BAW-1607, Rev 1

April 1980

CRYSTAL RIVER UNIT 3

— Cycle 3 Reload Report —

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BABCOCK & WILCOX
Power Generation Group
Nuclear Power Generation Division
P. O. Box 1260
Lynchburg, Virginia 24505

Babcock & Wilcox

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1. INTRODUCTION AND SUMMARY

This report justifies the operation of Crystal River Unit 3 (cycle 3) at a rated core power of 2544 MWt. Included are the required analyses to support cycle 3 operation; these analyses employ analytical techniques and design bases established in reports that have received technical approval by the USNRC (see references).

The design for cycle 3 raises the rated thermal power from 2452 to 2544 MWt; the latter corresponds to the ultimate core power level identified in the Crystal River Unit 3 FSAR.[1] The upgraded power was analyzed for cycle 2[2], but the upgrade was not implemented and cycle 2 was operated at 2452 MWt; many of the analyses are again summarized in this report for completeness. Each accident analyzed in the FSAR has been reviewed, and each review is summarized in this report. Some accidents were re-analyzed to include the reactor coolant pump power monitors[3], which are being installed during the refueling outage. It is worthy of note that several other Babcock & Wilcox cores of the same design are licensed for 2568 MWt. The Technical Specifications have been reviewed, and the modifications required for cycle 3 are justified in this report.

Based on the analyses performed, which take into account the postulated effects of fuel densification and the Final Acceptance Criteria for emergency core cooling systems (ECCS), it has been concluded that Crystal River 3, cycle 3, can be safely operated at the rated core power level of 2544 MWt.

2. OPERATING HISTORY

Cycle 1 of the Crystal River Unit 3 nuclear generating plant was completed on April 23, 1979, after 440 EFPD at 2452 MWt. Cycle 2, which achieved criticality on July 29, 1979, was completed on February 26, 1980, after approximately 166.5 EFPD at the current rated power level of 2452 MWt. No operating anomalies have occurred during previous cycle operations that would adversely affect fuel performance in Cycle 3. 1

Cycle 3 is scheduled to start operation in May 1980 with an upgraded rated power level of 2544 MWt. The design cycle length is 335 EFPD.

3. GENERAL DESCRIPTION

3.1. Plant Description

3.1.1. Reactor Coolant System Stress

In support of the power upgrade, reactor coolant system (RCS) stresses were reviewed. Since the Crystal River 3 (CR-3) functional specification did not analyze power levels up to 2544 MWt, a new document was issued. The revised document was reviewed by the applicable engineering groups, and it was determined that no hardware changes were required; however, a revision was issued to the RCS Stress Report.

3.1.2. Reactor Coolant Pump Power Monitors

In support of the power upgrade, reactor coolant pump power monitors (RCPPMs) are being added to CR-3 during the EOC-2 refueling outage.[3]

The RCPPM anticipates a loss or reduction of the reactor coolant flow by monitoring RC pump power and detecting abnormal power conditions indicative of an inoperable pump. The status of each pump is transmitted by the RCPPM to each of four reactor protection system (RPS) channels. Two RCPPMs are supplied to provide redundant pump status information to each RPS channel. Logic in the RPS will act on the pump status information and take appropriate action as follows:

1. With three or four RC pumps operating, no action is taken by the RCPPM. Reactor protection is provided by the nuclear overpower based on the RCS flow and axial power imbalance unit of the RPS.
2. With two or fewer RC pumps operating, the RCPPM trips the reactor.

As stated in the accident analyses of the CR-3 FSAR, in the event of a loss of reactor coolant flow due to failure of one or more of the RC pumps at the present licensed power level of 2452 MWt, the transient is terminated by the present RPS flux-flow trip. The present RPS action is quick enough to prevent the minimum DNBR going below 1.30 for the four-pump coastdown transient and below 1.00 for the locked-rotor transient.

However, at thermal power levels above 2500 MWt, RPS action by the flux-flow comparator is not fast enough — in the event of loss of more than one RC pump — to keep the minimun DNBR from going below the acceptance criterion. Therefore, for power levels above 2500 MWt, nuclear overpower based on RCPPMs must be added to the RPS trip functions to reduce the response time of the RPS and thereby terminate the transient quickly enough to ensure compliance with the minimum DNBR limits.

Each RCPPM string includes two current transformers and two potential transformers to measure the current and voltage on the RCP power feed lines. The transformers provide input to an electronic watt transducer, which produces an output signal proportional to real power. This power signal is fed into a bistable, which provides a contact output for selected overpower and underpower setpoints. The bistable output contact actuates four separate relays. A contact from each relay is wired to its respective RPS channel. Thus, one pump monitor string provides status information for one pump to each of four RPS channels. An identical redundant string using separate transformers and monitoring equipment again provides status information for the same pump to the four RPS channels. In the event of failure of one string, all four RPS channels would still have the necessary pump status information via the redundant string.

The complete RCPPM system is constructed so that equipment belonging to redundant strings is placed inside enclosures separated by barriers. Contact outputs from the RCPPM cabinets to the four RPS channels are arranged to provide adequate physical separation and electrical isolation of each channel. External signal cable and equipment separation for this installation complies with IEEE 384-1977 and Regulatory Guide 1.75. Where separation cannot be maintained, physical barriers are included.

RCPPM cabinets and equipment specified are seismically qualified and located in a Class I structure. All supports for engineered safeguards cable trays and conduits are designed for OBE and SSE using the acceleration floor response spectra developed for applicable levels of the containment building, auxiliary building, intermediate building, and control complex.

The current and potential transformers are not seismically qualified. However, separation of the cables carrying redundant transformer outputs to the RCPPM cabinets is provided in accordance with the separation criteria stated above. The current and potential transformers are not seismically qualified because they are not required to safely shutdown the reactor. The loss of the current or potential transformers would result in a "pump inoperable" signal to the RPS. Upon receipt of two such signals, whatever the cause, the RPS trips the reactor.

3.2. Core Description

The CR-3 reactor core is described in detail in Chapter 3 of the Final Safety Analysis Report for the unit.[1] The cycle 3 core consists of 177 fuel assemblies (FAs), each of which is a 15-by-15 array containing 208 fuel rods; 16 control rod guide tubes; and one incore instrument guide tube. The fuel assemblies in batches 2, 3, and 5 have an average nominal fuel loading of 463.6 kg of uranium, whereas the batch 4 assemblies maintain an average nominal fuel loading of 468.6 kg of uranium. The cladding is cold-worked Zircaloy-4 with an OD of 0.430 inch and a wall thickness of 0.0265 inch. The fuel consists of dished-end, cylindrical pellets of uranium dioxide (see Table 4-2 for data).

Figure 3-1 is the core loading diagram for cycle 3 of Crystal River 3. The initial enrichments of batches 2, 3, and 4 were 2.54, 2.83, and 2.64 wt % uranium-235, respectively. Fifty-two of the batch 2 assemblies will be discharged at the end of cycle 2. The batch 5 design enrichment is 2.62 wt % uranium-235. Batches 3 and 4 and the remaining batch 2 assemblies will be shuffled to new locations. The batch 5 assemblies will occupy the periphery of the core. Figure 3-2 is an eighth-core map showing the burnup of each assembly at the beginning of cycle 3 and its initial enrichment.

Core reactivity will be controlled by 61 full-length Ag-In-Cd control rod assemblies (CRAs) and soluble boron shim. In addition to the full-length CRAs, eight axial power shaping rods (APSRs) are provided for additional control of the axial power distribution. The cycle 3 locations of the 69 control rods and the group designations are unchanged from cycle 2 and are shown in Figure 3-3. Control rod group 7 will be withdrawn at 250 ± 10 EFPD of operation.

Figure 3-1. Core Loading Diagram for Crystal River 3, Cycle 3

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						5	5	5	5	5					
B				5	5	5	F7 3	C9 3	F9 3	5	5	5			
C			O13 4	5	D7 3	N3 4	L1 4	P8 2	L15 4	N13 4	D9 3	5	O3 4		
D		5	5	C7 3	N2 4	M2 4	D5 3	R8 4	D11 3	M14 4	N14 4	G13 3	5	5	
E		5	G4 3	B12 4	F6 2	K5 3	K1 4	L8 3	K15 4	K11 3	F10 2	B4 4	G12 3	5	
F	5	5	C12 4	B11 4	E9 3	E5 3	D6 3	B10 4	D10 3	E11 3	E7 3	B5 4	C4 4	5	5
G	5	G6 3	A10 4	E4 3	A9 4	F4 3	B6 4	D8 3	F14 4	F12 3	A7 4	E12 3	A6 4	G10 3	5
H	5	G3 3	H14 2	H15 4	H10 3	F2 4	H4 3	H8 2	H12 3	L14 4	H6 3	H1 4	H2 2	K13 3	5
K	5	K6 3	R10 4	M4 3	R9 4	L4 3	L2 4	N8 3	P10 4	L12 3	R7 4	M12 3	R6 4	K10 3	5
L	5	5	O12 4	P11 4	M9 3	M5 3	N6 3	P6 4	N10 3	M11 3	M7 3	P5 4	O4 4	5	5
M		5	K4 3	P12 4	L6 2	G5 3	G1 4	F8 3	G15 4	G11 3	L10 2	P4 4	K12 3	5	
N		5	5	K3 3	D2 4	E2 4	N5 3	A8 4	N11 3	E14 4	D14 4	O9 3	5	5	
O			C13 4	5	N7 3	D3 4	F1 4	B8 2	F15 4	D13 4	N9 3	5	C3 4		
P				5	5	5	L7 3	O7 3	L9 3	5	5	5			
R						5	5	5	5	5					

XX Cycle 2 Location
Y Batch Number

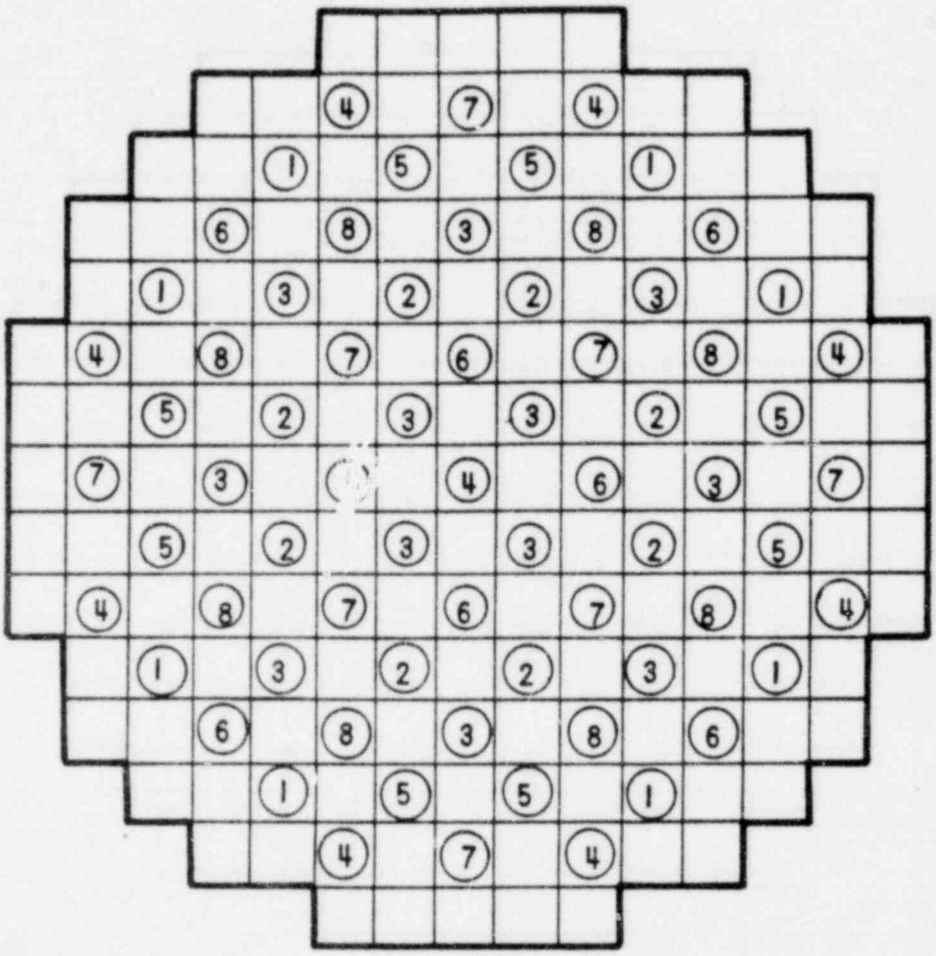
Figure 3-2 Enrichment and Burnup Distribution for Crystal River 3, Cycle 3

	8	9	10	11	12	13	14	15
H	2.54 17,015	2.83 16,512	2.64 5,691	2.83 19,762	2.64 3,085	2.54 16,741	2.83 15,949	2.62 0
K		2.64 5,690	2.83 12,950	2.64 3,452	2.83 17,364	2.64 3,206	2.83 15,308	2.62 0
L			2.83 14,095	2.83 13,590	2.64 4,923	2.64 6,051	2.62 0	2.62 0
M				2.54 17,460	2.64 3,639	2.83 17,466	2.62 0	
N					2.83 15,950	2.62 0	2.62 0	
O						2.64 4,231		
P								
R								

1

x.xx	Initial Enrichment
xx,xxx	BOC Burnup, MWd/mtU

Figure 3-3. Control Rod Locations



GROUP	NUMBER OF RODS	FUNCTION
1	8	SAFETY
2	8	SAFETY
3	12	SAFETY
4	9	SAFETY
5	8	CONTROL
6	8	CONTROL
7	8	CONTROL
8	8	APSRs
TOTAL	69	

4. FUEL SYSTEM DESIGN

4.1. Fuel Assembly Mechanical Design

The types of fuel assemblies and pertinent fuel design parameters for Crystal River Unit 3, cycle 3 are listed in Table 4-1. Two assemblies will contain primary neutron sources (PNS), and two assemblies will contain regenerative neutron sources (RNS) in cycle 3. The justification for the design and use of the retainers described in reference 4 is applicable to RNS and PNS retainers in the CR-3, cycle 3 fuel. All fuel assemblies are identical in concept and are mechanically interchangeable. All other results presented in the Crystal River 3 Cycle 2 Reload Report[2] are applicable to the reload fuel assemblies.

4.2. Fuel Rod Design

The fuel pellet end configuration has changed from a spherical dish for batches 1 through 4 to a truncated cone dish for batch 5; this minor change facilitates manufacturing. The mechanical evaluation of the fuel rod is discussed below.

4.2.1. Cladding Collapse

Creep collapse analyses were performed for three-cycle assembly power histories for Crystal River 3. Batches 2 and 3 are more limiting than batches 4 and 5 due to their previous incore exposure time. A batch 3 fuel assembly was determined to have the most limiting power history and was, therefore, analyzed for creep collapse.

The limiting power history was used to calculate the fast neutron flux level for the energy range >1 MeV. The collapse time for the most limiting assembly was conservatively determined to be greater than the three-cycle design life.

The collapse times reported in Table 4-1 are based on the procedures set forth in references 5 and 6.

4.2.2. Cladding Stress

The batch 2 and 3 reinserted fuel assemblies are the limiting batches from a cladding stress point of view because of their lower density and longer previous exposure time. Batches 2 and 3 have been analyzed and documented in the Crystal River Unit 3 Fuel Densification Report.[7]

4.2.3. Cladding Strain

The fuel design criteria specify a limit of 1.0% on cladding plastic circumferential strain. The pellet design is established for cladding plastic strain of less than 1% at values of maximum design pellet burnup and heat generation rate, which are considerably higher than the values the CR-3 fuel is expected to be. The strain analysis is also based on the maximum specification tolerance for the cladding ID.

4.3. Fuel Thermal Design

All fuel assemblies in this core are thermally similar. The fresh batch 5 fuel inserted for cycle 3 operation introduces no significant differences in fuel thermal performance relative to the fuel remaining in the core. The design minimum linear heat rate (LHR) capability and the average fuel temperature for each batch in cycle 3 are shown in Table 4-2. LHR capabilities are based on centerline fuel melt and were established using the TAFY-3 code with fuel densification to 96.5% of theoretical density.[18]

4.4. Operating Experience

Babcock & Wilcox operating experience with the Mark B 15 × 15 fuel assembly has verified the adequacy of its design. As of March 31, 1980, the following experience has been accumulated for the eight operating B&W 177-fuel assembly plants using the Mark-B fuel assembly: 1

Reactor	Current cycle	Maximum assembly burnup, MWd/mtU[(a)] Incore	Discharged	Cumulative net electrical output, MWh[(b)]	
Oconee 1	6	19,600	40,000	29,231,499	
Oconee 2	5	23,400	33,700	25,163,758	
Oconee 3	5	26,300	29,400	24,496,556	
TMI-1	4	32,400	32,200	23,840,053	
ANO-1	4	25,100	33,222	22,634,036	
Rancho Seco	3	37,729	29,378	20,110,890	1
Crystal River 3	2	23,194	23,194	10,391,640	
Davis Besse 1	1	14,600	--	6,170,578	

[(a)] As of March 31, 1980.

[(b)] As of December 31, 1979.

Table 4-1. Fuel Design Parameters and Dimensions

	Batch 2	3	4	5	
Fuel assembly type	Mark B-3	Mark B-3	Mark B-4	Mark B-4	
Number of assemblies	9	60	56	52	
Fuel rod OD, in.	0.430	0.430	0.430	0.430	
Fuel rod ID, in.	0.377	0.377	0.377	0.377	
Flexible spacers, type	Corrugated	Corrugated	Spring	Spring	
Rigid spacers, type	Ceramic	Ceramic	Zirc-4	Zirc-4	
Undensified active fuel length, in.	144	144	143.6	141.8	
Fuel pellet (mean specified), in.	0.370	0.370	0.3697	0.3686	
Fuel pellet initial density (mean specified), % TD	92.5	92.5	94.0	95.0	
Initial fuel enrichment, wt % ^{235}U	2.54	2.83	2.64	2.62	
Estimated residence time, EFPH	22,728	22,728	19,464	22,920	1
Cladding collapse time, EFPH	>25,000	>25,000	>30,000	>30,000	

Table 4-2. Fuel Thermal Analysis Parameters

	Batches 2/3	Batch 4	Batch 5
No. of assemblies	9/60	56	52
Nominal pellet density, % TD	92.5	94.0	95.0
Pellet diameter, in.	0.370	0.3697	0.3686
Stack height, in.	144.0	143.6	141.8
Densified Fuel Parameters[a]			
Pellet diameter, in.	0.3641	0.3648	0.3649
Fuel stack height, in.	141.1	141.8	140.74
Nominal LHR at 2568 MWt, kW/ft	5.77	5.74	5.79
Avg fuel temperature at nominal LHR, F	1330	1280	1310
LHR to centerline fuel melt, kW/ft	19.7	20.1	20.1
Core average densified LHR at 2544 MWt is 5.71 kW/ft			

[a] Densification to 96.5% TD assumed.

