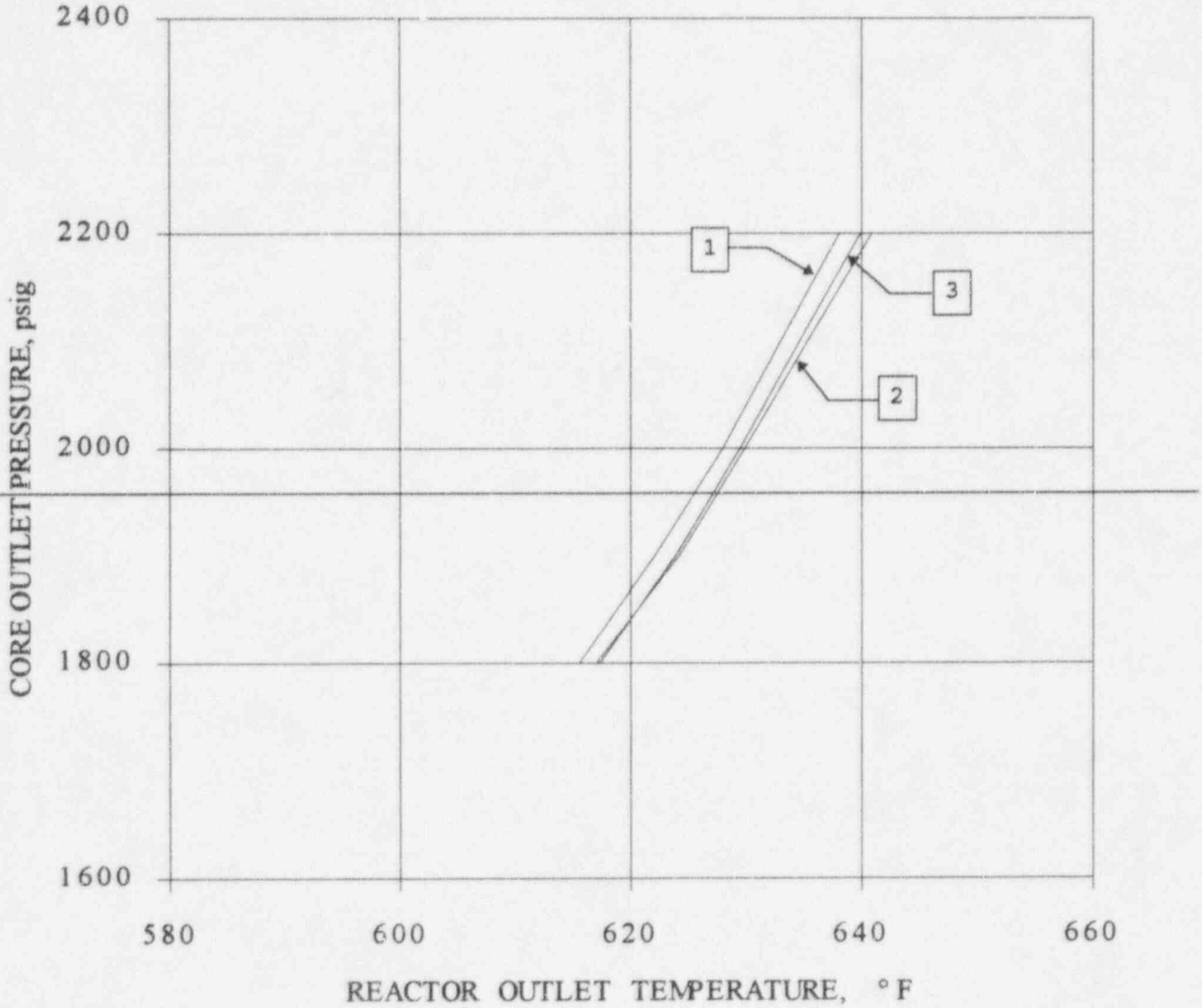
~~Core Protection Safety Limits - ANO 1~~

~~Figure 2.1-3~~



CURVE	GPM	POWER	PUMPS OPERATING (TYPE OF LIMIT)
1	374,880 (100%) *	112%	FOUR PUMPS (DNBR LIMIT)
2	280,035 (74.7%)	90.8%	THREE PUMPS (QUALITY LIMIT)
3	184,441 (49.2%)	63.7%	ONE PUMP IN EACH LOOP (QUALITY LIMIT)

* 106.5% OF DESIGN FLOW

pumps(s). The pump monitors also restrict the power level for the number of pumps in operation.

