RELOAD REPORT

Catawba Unit 2 Cycle 6

Duke Power Company
Nuclear Generation Department
Nuclear Engineering Section

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1. INTRODUCTION AND SUMMARY

This report justifies the operation of the sixth cycle of Catawba Nuclear Station, Unit 2 at the rated core power level of 3411 MW_{th}. Included are the required analyses as outlined in the USNRC document "Guidance for Proposed License Amendments Relating to Refueling," July 1975.

Cycle 6 for Catawba Unit 2 is the third Catawba cycle for which the reload fuel is supplied by B&W Fuel Company (BWFC). The incoming Batch 8 fuel assemblies are designated as Mark-BW. To support implementation of Mark-BW fuel in the McGuire and Catawba nuclear stations, Duke Power Company (DPC) has developed new methods and models to analyze the plants during normal and off-normal operation. The thermal-hydraulic analytical models are documented in topical report DPC-NE-3000P (Reference 11) for non-LOCA transients and BAW-10174 (Reference 13) for LOCA. Portions of the analytical methodology are documented in topical report DPC-NE-3001-PA (Reference 12) and DPC-NE-2004PA (Reference 8). The remaining Final Safety Analysis Report (FSAR) Chapter 15 non-LOCA system transient analysis methodology is documented in D .-NE-3002-A (Reference 16). The FSAR Chapter 15 LOCA system transient analysis methodology is documented in Reference 13. Approval of these topical reports has been completed.

Section 2 of this report is the operating history for fue Catawba Unit 2. Section 3 is a general description of the reacto e, and the fuel system design is provided in Section 4. Reactor system parameters and conditions are summarized in Sections 5, 6, and 7. Changes to the Technical Specifications, Core Operating Limits Report (COLR), and FSAR are provided in Section 8.

All of the accidents analyzed in the FSAR (Reference 1) have been reviewed for Cycle 6 operation, and many of the FSAR Chapter 15 system thermal-hydraulic accident analyses sensitive to reload core physics parameters have been reanalyzed using Duke Power methodology. These analyses are the same as those performed for the McGuire Unit 1, Cycle 8 and Catawba Unit 1, Cycle 7 reloads. Several bounding transients were analyzed in detail to demonstrate the capability of DPC calculational techniques. The results of these analyses were reported in DPC-NE-3001-PA. For the other reanalyzed transients, the approved methodology is documented in DPC-NE-3002-A. A further discussion of accident analysis is presented in Section 7 of this report. Other reanalyzed transients are included in Section 8 of this report.

Amendment Number 74 (Unit 1) and Amendment Number 68 (Unit 2) to the Catawba Nuclear Station Facility Operating License allow the removal of cycle-specific core parameter limits from Technical Specifications and require that these limits be included in a Core Operating Limits Report (COLR). The Core Operating Limits Report is submitted to the NRC upon issuance and does not require approval prior to implementation. Changes to the operating limits are made via the Core Operating Limits Report.

The Technical Specifications have been reviewed, and the modifications for Cycle 6 are justified in this report. Based on the analyses performed, it has been concluded that Catawba Unit 2 Cycle 6 can be safely operated at a core power level of 3411 MW_{th}.

2. OPERATING HISTORY

The current operating cycle for Catawba Unit 2 is Cycle 5, which achieved criticality on December 22, 1991 and reached 100% full power on December 28, 1991. Cycle 5 is scheduled to shut down in January 1993 after 375 EFPD. Catawba Unit 2 Cycle 5 and previous cycles operated entirely with fuel assemblies of Westinghouse design.

Catawba Unit 2, Cycle 6 is the first Catawba Unit 2 reload to contain a full reload batch of Mark-BW fuel assemblies (FAs). Catawba Unit 1 has had two batches of Mark-BW fuel assemblies, the first installed for Cycle 6 and the second for Cycle 7. Catawba Unit 2, Cycle 6 is scheduled to start up on April, 1, 1993 at a rated power level of 3411 MWt and has a design cycle length of 380 EFPD. No operating anomalies have occurred during previous cycle operations that would adversely affect fuel performance in Cycle 6.

3. GENERAL DESCRIPTION

The Catawba Unit 2 reactor core is described in detail in Chapter 4 of the FSAR (Reference 1). The core consists of 193 assemblies, each of which is a 17-by-17 array containing 264 fuel rods, 24 guide tubes, and 1 incore instrument tube. There are 117 burned FAs in the core, all of the Westinghouse Optimized Fuel Assembly design, and 76 fresh FA's consisting of the Mark-BW design (Reference 2). The fuel rod outside diameters are 0.360 and 0.374 inch, and the clad thicknesses are 0.0225 and 0.024 inch for the OFA and Mark-BW designs, respectively. The Mark-BW fuel consists of dished end, cylindrical pellets of uranium dioxide, (See Table 4-1 for data). The nominal fuel loadings are 423.5 kg of uranium per fuel assembly for the Westinghouse fuel in batches 5A, 6A and 7A; and 456.2 kg of uranium per fuel assembly for the Mark-BW fuel in batch 8A. The initial enrichments of batches 5A, 6A and 7A were 3.60, 3.50, and 3.75 wt% U_{235}. The design enrichment of the fresh batch 8A (Mark-BW) is 3.75 wt% U_{235}.

The 8 batch 5A, 33 batch 6A, and 76 batch 7A assemblies will be shuffled to new locations. One batch 6A FA will be inserted into the center assembly location. The 76 fresh batch 8A assemblies will be loaded into the core in a symmetric checkerboard pattern. Figure 3-1 shows the locations of the fresh fuel assemblies and the previous cycle location of the burned fuel assemblies. Figure 3-2 is a quarter core map showing the burnup and region number with corresponding initial enrichments of each assembly at the beginning of Cycle 6.

Cycle 6 will be operated in a feed-and-bleed mode. Core reactivity is controlled by 53 rod cluster control assemblies (RCCAs), 640 Mark-BW burnable absorbers, and soluble boron shim. Figure 3-3 shows the Cycle 6 fresh fuel locations with the Mark-BW BPRA clusters and number of pins (loaded with 3.0 wt% B_4C in Al_2O_3) in each location.

FIGURE 3-1
CORE LOADING PATTERN FOR CATAWBA UNIT 2 CYCLE 6

PREVIOUS CORE LOCATIONS
REGION NUMBERS

				J-04 7	Feed 8	F-05 7	F-15 7	K-05 7	Feed 8	G-04 7				
		L-04 7	K-13 7	Feed 8	J-14 7	Feed 8	H-09 7	Feed 8	G-14 7	Feed 8	F-13 7	E-04 7		
	M-05 7	H-05 7	Feed 8	B-05 7	Feed 8	D-13 7	H-15 7	M-13 7	Feed 8	P-05 7	Feed 8	E-08 7	D-05 7	
	C-06 7	Feed 8	B-10 6	Feed 8	G-10 6	Feed 8	H-03 6	Feed 8	J-10 6	Feed 8	K-14 6	Feed 8	N-06 7	
M-07 7	Feed 8	L-14 7	Feed 8	R-11 5	Feed 8	P-12 6	K-15 7	B-12 6	Feed 8	L-01 5	Feed 8	E-14 7	Feed 8	D-07 7
Feed 8	B-07 7	Feed 8	F-09 6	Feed 8	J-01 6	Feed 8	L-15 5	Feed 8	A-07 6	Feed 8	K-09 6	Feed 8	P-07 7	Feed 8
L-10 7	Feed 8	C-12 7	Feed 8	D-02 6	Feed 8	R-07 6	Feed 8	G-01 6	Feed 8	M-02 6	Feed 8	N-12 7	Feed 8	E-10 7
A-0[illegible] 7	G-08 7	A-08 7	N-08 6	A-10 7	A-11 5	Feed 8	B-06 6	Feed 8	R-05 5	R-06 7	C-08 6	R-08 7	J-08 7	R-10 7
L-06 7	Feed 8	C-04 7	Feed 8	D-14 6	Feed 8	J-15 6	Feed 8	A-09 6	Feed 8	M-14 6	Feed 8	N-04 7	Feed 8	E-06 7
Feed 8	B-09 7	Feed 8	F-07 6	Feed 8	R-09 6	Feed 8	E-01 5	Feed 8	G-15 6	Feed 8	K-07 6	Feed 8	P-09 7	Feed 8
M-09 7	Feed 8	L-02 7	Feed 8	E-15 5	Feed 8	P-04 6	F-01 7	B-04 6	Feed 8	A-05 5	Feed 8	E-02 7	Feed 8	D-09 7
	C-10 7	Feed 8	F-02 6	Feed 8	G-06 6	Feed 8	H-13 6	Feed 8	J-06 6	Feed 8	P-06 6	Feed 8	N-10 7	
	M-11 7	L-08 7	Feed 8	B-11 7	Feed 8	D-03 7	H-01 7	M-03 7	Feed 8	P-11 7	Feed 8	H-11 7	D-11 7	
		L-12 7	K-03 7	Feed 8	J-02 7	Feed 8	H-07 7	Feed 8	G-02 7	Feed 8	F-03 7	E-12 7		
				J-12 7	Feed 8	F-11 7	K-01 7	K-11 7	Feed 8	G-12 7				
R	P	N	M	L	K	J	H	G	F	E	D	C	B	A

Z-ZZ YY	CYCLE 5 LOCATION REGION NUMBER

FIGURE 3-2

ENRICHMENT AND BOC BURNUP DISTRIBUTION FOR CATAWBA 2 CYCLE 6

	H	G	F	E	D	C	B	A
8	32359.1 33803.6 6A	.0 .0 8A	30617.5 31884.8 5A	12374.2 16850.4 7A	30334.0 16850.4 6A	13196.1 17456.0 7A	19342.0 20531.8 7A	12374.2 16850.4 7A
9	.0 .0 8A	27803.5 30897.8 6A	.0 .0 8A	27963.7 33401.6 6A	.0 .0 8A	17603.9 20787.6 7A	.0 .0 8A	19729.6 21018.6 7A
10	30617.5 31884.8 5A	.0 .0 8A	27803.5 30897.8 6A	.0 .0 8A	29046.5 33395.0 6A	.0 .0 8A	18824.9 20794.5 7A	.0 .0 8A
11	12374.2 16850.4 7A	27963.7 33401.6 6A	.0 .0 8A	30617.5 32466.5 5A	.0 .0 8A	16960.3 20604.2 7A	.0 .0 8A	20161.3 21234.5 7A
12	30334.0 34090.8 6A	.0 .0 8A	29046.5 33395.0 6A	.0 .0 8A	32359.1 36717.9 6A	.0 .0 8A	20413.2 21436.7 7A	
13	13196.1 17456.0 7A	17603.9 20787.6 7A	.0 .0 8A	16930.3 20604.2 7A	.0 .0 8A	20054.3 21143.9 7A	20573.7 21735.3 7A	
14	19342.0 20531.8 7A	.0 .0 8A	18824.9 20794.5 7A	.0 .0 8A	20413.2 21436.7 7A	20573.7 21735.3 7A	Average Maximum Region #	
15	12374.2 16850.4 7A	19729.6 21018.8 7A	.0 .0 8A	20161.3 21234.5 7A				

REGION	ENRICHMENT w/o U-235	CYCLES BURNED	NUMBER OF ASSEMBLIES	BOC BURNUP MWD/MTU
5A	3.60	3	8	30618
6A	3.50	2	33	29141
7A	3.75	1	76	18201
8A	3.75	0	76	0
CORE			193	13419

FIGURE 3-3
CATAWBA UNIT 2 CYCLE 6
BURNABLE ABSORBER AND SOURCE ASSEMBLY LOCATIONS

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1						0				0					
2					4		12		12		4				
3				4		16				16		4			
4			4		12		12		12		12		4		
5		4		12		8				8		12		4	
6	0		16		8		8		8		8		16		0
7		12		12		8		8		8		12		12	
8			SS				8		8				SS		
9		12		12		8		8		8		12		12	
10	0		16		8		8		8		8		16		0
11		4		12		8				8		12		4	
12			4		12		12		12		12		4		
13				4		16				16		4			
14					4		12		12		4				
15						0				0					

NUMBER OF MkBW-BP PINS/ASSEMBLY	NUMBER OF BACKPLATE ASSEMBLIES
4	16
8	20
12	24
16	8
Total = 640 pins	Total = 68 backplates

SS = Secondary Source location

All BP Pins are 3.0 wt% $B_4C-Al_2O_3$ MkBW BP's.

4. FUEL SYSTEM DESIGN

4.1 Fuel Assembly Mechanical Design

The Catawba 2 Cycle 6 core will include 76 fresh Mark-BW fuel assemblies with an enrichment of 3.75 wt % U_{235}. The re-inserted fuel assemblies in Cycle 6 will be 117 Westinghouse optimized fuel assemblies. The Mark-BW 17 x 17 Zircaloy spacer grid fuel assembly is similar in design to the Westinghouse standard fuel assembly, Reference 2. The fuel rod outer diameter and guide tube top section, dashpot diameters, and instrument tube diameter are the same as the Westinghouse standard 17 x 17 design. The unique features of the Mark-BW design include the Zircaloy intermediate spacer grids, the spacer grid restraint system, and the use of Zircaloy grids with the standard lattice design. Mark-BW fuel design dimensions and parameters for Catawba 2 Cycle 6 are listed in Table 4-1.

4.2 Fuel Rod Design

Duke Power Company has performed generic Mark-BW mechanical analyses using the approved methodologies described in Reference 3. The generic analyses envelope the Cycle 6 design as discussed below.

4.2.1 Fuel Rod Cladding Collapse

The fuel rods were analyzed for creep collapse using the CROV computer code, Reference 4, and the methodology described in Reference 3. Internal pin pressures and clad temperatures used in CROV were calculated using the TACO2 computer code, Reference 5. A conservative power history which envelopes the predicted peaking for the Catawba 2 Cycle 6 fuel was analyzed. The collapse time was conservatively determined to be greater than the maximum predicted residence time for the Mark-BW fuel (Table 4-1).

4.2.2 Fuel Rod Cladding Stress

As described in Reference 3, Duke Power Company has performed a conservative generic fuel rod cladding stress analysis using the ASME pressure vessel stress intensity limits as guidelines. The maximum cladding stress intensities were shown to be within the ASME limits under all loading conditions. The generic Mark-BW cladding stress analysis includes the following conservatisms:

- Conservative cladding dimensions.
- High external pressure.
- Low internal pin pressure.
- High radial temperature gradient through the clad.

4.2.3 Fuel Rod Cladding Strain

Diametral cladding strain resulting from a local power transient is limited to 1.0%. A generic cladding strain analysis was performed using TACO2 to determine the maximum allowable local power change that the fuel could experience without exceeding the 1.0% limit. The maximum calculated local power change resulting from a worst case core maneuvering scenario was compared with the maximum allowable power change. This comparison demonstrated that margin exists to the 1.0% strain limit.

4.3 Thermal Design

The thermal performance of the Mark-BW fuel assemblies was evaluated using TACO2 with the methodology given in Reference 3. The nominal fuel parameters used to determine the generic linear heat rate to centerline melt (LHRTM) limits are given in Table 4-1. The LHRTM analysis included the following bounding conservatisms:

- Maximum gap based on as-fabricated pellet and clad data.
- Maximum incore densification based on resinter test results.