5. NUCLEAR DESIGN

5.1. Physics Characteristics

Table 5-1 compares the core physics parameters of cycles 2 and 3; these values were generated using PDQ07[8] for both cycles. Since the core has not yet reached an equilibrium cycle, differences in core physics parameters are expected between cycles. The longer cycle 3 will produce a larger cycle differential burnup than cycle 2. The accumulated average core burnup will be higher in cycle 3 than in cycle 2 because of the presence of the once-burned batch 4 and twice-burned batch 2 and 3 fuel. Figure 5-1 illustrates a representative relative power distribution for the beginning of the third cycle at full power with equilibrium xenon and normal rod positions.

The critical boron concentrations for cycle 3 are given in Table 5-1. Control rod worths are sufficient to attain the required shutdown margin as indicated in Table 5-2. The hot full power control rod worths vary little between cycles 2 and 3. The ejected rod worths for cycle 3 are similar to those in cycle 2 |1
for the same number of regulating banks inserted; however, values between cycles are difficult to compare since the isotopic distributions are different. Calculated ejected rod worths and their adherence to criteria are considered at all times in life and at all power levels in the development of the rod insertion limits presented in section 8. The maximum stuck rod worths for cycle 3 are less than those for cycle 2. The adequacy of the shutdown margin with |1
cycle 3 stuck rod worths is demonstrated in Table 5-2. The following conservatisms were applied for the shutdown calculations:

1. Poison material depletion allowance.
2. 10% uncertainty on net rod worth.
3. Flux redistribution penalty.

Flux redistribution was accounted for since the shutdown analysis was calculated using a two-dimensional model. The shutdown calculation at the end of cycle 3 was analyzed at 250 EFPD. This is the latest time (± 10 EFPD) in core life at which the transient bank is nearly fully inserted. After 250 EFPD, the transient bank will be almost fully withdrawn, thus, the available shutdown margin will be increased.

The cycle 3 power deficit from hot zero power to hot full power is identical to the cycle 2 deficit at BOC, but slightly more negative than the cycle 2 deficit at EOC. The Doppler coefficients and xenon worths are similar for the two cycles. The differential boron worths are similar for cycles 2 and 3. The effective delayed neutron fractions for both cycles show a decrease with burnup.

5.2. Changes in Nuclear Design

There is no major change between the designs of cycle 2 and cycle 3; the upgrading of the core power level to 2544 MWt was considered in cycle 2 design and will be implemented in cycle 3. The same calculational methods and design information were used to obtain the important nuclear design parameters. No significant operational or procedural changes exist with regard to axial or radial power shape control, xenon control, or tilt control. The operational limits and RPS limits (Technical Specification changes) for cycle 3 are presented in section 8.

Table 5-1. Physics Parameters, Crystal River Three, Cycle 3

	Cycle 2	Cycle 3[a]	
Design cycle length, EFPD	275	335	
Design cycle burnup, MWd/mtU	8,500	10,345	
Design average core burnup — EOC, MWd/mtU	17,364	17,922	
Design initial core loading, mtU	82.3	82.3	
Critical boron — BOC, ppm (no Xe)			
HZP[b], group 8 (37.5% wd)	1,260	1,430	
HZP, groups 7 and 8 inserted	1,185	1,351	
HFP[b], groups 7 and 8 inserted	991	1,185	
Critical boron — EOC, ppm (eq Xe)			
HZP, group 8 (37.5% wd)	305	287	
HFP, group 8 (37.5% wd)	51	27	
Control rod worths — HFP, BOC, %Δk/k			
Group 6	1.02	1.09	
Group 7	0.85	0.83	
Group 8 (37.5% wd)	0.49	0.48	
Control rod worths — HFP, EOC, %Δk/k			
Group 7	1.11[c]	1.07[d]	
Group 8 (37.5% wd)	0.48[c]	0.48[d]	
Max ejected rod worth[e] — HZP, %Δk/k			
BOC (N-12)	0.55	0.53	
EOC (N-12)	0.50[c]	0.59[d]	1
Max stuck rod worth — HZP, %Δk/k			
BOC (N-12)	1.82	1.76	
EOC (L-14)	1.88[c]	1.84[d]	
Power deficit, HZP to HFP, %Δk/k			
BOC	-1.30	-1.30	
EOC	-2.06	-2.12	
Doppler coeff — BOC, 10^{-5} (Δk/k/°F)			
100% power (0 Xe)	-1.50	-1.52	
Doppler coeff — EOC, 10^{-5} (Δk/k/°F)			
100% power (eq Xe)	-1.58	-1.61	
Moderator coeff — HFP, 10^{-4} (Δk/k/°F)			
BOC (0 Xe, critical ppm, group 8 inserted)	-0.65	-0.30	
EOC (eq Xe, 17 ppm, group 8 inserted)	-2.52	-2.63	
Boron worth — HFP, ppm/%Δk/k			
BOC	106	108	
EOC	94	94	
Xenon worth — HFP, %Δk/k			
BOC (4 EFPD)	2.67	2.63	
EOC (equilibrium)	2.74	2.74	

Table 5-1. (Cont'd)

	Cycle 2	Cycle 3	
Effective delayed neutron fraction — HFP			
BOC	0.00584	0.00597	1
EOC	0.00516	0.00519	

(a) Cycle 3 data are for the conditions stated in this report; the cycle 2 values given are at the core conditions identified in reference 2.

(b) HZP denotes hot zero power (532F T_{avg}); HFP denotes hot full power (579F T_{avg}).

(c) Rod worths for EOC-2 are calculated at 225 EFPD, the latest time in core life in which the transient bank is nearly full-in.

(d) Rod worths for EOC-3 are calculated at 250 EFPD, the latest time in core life in which transient bank is nearly full-in.

(e) Ejected rod worth for groups 5 thorugh 8 inserted.

Table 5-2. Shutdown Margin Calculation for Crystal River 3, Cycle 3

	BOC, %Δk/k	EOC[a], %Δk/k
Available Rod Worth		
Total rod worth, HZP[b]	9.37	9.29
Worth reduction due to burnup of poison material	-0.37	-0.42
Maximum stuck rod worth, HZP	-1.76	-1.84
Net worth	7.24	7.03
Less 10% uncertainty	-0.72	-0.70
Total available worth	6.52	6.33
Required Rod Worth		
Power deficit, HFP to HZP	1.30	2.08
Max allowable inserted rod worth	1.06	1.36
Flux redistribution	0.53	1.02
Total required worth	2.89	4.46
Shutdown Margin		
Total available minus total required	3.63	1.87

1

Note: Required shutdown margin is 1.00% Δk/k.

[a] For shutdown margin calculations, this is defined as ∿250 EFPD, the latest time in core life in which the transient bank is nearly full-in.

[b] HZP: hot zero power, HFP: hot full power.

Figure 5-1. BOC (4 EFPD), Cycle 3 Two-Dimensional Relative Power Distribution — HFP, Equilibrium Xenon, Banks 7 and 8 Inserted

	8	9	10	11	12	13	14	15
H	1.09	1.14	1.31	1.17	1.34	0.94	7 0.46	0.50
K		1.27	1.14	1.31	1.16	1.22	0.82	0.58
L			7 0.67	1.04	8 1.15	1.25	1.08	0.54
M				1.01	1.27	1.04	0.89	
N					1.10	1.08	0.61	
O						0.66		
P								
R								

1

x Inserted rod group No.

x.xx Relative power density

6. THERMAL-HYDRAULIC DESIGN

6.1. DNBR Evaluations

Crystal River 3 will be upgraded in power for cycle 3 operation from 2452 to 2544 MWt rated core power. Thermal-hydraulic design calculations in support of cycle 3 operation assumed a rated power level of 2568 MWt for consistency with other B&W reactors and used the analytical methods documented in the Final Safety Analysis Report[1] and updated in the Fuel Densification Report.[7] The following changes in thermal-hydraulic conditions or assumptions were made for cycle 2 and 3 evaluations.

1. The B&W-2 CHF correlation[9] was used instead of the W-3 correlation. The B&W-2 correlation, a realistic prediction of the burnout phenomenon, has been reviewed and approved for use with the Mark-B fuel assembly design. This correlation was used for the Crystal River 3, cycle 2 reload report[2] and is currently used to license all operating B&W plants with Mark-B fuel assembly cores.

2. The assumed system flow was changed from 105% (cycle 1) to 106.5% (cycles 2 and 3) of the design flow of 88,000 gpm/pump primarily to make the thermal-hydraulic design basis for Crystal River 3 consistent with that assumed for other B&W plants of similar design and rated power level (e.g., Oconee 1, 2, 3, ANO-1, and TMI-1). This assumption is fully justified by measured flow data from Crystal River 3, which indicates a system flow in excess of 109.5% of design flow, including allowance for measurement error.

3. The fresh incoming batch 5 fuel inserted for cycle 3 is the Mark B-4 assembly design. Batches 2 and 3 are Mark B-3 assemblies, while batch 4 is Mark B-4. The Mark B-4 fuel assemblies differ from the Mark B-3 assemblies primarily in the end fittings, which have been modified to reduce assembly pressure drop and increase holddown margin. The reduced assembly pressure drop causes a slight increase in flow through the B-4

assemblies relative to the B-3 design. No credit has been taken in thermal-hydraulic evaluations for any increase in B-4 assembly flow resulting from a mixed core that includes Mark B-3 assemblies. Similar core configurations (Mark B-3 in combination with Mark B-4 assemblies) have successfully operated in a number of B&W reactors, including Oconee 1, 2, 3, ANO-1, and TMI-1. Mark B-4 assemblies are currently in all B&W operating reactors.

4. A rod bow penalty has been calculated according to the procedure approved in reference 10. The burnup used is the maximum fuel assembly burnup of the batch that contains the limiting (maximum radial × local peak) fuel
assembly. For cycle 3, this burnup is 31,235 MWd/mtU in a batch 3 assem- 1
bly. The resultant net rod bow penalty after inclusion of the 1% flow
area reduction factor credit is 2.7% reduction in DNBR. The rod bow pen- 1
alty is more than offset by the 10.2% DNBR margin included in trip setpoints and operating limits.

5. A reference design radial × local power peaking factor ($F_{\Delta H}$) of 1.71 was used for cycle 2 and 3 evaluations. The cycle 1 $F_{\Delta H}$ of 1.78 was reduced to 1.71 in conjunction with ORA and BPRA removal.[11]

6. The densification power spike was eliminated from DNBR evaluations based on the NRC approval of this change in reference 12.

The cycle 1, 2, and 3 maximum design conditions and significant parameters are shown in Table 6-1.

6.2. Pressure-Temperature Limit Analysis

The pressure-temperature limit curves for four- and three-pump operations are shown in Figure 8-4. The most limiting of these curves (four-pump) provides the basis for the RPS variable-low-pressure trip function. The curves are based on a minimum DNBR of 1.433, which provides 10.2% margin to the CHF correlation limit. The margin is incorporated to provide flexibility for future cycle designs to avoid the potential need for revising setpoints on a cycle-by-cycle basis.

6.3. Flux/Flow Trip Setpoint Analysis

The flux/flow trip is designed to protect the plant during pump coastdowns from four-pump operation or to act as a high flux trip during partial-pump operation. Crystal River 3, cycle 3, will have redundant pump monitors on each pump, which will trip the reactor immediately upon the loss of power to two or more pumps. Therefore, the flux/flow trip setpoint need only protect the plant during a one-pump coastdown from four-pump operation.

The margin for any assumed flux/flow setpoint is determined with a transient analysis of a one-pump coastdown initiated from 102% indicated power (108% real power). The 6% full power difference between real power and indicated power accounts for 4% FP neutron power measurement error and a 2% FP heat balance error. Actual measured one-pump coastdown data are used in the analysis, and maximum additive trip delays are used betweeen the time trip conditions are reached and actual control rod motion starts. Once a flux/flow trip limit is found to be adequate by thermal-hydraulic analysis, error adjustments are made to account for flow measurement noise and instrument error before the actual trip setpoint is determined.

The recommended cycle 3 thermal-hydraulic flux/flow trip limit of 1.10 (actual in-plant setpoint of 1.07) resulted in a transient minimum DNBR of 1.75 (B&W-2) during the pump ocastdown. This represents >34% DNBR margin to the correlation limit of 1.30.

6.4. Loss-of-Coolant-Flow Transients

The one-pump coastdown analysis was discussed in conjunction with the flux/flow setpoint analysis in section 6.3. The four-pump coastdown and locked-rotor transients were also analyzed for 2568 MWt. The results of these analyses are discussed in section 7, "Accident and Transient Analysis."

Table 6-1. Cycle 1, 2, and 3 Thermal-Hydraulic Design Conditions

	Cycle 1 <268.8 EFPD	Cycle 1 >268.8 EFPD	Cycles 2&3 2544 MWt
Design power level, MWt	2452	2452	2568
System pressure, psia	2200	2200	2200
Reactor coolant flow, % design	105	105	106.5
Ref design radial × local power peaking factor, $F\Delta H$	1.78	1.71	1.71
Ref design axial flux shape	1.5 cosine	1.5 cosine	1.5 cosine
Hot channel factors			
Enthalpy rise	1.011	1.011	1.011
Heat flux	1.014	1.014	1.014
Flow area	0.98	0.98	0.98
Densified active length, in.	141.12	140.2[b]	140.2[b]
Avg heat flux at 100% power, Btu/h-ft^2	167×10^3	168×10^3	176×10^3
Max heat flux at 100% power, Btu/h-ft^2	446×10^3[a]	431×10^3	452×10^3
CHF correlation	W-3	B&W-2	B&W-2
Minimum DNBR (% power)	1.61 (114) 1.92 (102)	2.14 (112) 2.27 (108) 2.49 (102)	1.98 (112) 2.12 (108) 2.33 (102)

(a)The maximum heat fluxes shown are based on reference peaking and average flux. For cycle 1, thermal hydraulic calculations also included the densification spike factor in the DNBR calculations. B&W no longer considers this spike factor in DNBR calculations, as described in reference 7 and accepted in reference 12.

(b)140.2 inches is a conservative (minimum) value used in cycle 2 and 3 analyses; it is the minimum densified length for any B&W fuel. Specific densified lengths for CR-3 fuel are given in Table 4-2.

7. ACCIDENT AND TRANSIENT ANALYSIS

7.1. General Safety Analysis

Each FSAR[1] accident analysis has been examined with respect to changes in cycle 3 parameters to determine the effect of upgrading the reactor power from 2452 to 2544 MWt. Because the FSAR accident analysis, with the exception of the four-pump coastdown and locked-rotor accidents, was done at a higher power level than the requested upgrade (i.e., 2568 versus 2544 MWt), it was only necessary to examine the cycle 3 parameters relative to the FSAR values to ensure that the thermal performance during hypothetical transients is not degraded. Although the FSAR[1] states that all accidents were done at 2544 MWt, they were actually analyzed using the more conservative 2563 MWt.

The effects of fuel densification on the FSAR accident analysis results have been evaluated and are reported in reference 7. Since batch 5 reload fuel assemblies do not contain fuel rods whose theoretical density is lower than those considered in reference 7, the conclusions (with the exception of the four-pump coastdown and locked-rotor accidents) in reference 7 are still valid. These two accidents have been re-evaluated at 102% of 2568 MWt for consistency with other B&W reactors using the analytical methods documented in the FSAR[1] and updated in the Fuel Densification Report.[7] The input parameters used for these accidents are given in Table 6-1 and section 7.6. The letdown line rupture is analyzed in section 7.16, the environmental dose assessment for all accidents is summarized in section 7.18.

7.2. Accident Evaluation

The key parameters that have the greatest effect on determining the outcome of a transient can typically be classified in three major areas: core thermal parameters, thermal-hydraulic parameters, and kinetics parameters, including the reactivity feedback coefficients and control rod worths. Fuel thermal analysis parameters for each batch in cycle 3 are given in Table 4-2.

Table 6-1 compares the cycle 3 thermal-hydraulic maximum design conditions to the previous cycle values, and a comparison of the key kinetics parameters from the FSAR and cycle 3 is provided in Table 7-1. Table 7-2 is a tabulation showing the bounding values for allowable LOCA peak linear heat rates for Crystal River 3, cycle 3 fuel.

It is concluded from the loss-of-flow analysis (section 7.6) and by examination of cycle 3 core thermal and kinetics properties with respect to acceptable FSAR values that this core reload will not adversely affect the ability to safely operate the Crystal River 3 plant during cycle 3. Considering the previously accepted design basis used in the FSAR, the transient evaluation of cycle 3 is considered to be bounded by previously accepted analyses. The initial conditions of the transients in cycle 3 are bounded by the FSAR with the exception of the four pump coastdown and locked rotor accidents, which were redone at a core power of 102% of 2568 MWt.

7.3. Rod Withdrawal Accidents

This accident is defined as uncontrolled reactivity addition to the core due to withdrawal of control rods during startup conditions or from rated power conditions. Both types of incidents were analyzed in the FSAR.

The important parameters during a rod withdrawal accident are Doppler coefficient, moderator temperature coefficient, and the rate at which reactivity is added to the core. Only high-pressure and high-flux trips are accounted for in the FSAR analysis, ignoring multiple alarms, interlocks, and trips that normally preclude this type of incident.