C. RCS Pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure trip is reached before the nuclear overpower trip setpoint. The trip setting limit shown in ~~Figure~~ Table 2.3-1 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.(2)

The low pressure (1800 psig) and variable low pressure ~~(13.89T~~$_{\text{out}}$ ~~-6766)~~ (COLR) trip setpoint shown~~,~~ in ~~Figure~~ Table 2.3-1 have been established to maintain the DNB ratio greater than or equal to the minimum allowable DNB ratio for those design accidents that result in a pressure reduction.(2,3)

To account for the calibration and instrumentation errors, the accident analysis used the ~~safety~~ protective limit ~~of Figure 2.1-1~~ specified in the COLR.

D. Coolant Outlet Temperature

The high reactor coolant outlet temperature trip setting limit (618F) shown in ~~Figure~~ Table 2.3-1 has been established to prevent excessive core coolant temperatures in the operating range. Due to calibration and instrumentation errors, the safety analysis used a trip setpoint of 620 F.

E. Reactor Building Pressure

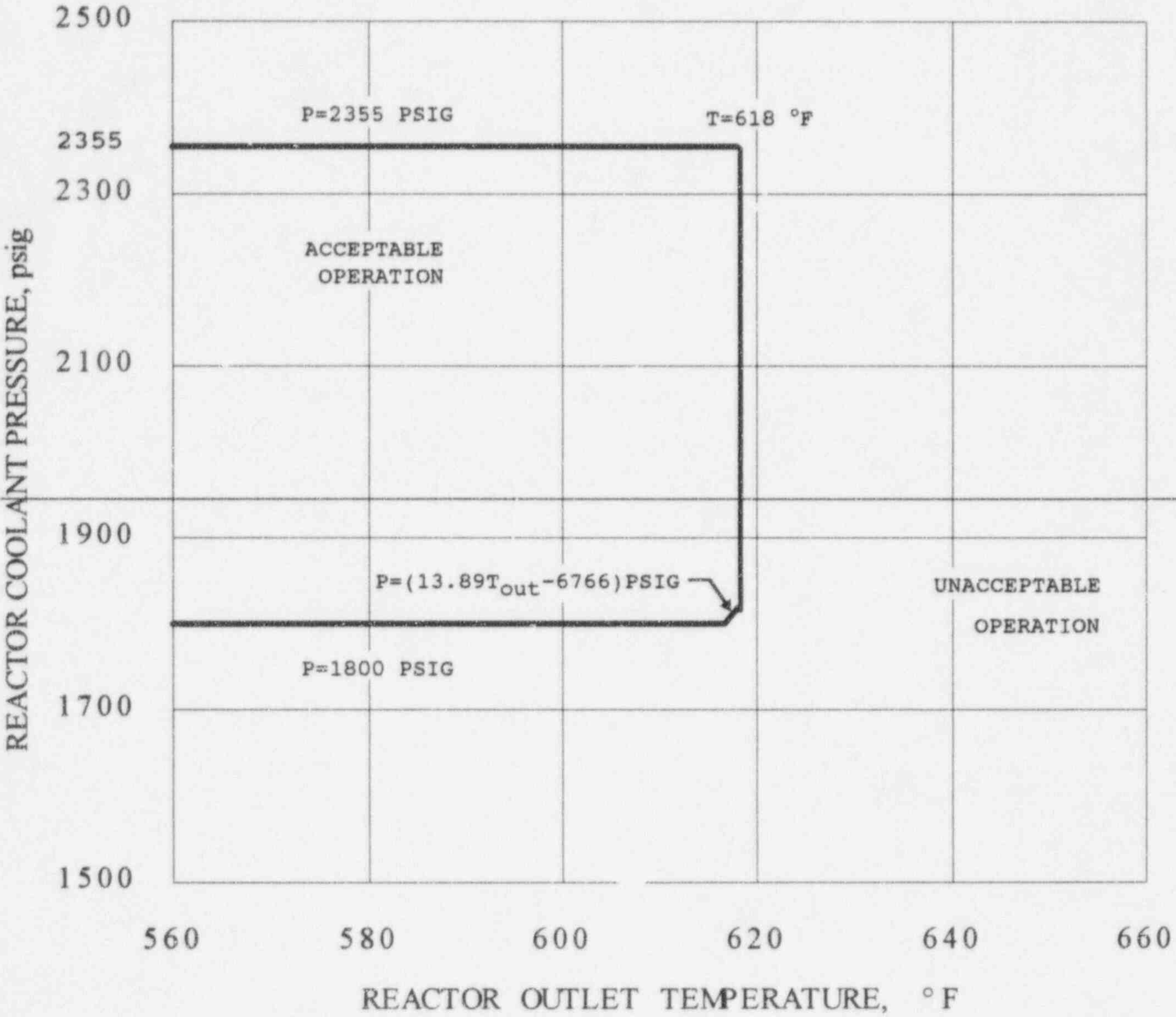
The high reactor building pressure trip setting limit (4 psig) provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the reactor building or a loss-of-coolant accident, even in the absence of a low reactor coolant system pressure trip.