The maximum predicted Mark-BW assembly burnup at EOC 6 (in Batch 8) is 18,992 MWD/MTU and the maximum predicted fuel rod burnup (in Batch 8) is 20,230 MWD/MTU. The fuel rod internal pressure has been evaluated for the highest burnup rod using TACO2 and a conservative pin power history. The maximum internal pin pressure is less than the nominal Reactor Coolant System pressure of 2250 psia.

4.4 Material Design

The Mark-BW fuel is not unique in concept, nor does it utilize different component materials. Thus, the chemical compatibility of all possible fuel-cladding-coolant-assembly interactions for the fresh fuel is identical to that of the present fuel.

4.5 Operating Experience

Experience with the Mark-BW 17 x 17 fuel assembly design started with the irradiation of four lead assemblies in McGuire 1 Cycle 5. McGuire 1 Cycle 7 was the third cycle of irradiation for three of the assemblies and the maximum predicted assembly burnup is 42,756 MWD/MTU. The lead assemblies were examined after each cycle and the fuel assembly bow, twist, growth, and holddown spring set were all within nominal bounds. Four other Mark-BW lead assemblies underwent their first cycle of irradiation in Trojan Cycle 13.

Catawba 2 Cycle 6 will be the fifth complete reload batch of Mark-BW 17 x 17 fuel. The first complete reload batch finished operation in Catawba 1 Cycle 6 in June 1992. The second, third and fourth batch are operating in both McGuire units and in Catawba 1 Cycle 7.

Table 4-1. Mark-BW Fuel Design Parameters and Dimensions

	Batch 8
Nominal fuel rod OD, in.	0.374
Nominal fuel rod ID, in.	0.326
Nominal active fuel length, in.	144.0
Nominal fuel pellet OD, in.	0.3195
Fuel pellet initial density, % TD	96.0
Initial fuel enrichment, wt. % U_{235}	3.75
Estimated residence time EOC 6, EFPH	9,120
Cladding collapse time, EFPH	>18,100
Nominal linear heat rate (LHR), kW/ft	5.43
Ave. fuel temperature @ nom. LHR, deg F	1360
Minimum LHR to melt, kW/ft	
0-1000 MWD/MTU	21.5
> 1000 MWD/MTU	21.8

5. NUCLEAR DESIGN

5.1 Physics Characteristics

Table 5-1 provides the core physics parameters for Cycles 5 and 6. The values for Cycle 6 were generated using the methodology described in DPC-NF-2010A (Reference 6) and are valid for the design cycle length (380 EFPD ± 10 EFPD). The values for Cycle 5 were generated by Westinghouse. Figure 5-1 illustrates a representative relative power distribution for the beginning of Cycle 6 at full power. This case was calculated as part of the design depletion using the PDQ07 methodology as described in DPC-NF-2010A (Reference 6). This case assumed equilibrium xenon and rods in the All Rods Out (ARO) position.

During verification of the control rod insertion limits specified in the COLR, calculated ejected rod worths and their adherence to acceptance criteria were considered. The adequacy of the shutdown margin with Cycle 6 stuck rod worths is demonstrated in Table 5-2. The shutdown margin calculations include a 10% uncertainty on available rod worth. The shutdown margin calculation at the end of Cycle 6 was analyzed at 390 EFPD.

5.2. Changes in Nuclear Design

No core design changes have been implemented in Cycle 6 which will impact the nuclear design parameters. The Cycle 6 physics parameters appearing in this report were calculated with the PDQ07 and EPRI-NODE-P codes. These codes and methods were approved by the NRC as documented in Reference 6. The PDQ07 calculations were performed in two dimensions; the EPRI-NODE-P calculations were performed in three dimensions. The Reactor Protection System (RPS) limits and operational limits for the core were verified by analyses for this fuel cycle using methodology approved by the NRC in Reference 7 and are provided in the Technical Specifications and the COLR. Revisions to these documents for Cycle 6 are presented in Section 8.

Table 5.1 Physics Parameters[a] Catawba 2 Cycles 5 and 6

	Cycle 5	Cycle 6
Design cycle length, EFPD	375	380
Design cycle burnup, MWD/MTU	15650	15390
Design average core burnup - EOC, MWD/MTU	28668	28831
Design initial core loading, MTU	81.7355	84.2207
Critical boron - BOC,ppmb, no Xe[b]		
HZP, ARO	1621	1713
HFP, ARO	1502	1569
Critical boron - EOC,ppmb		
HZP, No Xe, ARO	489	525
HFP, Eq Xe, ARO	0	0
Total Control Rod Worths - HZP, pcm		
BOC	6700	6996
EOC[c]	6981	7495
Max ejected rod worth(d) - HZP, pcm		
BOC	<780	373
EOC[c]	<900	552
Max stuck rod worth - HZP, pcm		
BOC	830	1194
EOC[c]	850	1202
Power deficit - HZP to HFP, pcm		
BOC	-1560	-1690
EOC[c]	-2920	-3018
Doppler coeff - HFP, pcm/°F		
BOC, no Xe	-0.91	-1.16
EOC[c], eq Xe	-2.90	-1.45
Moderator coeff - HFP, pcm/°F		
BOC, no Xe	-3.56	-2.93
EOC[c], eq Xe, 0 PPMB	-33.06	-32.50
Boron worth - HFP, pcm/ppmb		
BOC	-8.37	-7.89
EOC[c]	-10.34	-9.22

Table 5.1 Physics Parameters[a] Catawba 2 Cycles 5 and 6 (cont)

	Cycle 5	Cycle 6
Equilibrium Xenon worth - HFP, pcm		
BOC (4 EFPD)	2654	2604
EOC	2990	2816
Effective delayed neutron fraction - HFP		
BOC	0.00609	0.006242
EOC	>0.00440	0.005228

(a) Cycle 5 values obtained from Westinghouse analyses and cycle 6 values were obtained from Duke Power Company analyses.

(b) HZP denotes hot zero power (core average 557°F Tavg); HFP denotes hot full power (590.8°F vessel Tavg).

(c) EOC physics parameters calculated at design EOC plus 10 EFPD.

(d) Ejected rod worth for banks D, C, and B inserted to HZP RIL.

Table 5-2. Shutdown Margin Calculation for Catawba 2 Cycle 6

Control Rod Worth	BOC (PCM)	EOC[a] (PCM)
1. All rods inserted (ARI), HZP	6996	7495
2. ARI less most reactive stuck rod, HZP	5802	6293
3. Less 10% uncertainty	5222	5664
Required Rod Worth		
4. Rod insertion allowance (RIA)[b]	246	342
5. Power defect, HFP to HZP[b]	1992	3320
6. Shutdown margin (total available worth minus total required worth)	2984	2002

NOTE: Required shutdown margin is 1300 PCM.

(a) EOC physics parameters calculated at 390 EFPD, i.e., design EOC plus 10 EFPD.

(b) The rod insertion allowance and power defect include penalties to account for the effects of transient xenon conditions.

Figure 5-1: BOC (4 EFPD), Cycle 6 Two-Dimensional Relative Power Distribution - HFP, Equilibrioum Xenon

	H	G	F	E	D	C	B	A
8	0.9329 0.9460	1.2658 1.3456	0.9602 1.0004	1.1420 1.2410	0.9419 0.9611	1.2488 1.3421	1.0444 1.1829	0.6427 0.9584
9	1.2658 1.3456	1.0063 1.0775	1.2778 1.3646	0.9751 1.0887	1.2329 1.3738	1.2135 1.3149	1.1550 1.3682	1.6180 0.9024
10	0.9602 1.0004	1.2778 1.3646	1.0178 1.0796	1.2759 1.3638	0.9876 1.0668	1.2580 1.3828	1.0889 1.2459	0.7740 1.0910
11	1.1420 1.2410	0.9751 1.0887	1.2759 1.3638	0.9577 1.0083	1.2251 1.3686	1.2040 1.3312	1.1410 1.3799 4	0.4549 0.8203
12	0.9419 0.9611	1.2329 1.3738	0.9876 1.0668	1.2251 1.3686	0.9121 0.9856	1.2017 1.3729	0.6499 0.9876	
13	1.2488 1.3421	1.2135 1.3149	1.2580 1.3828	1.2040 1.3312	1.2017 1.3729	0.7361 0.9976	0.3372 0.6780	
14	1.0444 1.1829	1.1550 1.3682	1.0889 1.2459	1.1410 1.3794	0.9466 0.9876	0.3372 0.6780		
15	0.6427 0.9584	0.6180 0.9024	0.7740 1.0910	0.4549 0.8203				

XXXX YYYY	P(AVG) PEAK PIN

6. THERMAL-HYDRAULIC DESIGN

The generic and cycle-specific analyses supporting Cycle 6 operation were performed by Duke Power Company using the methodology described in Reference 8. Cycle 6 is the first Mark-BW transition cycle for Unit 2 and is analyzed using Duke's Statistical Core Design (SCD) methodology. Uncertainties on parameters that affect DNB performance are statistically combined to determine a Statistical DNBR limit (SDL). Using the BWCMV correlation, Reference 9, a generic SDL of 1.40 was calculated using a set of generic uncertainties given in Reference 8. The system parameter uncertainties used in Reference 8 and given in Table 6-1 bound the uncertainties specifically calculated for Catawba. Reactor core safety limits for Cycle 6 are based on a full Mark-BW core and a design $F\Delta H$ of 1.50. The Cycle 6 nominal thermal-hydraulic design conditions are given in Table 6-2.

The Mark-BW fuel assembly was designed to be hydraulically compatible with Westinghouse Optimized Fuel (OFA). BWFC has performed a series of flow tests to verify the compatibility of the two designs. The tests showed that the total pressure drop across the OFA fuel is 2.4 % higher than the pressure drop across the Mark-BW fuel, Reference 10. A generic transition core analysis was performed to determine the DNBR impact of this difference.

Since the Mark-BW fuel has a lower overall pressure drop than the OFA design, a Mark-BW assembly in a mixed core will tend to have more flow through it and consequently more DNB margin than the same assembly in an all Mark-BW core. Conversely, flow will be forced out of the OFA fuel in a mixed core; thus, the need to calculate a DNBR penalty for the OFA fuel. A generic transition core DNBR penalty was determined by modeling a conservative core configuration with one OFA assembly as the hot assembly. The rest of the core was modeled as Mark-BW fuel. A number of statepoints and peaking conditions were analyzed, yielding a maximum DNBR penalty of 3.8 % for the OFA fuel.

An anomalous flow condition has been observed in several Westinghouse plants, including both the Catawba units. The anomaly is a vortex that forms in the lower internals and re-distributes the flow into the core. The anomaly behavior was categorized based on measured plant data and the impact on DNBR in the core evaluated. As a result of the anomaly, a penalty has been assessed to account for periods during which the flow re-distribution occurs. This penalty is in terms of both a peaking penalty and a DNBR penalty and is applied to both the Catawba units.

To provide design flexibility, margin is added to the SDL to determine a design DNBR limit (DDL). For the generic Mark-BW and Catawba 2 Cycle 6 analyses, the DDL is 1.55 (10.7 % margin above the SDL). The DNBR penalties, such as the OFA transition core penalty, that must be assessed against the margin are given in Table 6-3.

Table 6-1
System Uncertainties Included in the
Statistical Core Design Analysis

Reference 8

Parameter	Uncertainty	Distribution
Core power	+/- 2 %	Normal
RCS flow	+/- 2.2 %	Normal
Core bypass flow	+/- 1.5 %	Uniform
Pressure	+/- 30 psi	Uniform
Inlet temperature	+/- 4 deg F	Uniform

Table 6-2 Nominal Thermal-Hydraulic Design Conditions

Catawba 2 Cycle 6

Core power, MWt	3411
Core exit pressure, psia	2280
Vessel ave. temperature, Deg F	590.8
RCS flow, gpm	385,000
Core bypass flow, %	7.5
Reference design FΔH	1.50
Reference design axial shape	1.55 Cosine
CHF correlation	BWCMV
Statistical DNBR limit	1.40
Design DNBR limit	1.55

Table 6-3. DNBR Penalties

Statistical DNBR limit	1.40
Design DNBR limit	1.55
DNBR margin	10.7 %

DNBR Penalty	Mark-BW	OFA
Transition core	0 %	3.8 %
Instrumentation/hardware	5.6 %	2.8 %
Rod bow	0 %	3.5 %
Flow anomaly	0.5 %	0.5 %
Total DNBR penalty	6.1 %	10.6 %
Available DNBR margin	**4.6 %**	**0.1 %**

Table 6-4. Flow Anomaly Peaking Penalties

The following penalties are applied to the Catawba maximum allowable total peaking limits to account for the RCS flow anomaly. The penalties apply to all peak magnitudes.

Position X/L	Penalty % Peak
0.01	1.5
0.1	1.5
0.2	1.2
0.3	0.9
0.4	0.6
0.5	0.3
0.6-1.0	0.0

7. ACCIDENT ANALYSIS

In order to determine the effects of this reload and to ensure that the thermal performance during hypothetical incidents is not degraded, each FSAR accident analysis sensitive to reload core physics parameters has been evaluated.

For the following FSAR Chapter 15 accidents, the licensing basis has been revised to reflect reanalysis by Duke Power Company of the thermal-hydraulic system transients:

- Steam line break
- Turbine trip
- Feedwater line break
- Partial loss of forced reactor coolant flow
- Complete loss of forced reactor coolant flow
- Reactor coolant pump locked rotor
- Uncontrolled bank withdrawal from subcritical or low power startup condition
- Uncontrolled bank withdrawal at power
- Dropped rod/rod bank
- Statically misaligned rod
- Single rod withdrawal
- Rod ejection
- Steam generator tube rupture

The analytical models and methodology for the statically misaligned rod accident are provided in approved topical reports, References 6, 8, and 16. For each of the remaining events, a single, generic, system thermal-hydraulic analysis is performed which bounds both Catawba Units 1 and 2, and McGuire Units 1 and 2. Since a single set of generic analyses has been performed for these events, the results for Catawba are identical to those submitted in the approved McGuire 1 Cycle 8 (Reference 17), McGuire 2 Cycle 8 (Reference 18) and Catawba 1 Cycle 7 (Reference 19) reload reports. The Catawba 2 Cycle 6 reload core physics parameter values have been reviewed with respect to the assumptions used in these analyses. The analysis methodology for these events, except for the steam line break, the dropped rod/rod bank, and the rod ejection events, has been approved in References 11 and 16. A minor change has been made to the operator action time value of 120 seconds presented in the feedwater line break analysis, Section 3.4.2.4 of Reference 16. This is a conservative change in the value of the input assumption, and is not a change in the methodology. The results of the analysis are within all acceptance criteria. The analysis methodology for the steam line break, the dropped rod/rod bank, and the rod ejection events has been approved in References 11 and 12.

For the remaining FSAR Chapter 15 system thermal-hydraulic accident analyses sensitive to reload core physics parameters, e.g. LOCA, the current licensing basis is being retained. In addition, the post-LOCA subcriticality evaluation and the boron precipitation evaluation have been performed by Duke Power Company as described in Chapter 15 and Chapter 6, respectively, of the Catawba FSAR, Reference 1. The Catawba 2 Cycle 6 parameter values have also been reviewed with respect to the assumptions used in the subcriticality analysis.