For positive reactivity addition indicative of these events, the most severe results occur for BOL conditions. The FSAR values of the key parameters for BOL conditions were -1.17×10^{-5} $\Delta k/k/°F$ for the Doppler coefficient, 0.0 $\Delta k/k/°F$ for the moderator temperature coefficient and rod group worths up to and including a 12.9% $\Delta/k/k$ rod worth. Comparable cycle 3 parametric values are -1.52×10^{-5} $\Delta k/k/°F$ for Doppler coefficient, -0.30×10^{-4} $\Delta k/k/°F$ for moderator temperature coefficient, and maximum rod bank worth of 9.37% $\Delta k/k$. 1

The FSAR analyses used an initial rated power level of 2568 MWt with a reactor trip at 112% of 2568 MWt. For the accidents that trip on high flux, this is more conservative than initializing the accident at 102% of 2544 MWt and tripping the reactor at 110% of 2544 since more energy is added to the system in the FSAR analysis. For the accidents that trip on high pressure, the pressure trip would occur a little sooner with the higher initial power level (2594 MWt = 102% of 2544 MWt) than with the lower initial power used in the FSAR (2568 MWt). Therefore, cycle 3 parameters are bounded by design values assumed for the FSAR analysis. Thus, for the rod withdrawal transients, the consequences will be no more severe than those presented in the FSAR and the Fuel Densification Report.

7.4. Moderator Dilution Accident

Boron in the form of boric acid is used to control excess reactivity. The boron content of the reactor coolant is periodically reduced to compensate for fuel burnup and transient xenon effects with dilution water supplied by the makeup and purification system. The moderator dilution transients considered are the pumping of water with zero boron concentration from the makeup tank to the RCS under conditions of full power operation, hot shutdown, and refueling.

The key parameters in this analysis are the initial boron concentration, boron reactivity worth, and moderator temperature coefficient for power cases.

For positive reactivity addition of this type, the most severe results occur for BOL conditions. The FSAR values of the key parameters for BOL conditions were 1150 ppm for the initial boron concentration, 100 ppm/1% $\Delta k/k$ boron reactivity worth and $+0.5 \times 10^{-4}$ $\Delta k/k/°F$ for the moderator temperature coefficient. Comparable cycle 3 values are 1185 ppm for the initial boron concentration, 108 ppm/1% $\Delta k/k$ boron reactivity worth and -0.30×10^{-4} $\Delta k/k/°F$ for the moderator temperature coefficient. The FSAR used an initial rated power level of 2568 MWt for these accidents. The effect of a higher initial power (i.e., 102% of 2544 MWt) is to cause the pressure trip to occur sooner. 1

The FSAR shows that the core and RCS are adequately protected during this event. Sufficient time for operator action to terminate this transient is also shown in the FSAR even with maximum dilution and minimum shutdown margin. **The predicted cycle 3 parameter values result in a slower reactivity addition rate than the rate in the FSAR analysis, thus, the analysis in the FSAR is valid.** 1

7.5. Cold Water (Pump Startup) Accident

The NSS contains no check or isolation valves in the RCS piping; therefore, the classical cold water accident is not possible. However, when the reactor is operated with one or more pumps not running, and the pumps are then started, the increased flow rate will cause the average core temperature to decrease. If the moderator temperature coefficient is negative, reactivity will be added to the core and a power increase will occur.

Protective interlocks and administrative procedures exist to prevent the starting of idle pumps if reactor power is above 22%. However, these restrictions were not assumed, and two-pump startup from 50% of 2568 MWt power was analyzed as the most severe transient. The initial power level of 50% of 2568 MWt is slightly more conservative than initializing the transient at 50% of 2544 MWt.

To maximize reactivity addition, the FSAR analysis assumed the most negative moderator temperature coefficient of -4.0×10^{-4} $\Delta k/k/°F$ and least negative Doppler coefficient of -1.17×10^{-4} $\Delta k/k/°F$. The corresponding most negative moderator temperature coefficient and least negative Doppler coefficient pre-
dicted for cycle 3 are -2.63×10^{-4} $\Delta k/k/°F$ and -1.52×10^{-5} $\Delta k/k/°F$, respec- 1
tively. As the predicted cycle 3 moderator temperature coefficient is less negative and the Doppler coefficient is more negative than the values used in the FSAR, the transient results would be less severe than those reported in the FSAR.

7.6. Loss of Coolant Flow (LOCF)

A reduction in reactor coolant flow can be caused by mechanical failure or a loss of electrical power to the pumps. The LOCF transients were re-analyzed for cycle 3 operation and assumed an initial power level of 102% of 2568 MWt for consistency with other B&W reactors.

7.6.1. Four-Pump Coastdown (4PCD)

The 4PCD transient has been analyzed under conditions that represent the most conservative that can occur for cycle 3 operation. These conditions include such key parameters as initial flow rate, flow rate versus time for the transient, initial power level, Doppler coefficient, moderator temperature coefficient, and reference design radial × local power peaking factor ($F\Delta H$). Table 7-3 compares the key parameters used in the analysis with those predicted for cycle 3. For all parameters, the value used in the analysis is either equal to the cycle 3 parameter or is more conservative.

The results of the analysis are shown on Figure 7-1. The minimum DNBR of 2.10 (BAW-2) obtained during the transient is well above the DNBR correlation limit of 1.30. The fuel and cladding temperatures are not shown since there was no increase in these parameters. It is therefore concluded that no fuel damage will occur.

Table 7-4 provides a comparison of MDNBRs between the FSAR, Fuel Densification Report, Cycle 2, and cycle 3 for both one- and four-pump coastdowns. Additional DNBR margin is shown for cycles 2 and 3 due to the use of the B&W-2 CHF correlation instead of the W-3 CHF correlation.

7.6.2. Locked Rotor (LR)

The locked-rotor accident has been analyzed under conditions that represent the most conservative that can occur for cycle 3 operation. These conditions are the same as those in section 7.6.1 (4 PCD). Table 7-3 compares the key parameters used in the analysis with those predicted for cycle 3. For all parameters, the value used in the analysis is either equal to the cycle 3 parameter or is more conservative.

The results of the analysis are shown on Figure 7-2. The maximum fuel temperature does not exceed the initial centerline fuel temperature of 4400F.

This temperature starts to decrease around 2 seconds into the accident. The analysis for the maximum transient cladding and fuel temperatures conservatively assumed film boiling at a DNBR of 1.43 instead of the correlation limit of 1.30 (refer to section 6). The DNBR reached the 1.43 value at approximately 1.2 seconds, after which the cladding temperature increased to a maximum of 1120F at 5.5 seconds after initiation of the accident. Less than 0.5% of the fuel pins in the core will experience a DNBR of less than 1.43, and no pins will experience a DNBR less than 1.00. For those pins that experience DNB, the cladding temperature will not exceed 1120F.

7.7. Stuck-Out, Stuck-In, or Dropped Control Rod Accident

If a control rod is dropped into the core while operating, a rapid decrease in neutron power would occur, accompanied by a decrease in core average coolant temperature. In addition, the power distribution may be distorted due to the new control rod insertions. Therefore, under these conditions, a return to rated power may lead to localized power densities and heat fluxes in excess of design limitations.

The key parameters for this transient are moderator temperature coefficient, worth of dropped rod, and local peaking factors. The FSAR analysis was based on 0.40% $\Delta k/k$ rod worth with a moderator temperature coefficient of -3.0×10^{-4} $\Delta k/k/°F$. For cycle 3, the maximum worth dropped rod at power is 0.20% $\Delta k/k$ and the moderator temperature coefficient is $-2.63 \times \Delta k/k/°F$. Since the predicted rod worth is less and the moderator temperature coefficient more positive, the consequences of this transient are less severe than the results presented in the FSAR.

The effect of initializing these accidents at 2568 MWt as done in the FSAR versus using 102% of 2544 MWt is judged insignificant or slightly beneficial since as shown in Figures 14-20 and -21 of the FSAR, the parameter of primary concern is low system pressure. Starting the accident at a higher power level (i.e., 102% of 2544 MWt) would yield slightly higher system pressures.

7.8. Loss of Electric Power

Two types of power losses were considered in the FSAR: a loss of load condition, caused by separation of the unit from the transmission system, and a hypothetical condition which results in a complete loss of all system and unit power except the unit batteries.

The FSAR analysis evaluated the loss of load with and without turbine runback. When there is no runback, a reactor trip occurs on high RC pressure or temperature. This case resulted in a non-limiting accident. The limiting accident for offsite dose considerations thus becomes the loss of all electrical power except unit batteries, and assuming operation with failed fuel and steam generator tube leakage. The environmental dose assessment is presented in section 7.18.

7.9. Steam Line Failure

A steam line failure is defined as a rupture of any of the steam lines from the steam generators. Upon initiation of the rupture, both steam generators start to blow down, causing a sudden decrease in primary system temperature, pressure, and pressurizer level. The temperature reduction leads to positive reactivity insertion and the reactor trips on high flux or low RC pressure.

The FSAR has identified a double-ended rupture of the steam line between the steam generator and steam stop valve as the worst-case situation at end-of-life conditions.

The key parameter for the core response is the moderator temperature coefficient which in the FSAR was assumed to be -3.0×10^{-4} $\Delta k/k/°F$. The cycle 3 predicted value of moderator temperature coefficient is -2.63×10^{-4} $\Delta k/k/°F$. This value is bounded by that used in the FSAR analysis; hence, the results in the FSAR represent the worst situation.

The FSAR used an initial power level of 2568 MWt for these accidents. This is more conservative than running the accident at 102% of 2544 MWt and tripping the reactor at 110% versus the current 112% setpoint since more energy is added to the system for the FSAR analysis.

7.10. Steam Generator Tube Failure

A rupture or leak in a steam generator tube allows reactor coolant and associated activity to pass to the secondary system. The FSAR analysis is based on complete severance of a steam generator tube. The primary concern for this incident is the potential radiological release. The environmental dose assessment is presented in section 7.18.

7.11. Fuel Handling Accident

The mechanical damage type of accident is considered the maximum potential source of activity release during fuel handling activity. The primary concern is over radiological releases. The environmental dose assessment is presented in section 7.18.

7.12. Rod Ejection Accident

For reactivity to be added to the core at a more rapid rate than by uncontrolled rod withdrawal, physical failure of a pressure barrier component in the CRDA must occur. Such a failure could cause a pressure differential to act on a CRA and rapidly eject the assembly from the core. This incident represents the most rapid reactivity insertion that can be reasonably postulated. The values used in the FSAR and densification report at BOL conditions of -1.17×10^{-5} $\Delta k/k/°F$ Doppler coefficient, 0.0 $\Delta k/k/°F$ moderator temperature coefficient, and ejected rod worth of 0.65% $\Delta k/k$ represented the maximum possible transient.

The use of a 0.65% Δk/k maximum rod worth is conservative in comparison to the cycle 3 predicted value of 0.59% Δk/k. Furthermore, the cycle 3 predicted values of -1.52×10^{-5} Δk/k/°F Doppler and -0.30×10^{-5} Δk/k/°F moderator temperature coefficient are both more negative than used in the FSAR analysis. 1

The FSAR used an initial rated power level of 2568 MWt for this accident. This is more conservative than initializing the accident at 102% of 2544 MWt and tripping the reactor at 110% versus the current 112% setpoint since more energy is added to the system for the FSAR analysis. For the accident which trip on high pressure, the effect of higher initial power level (i.e., 102% of 2544 MWt) is to cause the pressure trip to occur slightly sooner. Since the FSAR input bound the cycle 3 predicted values, the results in the FSAR and densification report are applicable to this reload.

7.13. Maximum Hypothetical Accident

There is no postulated mechanism whereby this accident can occur since this would require a multitude of failures in the engineered safeguards. The hypothetical accident is based solely on a gross release of radioactivity to the reactor building. The environmental dose assessment is presented in section 7.18.

7.14. Waste Gas Tank Rupture

The waste gas tank was assumed to contain the gaseous activity evolved from degassing all the reactor coolant following operation with 1% defective fuel. Rupture of the tank would result in the release of its radioactive contents to the plant ventilation system and to the atmosphere through the unit vent. The environmental dose assessment is presented in section 7.18.

7.15. LOCA Analysis

Generic LOCA analyses for B&W 177-FA lowered-loop NSSs have been performed using the Final Acceptance Criteria ECCS Evaluation Model. The large-break analysis is presented in a topical report[13], and is further substantiated in a letter report[14]. The small break analysis is presented in a letter report[15]. These analyses used the limiting values of key parameters for all plants in the category. Furthermore, the average fuel temperature as a function of linear heat rate and lifetime pin pressure data used in the LOCA limits analysis[13] are conservative compared to those calculated for this reload. Thus, these analyses and LOCA limits provide conservative results for the operation of Crystal River Unit 3 at 2544 MWt.

Crystal River Unit 3's proposed long-term ECCS modification for small break LOCA is presented in reference 16.

The LOCA analyses used a power level of 2772 MWt, which is conservative relative to the 2544 MWt rating. Table 7-2 shows the bounding values for allowable LOCA peak linear heat rates for Crystal River Unit 3, cycle 3.

7.16. Failure of Small Lines Carrying Primary Coolant Outside Containment

7.16.1. Identification of Causes

A break in fluid-bearing lines that penetrate the containment could result in the release of radioactivity to the environment. There are no instrument lines connected to the RCS that penetrate the containment. However, other piping lines from the RCS to the makeup and purification system and the decay heat removal system do penetrate the containment. Leakage through fluid penetrations not serving accident-consequence-limiting systems is minimized by a double-barrier design so that no single credible failure or malfunction of an active component will result in loss of isolation or intolerable leakage. The installed double barriers take the form of closed piping, both inside and outside the containment, and various types of isolation valves.

The most severe pipe rupture relative to radioactivity release during normal plant operation occurs in the makeup and purification system. This would be a rupture of the letdown line just outside the containment but upstream of the letdown control valves. A rupture at this point would result in a loss of reactor coolant until the RCS pressure dropped below its low pressure setpoint at 1500 psig. When this pressure is reached, the emergency injection signal initiates closure of the letdown isolation valve inside the containment, thus terminating the accident.

7.16.2. Analysis of Effects and Consequences

7.16.2.1. Safety Evaluation Criteria

The safety evaluation criterion for this accident is that resultant doses shall not exceed 10 CFR 100 limits.

7.16.2.2. Methods of Analysis

The CRAFT2 computer code[17] was used to determine the loss-of-coolant characteristics of this letdown line rupture accident. The multinode model included a detailed model of the RCS and additional noding simulating the letdown line piping, valves, and coolers. Before the accident, the reactor was assumed to be operating at 2603 MWt with a letdown flow of 140 gpm. A complete severance of the 2.5-inch letdown line between valves MU-V40 or MU-V41 and MU-V49 was assumed. Coincident with this accident, the makeup control valve was assumed to go to a full-open position so that the maximum makeup flow is available. This assumption extends the time to reactor trip/ESFAS actuation and increases the mass and energy releases to the auxiliary building. Termination of the accident was assumed following ESFAS actuation on low RC pressure (1500 psig) and closure of the letdown isolation valves inside the containment. An instrument error of 6% of full range was assumed for the ESFAS actuation pressure, and the letdown isolation valve was assumed closed 7.4 seconds after the ESFAS pressure setpoint was reached. The 7.4-second time period for the complete valve closure considers both the instrumentation response time and the actual valve closure time. Credit was not taken for a reduction in break flow during the time the isolation valves were closing.

7.16.2.3. Environmental Consequences

The time required for the RCS to reach the actuation pressure of 1350 psig (1500 psig minus 6% of 2500 psia) for the ESFAS to initiate isolation is conservatively calculated to be 752 seconds, including valve closure time. For the 2.5-inch letdown line, a total reactor coolant mass of 45,760 pounds is released into the auxiliary building. Ten percent of the iodine contained in the 45,760 pounds of reactor coolant was assumed to volatilize and become airborne in the auxiliary building. The remaining 90% was assumed to remain in the liquid which drains into the auxiliary building sump tank.

The airborne radioactive nuclides in the auxiliary building are filtered through HEPA and charcoal filters in the building's ventilation system before being exhausted to the environment. The analysis is based on a conservatively estimated charcoal filter iodine removal efficiency of 90%. The assumptions used in the evaluation of the offsite doses are summarized in Table 7-5. The atmospheric dispersion factors (X/Q) used to calculate the two-hour doses at the exclusion area boundary and the low population zone boundary are also listed in Table 7-5. The fission product activities released to the environment during the accident are listed in Table 7-6.

7.16.2.4. Results of the Analysis

The dose consequences of the letdown line rupture accident are presented in Table 7-7. The table presents (1) the thyroid dose due to inhalation of iodine activity, and (2) the whole body doses from gamma radiation due to immersion in the gas cloud for individuals located at the outer boundaries of either the exclusion area or the low population zone for the first two hours after the accident. The resulting doses are small fractions of the 10 CFR 100 limits. 1

7.17. Main Feedwater Line Break

A feedwater line failure is defined as a rupture of the feedwater line to the steam generator. The rupture results in a reduction in the heat removal from the primary coolant system. With this reduction the reactor coolant system pressure and temperature will increase until the reactor trips on high reactor coolant pressure at 11.8 seconds after the break. The FSAR analyzed the rupture of the main feedwater header at the steam generator inlet nozzles as the worse case, since this case results in the most rapid steam generator blowdown.

Because the feedwater accident is an overheating event, BOL values of Doppler and moderator coefficients represent the most positive reactivity addition to the core.

Table 7-1 shows that the FSAR value for these parameters are more positive than the cycle 3 value, (i.e., FSAR used -1.17×10^{-5} $\Delta k/k/F$ and 0 $\Delta k/k/F$ for the Doppler and moderator coefficients respectively, while cycle 3 predicts -1.52×10^{-5} $\Delta k/k/F$ and -0.30×10^{-4} $\Delta k/k/F$ for these two parameters). Therefore, the cycle 3 value is bounded by the FSAR analysis and the FSAR represents the worst situation. 1

The effect of a higher initial power level on this accident (i.e., 102% of 2544 MWt) is to cause the pressure trip to occur 0.14 seconds sooner and the peak system pressure to be 17 psi greater, still within the allowable code pressure limit. The reactor coolant system design will accommodate 14½ minutes of safe shutdown operation at the higher power level. Thereafter, the operator can provide a controlled cooldown of the plant utilizing the auxiliary feedwater system and steam relief through the atmospheric or condenser dump valves. Since core coverage can be maintained and reactor coolant system pressure remain within code allowable limits, the safety evaluation criteria are met.