F. 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 2.3-1. Two conditions are imposed when the bypass is used:

1. A nuclear overpower trip set point of ≤5.0 percent of rated power is automatically imposed during reactor shutdown.

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



~~PROTECTIVE SYSTEM MAXIMUM~~

~~ALLOWABLE SETPOINT~~

~~Figure 2.3-1~~

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Table 2.3-1

Reactor Protection System Trip Setting Limits

	Four Reactor Coolant Pumps Operating (Nominal Operating Power - 100%)	Three Reactor Coolant Pumps Operating (Nominal Operating Power, 75%)	One Reactor Coolant Pump Operating in Each Loop[~~e~~d] (Nominal Operating Power, 49%)	Shutdown Bypass
Nuclear power, % of rated, max	104.9	104.9	104.9	5.0[a]
Nuclear Power based on flow[b] and imbalance, % of rated, max	Protection System Maximum Allowable Setpoints for Axial Power Imbalance envelope in COLR	Protection System Maximum Allowable Setpoints for Axial Power Imbalance envelope in COLR	Protection System Maximum Allowable Setpoints for Axial Power Imbalance envelope in COLR	Bypassed
Nuclear Power based on pump monitors, % of rated, max[c]	NA	NA	55	Bypassed
High RC system pressure, psig, max	2355	2355	2355	1720[a]
Low RC system pressure, psig, min	1800	1800	1800	Bypassed
Variable low RC system pressure, psig, min	~~13.89 T_{out} -6766[d]~~ Specified in RCS P-T Prot. Max. All. Setpoints figure in COLR	~~13.89 T_{out} -6766[d]~~ Specified in RCS P-T Prot. Max. All. Setpoints figure in COLR	~~13.89 T_{out} -6766[d]~~ Specified in RCS P-T Prot. Max. All. Setpoints figure in COLR	Bypassed
RC temp, F, max	618	618	618	618
High reactor building pressure, psig, max	4(18.7 psia)	4(18.7 psia)	4(18.7 psia)	4(18.7 psia)

[a] Automatically set when other segments of the RPS (as specified) are bypassed.

[b] Reactor coolant system flow, %

[c] The pump monitors also produce a trip on (a) loss of two RC pumps in one RC loop, and (b) loss of one or two RC pumps during two-pump operation.

~~[d] T_{out} is in degrees Fahrenheit (F).~~

[~~e~~d] Operation with one Reactor Coolant Pump operating in each loop is limited to 24 hrs. with the reactor critical.

6.12.3 CORE OPERATING LIMITS REPORT

6.12.3.1 The core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT prior to each reload cycle or prior to any remaining part of a reload cycle for the following Specifications:

2.1	Safety Limits, Reactor Core - Axial Power Imbalance protective limits and Variable Low RCS Pressure-Temperature Protective Limits
2.3.1	Reactor Protection System trip setting limits - Protection System Maximum Allowable Setpoints for Axial Power Imbalance and Variable low RC system pressure
3.1.8.3	Minimum Shutdown Margin for Low Power Physics Testing
3.5.2.1	Allowable Shutdown Margin limit during Power Operation
3.5.2.2	Allowable Shutdown Margin limit during Power Operation with inoperable control rods
3.5.2.4	Quadrant power Tilt limit
3.5.2.5	Control Rod and APSR position limits
3.5.2.6	Reactor Power Imbalance limits

6.12.3.2 The analytical methods used to determine the core operating limits addressed by the individual Technical Specification shall be those previously reviewed and approved by the NRC in Babcock & Wilcox Topical Report BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses" (the approved revision at the time the reload analyses are performed). The approved revision number shall be identified in the CORE OPERATING LIMITS REPORT.

6.12.3.3 The core operating limits shall be determined so that all applicable limits (e.g. fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.12.3.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance for each reload cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

ANO-1 CYCLE 13 COLR

ENTERGY OPERATIONS
ARKANSAS NUCLEAR ONE - UNIT ONE
CYCLE 13
CORE OPERATING LIMITS REPORT

1.0 CORE OPERATING LIMITS

This Core Operating Limits Report for ANO-1 Cycle 13 has been prepared in accordance with the requirements of Technical Specification 6.12.3. The core operating limits have been developed using the methodology provided in the references.