The radiological consequences for the following events are reanalyzed due to differences between the Mark-BW fuel and OFA fuel fission

product core inventories, changes in the thermal-hydraulic analysis results, and changes in the dose analysis methodology.

- Reactor coolant pump locked rotor
- Single rod withdrawal
- Rod ejection

The above radiological consequence analyses are applicable to Catawba 2 Cycle 6. These dose analyses and resulting FSAR changes were submitted in the approved Catawba 1 Cycle 7 reload submittal (Reference 19).

Catawba 2 Cycle 6 reload core physics parameters were found to be bounded by the accident analysis assumptions for all accidents which are sensitive to core physics parameters, thus demonstrating conservative results for the operation of Catawba 2 Cycle 6.

8. PROPOSED MODIFICATIONS TO LICENSING BASIS DOCUMENTS

Revisions to the Technical Specifications and Core Operating Limits Report (COLR) have been proposed for Cycle 6 operation to accommodate the influence of the Cycle 6 core design on power peaking, reactivity, and control rod worths. The Technical Specification limits and COLR limits also reflect changes in reload analysis methodology beginning with this core. The Cycle 6 design analysis basis includes a low-leakage fuel cycle design and a mixed core containing both B&W Mark-BW and Westinghouse OFA fuel assemblies.

A cycle specific power distribution analysis of the final core design was conducted to generate the f(ΔI) limits for the Overpower ΔT and Overtemperature ΔT trip functions and the Limiting Conditions for Operation (control bank insertion and axial flux difference). The ΔT limits preserve the centerline fuel melt and steady-state DNBR limits. The Limiting Conditions for Operation preserve the maximum allowable LOCA and initial condition DNB peaking limits, ejected rod worth reactivity limits, and the shutdown margin reactivity limit. These limits were developed based on the NRC-approved methodology described in Reference 7. A peaking penalty for quadrant power tilt was taken in the analysis so that the resulting limits accommodate quadrant power tilt ratios up to a value of 1.02.

The maximum allowable LOCA peaking limits shown in Figure 4 of the COLR are based on the BWFC ECCS evaluation (References 13 and 14). A composite K(Z) limit was developed based on both large and small break analyses. Separate composite limits applicable to Mark-BW and OFA fuel were used in the power distribution analysis, and are specified in the COLR. These limits were used directly in determination of the control rod insertion and axial flux difference operating limits given in Technical Specifications 3.1.3.6 and 3.2.1. Technical Specification 3.2.2 provides the nuclear heat flux hot channel (F_Q) peaking limit.

The initial condition DNB maximum allowable peaking (MAP) limits shown in Table 4 of the COLR are based on core reference design peaking factors. The MAP limits provide allowable combinations of peaking factors that preserve DNBR performance equivalent to the design power distribution for a limiting loss of coolant flow transient. The initial condition MAPs are used as described in Reference 7 to calculate DNB peaking margins for determination of the control rod position and axial flux difference operating limits given in Technical Specifications 3.1.3.6 and 3.2.1. Technical Specification 3.2.3 provides the nuclear enthalpy rise hot channel ($F_{\Delta H}$) peaking limit.

The methodology for surveillance for the core hot channel peaking factors is described in Reference 7. In this application of the methodology, peaking margin calculations are performed whenever an incore flux map is taken for surveillance monitoring.

Specifications 4.2.2 and 4.2.3 have been written in a form that provides this capability, and the parameters required by this application of core monitoring are provided in the COLR. The core operating limits are provided in accordance with NRC Generic Letter 88-16 and Technical Specification 6.9.1.9. Table 8-1 lists the Technical Specification changes required for Cycle 6, and these changes are identical to those submitted in the approved Catawba 1 Cycle 7 reload

submittal (Reference 19), except those identified by an asterisk in Table 8-1. Table 8-2 lists the changes to the Core Operating Limits Report. These changes are being submitted to the NRC under separate covers. Parameters related to monitoring the core power distribution are defined in Reference 7, and are used by the plant computer software. These parameters will be supplied for inclusion in the COLR.

Based on the analysis and revisions to the Technical Specifications, COLR and FSAR described in this report, Cycle 6 of Catawba Unit 2 will operate within the 10CFR 50.46 ECCS acceptance criteria and within the thermal design criteria. The following pages contain the required Technical Specification, COLR and FSAR revisions.

Table 8-1 Technical Specification Changes

Specification		Description of Change
2.1.1		decreased $F_{\Delta H}$ for Mark-BW fuel changed CHF correlation reduced RCS minimum flow to 385,000 gpm
2.2.1		decreased $F_{\Delta H}$ for Mark-BW fuel removed power range neutron flux negative rate reactor trip
3/4.2.1		deleted baseload operation
3/4.2.2		changed F_Q methodology to reflect Duke nomenclature quantified surveillance requirements
3/4.2.3		changed $F_{\Delta H}$ methodology to reflect Duke nomenclature quantified surveillance requirements
3/4.2.4		increased the tilt ratio at which power reduction is required
3/4.2.5		incorporated RCS flow as DNB parameter deleted Figure 8, "RCS Flow vs. R-Four Loops in Operation" from COLR and remove unit specification from Technical Specification Figure 3.2-1. reduced RCS minimum measured flow to 385,000 gpm
3/4.3.1		removed power range neutron flux negative rate reactor trip
3/4.3.2		increased low steam line pressure setpoint increased feedwater isolation response time increased steam line isolation response time removed steam line pressure dynamic compensation
3/4.3.3.12	*	decreased Reactor Makeup Water Pump flowrate limit from 75 gpm to 70 gpm in Mode 5
3/4.4.2.1		increased pressurizer safety valve lift setpoint tolerance

3/4.4.2.2		increased pressurizer safety valve lift setpoint tolerance
3/4.6.3	*	changed steam generator main feedwater to auxiliary feedwater nozzle isolation valve, auxiliary nozzle temper valve, steam generator feedwater containment isolation valve, steam generator feedwater purge valve, main steam isolation valve, and main steam isolation bypass control valve stroke time from 5 seconds to Not Applicable.
3/4.7.1.4		increased main steam line isolation valve stroke time
6.9.1.9	*	Add NRC approved topical DPC-NE-1004A to list of analytical methods used to determine core operating limits

Note: "*" The proposed Technical Specification change was not included in the approved Catawba 1, Cycle 6 (Reference 20) or Catawba 1, Cycle 7 reload submittal (Reference 19). The items without the "*" are identical to those approved for Catawba 1, Cycle 6 or Catawba 1, Cycle 7.

Table 8-2 Core Operating Limits Report Changes

Specification	Description of Change
3/4.1.3.5	revised shutdown bank insertion limits to reflect a minimum rod withdrawal limit of 222 steps and a maximum rod withdrawal limit of 230 steps
3/4.1.3.6	revised control bank insertion limits to reflect a minimum rod withdrawal limit of 222 steps and a maximum rod withdrawal limit of 230 steps
3/4.2.1	revised AFD limits for Cycle 6 operation
3/4.2.2	revised for Cycle 6 operation to reflect a change in the heat flux hot channel factor F_Q methodology
3/4.2.3	revised for Cycle 6 operation to reflect a change in the nuclear enthalpy rise hot channel factor $F_{\Delta H}$ methodology
3/4.2.5	moved figure 3.2-1 from COLR to Technical Specifications

Note: The proposed Core Operating Limits Report changes are identical to those approved for Catawba 1 Cycle 7 reload submittal, Reference 19.

8.1 Changes to Technical Specifications

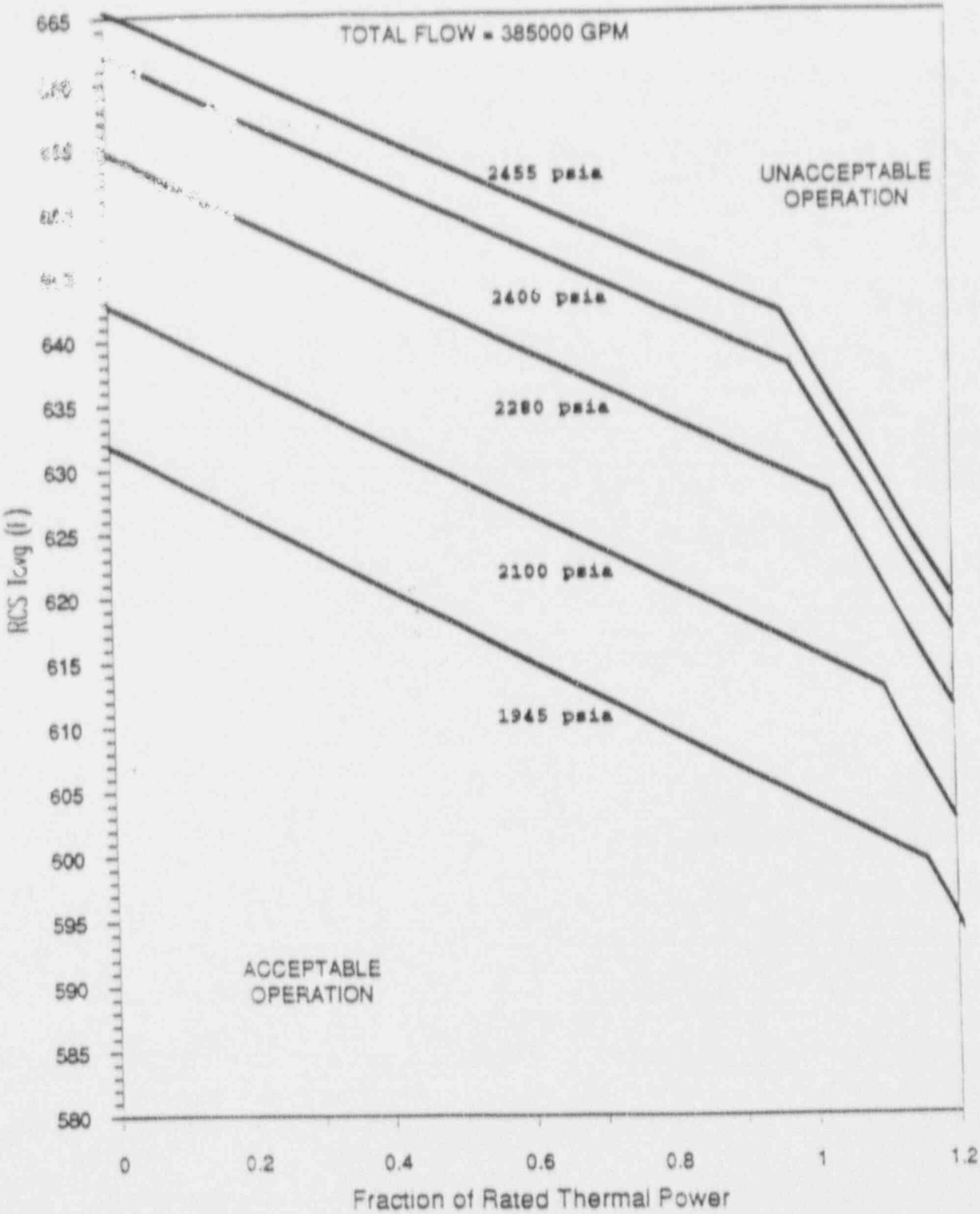

FIGURE 2.1-1a

REACTOR CORE SAFETY LIMITS - FOUR LOOPS IN OPERATION, ~~UNIT 1~~

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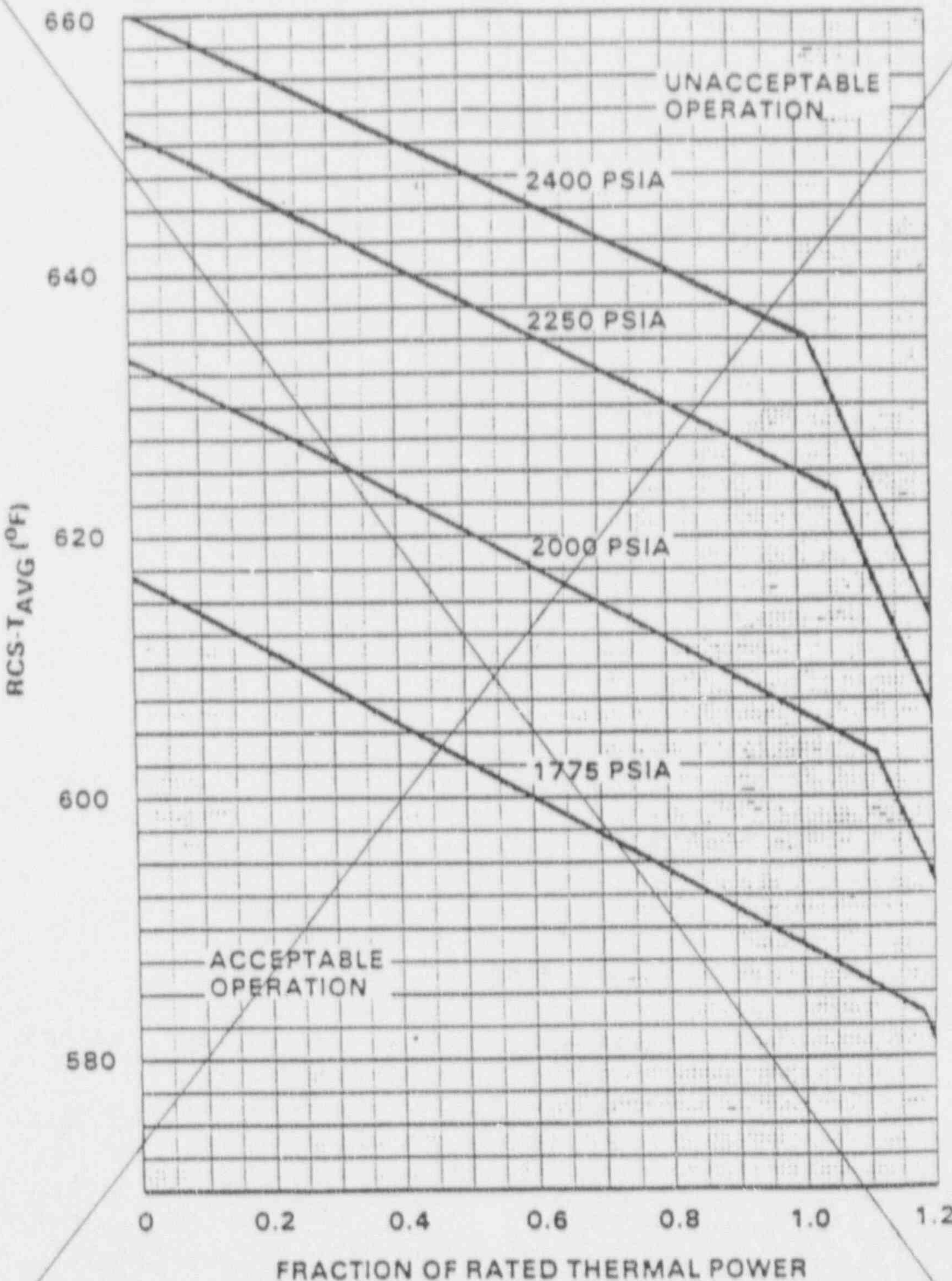