7.18. Dose Consequences of Accidents

Detailed dose calculations were performed for cycle 3, for the letdown line rupture accident and for the other FSAR accidents. The results are summarized in Tables 7-7 and 7-8. Table 7-7 presents for individuals located at either the exclusion area boundary (EAB) or the low population zone (LPZ) boundary, (1) the thyroid dose due to the inhalation of iodine activity, (2) the sum of the whole body doses from gamma radiation due to the immersion in the gas cloud and the skin doses from beta radiation due to the immersion in the cloud. The reload/FSAR dose ratios are tabulated in Table 7-7 for all accidents except the LOCA and MHA, which are discussed later in this section.

Using detailed cycle 3 fuel data have resulted in higher plutonium-to-uranium fission ratio than that assumed in the FSAR. Since plutonium has a higher iodine fission yield than uranium, more iodine activity is produced and thus the thyroid doses are expected to be higher than reported in the FSAR. The thyroid doses for the fuel handling accident increased by 30% due to this effect. Generally, the plutonium fission yield for noble gases is lower than for uranium which would result in lower noble gas inventories that would tend to lower the whole body doses below these reported in the FSAR unless the iodine release is large enough to result in an overall dose increase. 1

The FSAR doses for MHA and LOCA accidents were calculated using an iodine removal model associated with a sodium thiosulfate spray system. The spray system was later changed to a sodium hydroxide system which has a lower iodine removal rate. The MHA and LOCA doses for cycle 3 fuel were calculated using a sodium hydroxide spray system consistent with the NRC Safety Evaluation Report (Supplement 3 — December 1976). Table 7-8 compares the MHA and LOCA doses for the original FSAR assumptions, for the NRC SER and for the cycle 3 reload assumptions. The MHA cycle 3 thyroid doses at the EAB and at the LPZ are within 65 and 74% of the NRC SER doses. Similarly, the cycle 3 whole body doses are within 76 and 44% of the SER doses at the EAB and LPZ respectively. Even with the large increase in MHA doses in comparison to the FSAR doses due to the change in spray systems, the cycle 3 thyroid and whole body doses are well below the guidelines of 10 CFR 100.

Table 7-1. Comparison of Key Parameters for Accident Analysis

Parameter	FSAR[1], densif'n value[7]	Cycle 1[11]	Cycle 3 value
BOL Doppler coeff, 10^{-5} Δk/k/°F	-1.17	-1.47 (268 EFPD)	-1.52
EOL Doppler coeff, 10^{-5} Δk/k/°F	-1.30	-1.66 (510 EFPD)	-1.61
BOL moderator coeff, 10^{-4} Δk/k/°F	0[(a)]	-0.75 (268 EFPD)	-0.30
EOL moderator coeff, 10^{-4} Δk/k/°F	-4.0[(b)]	-2.42 (510 EFPD)	-2.63
All-rod bank worth at BOL, HZP, % Δk/k	12.9	9.12 (268 EFPD)	9.37
Boron reactivity worth (HFP), ppm/1% Δk/k	100	101	108
Max ejected rod worth (HFP), % Δk/k	0.65	0.55	0.49
Dropped rod worth (HFP), % Δk/k	0.65	0.20	0.20
Initial boron conc'n (HFP), ppm	1150	795	1185

1

(a) $+0.50 \times 10^{-4}$ Δk/k/°F was used for the moderator dilution accident.

(b) -3.0×10^{-4} Δk/k/°F was used for the steam line failure analysis and dropped rod accident analysis.

Table 7-2. Bounding Values for Allowable LOCA Peak Linear Heat Rates

Core elevation, ft	Allowable peak LHR, kW/ft
2	15.5
4	16.6
6	18.0
8	17.0
10	16.0

Table 7-3. Input Parameters to Loss-of-Coolant-Flow Transients

	Cycle 3 value	Value used in analysis
Initial flow rate, % of 352,000 gpm	>109.5	106.5
Flow rate Vs time	>Fig. 14-17, FSAR Fig. 14-19a, FSAR	Fig. 14-17, FSAR (4PCD) Fig. 14-19a, FSAR (LR)
Initial power level, MW	2544	102% of 2568
Doppler coeff, $\Delta k/k/°F$	-1.52×10^{-5}	-1.27×10^{-5}
Moderator temp coeff, $\Delta k/k/°F$	-0.30×10^{-4}	0
$F\Delta H$	1.47	1.71

1

Table 7-4. Summary of Minimum DNBR Results for Limiting Loss-of-Coolant-Flow Transients

Transient	Cycle 1: FSAR[1] (W-3)	Cycle 1: Densif'n report (W-3)	Cycle 2 (B&W-2)	Cycle 3 (B&W-2)
One-pump coastdown (flux/flow trip)	NR[(a)]	NR	1.75	1.75
Four-pump coastdown (flux/flow trip, cycle 1; pump monitor trip, cycles 2 and 3)	1.45	1.39	2.10	2.10

(a) NR: not reported.

Table 7-5. Analysis Assumptions for MU&PS Letdown Line Rupture Accident

Data and Assumptions Used to Estimate Radioactive Source

Power level, MWT	2544
Percent of fuel rods leaking, %	1.0
Escape rate coeff (see FSAR[1] table)	11-1
Reactor coolant activities	

Nuclide	Activity, μCi/cc
$^{85}Kr^m$	1.48
^{85}Kr	4.36
^{87}Kr	0.779
^{88}Kr	2.41
$^{131}Xe^m$	1.63
$^{133}Xe^m$	2.58
^{133}Xe	238.0
$^{135}Xe^m$	0.294
^{135}Xe	4.88
^{138}Xe	0.421
^{131}I	3.47
^{132}I	1.17
^{133}I	3.70
^{134}I	0.461
^{135}I	1.88

1

Data and Assumptions Used to Estimate Radioactivity Released

Total mass of reactor coolant released to auxiliary building, lb	45,760
Charcoal filter efficiency for	
Iodine, %	90
Noble gas, %	0
Fraction of iodine airborne	0.1

Dispersion Data

EAB, m	1340
LPZ boundary, m	8047
Atmospheric dispersion percentile, %	5
0-2 h atmospheric dispersion factors, s/m^3	
at EAB	1.6×10^{-4}
at LPZ boundary	1.4×10^{-5}

Table 7-6. Activity Released to Environment Due to Rupture of MU&PS Letdown Line

Nuclide	Activity, Ci
$^{85}Kr^m$	44.6
^{85}Kr	131.0
^{87}Kr	23.5
^{88}Kr	72.6
$^{131}Xe^m$	49.1
$^{133}Xe^m$	77.7
^{133}Xe	7170.0
$^{135}Xe^m$	8.85
^{135}Xe	147.0
^{138}Xe	12.7
^{131}I	10.4
^{132}I	3.52
^{133}I	11.1
^{134}I	1.39
^{135}I	5.66

1

Table 7-7. Comparison of FSAR Accident Doses to Cycle 3 Reload Doses

Accident	FSAR dose, Rem	Cycle 3 reload dose, Rem	Ratio — reload/FSAR	
Steam line failure				
Thyroid dose at EAB[a]	0.488	0.503	1.03	
Whole body dose at EAB	0.0044	0.0033	0.75	
Steam generator tube failure				
Thyroid dose at EAB	0.00225	0.0023	1.04	
Whole body dose at EAB	0.162	0.130	0.80	
Fuel handling accident — conservative case				
Thyroid dose at EAB	10.6	14.0	1.32	
Whole body dose at EAB	0.16	0.190	1.19	
Thyroid dose at site boundary	4.64	6.08	1.31	
Whole body dose at site boundary	0.07	0.084	1.20	
Thyroid dose at LPZ[a]	0.04	0.528	1.32	
Whole body dose at LPZ	0.006	0.0072	1.20	1
Rod ejection accident				
Thyroid dose at EAB	1.67	0.65	0.39	
Whole body dose at EAB	0.003	0.0008	0.27	
Thyroid dose at LPZ	0.878	0.35	0.40	
Whole body dose at LPZ	0.002	0.0005	0.26	
Waste gas tank rupture				
Thyroid dose at EAB	1.44	1.43	0.99	
Whole body dose at EAB	1.08	0.92	0.85	
Letdown line rupture[b]				
Thyroid dose at EAB	0.111	0.115	1.04	
Whole body dose at EAB	0.082	0.066	0.80	
Thyroid dose at LPZ	0.0098	0.0101	1.04	
Whole body dose at LPZ	0.0072	0.0058	0.80	

[a] EAB: exclusion area boundary, LPZ: low-population zone outer boundary.

[b] Letdown line rupture was not addressed in the FSAR; therefore, the doses in the FSAR column are really from the cycle 2 reload report (BAW-1521).

Table 7-8. MHA and LOCA Doses for Cycle 3, Rems

Accident	FSAR dose[a]	NRC[b] SER dose	Cycle 3[c] reload dose	Ratio — reload/SER	
LOCA					
Thyroid dose at EAB	0.549	--	2.19	--	
Whole body dose at EAB	0.0174	--	0.016	--	
Thyroid dose at LPZ	0.073	--	0.517	--	
Whole body dose at LPZ	0.011	--	0.0081	--	
MHA					
Thyroid dose at EAB	26.1	133	86.8	0.65	1
Whole body dose at EAB	2.02	3	2.28	0.76	
Thyroid dose at LPZ	2.89	25	18.4	0.74	
Whole body dose at LPZ	0.29	<1	0.44	0.44	

[a] FSAR dose based on sodium thiosulfate spray system.

[b] NRC Safety Evaluation Report, Supplement 3 (December 30, 1976), based on sodium hydroxide spray system.

[c] Reload dose based on sodium hydroxide spray system with the following iodine removal rates for 2 hours: elemental iodine — 4.86 h^{-1}, and particulate iodine — 0.45 h^{-1}.

Figure 7-1. Four-Pump Coastdown — Hot Channel MDNBR Vs Time, Crystal River 3

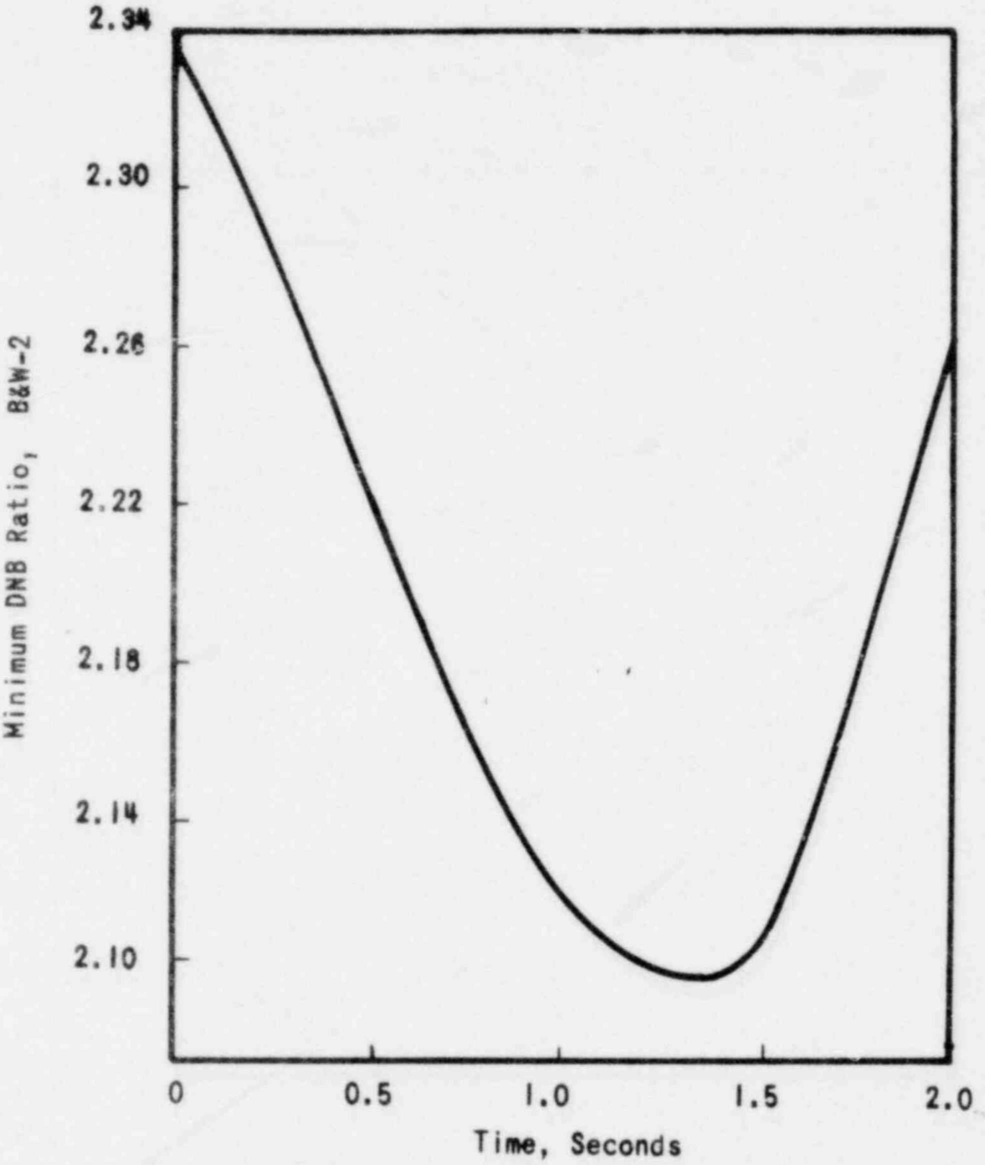
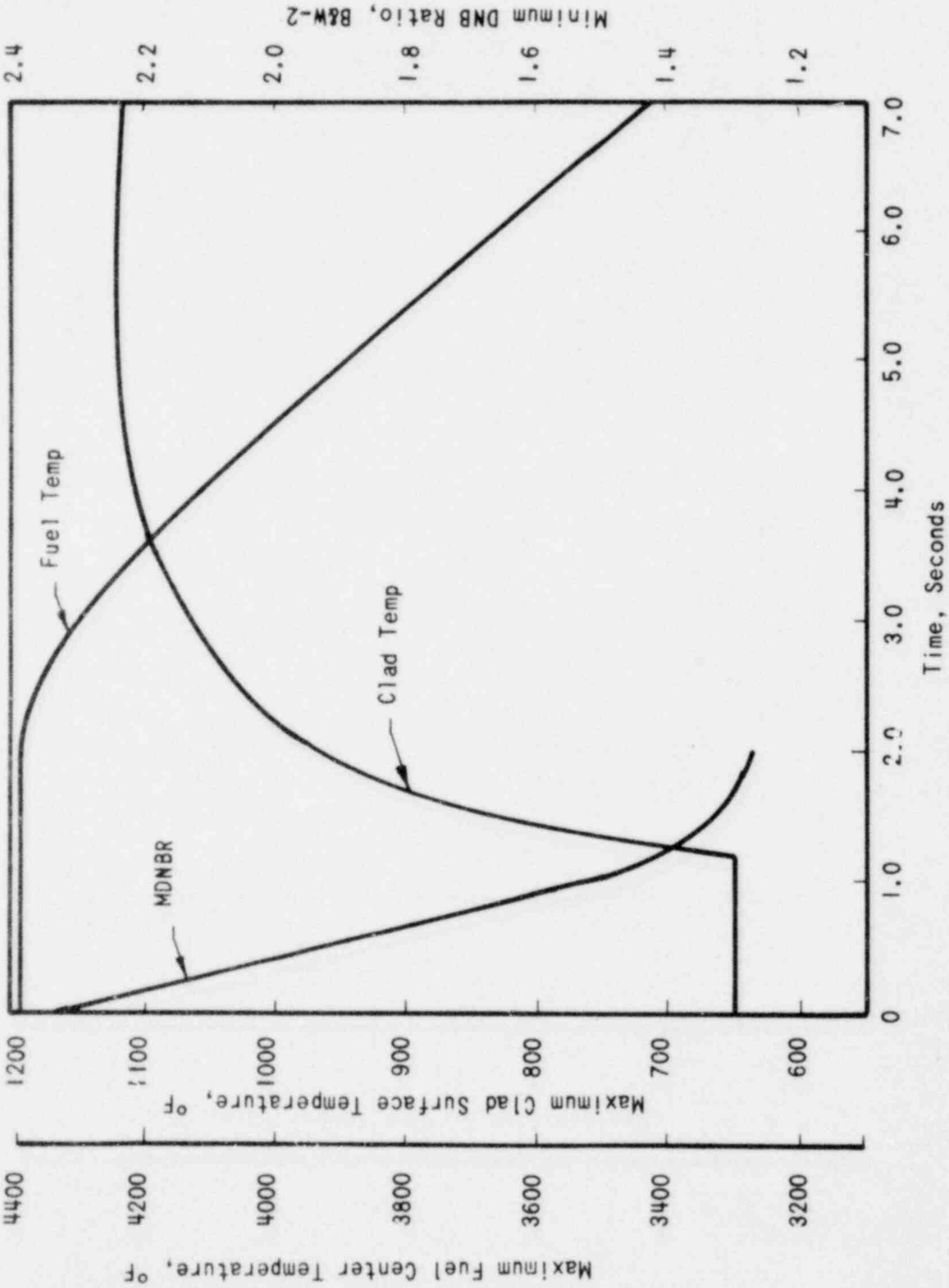


Figure 7-2. Locked-Rotor, Crystal River 3



8. PROPOSED MODIFICATIONS TO TECHNICAL SPECIFICATIONS

All technical specifications have been reviewed by Florida Power Corporation and B&W and some were revised for cycle 3 operation. The Technical Specification sections to which modifications have been made are listed in Table 8-1 and are shown on the following pages. The reanalysis of Technical Specifications for cycle 3 operation used the same analytical techniques as the cycle 2 design.[2]

The review of the Technical Specifications based on the analyses presented in this report, and the proposed modifications contained in this section, ensure that the Final Acceptance Criteria ECCS limits will not be exceeded nor will the thermal design criteria be violated.