The following cycle - specific core operating limits are included in this report:

1) Regulating control rod position setpoints,

2) Reactor power imbalance setpoints,

3) LOCA limited maximum allowable linear heat rate limits,

4) Axial power imbalance protective limits,

5) Protection system maximum allowable setpoints for axial power imbalance,

6) KW/ft limit for axial power imbalance protective limits,

7) Minimum shutdown margin,

8) Axial power shaping rod insertion limits,

9) Quadrant power tilt limits,

10) Variable Low RCS Pressure-Temperature (P-T) Protective Limits,

11) RCS Pressure-Temperature (P-T) Protective Maximum Allowable Setpoints,

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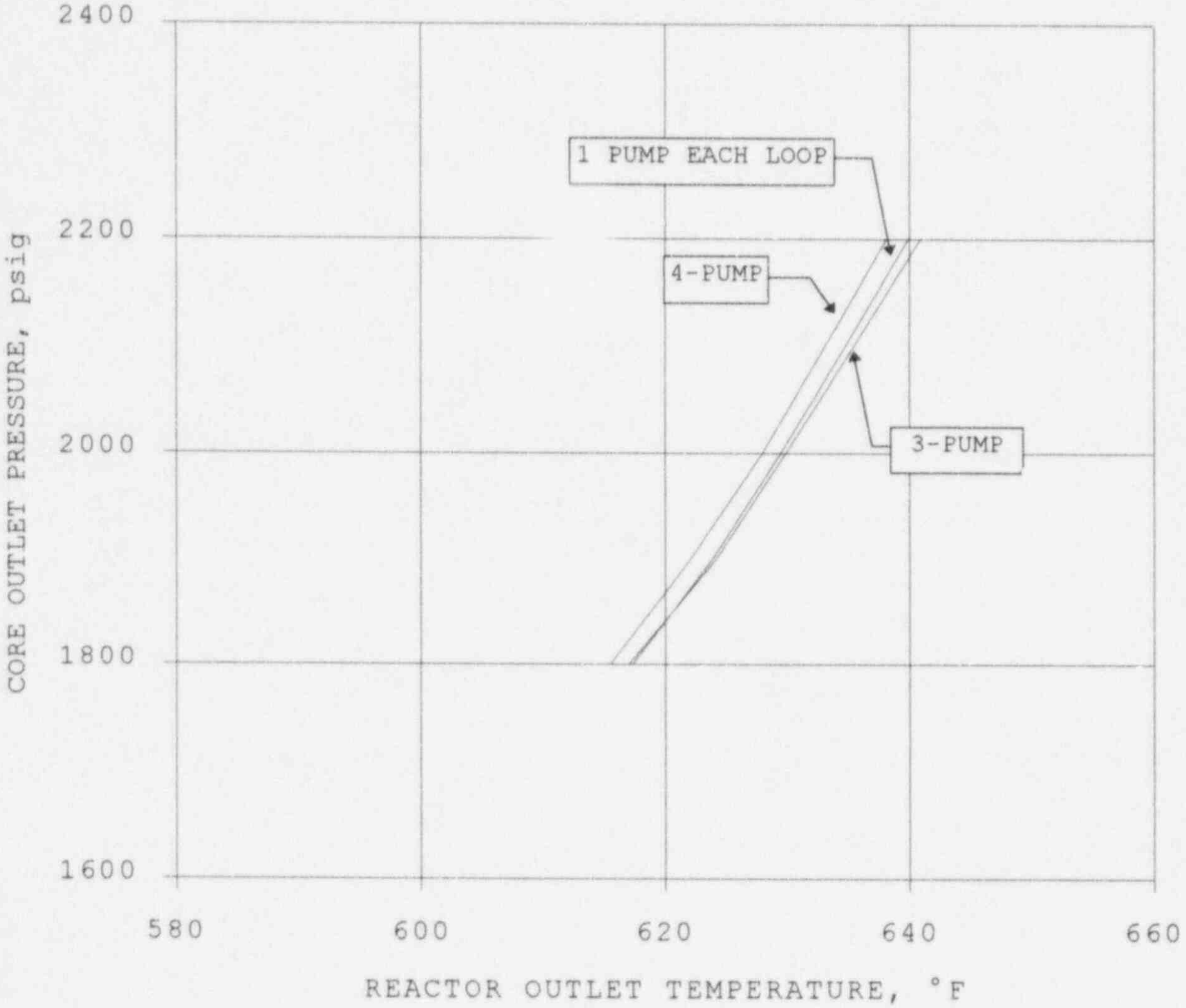
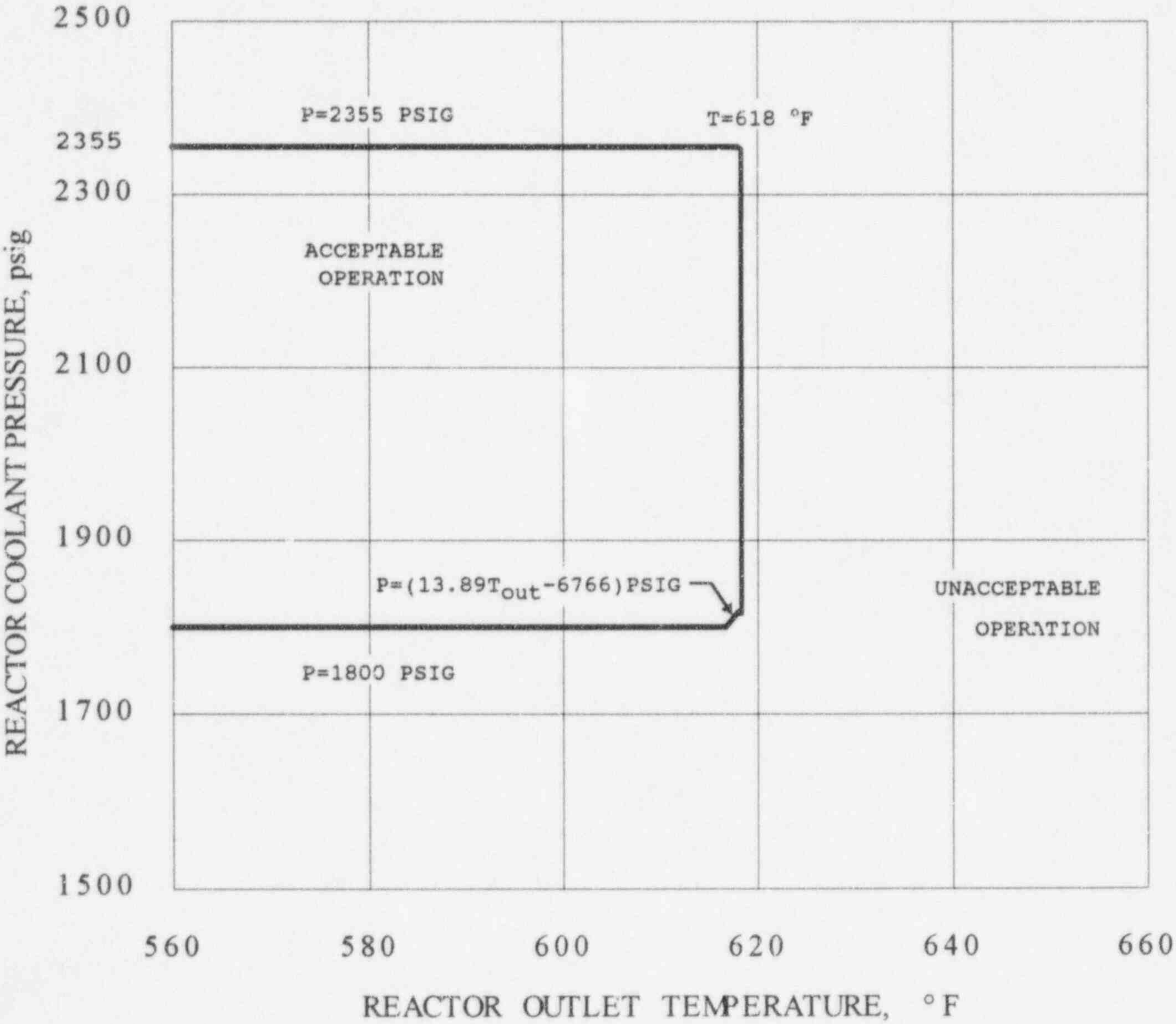


PUMPS OPERATING (TYPE OF LIMIT)	GPM	POWER
FOUR PUMPS (DNBR LIMIT)	374,880 (100%) *	112%
THREE PUMPS (QUALITY LIMIT)	280,035 (74.7%)	90.8%
ONE PUMP IN EACH LOOP (QUALITY LIMIT)	184,441 (49.2%)	63.7%

* 106.5% OF DESIGN FLOW

Figure is referred to by
Technical Specification Table 2.3-1

Figure 11 RCS Pressure-Temperature Protective Maximum Allowable Setpoints



Limit is referred to by
Technical Specificaton 2.1 Bases

Design Nuclear Power Peaking Factors

Maximum Radial	($F^N_{\Delta H}$)	1.71
Maximum Axial	(F^N_z)	1.65
Maximum Total	(F^N_q)	2.83