FIGURE 2.1-1b

REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION, UNIT 2

TABLE 2.2-1 ~~FOR UNIT 1~~

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.
2. Power Range, Neutron Flux		
a. High Setpoint	≤109% of RTP*	≤110.9% of RTP*
b. Low Setpoint	≤25% of RTP*	≤27.1% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	≤5% of RTP* with a time constant ≥ 2 seconds	≤6.3% of RTP* with a time constant ≥ 2 seconds
4. Intermediate Range, Neutron Flux	≤25% of RTP*	≤31% of RTP*
5. Source Range, Neutron Flux	$\leq 10^5$ cps	$\leq 1.4 \times 10^5$ cps
6. Overtemperature ΔT	See Note 1	See Note 2
7. Overpower ΔT	See Note 3	See Note 4
8. Pressurizer Pressure-Low	≥1945 psig	≥1938 psig***
9. Pressurizer Pressure-High	≤2385 psig	≤2399 psig
10. Pressurizer Water Level-High	≤92% of instrument span	≤93.8% of instrument span
11. Reactor Coolant Flow-Low	≥90% of loop minimum measured flow**	≥88.9% of loop minimum measured flow**

*RTP = RATED THERMAL POWER

**Loop minimum measured flow = ~~96,900 gpm (Unit 2),~~ 96,250 gpm ~~(Unit 1)~~

***Time constants utilized in the lead-lag controller for Pressurizer Pressure-Low are 2 seconds for lead and 1 second for lag. Channel calibration shall ensure that these time constants are adjusted to these values.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
12. Steam Generator Water Level Low-Low		
a. Unit 1	≥17% of span from 0% to 30% RTP* increasing linearly to ≥ 40.0% of span from 30% to 100% RTP*	≥15.3% of span from 0% to 30% RTP* increasing linearly to ≥38.3% of span from 30% to 100% RTP*
b. Unit 2	≥36.8% of narrow range span	≥35.1% of narrow range span
13. Undervoltage - Reactor Coolant Pumps	≥77% of bus voltage (5082 volts) with a 0.7s response time	≥76% (5016 volts)
14. Underfrequency - Reactor Coolant Pumps	≥56.4 Hz with a 0.2s response time	≥55.9 Hz
15. Turbine Trip		
a. Stop Valve EH Pressure Low	≥550 psig	≥500 psig
b. Turbine Stop Valve Closure	≥1% open	≥1% open
16. Safety Injection Input from ESF	N.A.	N.A.

*RTP = RATED THERMAL POWER

~~UNIT 1~~

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
17. Reactor Trip System Interlocks		
a. Intermediate Range Neutron Flux, P-6	$\geq 1 \times 10^{-10}$ amps	$\geq 6 \times 10^{-11}$ amps
b. Low Power Reactor Trips Block, P-7		
1) P-10 input	$\leq$10% of RTP*	$\leq$12.2% of RTP*
2) P-13 input	$\leq$10% RTP* Turbine Impulse Pressure Equivalent	$\leq$12.2% RTP* Turbine Impulse Pressure Equivalent
c. Power Range Neutron Flux, P-8	$\leq$48% of RTP*	$\leq$50.2% of RTP*
d. Power Range Neutron Flux, P-9	$\leq$69% of RTP*	$\leq$70% of RTP*
e. Power Range Neutron Flux, P-10	$\geq$10% of RTP*	$\geq$7.8% of RTP*
f. Power Range Neutron Flux, Not P-10	$\leq$10% of RTP*	$\leq$12.2% of RTP*
g. Turbine Impulse Chamber Pressure, P-13	$\leq$10% RTP* Turbine Impulse Pressure Equivalent	$\leq$12.2% RTP* Turbine Impulse Pressure Equivalent
18. Reactor Trip Breakers	N.A.	N.A.
19. Automatic Trip and Interlock Logic	N.A.	N.A.

*RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_o \left\{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where:

ΔT	=	Measured ΔT by Loop Narrow Range RTDs;
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT;
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, τ_1 = 12 s, τ_2 = 3 s;
$\frac{1}{1 + \tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constant utilized in the lag compensator for ΔT, τ_3 = 0;
ΔT_o	=	Indicated ΔT at RATED THERMAL POWER;
K_1	=	1.1953
K_2	=	0.03163/°F
$\frac{1 + \tau_4 S}{1 + \tau_5 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
τ_4, τ_5	=	Time constants utilized in the lead-lag compensator for T_{avg}, τ_4 = 22 s, τ_5 = 4 s;
T	=	Average temperature, °F;
$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0;

TABLE 2.2-1 (Continued)
TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

T'	$\leq$	590.8°F (Nominal T_{avg} allowed by Safety Analysis);
K_3	=	0.001414;
P	=	Pressurizer pressure, psig;
P'	=	2235 psig (Nominal RCS operating pressure);
S	=	Laplace transform operator, s^{-1};

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant STARTUP tests such that:

(i) For $q_t - q_b$ between -39.9% and +3.0%, $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) For each percent ΔI that the magnitude of $q_t - q_b$ is more negative than -39.9%, the ΔT Trip Setpoint shall be automatically reduced by 3.910% of ΔT_o; and

(iii) For each percent ΔI that the magnitude of $q_t - q_b$ is more positive than +3.0%, the ΔT Trip Setpoint shall be automatically reduced by 2.316% of ΔT_o.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3.0%.

TABLE 2.2-1 (Continued)
TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \le \Delta T_0 \left\{K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right) \left(\frac{1}{1 + \tau_6 S}\right) T - K_6 \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T''\right] - f_2(\Delta I)\right\}$$

Where:

ΔT	=	As defined in Note 1,
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	As defined in Note 1,
τ_1, τ_2	=	As defined in Note 1,
$\frac{1}{1 + \tau_3 S}$	=	As defined in Note 1,
τ_3	=	As defined in Note 1,
ΔT_0	=	As defined in Note 1,
K_4	=	1.0819
K_5	=	0.02/°F for increasing average temperature and 0 for decreasing average temperature,
$\frac{\tau_7 S}{1 + \tau_7 S}$	=	The function generated by the rate-lag controller for T_{avg} dynamic compensation,
τ_7	=	Time constant utilized in the rate-lag controller for T_{avg}, $\tau_7 = 10$ s,
$\frac{1}{1 + \tau_6 S}$	=	As defined in Note 1,
τ_6	=	As defined in Note 1,

TABLE 2.2-1 (Continued)
TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6 = 0.001291/°F for T > 590.8°F and K_6 = 0 for T $\leq$ 590.8°F,

T = As defined in Note 1,

T" = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 590.8°F),

S = As defined in Note 1,

and f_2 (ΔI) is a function of the indicated differences between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between -35% and +35% ΔI; f_2 (ΔI) = 0, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent ΔI that the magnitude of $q_t - q_b$ is more negative than -35% ΔI, the ΔT Trip Setpoint shall be automatically reduced by 7.0% of ΔT_o; and

(iii) for each percent ΔI that magnitude of $q_t - q_b$ is more positive than +35% ΔI, the ΔT Trip Setpoint shall be automatically reduced by 7.0% of ΔT_o.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.8%.

TABLE 2.2.-1 FOR UNIT 2

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.
2. Power Range, Neutron Flux		
a. High Setpoint	$\leq$109% of RTP*	$\leq$110.9% of RTP*
b. Low Setpoint	$\leq$25% of RTP*	$\leq$27.1% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	$\leq$5% of RTP* with a time constant $\geq$ 2 seconds	$\leq$6.3% of RTP* with a time constant $\geq$ 2 seconds
4. Power Range, Neutron Flux High Negative Rate	$\leq$5% of RTP* with a time constant $\geq$2 seconds	$\leq$6.3% of RTP* with a time constant $\geq$2 seconds
5. Intermediate Range, Neutron Flux	$\leq$25% of RTP*	$\leq$31% of RTP*
6. Source Range, Neutron Flux	$\leq 10^5$ cps	$\leq 1.4 \times 10^5$ cps
7. Overtemperature ΔT	See Note 1	See Note 2
8. Overpower ΔT	See Note 3	See Note 4
9. Pressurizer Pressure-Low	$\geq$1945 psig	$\geq$1938 psig***
10. Pressurizer Pressure-High	$\leq$2385 psig	$\leq$2399 psig
11. Pressurizer Water Level-High	$\leq$92% of instrument span	$\leq$93.8% of instrument span
12. Reactor Coolant Flow-Low	$\geq$90% of loop minimum measured flow**	$\geq$88.9% of loop minimum measured flow**

*RTP = RATED THERMAL POWER

**Loop minimum measured flow = 96,900 gpm (Unit 2), 96,250 gpm (Unit 1)

***Time constants utilized in the lead-lag controller for Pressurizer Pressure-Low are 2 seconds for lead and 1 second for lag. Channel calibration shall ensure that these time constants are adjusted to these values.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
13. Steam Generator Water Level Low-Low		
a. Unit 1	$\geq$17% of span from 0% to 30% RTP* increasing linearly to $\geq$ 40.0% of span from 30% to 100% RTP*	$\geq$15.3% of span from 0% to 30% RTP* increasi g linearly to $\geq$38.3% of span from 30% to 100% RTP*
b. Unit 2	$\geq$36.8% of narrow range span	$\geq$35.1% of narrow range span
14. Undervoltage - Reactor Coolant Pumps	$\geq$77% of bus voltage (5082 volts) with a 0.7s response time	$\geq$76% (5016 volts)
15. Underfrequency - Reactor Coolant Pumps	$\geq$56.4 Hz with a 0.2s response time	$\geq$55.9 Hz
16. Turbine Trip		
a. Stop Valve EH Pressure Low	$\geq$550 psig	$\geq$500 psig
b. Turbine Stop Valve Closure	$\geq$1% open	$\geq$1% open
17. Safety Injection Input from ESF	N.A.	N.A.

*RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
18. Reactor Trip System Interlocks		
a. Intermediate Range Neutron Flux, P-6	$\geq 1 \times 10^{-10}$ amps	$\geq 6 \times 10^{-11}$ amps
b. Low Power Reactor Trips Block, P-7		
1) P-10 input	$\leq$10% of RTP*	$\leq$12.2% of RTP*
2) P-13 input	<10% RTP* Turbine Impulse Pressure Equivalent	<12.2% RTP* Turbine Impulse Pressure Equivalent
c. Power Range Neutron Flux, P-8	$\leq$48% of RTP*	$\leq$50.2% of RTP*
d. Power Range Neutron Flux, P-9	$\leq$69% of RTP*	$\leq$70% of RTP*
e. Power Range Neutron Flux, P-10	$\geq$10% of RTP*	$\geq$7.6% of RTP*
f. Power Range Neutron Flux, Not P-10	$\leq$10% of RTP*	$\leq$12.2% of RTP*
g. Turbine Impulse Chamber Pressure, P-13	<10% RTP* Turbine Impulse Pressure Equivalent	<12.2% RTP* Turbine Impulse Pressure Equivalent
19. Reactor Trip Breakers	N.A	N.A.
20. Automatic Trip and Interlock Logic	N.A.	N.A.

*RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_o \left\{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where:

ΔT	=	Measured ΔT by Loop Narrow Range RTDs;
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT;
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, τ_1 = 12 s, τ_2 = 3 s;
$\frac{1}{1 + \tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constant utilized in the lag compensator for ΔT, $\tau_3 = 0$;
ΔT_o	=	Indicated ΔT at RATED THERMAL POWER;
K_1	=	1.38;
K_2	=	0.02401/°F;
$\frac{1 + \tau_4 S}{1 + \tau_5 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
τ_4, τ_5	=	Time constants utilized in the lead-lag compensator for T_{avg}, τ_4 = 22 s, τ_5 = 4 s;
T	=	Average temperature, °F;
$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$;

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

T'	$\leq$	590.8°F (Nominal T_{avg} allowed by Safety Analysis);
K_3	=	0.001189;
P	=	Pressurizer pressure, psig;
P'	=	2235 psig (Nominal RCS operating pressure);
S	=	Laplace transform operator, s^{-1};

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant STARTUP tests such that:

(i) For $q_t - q_b$ between -22.5% and -6.5%, $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) For each percent ΔI that the magnitude of $q_t - q_b$ is more negative than -22.5%, the ΔT Trip Setpoint shall be automatically reduced by 3.151% of ΔT_o; and

(iii) For each percent ΔI that the magnitude of $q_t - q_b$ is more positive than -6.5%, the ΔT Trip Setpoint shall be automatically reduced by 2.414% of ΔT_o.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.3%.

TABLE 2.2-1 (Continued)
TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_o \{K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right) \left(\frac{1}{1 + \tau_6 S}\right) T - K_6 [T \left(\frac{1}{1 + \tau_6 S}\right) - T''] - f_2(\Delta I)\}$$

Where: ΔT = As defined in Note 1,

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = As defined in Note 1,

τ_1, τ_2 = As defined in Note 1,

$\frac{1}{1 + \tau_3 S}$ = As defined in Note 1,

τ_3 = As defined in Note 1,

ΔT_o = As defined in Note 1,

K_4 = 1.0704,

K_5 = 0.02/°F for increasing average temperature and 0 for decreasing average temperature,

$\frac{\tau_7 S}{1 + \tau_7 S}$ = The function generated by the rate-lag controller for T_{avg} dynamic compensation,

τ_7 = Time constant utilized in the rate-lag controller for T_{avg}, τ_7 = 10 s,

$\frac{1}{1 + \tau_6 S}$ = As defined in Note 1,

τ_6 = As defined in Note 1,

TABLE 2.2-1 (Continued)
TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6	=	0.001707/°F or $T > 590.8$°F and $K_6 = 0$ for $T \leq 590.8$°F,
T	=	As defined in Note 1,
T''	=	Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, ≤ 590.8°F),
S	=	As defined in Note 1, and
$f_2(\Delta I)$	=	0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.8%.

2.1.1 REACTOR CORE ~~(FOR UNIT 1)~~

The restrictions of this Safety Limit prevent overheating of the fuel 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 could 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 BWCMV correlation. The BWCMV 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), is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, and is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95% probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the BWCMV correlation in this application). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95% probability with 95% confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.

In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters, and the BWCMV DNB correlation are considered statistically such that there is at least a 95% confidence that the minimum DNBR for the limited rod is greater than or equal to the DNBR limit. The uncertainties in the above parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty is used to establish a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature below which the calculated DNBR is no less than the design DNBR value, or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

These curves are based on a nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, of 1.50 and a reference cosine with a peak of 1.55. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.50 \, [1 + 1/RRH \, (1-P)]$$

Where P is the fraction of RATED THERMAL POWER.
RRH is given in the COLR.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature ΔT trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the Setpoints to provide protection consistent with core Safety Limits.