Table 8-1. Technical Specification Changes

Tech Spec No. (figure, table Nos.)	Report page Nos. (figure Nos.)	Reason for change
1.3	8-3	Rated thermal power increased to 2544 MWt.
2.0 Bases for 2.0	8-4 thru 8-11 (8-1 thru -4) Table 8-2	Because of the large number of changes to section 2.0 in cycles 1, 2, and 3, the entire section is presented here to avoid confusion. The flux/Δflux envelopes changed due to the power upgrade. Flux/flow trips changed with the addition of the RC pump monitors; the trips are now based on a one-pump versus four-pump coastdown.
3.1.3.6 (3.1-1, -2, -3, -4)	Figures 8-5 thru 8-8	Specs 3.1.3.6, 3.1.3.9, and 3.2.1 reflect revised nuclear parameters as a result of the cycle 3 reload, including the power upgrade.
3.1.3.9 (3.1-9, -10	Figures 8-9, 8-10	
3.2.1 (3.2-1, -2)	Figures 8-11, 8-12	
3.2.4 (Table 3.2-2)	Table 8-3	Tilt limits were reduced to reflect increased detector depletion.
3.2.5 (Table 3.2-1)	Table 8-4	Flow rates were recalculated based on 2544 MWt.
3/4.1.1, 3.1.2.7, 3.1.2.9, and Bases	8-12 thru 8-19	Shutdown margin requirements for modes 4 and 5 were increased to account for the inadvertent deboration by sodium hydroxide addition. 1

DEFINED TERMS

1.1 The DEFINED TERMS of this section appear in capitalized type and are applicable throughout these Technical Specifications.

THERMAL POWER

1.2 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

RATED THERMAL POWER

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2544 MWt.

OPERATIONAL MODE

1.4 An OPERATIONAL MODE shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.1.

ACTION

1.5 ACTION shall be those additional requirements specified as corollary statements to each principal specification and shall be part of the specifications.

OPERABLE – OPERABILITY

1.6 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment, that are required for the system, subsystem, train, component or device to perform its function(s), are also capable of performing their related support function(s).

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1. SAFETY LIMITS

REACTOR CORE

2.1.1. The combination of the reactor coolant core outlet pressure and outlet temperature shall not exceed the safety limit shown in Figure 2.1-1.

APPLICABILITY: MODES 1 and 2.

ACTION:

When the point defined by the combination of reactor coolant core outlet pressure and outlet temperature has exceeded the safety limit, be in HOT STANDBY within one hour.

REACTOR CORE

2.1.2 The combination of reactor THERMAL POWER and AXIAL POWER IMBALANCE shall not exceed the safety limit shown in Figure 2.1-2 for the various combinations of three and four reactor coolant pump operation.

APPLICABILITY: MODE 1.

ACTION:

Whenever the point defined by the combination of Reactor Coolant System flow, AXIAL POWER IMBALANCE and THERMAL POWER has exceeded the appropriate safety limit, be in HOT STANDBY within one hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The Reactor Coolant System pressure shall not exceed 2750 psig.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2 Whenever the Reactor Coolant System pressure has exceeded 2750 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within one hour.

MODES 3 and 4 Whenever the Reactor Coolant System pressure has exceeded 2750 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

2.2. LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTION SYSTEM STEPOINTS

2.2.1 The Reactor Protection System instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a Reactor Protection System instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

2.1.1 and 2.1.2 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime would result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and reactor coolant temperature and pressure have been related to DNB through the BAW-2 DNB correlation. The DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local 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. This value 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.

The curve presented in Figure 2.1-1 represents the conditions at which a minimum DNBR ≥ 1.30 is predicted for the maximum possible thermal power, 112% when the reactor coolant flow is 139.7×10^6 lb/h, which is 106.5% of the design flow rate for four operating reactor coolant pumps. This curve is based on the following nuclear power peaking factors with potential fuel densification effects:

$$F_Q^N = 2.57; \qquad F_{\Delta H}^N = 1.71; \qquad F_Z^N = 1.50.$$

The design limit power peaking factors are the most restrictive calculated at full power for the range from all control rods fully withdrawn to minimum allowable control rod withdrawal, and form the core DNBR design basis.

BASES

The reactor trip envelope appears to approach the safety limit more closely than it actually does because the reactor trip pressures are measured at a location where the indicated pressure is about 30 psi less than core outlet pressure, providing a more conservative margin to the safety limit.

The curves of Figure 2.1-2 are based on the more restrictive of two thermal limits and account for the effects of potential fuel densification and potential fuel rod bow:

1. The 1.30 DNBR limit produced by a nuclear power peaking factor of $F_Q^N = 2.57$ or the combination of the radial peak, axial peak and position of the axial peak that yields no less than a 1.30 DNBR.

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 19.7 kW/ft.

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 specified flow rates for curves 1 and 2 of Figure 2.1-2 correspond to the expected minimum flow rates with four pumps and three pumps respectively.

The curve of Figure 2.1-1 is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in BASES Figure 2.1. The curves of BASES Figure 2.1 represent the conditions at which a minimum DNBR of 1.30 is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation.

These curves include the potential effects of fuel rod bow and fuel densification.

The DNBR as calculated by the BAW-2 DNB correlation continually increases from point of minimum DNBR, so that the exit DNBR is always higher. Extrapolation of the correlation beyond its published quality range of 22% is justified on the basis of experimental data.

For each curve of BASES Figure 2.1, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 or a local quality at the point of minimum DNBR less than 22% for that particular reactor coolant pump situation. The 1.30 DNBR curve for four pump operation is more restrictive than any other reactor coolant pump combination 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.

2.1.3 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Boiler and Pressure Vessel Code which permits a maximum transient pressure of 110%, 2750 psig, of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to USAS B 31.7, February, 1968 Draft Edition, which permits a maximum transient pressure of 110%, 2750 psig, of component design pressure. The Safety Limit of 2750 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Protection System Instrumentation Trip Setpoint specified in Table 2.2-1 are the values at which the Reactor Trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits. Operation with a trip setpoint less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

The Shutdown Bypass provides for bypassing certain functions of the Reactor Protection System in order to permit control rod drive tests, zero power PHYSICS TESTS and certain startup and shutdown procedures. The purpose of the Shutdown Bypass RCS Pressure-High trip is to prevent normal operation with Shutdown Bypass activated. This high pressure trip setpoint is lower than the normal low pressure trip setpoint so that the reactor must be tripped before the bypass is initiated. The Nuclear Overpower Trip Setpoint of $\leq$ 5.0% prevents any significant reactor power from being produced. Sufficient natural circulation would be available to remove 5.0% of RATED THERMAL POWER if none of the reactor coolant pumps were operating.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic Reactor Protection System instrumentation channels and provides manual reactor trip capability.

Nuclear Overpower

A Nuclear Overpower trip at high power level (neutron flux) provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry.

During normal station operation, reactor trip is initiated when the reactor power level reaches 105.5% of rated power. Due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which was used in the safety analysis.

LIMITING SAFETY SYSTEM SETTINGS

BASES

RCS Outlet Temperature - High

The RCS outlet temperature high trip ≤ 619°F prevents the reactor outlet temperature from exceeding the design limits and acts as a backup trip for all power excursion transients.

Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE

The power level trip setpoint produced by the reactor coolant system flow is based on a flux-to-flow ratio which has been established to accommodate flow decreasing transients from high power.

The power level trip setpoint 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 setpoint 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 2.2-1 are as follows:

1. Trip would occur when four reactor coolant pumps are operating if power is ≥ 107.0% and reactor flow rate is 100%, or flow rate is ≤ 93.5% and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is ≥ 79.9% and reactor flow rate is 74.7%, or flow rate is ≤ 69.9% and power is 75%. 1

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

The AXIAL 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 AXIAL POWER IMBALANCE reduces the power level trip produced by the flux-to-flow ratio so that the boundaries of Figure 2.2-1 are produced. The flux-to-flow ratio reduces the power level trip and associated reactor power-reactor power-imbalance boundaries by 1.07% for a 1% flow reduction.

RCS Pressure - Low, High and Variable Low

The high and low trips are provided to limit the pressure range in which reactor operation is permitted.

During a slow reactivity insertion startup accident from low power or a slow reactivity insertion from high power, the RCS pressure-high setpoint is reached before the nuclear overpower trip setpoint. The trip setpoint for RCS pressure-high, 2300 psig, has been established to maintain the system pressure below the safety limit, 2750 psig, for any design transient. The RCS pressure-high trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves, 2500 psig. The RCS pressure-high trip also backs up the nuclear overpower trip.

The RCS pressure-low, 1800 psig, and RCS pressure-variable low (11.80 T_{out}°F-5209.2) psig, trip setpoints have been established to maintain the DNB ratio greater than or equal to 1.30 for those design accidents that result in a pressure reduction. It also prevents reactor operation at pressures below the valid range of DNB correlation limits, protecting against DNB.

Due to the calibration and instrumentation errors, the safety analysis used an RCS pressure-variable low trip setpoint of (11.80 T_{out}°F-5249.2) psig.

Reactor Containment Vessel Pressure - High

The reactor containment vessel pressure-high trip setpoint, ≤ 4 psig, provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the containment vessel or a loss-of-coolant accident, even in the absence of an RCS pressure-low trip.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.1.1.1.2 The SHUTDOWN MARGIN shall be $\geq$ 3.0% $\Delta k/k$.

APPLICABILITY: MODES 4 and 5.

ACTION:

With the SHUTDOWN MARGIN < 3.0% $\Delta k/k$, immediately initiate and continue boration at $\geq$ 10 gpm of 11,600 ppm boric acid solution or its equivalent, until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS 1

4.1.1.1.2.1 The SHUTDOWN MARGIN shall be determined to be $\geq$ 3.0% $\Delta k/k$:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above-required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rod(s).

b. At least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. Control rod position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition. During modes 1 and 2 the SHUTDOWN MARGIN is known to be within limits if all control rods are OPERABLE and withdrawn to or beyond the insertion limits.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition for modes 1, 2, and 3 occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident a minimum SHUTDOWN MARGIN of 0.60% $\Delta k/k$ is initially required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN required is based on this limiting condition and is consistent with FSAR safety analysis assumptions. The most restrictive condition for modes 4 and 5 occurs at BOL and is associated with deboration due to inadvertent injection of sodium hydroxide. The higher requirement for these modes ensures that the accident will not result in criticality. 1

3/4.1.1.2 BORON DILUTION

A minimum flow rate of at least 2700 gpm provides adequate mixing, prevents stratification, and ensures that reactivity changes will be gradual through the reactor coolant system in the core during boron concentration reductions in the reactor coolant system. A flow rate of at least 2700 gpm will circulate an equivalent reactor coolant system volume of 12,000 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron concentration reduction will be within the capability for operator recognition and control.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC vlaue is within its limit provides assurance that the coefficient will be maintained within acceptable values throughout each fuel cycle.

REACTIVITY CONTROL SYSTEMS

BORIC ACID PUMPS – OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.7 At least one boric acid pump in the boron injection flow path required by Specification 3.1.2.2a shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if the flow path through the boric acid pump in Specification 3.1.2.2a is OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

MODES 1, 2, and 3:

With no boric acid pump OPERABLE, restore at least one boric acid pump to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to 1% $\Delta k/k$ at 200F within the next 6 hours; restore at least one boric acid pump to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

MODE 4:

With no boric acid pump OPERABLE, restore at least one boric acid pump to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours. 1

SURVEILLANCE REQUIREMENTS

4.1.2.7 No additional Surveillance Requirements other than those required by Specification 4.0.5.

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES – OPERATING

3.1.2.9 Each of the following borated water sources shall be OPERABLE:

a. The concentrated boric acid storage system and associated heat tracing with:

1. A minimum contained borated water volume of 6615 gallons,
2. Between 11,600 and 14,000 ppm of boron, and
3. A minimum solution temperature of 105F.

b. The borated water storage tank (BWST) with:

1. A contained borated water volume of between 415,200 and 449,000 gallons,
2. Between 2270 and 2450 ppm of boron, and
3. A minimum solution temperature of 40F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

MODES 1, 2, and 3:

a. With the concentrated boric acid storage system inoperable, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to 1% Δk/k at 200F within the next 6 hours; restore the concentrated boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours. 1

b. With the borated water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

MODE 4:

a. With the concentrated boric acid storage system inoperable, restore the storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the borated water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in COLD SHUTDOWN within 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.9 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration in each water source,
2. Verifying the contained borated water volume of each water source, and
3. Verifying the concentrated boric acid storage system solution temperature.

b. At least once per 24 hours by verifying the BWST temperature when the outside air temperature is < 40F.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the reactor coolant system average temperature less than 525F. This limitation is required to ensure that (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the protective instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include (1) borated water sources, (2) makeup or DHR pumps, (3) separate flow paths, (4) boric acid pumps, (5) associated heat tracing systems, and (6) an emergency power supply from OPERABLE emergency busses.

With the RCS average temperature above 200F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period. 1

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from all operating conditions of 3.0% $\Delta k/k$ after xenon decay and cooldown to 200F. The maximum boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires either 6615 gallons of 11,600 ppm boric acid solution from the boric acid storage tanks or 45,421 gallons of 2270 ppm borated water from the borated water storage tank.

The requirements for a minimum contained volume of 415,200 gallons of borated water in the borated water storage tank ensures the capability for borating the RCS to the desired level. The specified quantity of borated water is consistent with the ECCS requirements of Specification 3.5.4. Therefore, the larger volume of borated water is specified.

With the RCS temperature below 200F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

The boron capability required below 200F is sufficient to provide a SHUTDOWN MARGIN of 3.0% $\Delta k/k$ after xenon decay and cooldown from 200F to 140F. This condition requires either 300 gallons of 11,600 ppm boric acid solution from the boric acid storage system or 1608 gallons of 2270 ppm borated water from the borated water storage tank. To envelop future cycle BWST contained borated water volume requirements, a minimum volume of 13,500 gallons is specified.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics. The limits on contained water volume and boron concentration ensure a pH value of between 7.2 and 11.0 of the solution sprayed within the containment after a design basis accident. The pH band minimizes the evolution of iodine and minimizes the effect of chlorides and caustic stress corrosion cracking on mechanical systems and components.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section (1) ensure that acceptable power distribution limits are maintained, (2) ensure that the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential effects of a rod ejection accident. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met. For example, misalignment of a safety or regulating rod requires a restriction in THERMAL POWER. The reactivity worth of a misaligned rod is limited for the remainder of the fuel cycle to prevent exceeding the assumptions used in the safety analysis.

The position of a rod declared inoperable due to misalignment should not be included in computing the average group position for determining the OPERABILITY of rods with lesser misalignments. 1

Table 8-2. RPS Trip Setpoints

Table 2.2-1. Reactor Protection System Instrumentation Trip Setpoints

Functional unit	Trip setpoint	Allowable values
1. Manual reactor trip	Not applicable	Not applicable
2. Nuclear overpower	≤ 105.5% of RATED THERMAL POWER with four pumps operating ≤ 79.9% of RATED THERMAL POWER with three pumps operating	≤ 105.5% of RATED THERMAL POWER with four pumps operating ≤ 79.9% of RATED THERMAL POWER with three pumps operating
3. RCS outlet temp-high	≤ 619°F	≤ 619°F
4. Nuclear overpower based on RCS flow and AXIAL POWER IMBALANCE[a]	Trip setpoint not to exceed the limit line of Figure 2.2-1	Allowable values not to exceed the limit line of Figure 2.2-1
5. RCS pressure-low[a]	≥ 1800 psig	≥ 1800 psig
6. RCS pressure-high	≤ 2300 psig	≤ 2300 psig
7. RCS pressure-variable-low[a]	≥ (11.80 T_{out}°F-5209.2) psig	≥ (11.80 T_{out}°F-5209.2) psig
8. Nuclear overpower based on RCPPMs[a]	More than one pump inoperable.	More than one pump inoperable.
9. Reactor containment vessel	≤ 4 psig	≤ 4 psig

[a]Trip may be manually bypassed when RCS pressure ≤ 1720 psig by actuating the shutdown bypass, provided that (1) the nuclear overpower trip setpoint is ≤ 5% of RATED THERMAL POWER, (2) the shutdown bypass RCS pressure-high trip setpoint of ≤ 1720 psig is imposed, and (3) the shutdown bypass is removed when RCS pressure > 1800 psig.

Table 8-3. Quadrant Power Tilt Limits

Table 3.2-2. Quadrant Power Tilt Limits

	Steady-state limit	Transient limit	Maximum limit
QUADRANT POWER TILT as measured by:			
Symmetrical incore detector system	3.31	8.81	20.0
Power range channels	1.96	6.96	20.0
Minimum incore detector system	1.90	4.40	20.0

1

Table 8-4. DNBR Limits

Table 3.2-1. DNB Margin

Parameter	Four RC pumps operating	Three RC pumps operating
Reactor coolant hot leg temperature, T_H, °F	≤ 604.6	≤ 604.6[a]
Reactor coolant pressure, psig[b]	≥ 2,061.6	≥ 2,057.2[a]
Reactor coolant flow rate, lbm/hr	≥ 139.7×10^6	≥ 104.4×10^6

[a]Applicable to the loop with two RC pumps operating.

[b]Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step increase of greater than 10% of RATED THERMAL POWER.

Figure 8-1. Reactor Core Safety Limits

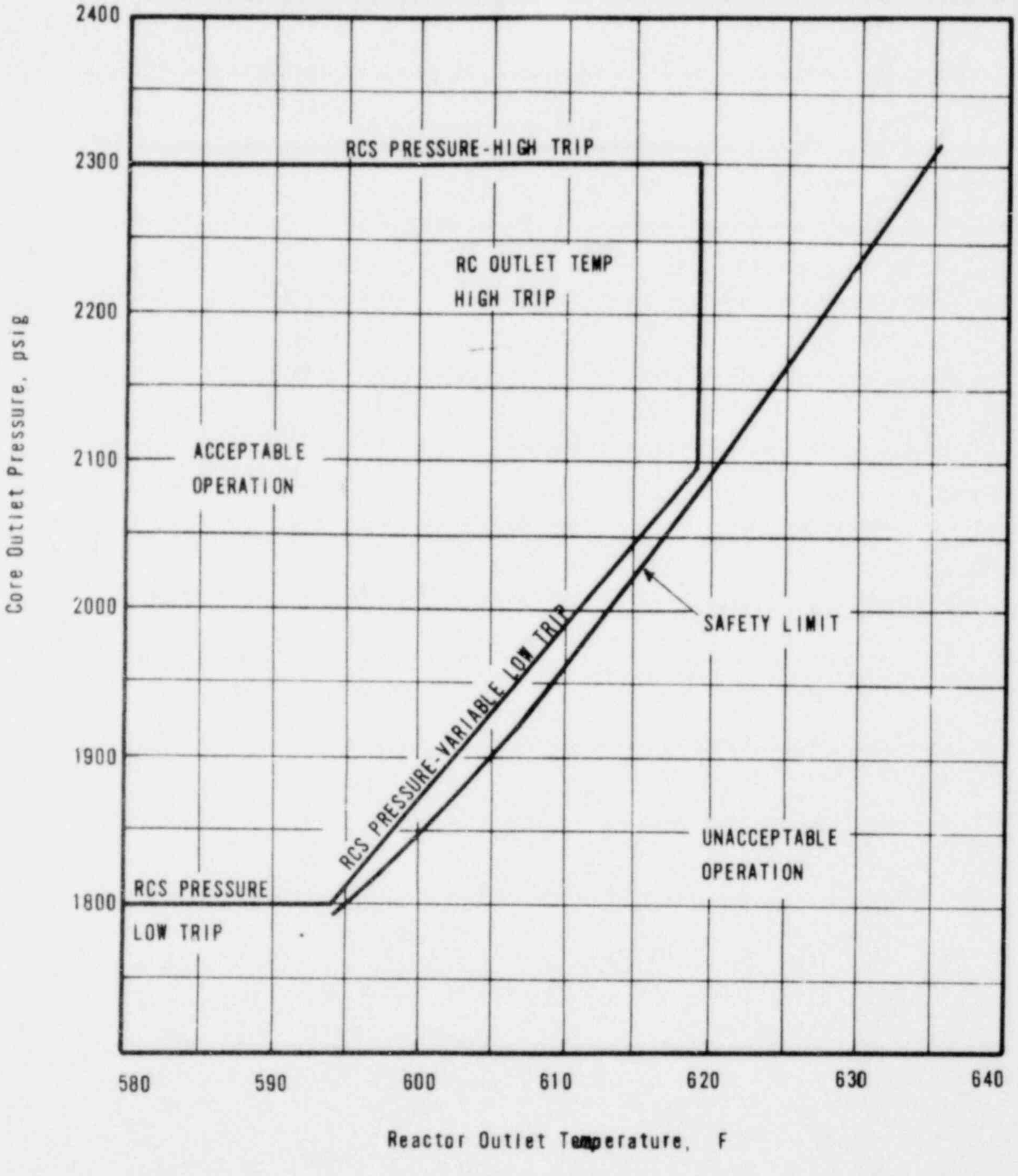
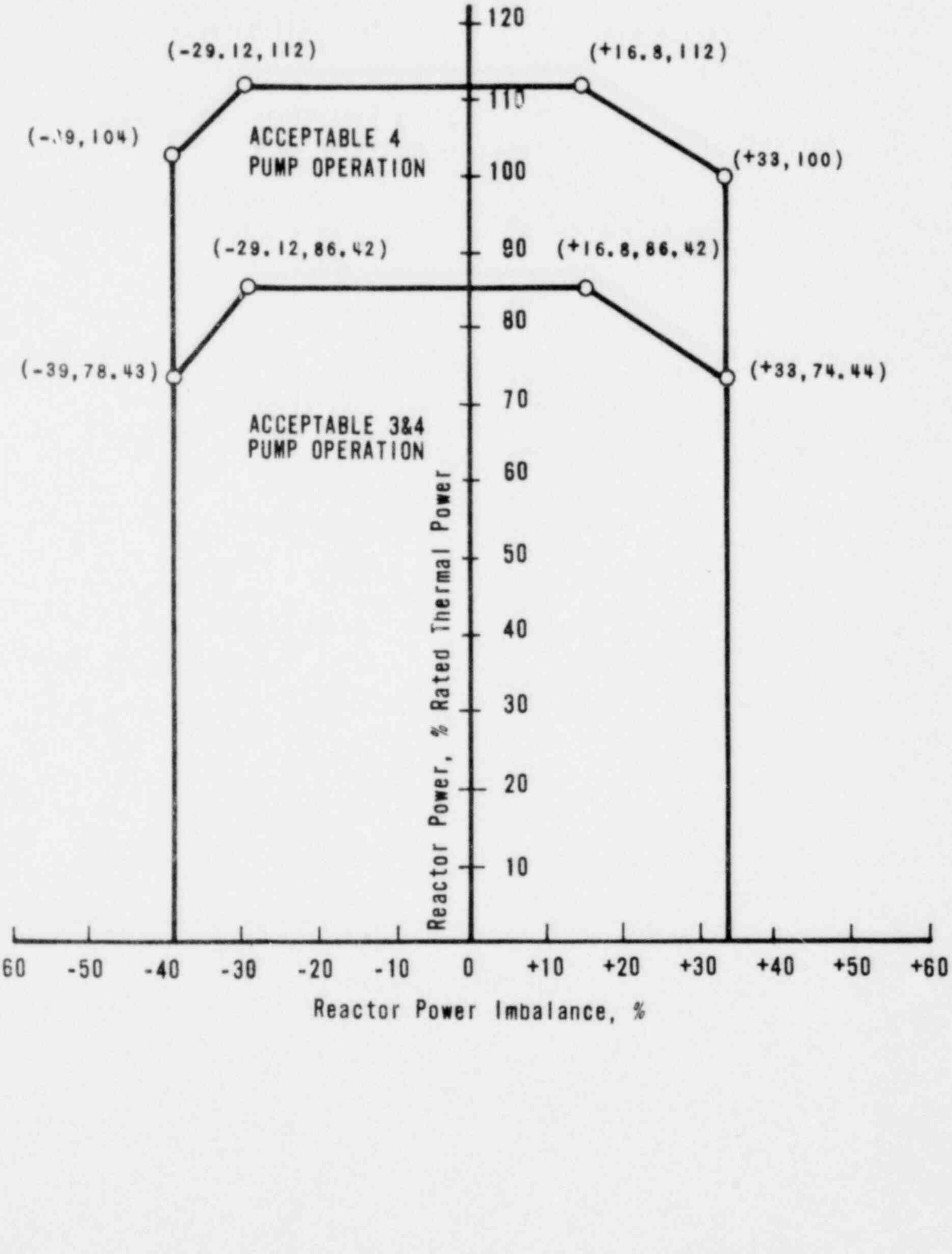
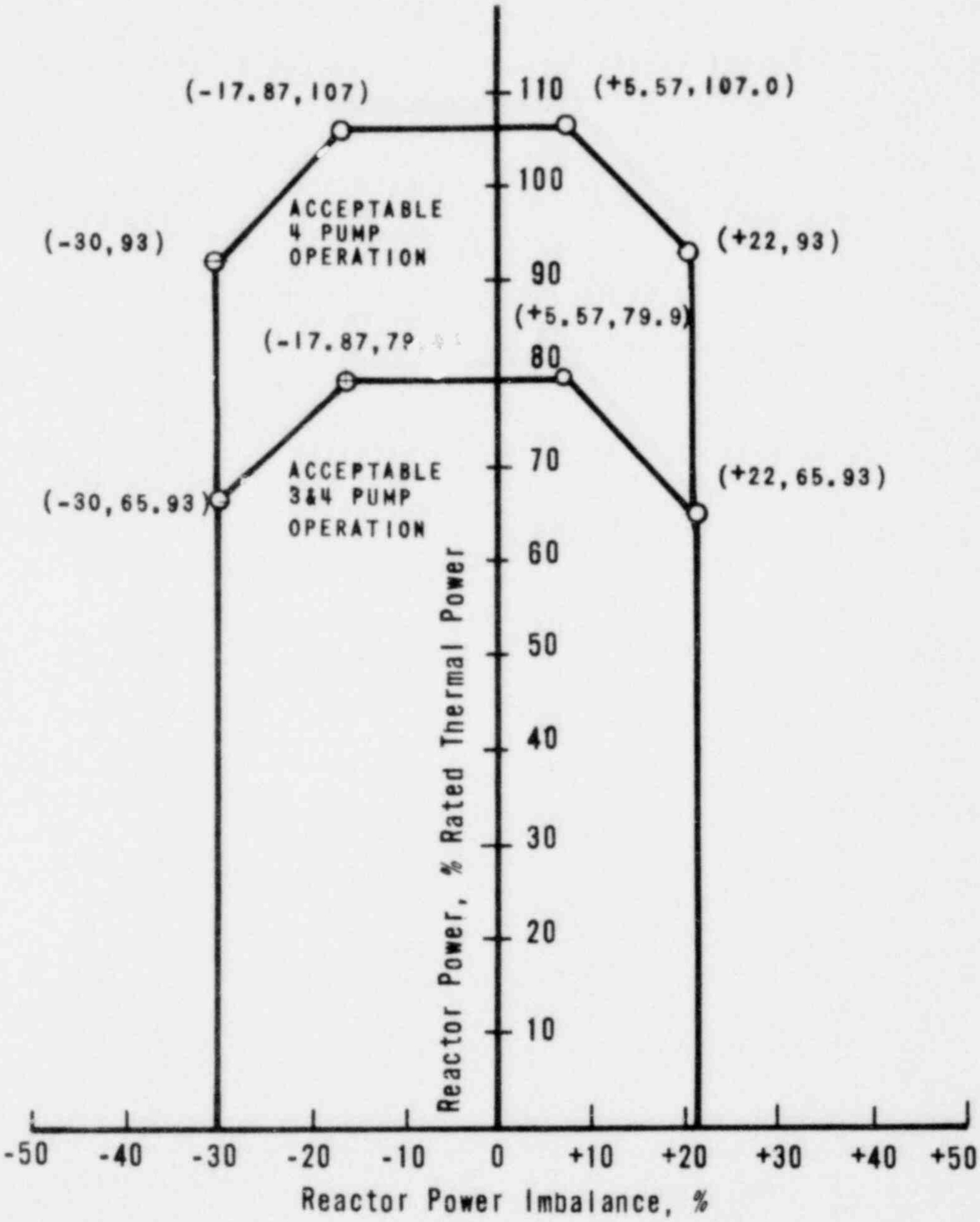