~~2.1.1 REACTOR CORE (FOR UNIT 2)~~

~~The restrictions of this Safety Limit prevent overheating of the fuel 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 could 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 WRB-1 correlation. The WRB-1 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), is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, and is indicative of the margin to DNB.~~

~~The DNB design basis is as follows: there must be at least a 95% probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-1 correlation in this application). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95% probability with 95% confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.~~

In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95% confidence that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature below which the calculated DNBR is no less than the design DNBR value, or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

This curve is based on a nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, of 1.49 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.49\ [1 + 0.3\ (1-P)]$$

Where P is the fraction of RATED THERMAL POWER.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the Setpoints to provide protection consistent with core Safety Limits.

2.1.2 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 vessel, pressurizer, and the Reactor Coolant System piping, valves, and fittings are designed to Section III of the ASME Code for Nuclear Power Plants which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated Code requirements.

The entire Reactor Coolant System is hydrotested at 125% (3110 psig) of design pressure, to demonstrate integrity prior to initial operation.

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

The various Reactor trip circuits automatically open the Reactor trip breakers whenever a condition monitored by the Reactor Trip System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides a Reactor Trip System which monitors numerous system variables, therefore providing Trip System functional diversity. The functional capability at the specified trip setting is required for those anticipatory or diverse Reactor trips for which no direct credit was assumed in the accident analysis to enhance the overall reliability of the Reactor Trip System. The Reactor Trip System initiates a Turbine trip signal whenever Reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive Reactor Coolant System cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

Manual Reactor Trip

The Reactor Trip System includes manual Reactor trip capability.

Power Range, Neutron Flux

In each of the Power Range Neutron Flux channels there are two independent bistables, each with its own trip setting used for a High and Low Range trip setting. The Low Setpoint trip provides protection during subcritical and low power operations to mitigate the consequences of a power excursion beginning from low power, and the High Setpoint trip provides protection during power operations to mitigate the consequences of a reactivity excursion from all power levels.

The Low Setpoint trip may be manually blocked above P-10 (a power level of approximately 10% of RATED THERMAL POWER) and is automatically reinstated below the P-10 Setpoint.

Positive Rate

Power Range, Neutron Flux, ~~High Rates~~

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of a rupture of a control rod drive housing. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for all rod ejection accidents.

~~The Power Range Negative Rate trip provides protection for control rod drop accidents. At high power a rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate trip for those control rod drop accidents for which DNBRs will be greater than the applicable design limit DNBR value for each fuel type.~~

~~The Power Range Negative Rate Trip has been deleted for Units 1.~~

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the acceptable limits specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODE 1, above 50% of RATED THERMAL POWER.* ~~(Unit 1)~~

ACTION:

a. For operation with the indicated AFD outside of the limits specified in the COLR,

   1. Either restore the indicated AFD to within the COLR limits within 15 minutes, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux-High Trip setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

b. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the limits specified in the COLR.

*See Special Test Exceptions Specification 3.10.2.

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 50% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

   1) At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

   2) At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.

b. Monitoring and logging the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AFD Monitor Alarm is inoperable. The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

c. The provisions of Specification 4.0.4 are not applicable.

4.2.1.2 The indicated AFD shall be considered outside of its limits when at least two OPERABLE excore channels are indicating the AFD to be outside the limits.

POWER DISTRIBUTION LIMITS

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(X,Y,Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(X,Y,Z)$ shall be limited by imposing the following relationships:

$$F_Q^{MA}(X,Y,Z) \leq \frac{F_Q^{RTP}}{P} K(Z) \text{ for } P > 0.5$$

$$F_Q^{MA}(X,Y,Z) \leq \frac{F_Q^{RTP}}{0.5} K(Z) \text{ for } P \leq 0.5$$

Where: F_Q^{RTP} = the F_Q Limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR),

$F_Q^{MA}(X,Y,Z)$ = the measured heat flux hot channel factor $F_Q^M(X,Y,Z)$, with adjustments as specified in 4.2.2.3,

$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

K(Z) = the normalized $F_Q(X,Y,Z)$ limit specified in the COLR for the appropriate fuel types.

APPLICABILITY: MODE 1. ~~(Unit 1)~~

ACTION:

With $F_Q(X,Y,Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q^{MA}(X,Y,Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours, and

b. Control the AFD to within new AFD limits which are determined by reducing the allowable power at each point along the AFD limit lines of Specification 3.2.1 at least 1% for each 1% $F_Q^{MA}(X,Y,Z)$ exceeds the limit within 15 minutes and reset the AFD alarm setpoints to the modified limits within 8 hours, and

c. POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints (value of K_4) have been reduced at least 1% (in ΔT span) for each 1% $F_Q^{MA}(X,Y,Z)$ exceeds the limit, and

d. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by ACTION a., above; THERMAL POWER may then be increased provided $F_Q(X,Y,Z)$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 $F_Q^M(X,Y,Z)$[1] shall be evaluated to determine whether $F_Q(X,Y,Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Measuring $F_Q^M(X,Y,Z)$ at the earliest of:

1. At least once per 31 Effective Full Power Days, or
2. Upon reaching equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q^M(X,Y,Z)$ was last determined[2], or
3. At each time the QUADRANT POWER TILT RATIO indicated by the excore detectors is normalized using incore detector measurements.

c. Performing the following calculations:

1. For each location, calculate the % margin to the maximum allowable design as follows:

$$\%\ \text{Operational Margin} = \left(1 - \frac{F_Q^M(X,Y,Z)}{[F_Q^L(X,Y,Z)]^{OP}}\right) \times 100\%$$

$$\%\ \text{RPS Margin} = \left(1 - \frac{F_Q^M(X,Y,Z)}{[F_Q^L(X,Y,Z)]^{RPS}}\right) \times 100\%$$

where $[F_Q^L(X,Y,Z)]^{OP}$ and $[F_Q^L(X,Y,Z)]^{RPS}$ are the Operational and RPS design peaking limits defined in the COLR.

2. Find the minimum Operational Margin of all locations examined in 4.2.2.2.c.1 above. If any margin is less than zero, then either of the following actions shall be taken:

---

(1) No additional uncertainties are required in the following equations for $F_Q^M(X,Y,Z)$, because the limits include uncertainties.

(2) During power escalation at the beginning of each cycle, THERMAL POWER may be increased until a power level for extended operation has been achieved and a power distribution map obtained.

(a) Within 15 minutes:

(1) Control the AFD to within new AFD limits that are determined by:

$$(\text{AFD Limit})^{\text{reduced}}_{\text{negative}} = (\text{AFD Limit})^{\text{COLR}^{(3)}}_{\text{negative}}$$

$$+ [\text{Margin}^{\text{min}}_{\text{OP}}] \text{ absolute value}$$

$$(\text{AFD Limit})^{\text{reduced}}_{\text{positive}} = (\text{AFD Limit})^{\text{COLR}^{(3)}}_{\text{positive}}$$

$$- [\text{Margin}^{\text{min}}_{\text{OP}}] \text{ absolute value}$$

where $\text{Margin}^{\text{min}}_{\text{OP}}$ is the minimum margin from 4.2.2.2.c.1, and

(2) Within 8 hours, reset the AFD alarm setpoints to the modified limits of 4.2.2.2.c.2.a, or

(b) Comply with the ACTION requirements of Specification 3.2.2, treating the margin violation in 4.2.2.2.c.1 above as the amount by which F_Q^{MA} is exceeding its limit.

3. Find the minimum RPS Margin of all locations examined in 4.2.2.2.c.1 above. If any margin is less than zero, then the following action shall be taken:

Within 72 hours, reduce the K_1 value for OTΔT by:

$$K_1 \text{ adjusted} = K_1^{(4)} - [\text{KSLOPE}^{(3)} \times \text{Margin}^{\text{min}}_{\text{RPS}}] \text{absolute value}$$

where $\text{MARGIN}^{\text{min}}_{\text{RPS}}$ is the minimum margin from 4.2.2.2.c.1.

(3) Defined and specified in the COLR per Specification 6.9.1.9.

(4) K_1 value from Table 2.2-1.

d. Extrapolating[5] at least two measurements to 31 Effective Full Power Days beyond the most recent measurement and if:

$$(F_Q^M(X,Y,Z)) \text{ (extrapolated)} \geq (F_Q^L(X,Y,Z))^{OP} \text{ (extrapolated), and}$$

$$\frac{(F_Q^M(X,Y,Z)) \text{ (extrapolated)}}{(F_Q^L(X,Y,Z))^{OP} \text{ (extrapolated)}} > \frac{(F_Q^M(X,Y,Z))}{(F_Q^L(X,Y,Z))^{OP}}$$

or

$$(F_Q^M(X,Y,Z)) \text{ (extrapolated)} \geq (F_Q^L(X,Y,Z))^{RPS} \text{ (extrapolated), and}$$

$$\frac{(F_Q^M(X,Y,Z)) \text{ (extrapolated)}}{(F_Q^L(X,Y,Z))^{RPS} \text{ (extrapolated)}} > \frac{(F_Q^M(X,Y,Z))}{(F_Q^L(X,Y,Z))^{RPS}}$$

either of the following actions shall be taken:

1. $F_Q^M(X,Y,Z)$ shall be increased by 2 percent over that specified in 4.2.2.2.a, and the calculations of 4.2.2.2.c repeated, or

2. A movable incore detector power distribution map shall be obtained, and the calculations of 4.2.2.2.c.1 shall be performed no later than the time at which the margin in 4.2.2.2.c.1 is extrapolated to be equal to zero.

e. The limits in Specifications 4.2.2.2.c and 4.2.2.2.d are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:

1. Lower core region from 0 to 15%, inclusive.

2. Upper core region from 85 to 100%, inclusive.

4.2.2.3 When a full core power distribution map is taken for reasons other than meeting the requirements of Specification 4.2.2.2, an overall $F_Q^M(X,Y,Z)$ shall be determined, then increased by 3% to account for manufacturing tolerances, further increased by 5% to account for measurement uncertainty, and further increased by the radial-local peaking factor to obtain a maximum local peak. This value shall be compared to the limit in Specification 3.2.2.

(5) Extrapolation of F_Q^M for the initial flux map taken after reaching equilibrium conditions is not required since the initial flux map extablishes the baseline measurement for future trending. Also, extrapolation of F_Q^M limits are not valid for core locations that were previously rodded, or for core locations that were previously within ±2% of the core height about the demand position of the rod tip.

POWER DISTRIBUTION LIMITS

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR $F\Delta H(X,Y)$

LIMITING CONDITION FOR OPERATION

3.2.3 $F_{\Delta H}(X,Y)$ shall be limited by imposing the following relationship:

$$F\Delta H^{M}(X,Y) \leq [F\Delta H^{L}(X,Y]^{LCO}$$

Where: $F\Delta H^{M}(X,Y)$ = the measured radial peak.

$[F\Delta H^{L}(X,Y)]^{LCO}$ = the maximum allowable radial peak as defined in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODE 1. ~~(UNIT 1)~~

ACTION:

With $F_{\Delta H}(X,Y)$ exceeding its limit:

a. Within 2 hours, reduce the allowable THERMAL POWER from RATED THERMAL POWER at least RRH%[1] for each 1% that $F\Delta H^{M}(X,Y)$ exceeds the limit, and

b. Within 6 hours either:

1. Restore $F\Delta H^{M}(X,Y)$ to within the limit of Specification 3.2.3 for RATED THERMAL POWER, or

2. Reduce the Power Range Neutron Flux-High Trip Setpoint in Table 2.2-1 at least RRH% for each 1% that $F\Delta H^{M}(X,Y)$ exceeds that limit, and

c. Within 72 hours of initially being outside the limit of Specification 3.2.3, either:

1. Restore $F\Delta H^{M}(X,Y)$ to within the limit of Specification 3.2.3 for RATED THERMAL POWER, or

2. Perform the following actions:

(a) Reduce the OTΔT K_1 term in Table 2.2-1 by at least TRH[2] for each 1% that $F\Delta H^{M}(X,Y)$ exceeds the limit, and

(b) Verify through incore mapping that $F\Delta H^{M}(X,Y)$ is restored to within the limit for the reduced THERMAL POWER allowed by ACTION a, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

---

[1] RRH is the amount of THERMAL POWER reduction required to compensate for each 1% that $F\Delta H^{M}(X,Y)$ exceeds the limits of Specification 3.2.3 provided in the COLR per Specification 6.9.1.9.

[2] TRH is the amount of OTΔT K_1 setpoint reduction required to compensate for each 1% that $F\Delta H^{M}(X,Y)$ exceeds the limit of Specification 3.2.3, provided in the COLR per Specification 6.9.1.9.

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

d. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a. and/or c.2., above; subsequent POWER OPERATION may proceed provided that $F\Delta H^M(X,Y)$ is demonstrated, through incore flux mapping, to be within the limit specified in the COLR prior to exceeding the following THERMAL POWER levels:

1) 50% of RATED THERMAL POWER,

2) 75% of RATED THERMAL POWER, and

3) Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 $F\Delta H^M(X,Y)$ shall be evaluated to determine whether $F_{\Delta H}(X,Y)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Measuring $F\Delta H^M(X,Y)$ according to the following schedule:

1. Upon reaching equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F\Delta H^M(X,Y)$ was last determined[3], or

2. At least once per 31 Effective Full Power Days, or

3. At each time the QUADRANT POWER TILT RATIO indicated by the excore detectors is normalized using incore detector measurements.

c. Performing the following calculations:

1. For each location, calculate the % margin to the maximum allowable design as follows:

$$\%F_{\Delta H}\ \text{Margin} = 1 - \left[\frac{F\Delta H^M(X,Y)}{[F\Delta H^L(X,Y)]^{SURV}}\right] \times 100\%$$

Where $[F\Delta H^L(X,Y)]^{SURV}$ is the design peaking limit defined in the COLR.

No additional uncertainties are required for $F\Delta H^M(X,Y)$, because $[F\Delta H^L(X,Y)]^{SURV}$, includes uncertainties.

(3) During power escalation at the beginning of each cycle, THERMAL POWER may be increased until a power level for extended operation has been achieved and a power distribution map obtained.

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

2. Find the minimum margin of all locations examined in 4.2.3.2.c.1 above. If any margin is less than zero, comply with the ACTION requirements of Specification 3.2.3 as if $[F_{\Delta H}{}^{L}(X,Y)]^{SURV}$ is the same as $[F_{\Delta H}{}^{L}(X,Y)]^{LCO}$.

d. Extrapolating[4] at least two measurements to 31 Effective Full Power Days beyond the most recent measurement and if:

$F^{M}_{\Delta H}(X,Y)$ (extrapolated) $\geq (F^{L}_{\Delta H}(X,Y))^{SURV}$ (extrapolated), and

$$\frac{F^{M}_{\Delta H}(X,Y) \text{ (extrapolated)}}{(F^{L}_{\Delta H}(X,Y))^{SURV} \text{(extrapolated)}} > \frac{F^{M}_{\Delta H}(X,Y)}{(F^{L}_{\Delta H}(X,Y))^{SURV}}$$

either of the following actions shall be taken

1. $F^{M}_{\Delta H}(X,Y)$ shall be increased by 2 percent over that specified in 4.2.3.2.a, and the calculations of 4.2.3.2.c repeated, or

2. A movable incore detector power distribution map shall be obtained, and the calculations of 4.2.3.2.c shall be performed no later than the time at which the margin in 4.2.3.2.c is extrapolated to be equal to zero.

---

(4) Extrapolation of $F^{M}_{\Delta H}$ for the initial flux map taken after reaching equilibrium conditions is not required since the initial flux map establishes the baseline measurement for future trending.

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1 above 50% of RATED THERMAL POWER ~~(Unit 1).~~ *,**

ACTION:

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but less than or equal to 1.09:

   1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

      a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

      b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

   2. Within 2 hours either:

      a) Reduce the QUADRANT POWER TILT RATIO to within its limit, or

      b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.02 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

   3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

   4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

---

*See Special Test Exceptions Specification 3.10.2.

**Not applicable until calibration of the excore detectors is completed subsequent to refueling.

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

b. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to misalignment of either a shutdown or control rod:

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

2. Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.02, within 30 minutes;

3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

c. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to causes other than the misalignment of either a shutdown or control rod:

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

3. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified at 95% or greater RATED THERMAL POWER.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

a. Calculating the ratio at least once per 7 days when the alarm is OPERABLE, and

b. Calculating the ratio at least once per 12 hours during steady-state operation when the alarm is inoperable.

c. The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75% of RATED THERMAL POWER with one Power Range channel inoperable by using the movable incore detectors to confirm that the normalized symmetric power distribution, obtained from two sets of four symmetric thimble locations or full-core flux map, is consistent with the indicated QUADRANT POWER TILT RATIO at least once per 12 hours.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg},

b. Pressurizer Pressure,

c. Reactor Coolant System Total Flow Rate.

APPLICABILITY: MODE 1. ~~(Unit 1)~~

ACTION:

a. With either of the parameters identified in 3.2.5a. and b. above exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

b. With the combination of Reactor Coolant System total flow rate and THERMAL POWER within the region of restricted operation specified on Figure 3.2-1, within 6 hours reduce the Power Range Neutron Flux-High Trip Setpoint to below the nominal setpoint by the same amount (% RTP) as the power reduction required by Figure 3.2-1.

c. With the combination of Reactor Coolant System total flow rate and THERMAL POWER within the region of prohibited operation specified on Figure 3.2-1:

1. Within 2 hours either:

a) Restore the combination of Reactor Coolant System total flow rate and THERMAL POWER to within the region of permissible operation, or

b) Restore the combination of Reactor Coolant System total flow rate and THERMAL POWER to within the region of restricted operation and comply with action b. above, or

c) Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

2. Within 24 hours of initially being within the region of prohibited operation specified on Figure 3.2-1, verify that the combination of THERMAL POWER and Reactor Coolant System total flow rate are restored to within the regions of restricted or permissible operation, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 The Reactor Coolant System total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months. The measurement instrumentation shall be calibrated within 7 days prior to the performance of the calorimetric flow measurement.

4.2.5.3 The Reactor Coolant System total flow rate shall be determined by precision heat balance measurement at least once per 18 months.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER		LIMITS Four Loops in Operation
Average Temperature		
Meter Average	- 4 channels:	$\leq$ 592°F
	- 3 channels:	$\leq$ 592°F
Computer Average	- 4 channels:	$\leq$ 593°F
	- 3 channels:	$\leq$ 593°F
Pressurizer Pressure		
Meter Average	- 4 channels:	$\geq$ 2227 psig*
	- 3 channels:	$\geq$ 2230 psig*
Computer Average	- 4 channels:	$\geq$ 2222 psig*
	- 3 channels:	$\geq$ 2224 psig*
Reactor Coolant System Total Flow Rate		Figure 3.2-1

*Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RATED THERMAL POWER.

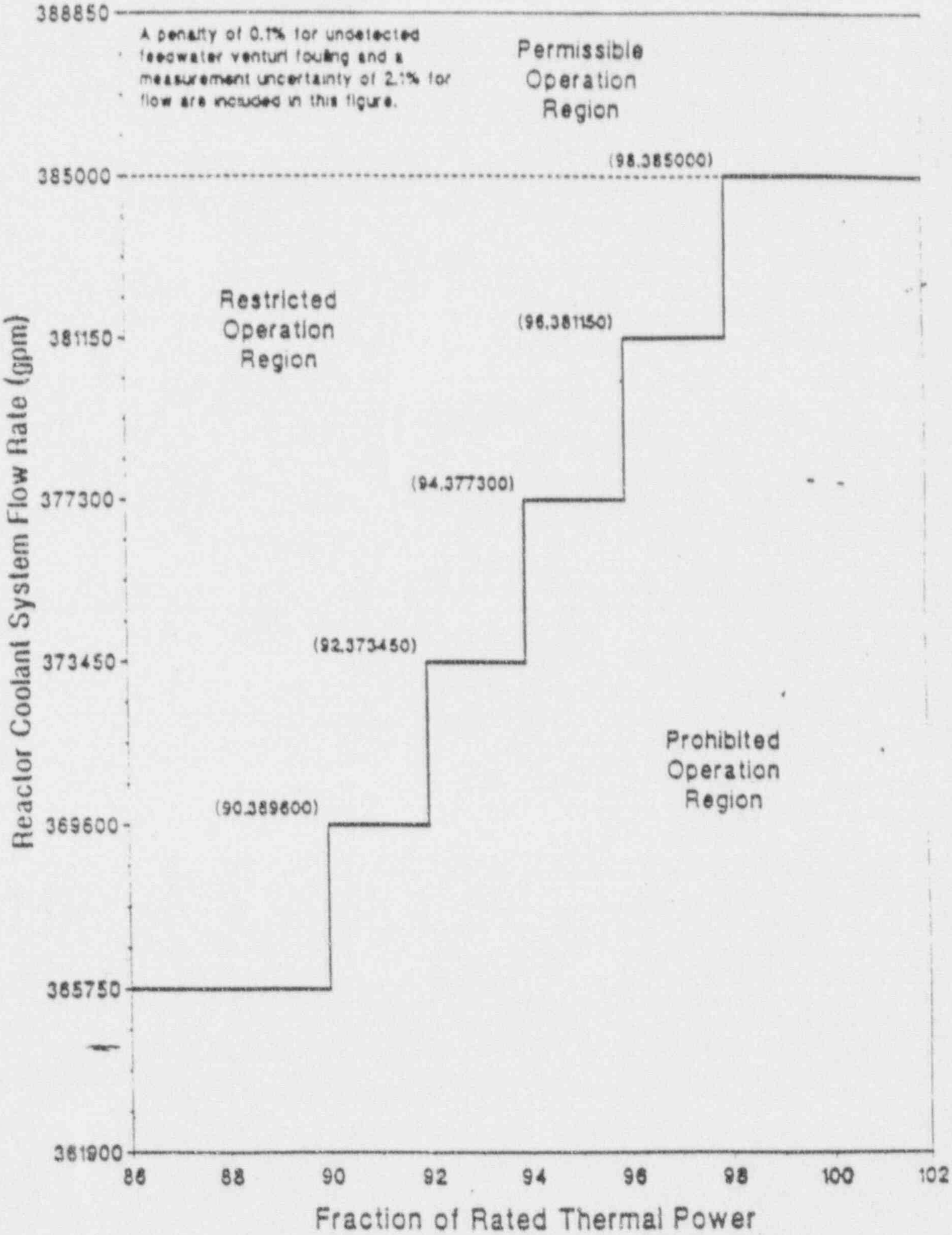

Figure 3.2-1 Reactor Coolant System Total Flow Rate Versus Rated Thermal Power - Four Loops in Operation ~~(Unit 1)~~

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within:

a. the allowed operational space as specified in the CORE OPERATING LIMITS REPORT (COLR) for RAOC operation, or

b. within the target band specified in the COLR about the target flux difference during baseload operation.

APPLICABILITY: MODE 1, above 50% of RATED THERMAL POWER* (Unit 2)

ACTION:

a. For RAOC operation with the indicated AFD outside of the limits specified in the COLR,

   1. Either restore the indicated AFD to within the COLR limits within 15 minutes, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux-High Trip setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

b. For Base Load operation above APL^{ND}** with the indicated AXIAL FLUX DIFFERENCE outside of the applicable target band about the target flux difference:

   1. Either restore the indicated AFD to within the COLR specified target band limits within 15 minutes, or

   2. Reduce THERMAL POWER to less than APL^{ND} of RATED THERMAL POWER and discontinue Base Load operation within 30 minutes.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the limits specified in the COLR.

---

*See Special Test Exceptions Specification 3.10.2.

**APL^{ND} is the minimum allowable (nuclear design) power level for base load operation and is specified in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.9.

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 50% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1) At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2) At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.

b. Monitoring and logging the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AFD Monitor Alarm is inoperable. The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

c. The provisions of Specification 4.0.4 are not applicable.

4.2.1.2 The indicated AFD shall be considered outside of its limits when at least two OPERABLE excore channels are indicating the AFD to be outside the limits.

4.2.1.3 When in Base Load operation, the target axial flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 When in Base Load operation, the target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference in conjunction with the surveillance requirements of Specification 3/4.2.2 or by linear interpolation between the most recently measured values and the calculated value at the end of cycle life. The provisions of Specification 4.0.4 are not applicable.

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \frac{F_Q^{RTP}}{P} K(Z) \text{ for } P > 0.5$$

$$F_Q(Z) \leq \frac{F_Q^{RTP}}{0.5} K(Z) \text{ for } P \leq 0.5$$

Where: F_Q^{RTP} = the F_Q Limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR),

$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

$K(Z)$ = the normalized $F_Q(Z)$ for a given core height specified in the COLR.

APPLICABILITY: MODE 1. (Unit 2)

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints (value of K_4) have been reduced at least 1% (in ΔT span) for each 1% $F_Q(Z)$ exceeds the limit, and

b. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by ACTION a., above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 For RAOC operation, $F_Q(z)$ shall be evaluated to determine if $F_Q(z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured $F_Q(z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. Verify the requirements of Specification 3.2.2 are satisfied.

c. Satisfying the following relationship:

$$F_Q^M(z) \leq \frac{F_Q^{RTP} \times K(z)}{P \times W(z)} \text{ for } P > 0.5$$

$$F_Q^M(z) \leq \frac{F_Q^{RTP} \times K(z)}{W(z) \times 0.5} \text{ for } P \leq 0.5$$

where $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, F_Q^{RTP} is the F_Q limit, $K(z)$ is the normalized $F_Q(z)$ as a function of core height, P is the relative THERMAL POWER, and $W(z)$ is the cycle dependent function that accounts for power distribution transients encountered during normal operation. F_Q^{RTP}, $K(z)$, and $W(z)$ are specified in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.9.

d. Measuring $F_Q^M(z)$ according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(z)$ was last determined,* or

2. At least once per 31 Effective Full Power Days, whichever occurs first.

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*During power escalation at the beginning of each cycle, power level may be increased until a power level for extended operation has been achieved and a power distribution map obtained.

SURVEILLANCE REQUIREMENTS (Continued)

e. With measurements indicating

$$\underset{\text{over } z}{\text{maximum}} \quad \frac{F_Q^M(z)}{K(z)}$$

has increased since the previous determination of $F_Q^M(z)$ either of the following actions shall be taken:

1) $F_Q^M(z)$ shall be increased by 2% over that specified in Specification 4.2.2.2c., or

2) $F_Q^M(z)$ shall be measured at least once per 7 Effective Full Power Days until two successive maps indicate that

$$\underset{\text{over } z}{\text{maximum}} \quad \frac{F_Q^M(z)}{K(z)} \quad \text{is not increasing.}$$

f. With the relationships specified in Specification 4.2.2.2c. above not being satisfied:

1) Calculate the percent $F_Q(z)$ exceeds its limit by the following expression:

$$\left\{\left(\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(z) \times W(z)}{\frac{F_Q^{RTP}}{P} \times K(z)}\right]\right) - 1\right\} \times 100 \text{ for } P \geq 0.5$$

$$\left\{\left(\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(z) \times W(z)}{\frac{F_Q^{RTP}}{0.5} \times K(z)}\right]\right) - 1\right\} \times 100 \text{ for } P < 0.5$$

2) One of the following actions shall be taken:

a) Within 15 minutes, control the AFD to within new AFD limits which are determined by reducing the AFD limits of Specification 3.2.1 by 1% AFD for each percent $F_Q(z)$ exceeds its limits as determined in Specification 4.2.2.2f.1). Within 8 hours, reset the AFD alarm setpoints to these modified limits, or

b) Comply with the requirements of Specification 3.2.2 for $F_Q(z)$ exceeding its limit by the percent calculated above, or

c) Verify that the requirements of Specification 4.2.2.3 for Base Load operation are satisfied and enter Base Load operation.

g. The limits specified in Specifications 4.2.2.2c., 4.2.2.2e., and 4.2.2.2f., above are not applicable in the following core plane regions:

1. Lower core region from 0 to 15%, inclusive
2. Upper core region from 85 to 100%, inclusive.

4.2.2.3 Base Load operation is permitted at powers above APL^{ND*} if the following conditions are satisfied:

a. Prior to entering Base Load operation, maintain THERMAL POWER above APL^{ND} and less than or equal to that allowed by Specification 4.2.2.2 for at least the previous 24 hours. Maintain Base Load operation surveillance (AFD within the target band about the target flux difference of Specification 3.2.1) during this time period. Base Load operation is then permitted providing THERMAL POWER is maintained between APL^{ND} and APL^{BL} or between APL^{ND} and 100% (whichever is most limiting) and FQ surveillance is maintained pursuant to Specification 4.2.2.4. APL^{BL} is defined as:

$$APL^{BL} = \underset{\text{over } Z}{\text{minimum}} \left[\frac{F_Q^{RTP} \times K(Z)}{F_Q^M(Z) \times W(Z)_{BL}} \right] \times 100\%$$

where: $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty. F_Q^{RTP} is the F_Q limit, K(z) is the normalized $F_Q(Z)$ as a function of core height. $W(z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during Base Load operation. F_Q^{RTP}, K(z), and $W(Z)_{BL}$ are specified in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.9.

b. During Base Load operation, if the THERMAL POWER is decreased below APL^{ND} then the conditions of 4.2.2.3a shall be satisfied before re-entering Base Load operation.

4.2.2.4 During Base Load Operation $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER above APL^{ND}.

b. Increasing the measured $F_Q(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. Verify the requirements of Specification 3.2.2 are satisfied.

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*APL^{ND} is the minimum allowable (nuclear design) power level for Base Load operation in Specification 3.2.1.