Figure 8-2. Reactor Core Safety Limits



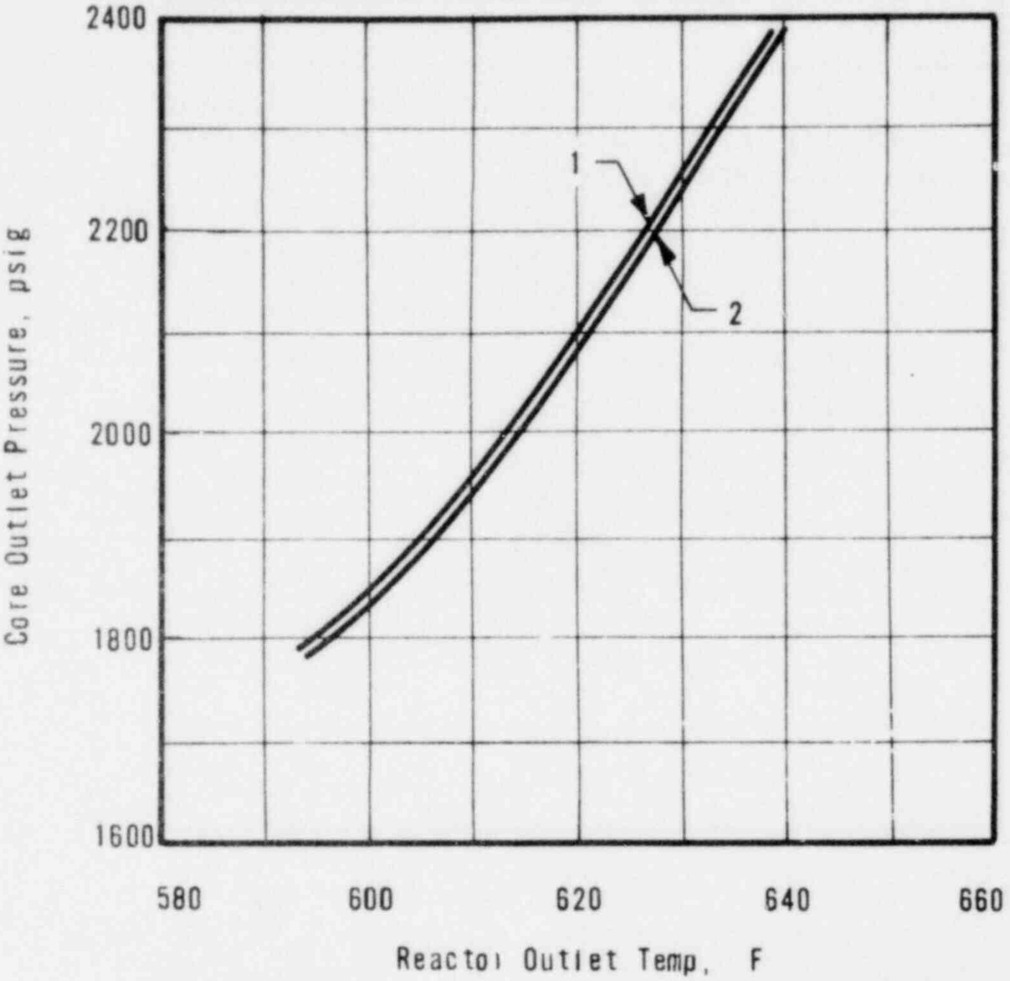
1

Figure 8-3. Reactor Trip Setpoints



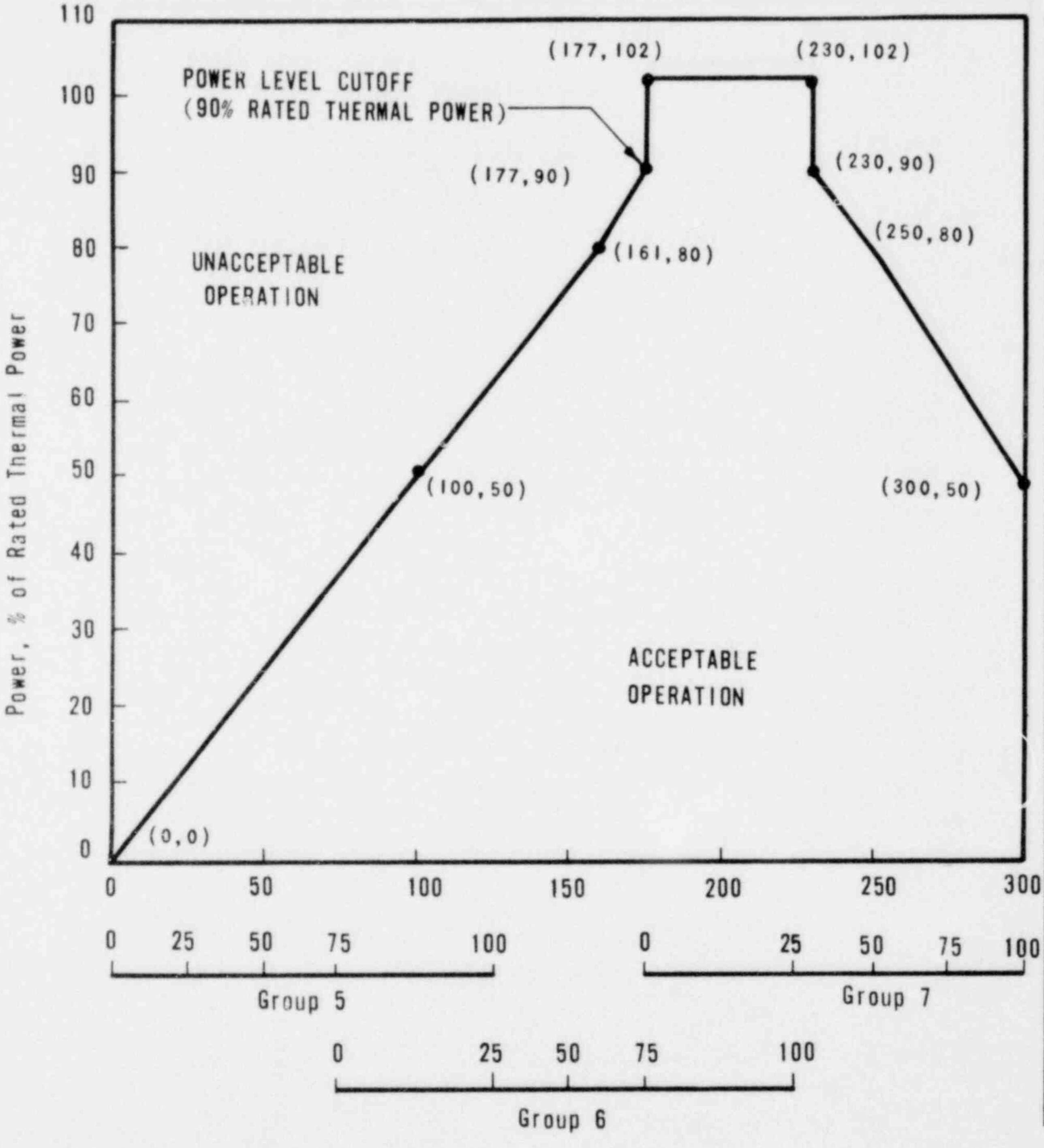
1

Figure 8-4. Pressure/Temperature Limits



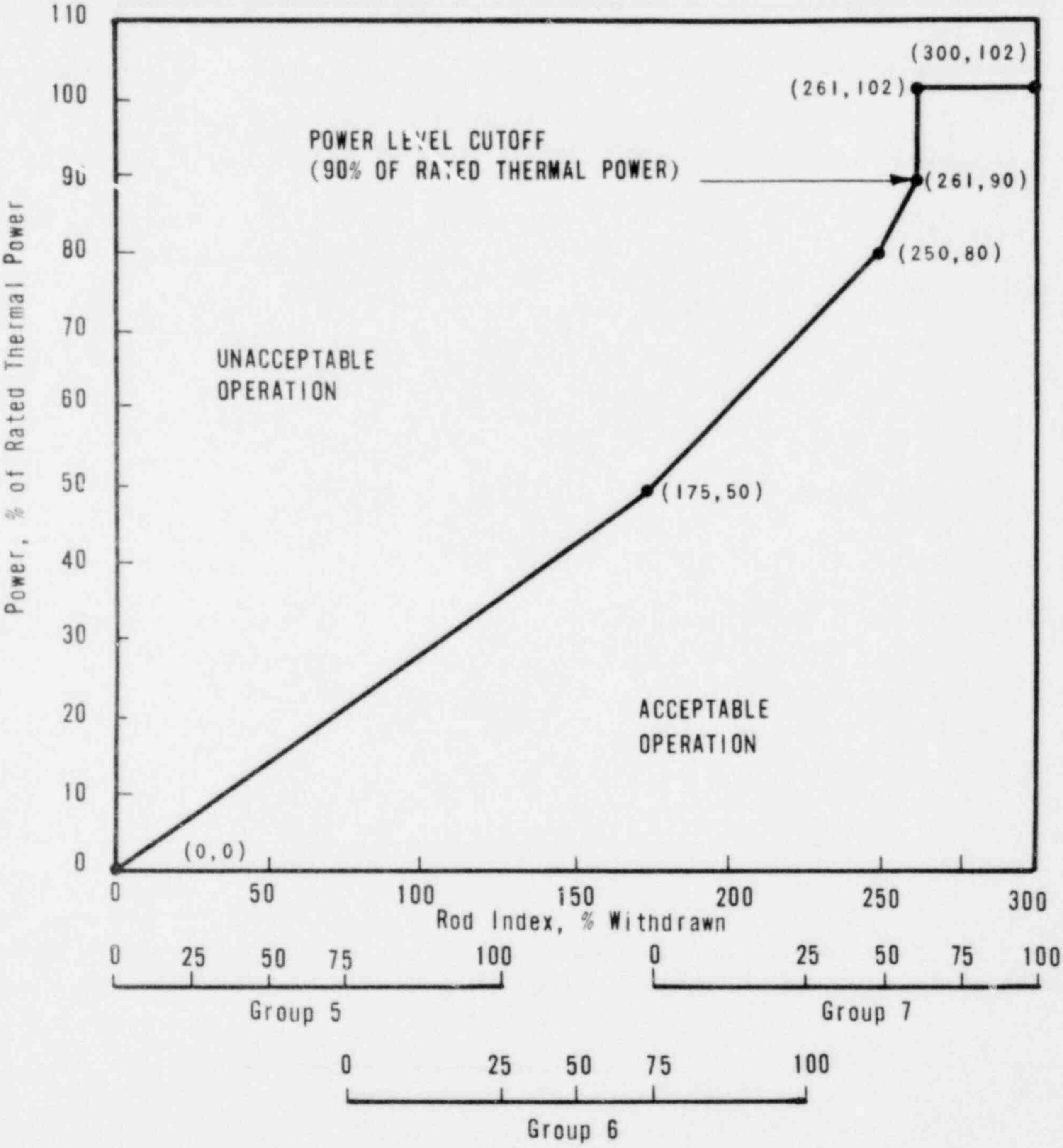
CURVE	FLOW (% DESIGN)	POWER (% OF 2568 MWt)	PUMPS OPERATING (TYPE OF LIMIT)
1	106.5	112	4 PUMPS (DNBR)
2	74.7	86.4	3 PUMPS (DNBR)

Figure 8-5. Regulating Rod Group Insertion Limits for Four-Pump Operation From 0 to 250 ± 10 EFPD



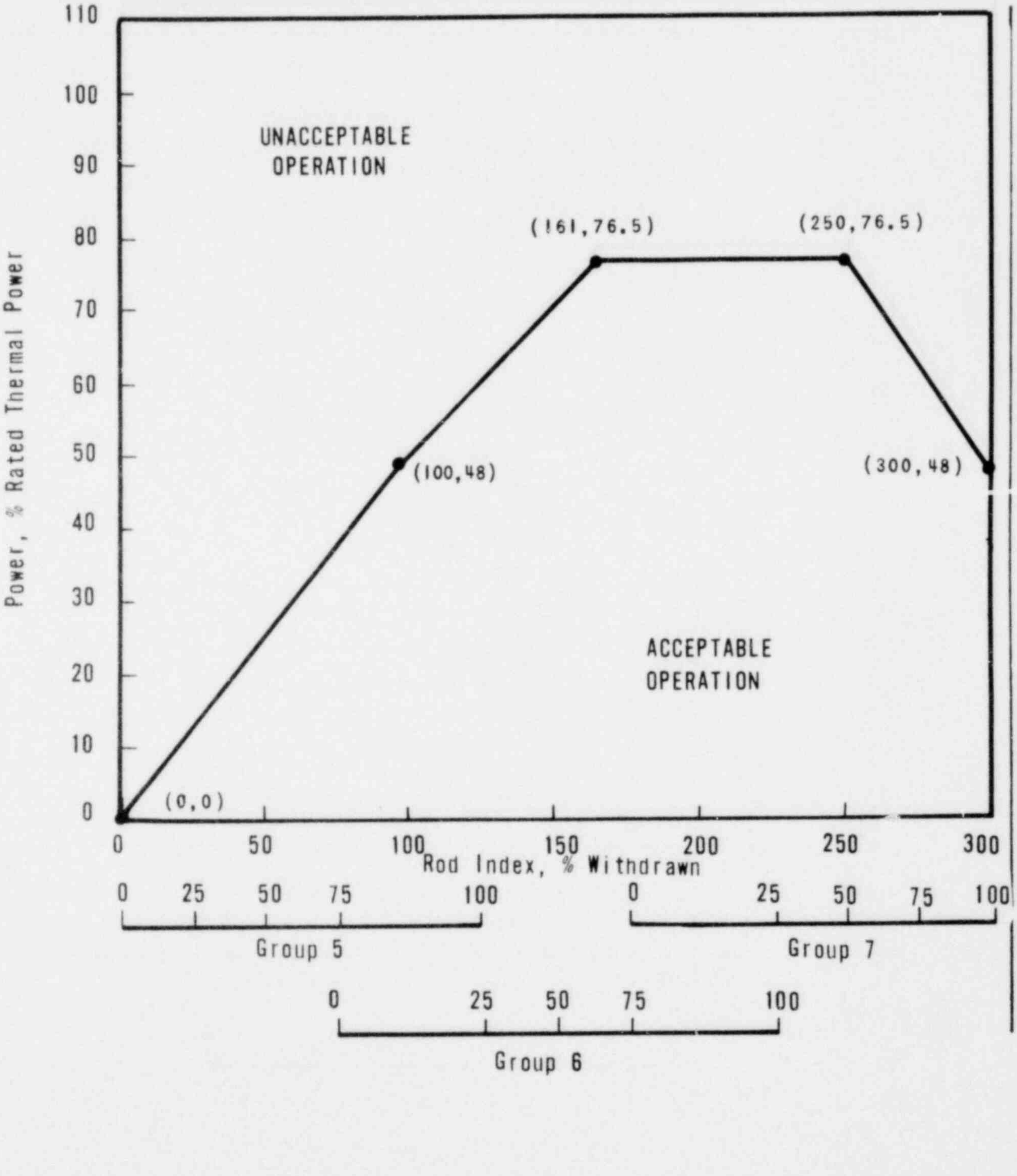
1

Figure 8-6. Regulating Rod Group Insertion Limits for Four-Pump Operation After 250 ± 10 EFPD



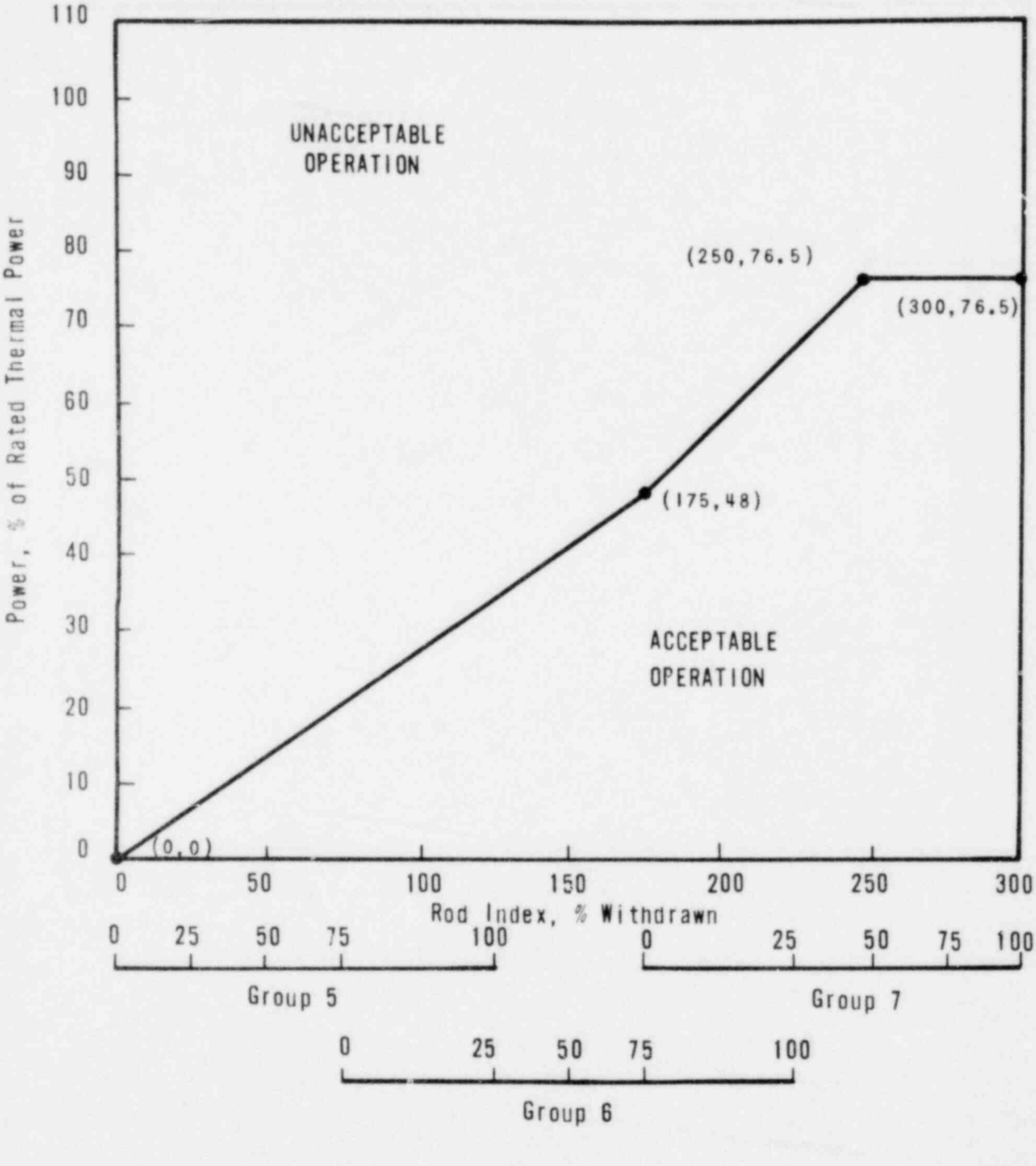
1

Figure 8-7. Regulating Rod Group Insertion Limits for Three-Pump Operation From 0 to 250 ± 10 EFPD



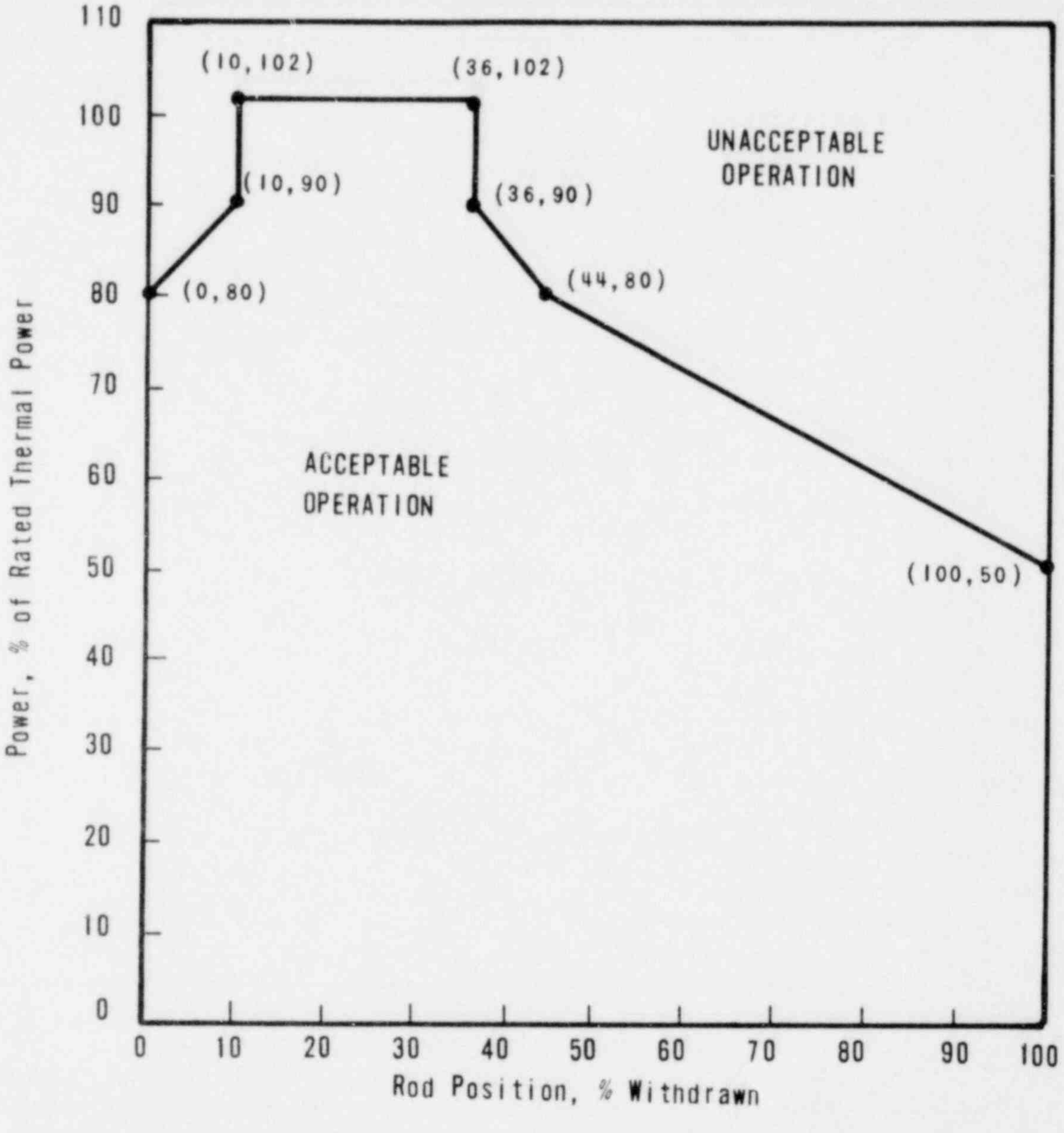
1

Figure 8-8. Regulating Rod Group Insertion Limits for Three-Pump Operation After 250 ± 10 EFPD



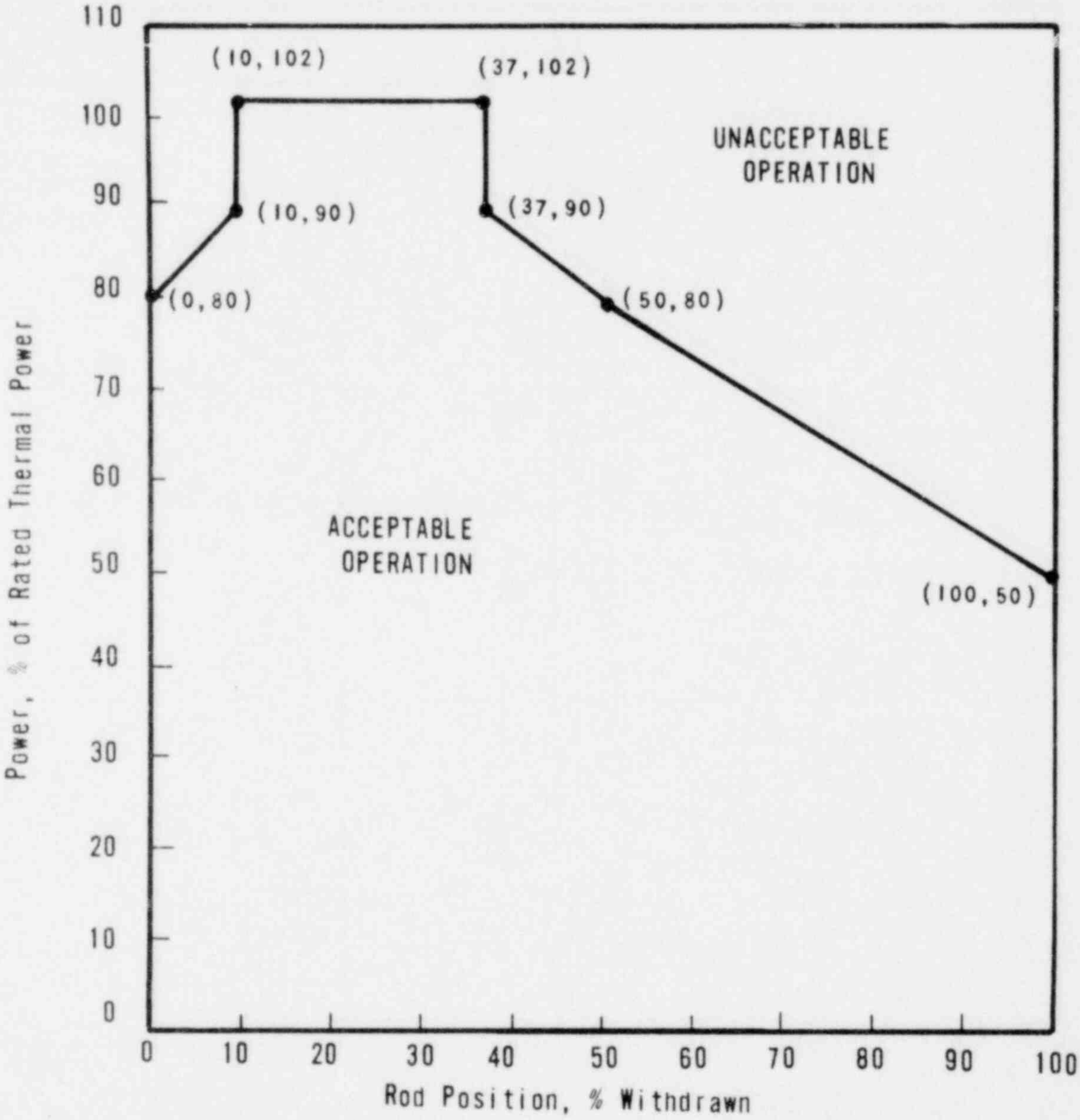
1

Figure 8-9. APSR Position Limits for 0 to 250 ± 10 EFPD, Crystal River 3



1

Figure 8-10. APSR Position Limits After 250 ± 10
EFPD, Crystal River 3



1

Figure 8-11. Axial Power Imbalance Envelope for Operation From 0 to 250 ± 10 EFPD