SURVEILLANCE REQUIREMENTS (Continued)

c. Satisfying the following relationship:

$$F_Q^M(Z) \leq \frac{F_Q^{RTP} \times K(Z)}{P \times W(Z)_{BL}} \text{ for } P > APL^{ND}$$

where: $F_Q^M(Z)$ is the measured $F_Q(Z)$. F_Q^{RTP} is the F_Q limit. $K(Z)$ is the normalized $F_Q(Z)$ as a function of core height. P is the relative THERMAL POWER. $W(Z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during Base Load operation. F_Q^{RTP}, $K(Z)$, and $W(Z)_{BL}$ are specified in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.9.

d. Measuring $F_Q^M(Z)$ in conjunction with target flux difference determination according to the following schedule:

1. Prior to entering Base Load operation after satisfying surveillance 4.2.2.3 unless a full core flux map has been taken in the previous 31 EFPD with the relative thermal power having been maintained above APL^{ND} for the 24 hours prior to mapping, and

2. At least once per 31 effective full power days.

e. With measurements indicating

$$\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(z)}{K(z)}\right]$$

has increased since the previous determination $F_Q^M(Z)$ either of the following actions shall be taken:

1. $F_Q^M(Z)$ shall be increased by 2 percent over that specified in 4.2.2.4c, or

2. $F_Q^M(Z)$ shall be measured at least once per 7 EFPD until 2 successive maps indicate that

$$\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(z)}{K(z)}\right] \text{ is not increasing.}$$

f. With the relationship specified in 4.2.2.4c above not being satisfied, either of the following actions shall be taken:

1. Place the core in an equilibrium condition where the limit in 4.2.2.2c is satisfied, and remeasure $F_Q^M(Z)$, or

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

2. Comply with the requirements of Specification 3.2.2 for $F_Q(Z)$ exceeding its limit by the percent calculated with the following expression:

$$[(\text{max. over } z \text{ of } [\frac{F_Q^M(Z) \times W(Z)_{BL}}{\frac{F_Q^{RTP}}{P} \times K(Z)}]) -1] \times 100 \text{ for } P \geq APL^{ND}$$

g. The limits specified in 4.2.2.4c., 4.2.2.4e., and 4.2.2.4f. above are not applicable in the following core plan regions:

1. Lower core region 0 to 15 percent, inclusive.

2. Upper core region 85 to 100 percent, inclusive.

4.2.2.5 When $F_Q(Z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.2 an overall measured $F_Q(z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

3/4.2.3 REACTOR COOLANT SYSTEM FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System total flow rate and R shall be maintained within the region of permissible operation specified in the CORE OPERATING LIMITS REPORT (COLR) for four loop operation.

Where:

a. $R = \dfrac{F^N_{\Delta H}}{F^{RTP}_{\Delta H}[1.0 + MF_{\Delta H}\ (1.0 - P)]}$,

b. $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$,

c. $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^N_{\Delta H}$ shall be used to calculate R since the figure specified in the COLR includes penalties for undetected feedwater venturi fouling of 0.1% and for measurement uncertainties of 2.1% for flow and 4% for incore measurement of $F^N_{\Delta H}$,

d. $F^{RTP}_{\Delta H}$ = The $F^N_{\Delta H}$ limit at RATED THERMAL POWER (RTP) specified in the COLR, and

e. $MF_{\Delta H}$ = The power factor multiplier specified in the COLR.

APPLICABILITY: MODE 1 (UNIT 2).

ACTION:

a. With the combination of Reactor Coolant System total flow rate and R within the region of restricted operation within 6 hours reduce the Power Range Neutron Flux-High Trip Setpoint to below the nominal setpoint by the same amount (% RTP) as the power reduction required by the figure specified in the COLR.

b. With the combination of Reactor Coolant System total flow rate and R within the region of prohibited operation specified in the COLR:

1. Within 2 hours either:

   a) Restore the combination of Reactor Coolant System total flow rate and R to within the region of permissible operation, or

   b) Restore the combination of Reactor Coolant System total flow rate and R to within the region of restricted operation and comply with action a. above, or

POWER DISTRIBUTION LIMITS

3/4.2.3 REACTOR COOLANT SYSTEM FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c) Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

2. Within 24 hours of initially being within the region of prohibited operation specified in the COLR, verify through incore flux mapping and Reactor Coolant System total flow rate comparison that the combination of R and Reactor Coolant System total flow rate are restored to within the regions of restricted or permissible operation, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

3. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION b.1.c) and/or b.2., above; subsequent POWER OPERATION may proceed provided that the combination of R and indicated Reactor Coolant System total flow rate are demonstrated, through incore flux mapping and Reactor Coolant System total flow rate comparison, to be within the regions of restricted or permissible operation specified in the COLR prior to exceeding the following THERMAL POWER levels:

   a) A nominal 50% of RATED THERMAL POWER,

   b) A nominal 75% of RATED THERMAL POWER, and

   c) Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated Reactor Coolant System total flow rate determined by process computer readings or digital voltmeter measurement and R shall be determined to be within the regions of restricted or permissible operation specified in the COLR:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

4.2.3.3 The indicated Reactor Coolant System total flow rate shall be verified to be within the regions of restricted or permissible operation specified in the COLR at least once per 12 hours when the most recently obtained value of R, obtained per Specification 4.2.3.2, is assumed to exist.

4.2.3.4 The Reactor Coolant System total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months. The measurement instrumentation shall be calibrated within 7 days prior to the performance of the calorimetric flow measurement.

4.2.3.5 The Reactor Coolant System total flow rate shall be determined by precision heat balance measurement at least once per 18 months.

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1, *above 50% of RATED THERMAL POWER (Unit 2)

ACTION:

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but less than or equal to 1.09:

  1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

     a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

     b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

  2. Within 2 hours either:

     a) Reduce the QUADRANT POWER TILT RATIO to within its limit, or

     b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

  3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

  4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

*See Special Test Exceptions Specification 3.10.2.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

b. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to misalignment of either a shutdown or control rod:

   1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

      a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

      b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

   2. Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1, within 30 minutes;

   3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

   4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

c. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to causes other than the misalignment of either a shutdown or control rod:

   1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

      a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

      b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

3. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified at 95% or greater RATED THERMAL POWER.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

a. Calculating the ratio at least once per 7 days when the alarm is OPERABLE, and

b. Calculating the ratio at least once per 12 hours during steady-state operation when the alarm is inoperable.

c. The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75% of RATED THERMAL POWER with one Power Range channel inoperable by using the movable incore detectors to confirm that the normalized symmetric power distribution, obtained from two sets of four symmetric thimble locations or full-core flux map, is consistent with the indicated QUADRANT POWER TILT RATIO at least once per 12 hours.

POWER DISTRIBUTION LIMITS

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg}, and

b. Pressurizer Pressure.

APPLICABILITY: MODE 1, Unit 2 only.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER		LIMITS Four Loops in Operation
Average Temperature		
Meter Average	- 4 channels:	$\leq$ 592°F
	- 3 channels:	$\leq$ 592°F
Computer Average	- 4 channels:	$\leq$ 593°F
	- 3 channels:	$\leq$ 593°F
Pressurizer Pressure		
Meter Average	- 4 channels:	$\geq$ 2227 psig*
	- 3 channels:	$\geq$ 2230 psig*
Computer Average	- 4 channels:	$\geq$ 2222 psig*
	- 3 channels:	$\geq$ 2224 psig*

*Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RATED THERMAL POWER.

TABLE 3.3-1 ~~FOR UNIT 1~~

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	10
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2
b. Low Setpoint	4	2	3	1###, 2	2
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2
4. Intermediate Range, Neutron Flux	2	1	2	1###, 2	3
5. Source Range, Neutron Flux					
a. Startup	2	1	2	2##	4
b. Shutdown	2	1	2	3*, 4*, 5*	10
6. Overtemperature ΔT Four Loop Operation	4	2	3	1, 2	6
7. Overpower ΔT Four Loop Operation	4	2	3	1, 2	6
8. Pressurizer Pressure-Low	4	2	3	1	6**

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
9. Pressurizer Pressure-High	4	2	3	1, 2	6**
10. Pressurizer Water Level-High	3	2	2	1	6
11. Reactor Coolant Flow-Low					
a. Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	6
b. Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop each operating loop	1	6
12. Steam Generator Water Level--Low-Low	4/stm gen	2/stm gen in any operating stm gen	3/stm gen each operating stm gen	1, 2	6**
13. Undervoltage-Reactor Coolant Pumps (Above P-7)	4-1/bus	2	3	1	6
14. Underfrequency-Reactor Coolant Pumps (Above P-7)	4-1/bus	2	3	1	6
15. Turbine Trip					
a. Stop Valve EH Pressure - Low	4	2	3	1####	6
b. Turbine Stop Valve Closure	4	4	1	1####	11
16. Safety Injection Input from ESF	2	1	2	1, 2	9

~~UNIT 1~~

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
17. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2	1	2	2##	8
b. Low Power Reactor Trips Block, P-7					
P-10 Input	4	2	3	1	8
or					
P-13 Input	2	1	2	1	8
c. Power Range Neutron Flux, P-8	4	2	3	1	8
d. Power Range Neutron Flux, P-9	4	2	3	1	8
e. Power Range Neutron Flux, P-10	4	2	3	1	8
f. Power Range Neutron Flux, Not P-10	4	3	4	1, 2	8
g. Turbine Impulse Chamber Pressure, P-13	2	1	2	1	8
18. Reactor Trip Breakers	2	1	2	1, 2	9, 12
	2	1	2	3*, 4*, 5*	10
19. Automatic Trip and Interlock Logic	2	1	2	1, 2	9
	2	1	2	3*, 4*, 5*	10
20. Reactor Trip Bypass Breakers	N.A.	N.A.	N.A.	1,2,3*,4*,5*	13

Amendment No. 101 (Unit 1)
Amendment No. 95 (Unit 2)

TABLE NOTATIONS

*Only if the Reactor Trip System breakers happen to be in the closed position and the Control Rod Drive System is capable of rod withdrawal.

**Comply with the provisions of Specification 3.3.2, for any portion of the channel required to be OPERABLE by Specification 3.3.2.

##Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

###Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

####Above the P-9 (Reactor Trip on Turbine Trip Interlock) Setpoint.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours,

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1, and

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.2.

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint; or

b. Above the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint but below 10% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED THERMAL POWER.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - Delete

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1.

ACTION 7 - Delete

ACTION 8 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive status light(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 10 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor trip breakers within the next hour.

ACTION 11 - With the number of OPERABLE channels less than the Total Number of Channels, operation may continue provided the inoperable channels are placed in the tripped condition within 6 hours.

ACTION 12 - With one of the diverse trip features (Undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 9. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status. With the breaker bypassed, apply ACTION 9.

ACTION 13 - With any reactor trip bypass breaker inoperable, restore the bypass breaker to OPERABLE status prior to placing it in service.

TABLE 3.3-2 ~~FOR UNIT 1~~

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	N.A.
2. Power Range, Neutron Flux	≤ 0.5 second*
3. Power Range, Neutron Flux, High Positive Rate	N.A.
4. Intermediate Range, Neutron Flux	N.A.
5. Source Range, Neutron Flux	N.A.
6. Overtemperature ΔT	≤ 8 seconds*
7. Overpower ΔT	≤ 8 seconds*
8. Pressurizer Pressure-Low	≤ 2 seconds
9. Pressurizer Pressure-High	≤ 2 seconds
10. Pressurizer Water Level-High	N.A.

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

TABLE 3.3-2 ~~FOR UNIT 1~~ (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
11. Low Reactor Coolant Flow	
a. Single Loop (Above P-8)	$\leq$ 1 second
b. Two Loops (Above P-7 and below P-8)	$\leq$ 1 second
12. Steam Generator Water Level-Low-Low	
a. Unit 1	$\leq$ 3.5 seconds
b. Unit 2	$\leq$ 2.0 seconds
13. Undervoltage-Reactor Coolant Pumps	$\leq$ 1.5 seconds
14. Underfrequency-Reactor Coolant Pumps	$\leq$ 0.6 second
15. Turbine Trip	
a. Stop Valve EH Pressure-Low	N.A.
b. Turbine Stop Valve Closure	N.A.
16. Safety Injection Input from ESF	N.A.
17. Reactor Trip System Interlocks	N.A.
18. Reactor Trip Breakers	N.A.
19. Automatic Trip and Interlock Logic	N.A.

TABLE 4.3-1 ~~FOR UNIT 1~~

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R(14)	N.A.	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux						
a. High Setpoint	S	D(2, 4), M(3, 4), Q(4, 6), R(4, 5)	M	N.A.	N.A.	1, 2
b. Low Setpoint	S	R(4)	M	N.A.	N.A.	1###, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(4)	M	N.A.	N.A.	1, 2
4. Intermediate Range, Neutron Flux	S	R(4, 5)	S/U(1),M	N.A.	N.A.	1###, 2
5. Source Range, Neutron Flux	S	R(4, 5)	S/U(1),M(9)	N.A.	N.A.	2##, 3, 4, 5
6. Overtemperature ΔT	S	R	M	N.A.	N.A.	1, 2
7. Overpower ΔT	S	R	M	N.A.	N.A.	1, 2
8. Pressurizer Pressure-Low	S	R	M	N.A.	N.A.	1
9. Pressurizer Pressure-High	S	R	M	N.A.	N.A.	1, 2
10. Pressurizer Water Level-High	S	R	M	N.A.	N.A.	1
11. Reactor Coolant Flow-Low	S	R	M	N.A.	N.A.	1

TABLE 4.3-1 ~~FOR UNIT 1~~ (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
12. Steam Generator Water Level-Low-Low	S	R(13)	M	N.A.	N.A.	1, 2
13. Undervoltage - Reactor Coolant Pumps	N.A.	R	N.A.	M	N.A.	1
14. Underfrequency - Reactor Coolant Pumps	N.A.	R	N.A.	M	N.A.	1
15. Turbine Trip						
a. Stop Valve EH Pressure - Low	N.A.	R	N.A.	S/U(1, 10)	N.A.	1#
b. Turbine Stop Valve Closure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1#
16. Safety Injection Input from ESF	N.A.	N.A.	N.A.	R**	N.A.	1, 2
17. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	R(4)	M	N.A.	N.A.	2##
b. Low Power Reactor Trips Block, P-7	N.A.	R(4)	M(8)	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	R(4)	M(8)	N.A.	N.A.	1
d. Low Power Range Neutron Flux, P-9	N.A.	R(4)	M(8)	N.A.	N.A.	1

** This surveillance need not be performed until prior to entering STARTUP following the Unit 1 first refueling.

TABLE 4.3-1 ~~FOR UNIT 1~~ (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
17. Reactor Trip System Interlocks (Continued)						
e. Power Range Neutron Flux, P-10	N.A.	R(4)	M(8)	N.A.	N.A.	1
f. Power Range Neutron Flux, Not P-10	N.A.	R(4)	M(8)	N.A.	N.A.	1, 2
g. Turbine Impulse Chamber Pressure, P-13	N.A.	R	M(8)	N.A.	N.A.	1
18. Reactor Trip Breaker	N.A.	N.A.	N.A.	M(7, 11)	N.A.	1, 2, 3*, 4*, 5*
19. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M(7)	1, 2, 3*, 4*, 5*
20. Reactor Trip Bypass Breakers	N.A.	N.A.	N.A.	M(7,15)R(16)	N.A	1, 2, 3*, 4*, 5*

TABLE 4.3-1 ~~FOR UNIT 1~~ (Continued)

TABLE NOTATIONS

* Only if the Reactor Trip System breakers happen to be closed and the Control Rod Drive System is capable of rod withdrawal.

Above P-9 (Reactor Trip on Turbine Trip Interlock) Setpoint.

Below P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

(1) If not performed in previous 7 days.