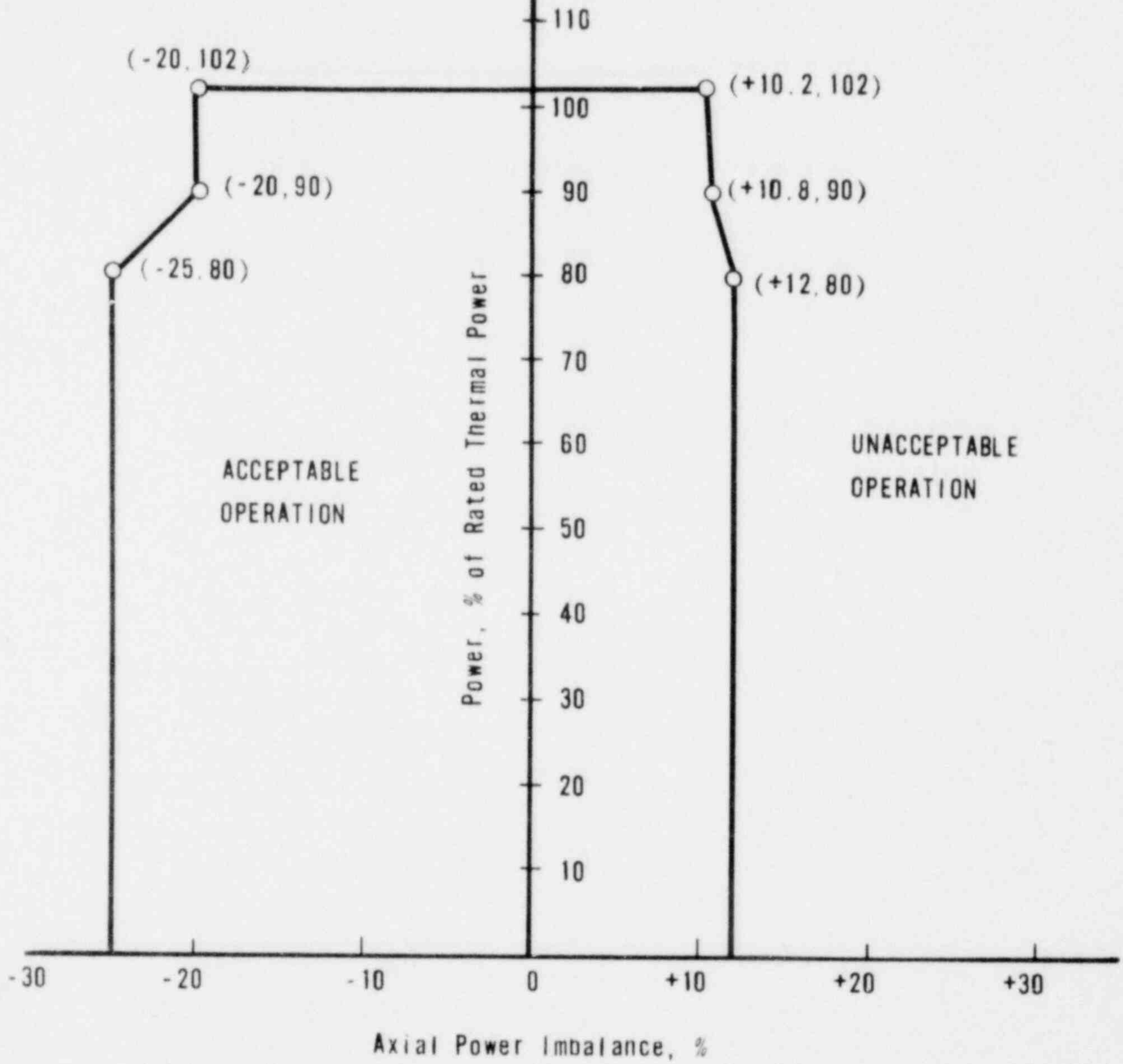
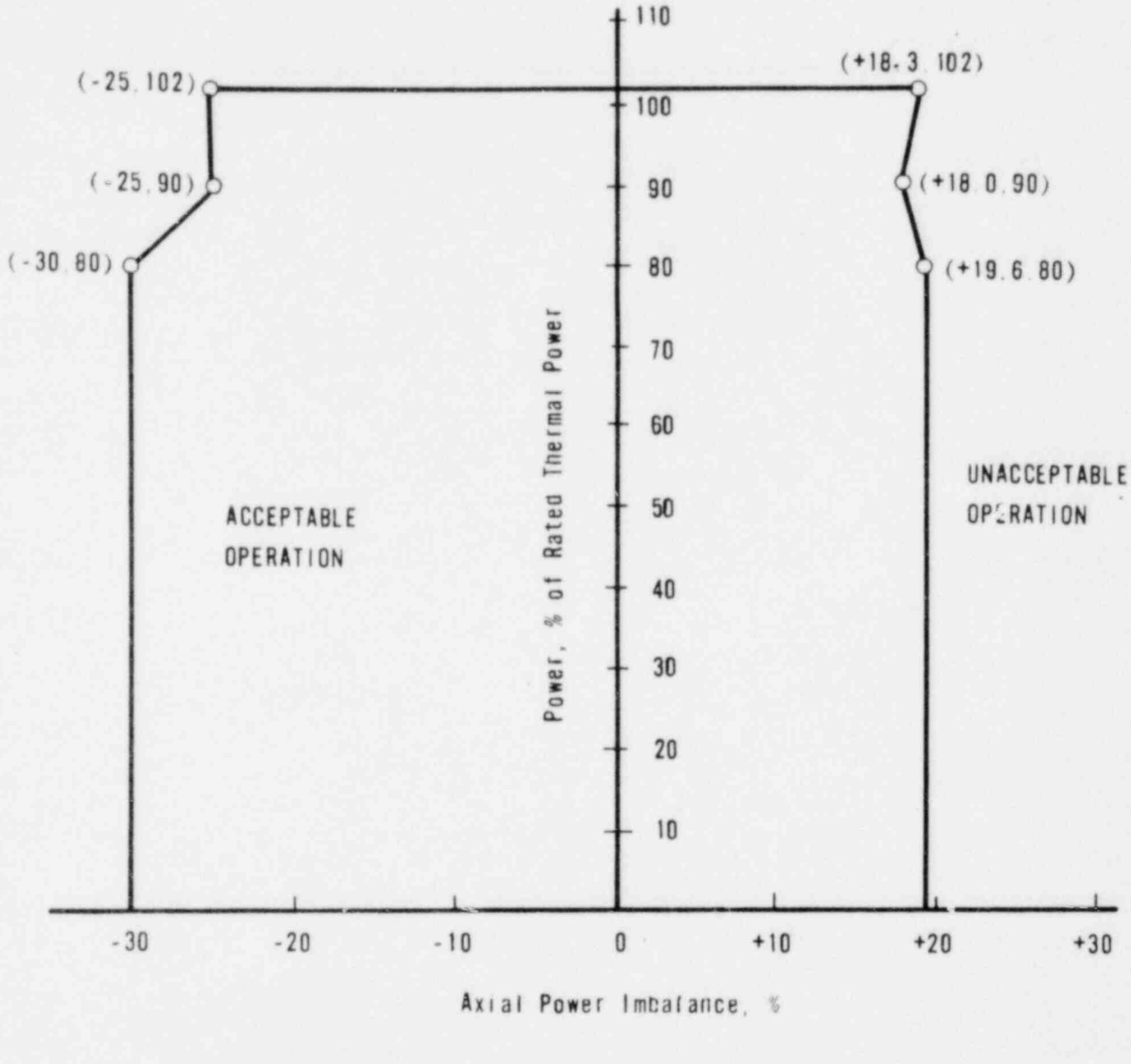


Figure 8-12. Axial Power Imbalance Envelope for Operation After 250 ± 10 EFPD



9. STARTUP PROGRAM — PHYSICS TESTING

The planned startup test program associated with core performance is outlined below. These tests verify that core performance is within the assumptions of the safety analysis and provide confirmation for continued safe operation of the unit.

9.1. Precritical Tests

9.1.1. Control Rod Trip Test

Precritical control rod drop times are recorded for all control rods at hot full-flow conditions before zero power physics testing begins. Acceptable criteria state that the rod drop time from fully withdrawn to 75% inserted shall be less than 1.66 seconds at the conditions above.

It should be noted that safety analysis calculations are based on a rod drop time of 1.40 seconds from fully withdrawn to two-thirds inserted. Since the most accurate position indication is obtained from the zone reference switch at the 75%-inserted position, this position is used instead of the two-thirds inserted position for data gathering. The acceptance cirterion of 1.40 seconds corrected to a 75%-inserted position (by rod insertion versus time correlation) is 1.66 seconds.

9.1.2. RC Flow

RC Flow with four RC pumps running will be measured at hot zero power, steady-state conditions. Acceptance criteria require that the measured flow be within allowabl limits.

9.1.3. RC Flow Coastdown

The coastdown of RC flow from the tripping of the RC pump with highest flow from four RC pumps running will be measured at hot zero power conditions. The coastdown of RC flow versus time will then be compared to the required RC flow versus time. Acceptance criteria require that the measured flow rate exceed the minimum.

9.2. Zero Power Physics Tests

9.2.1. Critical Boron Concentration

Criticality is obtained by deboration at a constant dilution rate. Once criticality is achieved, equilibrium boron is obtained and the critical boron concentration determined. The critical boron concentration is calculated by correcting for any rod withdrawal required in achieving equilibrium boron. The acceptance criterion placed on critical boron concentration is that the actual boron concentration must be within ±100 ppm boron of the predicted value.

9.2.2. Temperature Reactivity Coefficient

The isothermal temperature coefficient is measured at approximately the all-rods-out configuration and at the hot zero power rod insertion limit. The average coolant temperature is varied by first decreasing then increasing temperature by 5°F. During the change in temperature, reactivity feedback is compensated by discrete change in rod motion, the change in reactivity is then calculated by the summation of reactivity (obtained from reactivity calculation on a strip chart recorder) associated with the temperature change. Acceptance criteria state that the measured value shall not differ from the predicted value by more than $\pm 0.4 \times 10^{-4}$ $(\Delta k/k)/°F$ (predicted value obtained from Physics Test Manual curves).

The moderator coefficient of reactivity is calculated in conju ction with the temperature coefficient measurement. After the temperature coefficient has been measured, a predicted value of fuel Doppler coefficient of reactivity is added to obtain moderator coefficient. This value must not be in excess of the acceptance criteria limit of $+0.9 \times 10^{-4}$ $(\Delta k/k)/°F$.

9.2.3. Control Rod Group Reactivity Worth

Control bank group reactivity worths (groups 5, 6, and 7) are measured at hot zero power conditions using the boron/rod swap method. The boron/rod swap method consists of establishing a deboration rate in the reactor coolant system and compensating for the reactivity changes of this deboration by inserting control rod groups 7, 6, and 5 incremental steps. The reactivity changes that occur during these measurements are calculated based on Reactimeter data, and differential rod worths are obtained from the measured reactivity worth versus the change in rod group position. The differential rod worths of each of the controlling groups are then summed to obtain integral rod group worths. The acceptance criteria for the control bank group worths are as follows:

1. Individual bank 5, 6, 7 worth:

$$\left|\frac{\text{predicted value - measured value}}{\text{measured value}} \times 100\right| \leq 15$$

2. Sum of groups 5, 6, and 7:

$$\left|\frac{\text{predicted value - measured value}}{\text{measured value}} \times 100\right| \leq 10$$

9.2.4. Ejected Control Rod Reactivity Worth

After CRA groups 7, 6, and 5 have been positioned near the minimum rod insertion limit, the ejected rod is borated to 100% withdrawn and the worth obtained by adding the incremental changes in reactivity by boration.

After the ejected rod has been borated to 100% withdrawn and equilibrium boron established, the ejected rod is then swapped in versus the controlling rod group and the worth determined by the change in the previously calibrated controlling rod group position. The boron swap and rod swap values are averaged and error-adjusted to determine ejected rod worth. Acceptance criteria for the ejected rod worth test are as follows:

1. $\left|\frac{\text{predicted value - measured value}}{\text{measured value}} \times 100\right| \leq 20$

2. Measured value (error-adjusted) $\leq 1.0\% \ \Delta k/k$

The predicted ejected rod worth is given in the Physics Test Manual.

9.3. Power Escalation Tests

9.3.1. Core Power Distribution Verification at $\sim$40, 75, and 100% FP With Nominal Control Rod Position

Core power distribution tests are performed at 40, 75, and 100% full power (FP). The test at 40% FP is essentially a check on power distribution in the core to identify any abnormalities before escalating to the 75% FP plateau. Rod index is established at a nominal full power rod configuration at which the core power distribution was calculated. APSR position is established to provide a core power imbalance corresponding to the imbalance at which the core power distribution calculations were performed.

The following acceptance criteria are placed on the 40% FP test:

1. The worst-case maximum linear heat rate must be less than the LOCA limit.
2. The minimum DNBR must be greater than 1.30.
3. The value obtained from the extrapolation of the minimum DNBR to the next power plateau overpower trip setpoint must be greater than 1.30 or the extrapolated value of imbalance must fall outside the RPS power/imbalance/flow trip envelope.
4. The value obtained from the extrapolation of the worst-case maximum linear heat rate to the next power plateau overpower trip setpoint must be less than the fuel melt limit or the extrapolated value of imbalance must fall outside the RPS power/imbalance/flow trip envelope.
5. The quadrant power tilt shall not exceed the limits specified in the Technical Specifications.
6. The highest measured and predicted radial peaks shall be within the following limits:

$$\left| \frac{\text{predicted value - measured value}}{\text{measured value}} \times 100 \right| \leq 8$$

7. The highest measured and predicted total peaks shall be within the following limits:

$$\left| \frac{\text{predicted value - measured value}}{\text{measured value}} \times 100 \right| \leq 12$$

Items 1, 2, 5, 6, and 7 above are established to verify core nuclear and thermal calculational models, thereby verifying the acceptability of data from these models for input to safety evaluations.

Items 3 and 4 establish the criteria whereby escalation to the next power plateau may be accomplished without exceeding the safety limits specified by the safety analysis with regard to DNBR and linear heat rate.

The power distribution tests performed at 75 and 100% FP are identical to the 40% FP test except that core equilibrium xenon is established prior to the 75 and 100% FP tests. Accordingly, the 75 and 100% FP measured peak acceptance criteria are as follows:

1. The highest measured and predicted radial peaks shall be within the following limits:

$$\left|\frac{\text{predicted value} - \text{measured value}}{\text{measured value}} \times 100\right| \leq 5$$

2. The highest measured and predicted total peaks shall be within the following limits:

$$\left|\frac{\text{predicted value} - \text{measured value}}{\text{measured value}} \times 100\right| \leq 7.5$$

9.3.2. Incore Vs Excore Detector Imbalance Correlation Verification at ∿40% FP

Imbalances are set up in the core by control rod positioning. Imbalances are read simultaneously on the incore detectors and excore power range detectors for various imbalances. The excore detector offset versus incore detector offset slope must be at least 1.15. If the excore detector offset versus incore detector offset slope criterion is not met, gain amplifiers on the excore detector signal processing equipment are adjusted to provide the required gain.

9.3.3. Temperature Reactivity Coefficient at ∿100% FP

The average reactor coolant temperature is decreased and then increased by about 5°F at constant reactor power. The reactivity associated with each temperature change is obtained from the change in the controlling rod group position. Controlling rod group worth is measured by the fast insert/withdraw method. The temperature reactivity coefficient is calculated from the measured changes in reactivity and temperature.

Acceptance criteria state that the moderator temperature coefficient shall be negative.

9.3.4. Power Doppler Reactivity Coefficient at ∿100% FP

Reactor power is decreased and then increased by about 5% FP. The reactivity change is obtained from the change in controlling rod group position. Control rod group worth is measured using the fast insert/withdraw method. Reactivity corrections are made for changes in xenon and reactor coolant temperature that occur during the measurement. The power doppler reactivity coefficient is calculated from the measured reactivity change, adjusted as stated above, and the measured power change.

The predicted value of the power doppler reactivity coefficient is given in the Physics Test Manual. Acceptance criteria state that the measured value shall be more negative than -0.55×10^{-4} $(\Delta k/k)/\%$ FP.

9.4. Procedure for Failure to Meet Acceptance Criteria

Florida Power reviews the results of all startup tests to ensure that all acceptance criteria are met. If the review of the test indicates that the results are well within the acceptance criteria, no further evaluation is conducted. If the review indicates that the results are approaching or close to the acceptance criteria limits, further evaluation of that particular test or other supporting tests is performed to look for trends. This evaluation will determine whether additional support data are required to discover any abnormal conditions. If acceptance criteria for any test are not met, an evaluation is performed before the test program is continued. This evaluation is performed by site test personnel with participation by Babcock & Wilcox technical personnel as required. Further specific actions depend on evaluation results. These actions can include repeating the tests with more detailed attention to test prerequisites, added tests to search for anomalies, or design personnel performing detailed analyses of potential safety problems because of parameter deviation. Power is not escalated until evaluation shows that plant safety will not be compromised by such escalation.

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