(2) Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference is greater than 2%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(3) Single point comparison of incore to excore axial flux difference above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(4) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) Detector plateau curves shall be obtained, evaluated and compared to manufacturer's data. For the Intermediate Range and Power Range Neutron Flux channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(6) Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(7) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(8) With power greater than or equal to the interlock setpoint the required ANALOG CHANNEL OPERATIONAL TEST shall consist of verifying that the interlock is in the required state by observing the permissive status light.

(9) Monthly surveillance in MODES 3*, 4*, and 5* shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive status light.

(10) Setpoint verification is not applicable.

(11) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include independent verification of the operability of the Undervoltage and Shunt trips.

(12) Deleted

(13) For Unit 1, CHANNEL CALIBRATION shall ensure that the filter time constant associated with Steam Generator Water Level Low-Low is adjusted to a value less than or equal to 1.5 seconds.

(14) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip circuits for the Manual Reactor Trip Function. The test shall also verify the OPERABILITY of the Bypass Breaker trip circuit(s).

TABLE 4.3-1 ~~FOR UNIT 1~~ (Continued)

TABLE NOTATIONS

(15) A local manual shunt trip on the bypass breakers shall be performed prior to placing breaker in service.

(16) The automatic undervoltage trip capability shall be verified OPERABLE.

TABLE 3.3-1 FOR UNIT 2

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2 2	1 1	2 2	1, 2 3*, 4*, 5*	1 10
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2
b. Low Setpoint	4	2	3	1###, 2	2
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2
5. Intermediate Range, Neutron Flux	2	1	2	1###, 2	3
6. Source Range, Neutron Flux					
a. Startup	2	1	2	2##	4
b. Shutdown	2	1	2	3*, 4*, 5*	10
7. Overtemperature ΔT Four Loop Operation	4	2	3	1, 2	6
8. Overpower ΔT Four Loop Operation	4	2	3	1, 2	6
9. Pressurizer Pressure-Low	4	2	3	1	6**

TABLE 3.3-1 FOR UNIT 2 (Continued)

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
10. Pressurizer Pressure-High	4	2	3	1, 2	6**
11. Pressurizer Water Level-High	3	2	2	1	6
12. Reactor Coolant Flow-Low					
a. Single Loop (Above P-8)	3/loop any operating loop	2/loop in each operating loop	2/loop in	1	6
b. Two Loops (Above P-7 and below P-8)	3/loop two operating loops	2/loop in each operating loop	2/loop	1	6
13. Steam Generator Water Level--Low-Low	4/stm gen	2/stm gen in any operating stm gen	3/stm gen each operating stm gen	1, 2	6**
14. Undervoltage-Reactor Coolant Pumps (Above P-7)	4-1/bus	2	3	1	6
15. Underfrequency-Reactor Coolant Pumps (Above P-7)	4-1/bus	2	3	1	6
16. Turbine Trip					
a. Stop Valve EH Pressure - Low	4	2	3	1####	6
b. Turbine Stop Valve Closure	4	4	1	1####	11
17. Safety Injection Input from ESF	2	1	2	1, 2	9

TABLE 3.3-1 FOR UNIT 2 (Continued)

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2	1	2	2##	8
b. Low Power Reactor Trips Block, P-7					
P-10 Input	4	2	3	1	8
or					
P-13 Input	2	1	2	1	8
c. Power Range Neutron Flux, P-8	4	2	3	1	8
d. Power Range Neutron Flux, P-9	4	2	3	1	8
e. Power Range Neutron Flux, P-10	4	2	3	1	8
f. Power Range Neutron Flux, Not P-10	4	3	4	1, 2	8
g. Turbine Impulse Chamber Pressure, P-13	2	1	2	1	8
19. Reactor Trip Breakers	2	1	2	1, 2	9, 12
	2	1	2	3*, 4*, 5*	10
20. Automatic Trip and Interlock Logic	2	1	2	1, 2	9
	2	1	2	3*, 4*, 5*	10
21. Reactor Trip Bypass Breakers	N.A.	N.A.	N.A.	1, 2, 3*, 4*, 5*	13

TABLE NOTATIONS

*Only if the Reactor Trip System breakers happen to be in the closed position and the Control Rod Drive System is capable of rod withdrawal.

**Comply with the provisions of Specification 3.3.2, for any portion of the channel required to be OPERABLE by Specification 3.3.2.

##Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

###Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

####Above the P-9 (Reactor Trip on Turbine Trip Interlock) Setpoint.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours,

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1, and

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.2.

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint; or

b. Above the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint but below 10% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED THERMAL POWER.

TABLE 3.3-1 FOR UNIT 2 (Continued)

ACTION STATEMENTS (Continued)

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - Delete

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1.

ACTION 7 - Delete

ACTION 8 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive status light(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 10 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor trip breakers within the next hour.

ACTION 11 - With the number of OPERABLE channels less than the Total Number of Channels, operation may continue provided the inoperable channels are placed in the tripped condition within 6 hours.

ACTION 12 - With one of the diverse trip features (Undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 9. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status. With the breaker bypassed, apply ACTION 9.

ACTION 13 - With any reactor trip bypass breaker inoperable, restore the bypass breaker to OPERABLE status prior to placing it in service.

TABLE 3.3-2 FOR UNIT 2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	N.A.
2. Power Range, Neutron Flux	$\leq$ 0.5 second*
3. Power Range, Neutron Flux, High Positive Rate	N.A.
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 0.5 second*
5. Intermediate Range, Neutron Flux	N.A.
6. Source Range, Neutron Flux	N.A.
7. Overtemperature ΔT	$\leq$ 8 seconds*
8. Overpower ΔT	$\leq$ 8 seconds*
9. Pressurizer Pressure-Low	$\leq$ 2 seconds
10. Pressurizer Pressure-High	$\leq$ 2 seconds
11. Pressurizer Water Level-High	N.A.

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

TABLE 3.3-2 FOR UNIT 2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
12. Low Reactor Coolant Flow	
a. Single Loop (Above P-8)	$\leq$ 1 second
b. Two Loops (Above P-7 and below P-8)	$\leq$ 1 second
13. Steam Generator Water Level-Low-Low	
a. Unit 1	$\leq$ 3.5 seconds
b. Unit 2	$\leq$ 2.0 seconds
14. Undervoltage-Reactor Coolant Pumps	$\leq$ 1.5 seconds
15. Underfrequency-Reactor Coolant Pumps	$\leq$ 0.6 second
16. Turbine Trip	
a. Stop Valve EH Pressure-Low	N.A.
b. Turbine Stop Valve Closure	N.A.
17. Safety Injection Input from ESF	N.A.
18. Reactor Trip System Interlocks	N.A.
19. Reactor Trip Breakers	N.A.
20. Automatic Trip and Interlock Logic	N.A.

TABLE 4.3-1 FOR UNIT 2

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R(14)	N.A.	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux						
a. High Setpoint	S	D(2, 4), M(3, 4), Q(4, 6), R(4, 5)	M	N.A.	N.A.	1, 2
b. Low Setpoint	S	R(4)	M	N.A.	N.A.	1###, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(4)	M	N.A.	N.A.	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(4)	M	N.A.	N.A.	1, 2
5. Intermediate Range, Neutron Flux	S	R(4, 5)	S/U(1),M	N.A.	N.A.	1###, 2
6. Source Range, Neutron Flux	S	R(4, 5)	S/U(1),M(9)	N.A.	N.A.	2##, 3, 4, 5
7. Overtemperature ΔT	S	R	M	N.A.	N.A.	1, 2
8. Overpower ΔT	S	R	M	N.A.	N.A.	1, 2
9. Pressurizer Pressure-Low	S	R	M	N.A.	N.A.	1
10. Pressurizer Pressure-High	S	R	M	N.A.	N.A.	1, 2
11. Pressurizer Water Level-High	S	R	M	N.A.	N.A.	1
12. Reactor Coolant Flow-Low	S	R	M	N.A.	N.A.	1

TABLE 4.3-1 FOR UNIT 2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Steam Generator Water Level-Low-Low	S	R(13)	M	N.A.	N.A.	1, 2
14. Undervoltage - Reactor Coolant Pumps	N.A.	R	N.A.	M	N.A.	1
15. Underfrequency - Reactor Coolant Pumps	N.A.	R	N.A.	M	N.A.	1
16. Turbine Trip						
a. Stop Valve EH Pressure - Low	N.A.	R	N.A.	S/U(1, 10)	N.A.	1#
b. Turbine Stop Valve Closure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1#
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	R**	N.A.	1, 2
18. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	R(4)	M	N.A.	N.A.	2##
b. Low Power Reactor Trips Block, P-7	N.A.	R(4)	M(8)	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	R(4)	M(8)	N.A.	N.A.	1
d. Low Power Range Neutron Flux, P-9	N.A.	R(4)	M(8)	N.A.	N.A.	1

** This surveillance need not be performed until prior to entering STARTUP following the Unit 1 first refueling.

TABLE 4.3-1 FOR UNIT 2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
18. Reactor Trip System Interlocks (Continued)						
e. Power Range Neutron Flux, P-10	N.A.	R(4)	M(8)	N.A.	N.A.	1
f. Power Range Neutron Flux, Not P-10	N.A.	R(4)	M(8)	N.A.	N.A.	1, 2
g. Turbine Impulse Chamber Pressure, P-13	N.A.	R	M(8)	N.A.	N.A.	1
19. Reactor Trip Breaker	N.A.	N.A.	N.A.	M(7, 11)	N.A.	1, 2, 3*, 4*, 5*
20. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M(7)	1, 2, 3*, 4*, 5*
21. Reactor Trip Bypass Breakers	N.A.	N.A.	N.A	M(7,15)R(16)	N.A	1, 2, 3*, 4*, 5*

TABLE 4.3-1 FOR UNIT 2 (Continued)

TABLE NOTATIONS

* Only if the Reactor Trip System breakers happen to be closed and the Control Rod Drive System is capable of rod withdrawal.

\# Above P-9 (Reactor Trip on Turbine Trip Interlock) Setpoint.

\## Below P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

\### Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

(1) If not performed in previous 7 days.

(2) Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference is greater than 2%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(3) Single point comparison of incore to excore axial flux difference above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(4) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) Detector plateau curves shall be obtained, evaluated and compared to manufacturer's data. For the Intermediate Range and Power Range Neutron Flux channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(6) Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(7) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(8) With power greater than or equal to the interlock setpoint the required ANALOG CHANNEL OPERATIONAL TEST shall consist of verifying that the interlock is in the required state by observing the permissive status light.

(9) Monthly surveillance in MODES 3*, 4*, and 5* shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive status light.

(10) Setpoint verification is not applicable.

(11) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include independent verification of the Undervoltage and Shunt trips.

(12) Deleted

(13) For Unit 1, CHANNEL CALIBRATION shall ensure that the filter time constant associated with Steam Generator Water Level Low-Low is adjusted to a value less than or equal to 1.5 seconds.

(14) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip circuits for the Manual Reactor Trip Function. The test shall also verify the OPERABILITY of the Bypass Breaker trip circuit(s).

TABLE 4.3-1 FOR UNIT 2 (Continued)

TABLE NOTATIONS

(15) A local manual shunt trip on the bypass breakers shall be performed prior to placing breaker in service.

(16) The automatic undervoltage trip capability shall be verified OPERABLE.

TABLE 3.3-4 ~~For Unit 1~~

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
1. Safety Injection (Reactor Trip, Phase "A" Isolation, Feedwater Isolation, Control Room Area Ventilation Operation, Auxiliary Feedwater-Motor-Driven Pump, Purge & Exhaust Isolation, Annulus Ventilation Operation, Auxiliary Building Filtered Exhaust Operation, Emergency Diesel Generator Operation, Component Cooling Water, Turbine Trip, and Nuclear Service Water Operation)		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure-High	$\leq$ 1.2 psig	$\leq$ 1.4 psig
d. Pressurizer Pressure-Low	$\geq$ 1845 psig	$\geq$ 1839 psig
e. Steam Line Pressure-Low	$\geq$ 775 psig	$\geq$ 744 psig
2. Containment Spray		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure-High-High	$\leq$ 3 psig	$\leq$ 3.2 psig

TABLE 3.3-4 ~~For Unit 1~~ (Continued)

ENGINEERED SAFETY FEATURES A TUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
3. Containment Isolation		
a. Phase "A" Isolation		
1) Manual Initiation	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.	
b. Phase "B" Isolation (Nuclear Service Water Operation)		
1) Manual Initiation	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Containment Pressure-High-High	$\leq$ 3 psig	$\leq$ 3.2 psig
c. Purge and Exhaust Isolation		
1) Manual Initiation	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.	

TABLE 3.3-4 ~~For Unit 1~~ (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
4. Steam Line Isolation		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure-High-High	≤ 3 psig	≤ 3.2 psig
d. Steam Line Pressure - Low	≥ 775 psig	≥ 744 psig
e. Steam Line Pressure-Negative Rate - High	≤ 100 psi	≤ 122.8 psi**
5. Feedwater Isolation		
a. Automatic Actuation Logic Actuation Relays	N.A.	N.A.
b. Steam Generator Water Level-High-High (P-14)		
1. Unit 1	≤ 82.4% of narrow range instrument span	≤ 84.2% of narrow range instrument span
2. Unit 2	≤ 77.1% of narrow range instrument span	≤ 78.9% of narrow range instrument span
c. T_{avg}-Low	≥ 564°F	≥ 561°F
d. Doghouse Water Level-High	11 inches above 577' floor level	12 inches above 577' floor level
e. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values	

TABLE 3.3-4 ~~for Unit 1~~ (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
6. Turbine Trip		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Steam Generator Water Level-High-High (P-14)		
1. Unit 1	$\leq$ 82.4% of narrow range instrument span	$\leq$ 84.2% of narrow range instrument span
2. Unit 2	$\leq$ 77.1% of narrow range instrument span	$\leq$ 78.9% of narrow range instrument span
d. Trip of All Main Feedwater Pumps	N.A.	N.A.
e. Reactor Trip (P-4)	N.A.	N.A.
f. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.	
7. Containment Pressure Control System		
a. Start Permissive	$\leq$ 0.4 psid	$\leq$ 0.45 psid
b. Termination	$\geq$ 0.3 psid	$\geq$ 0.25 psid
8. Auxiliary Feedwater		
a. Manual Initiation	N.A	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.

TABLE 3.3-4 ~~For Unit 1~~ (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
8. Auxiliary Feedwater (Continued)		
c. Steam Generator Water Level - Low-Low		
1) Unit 1	$\geq$ 17% of span from 0% to 30% RTP increasing linearly to $\geq$ 40.0% of span from 30% to 100% RTP	$\geq$ 15.3% of span from 0% to 30% RTP increasing linearly to $\geq$ 38.3% of span from 30% to 100% RTP
2) Unit 2	$\geq$ 36.8% of narrow range span	$\geq$ 35.1% of narrow range instrument span
d. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.	
e. Loss-of-Offsite Power	$\geq$ 3500 V	$\geq$ 3200 V
f. Trip of All Main Feedwater Pumps	N.A.	N.A.
g. Auxiliary Feedwater Suction Pressure-Low		
1) CAPS 5220, 5221, 5222	$\geq$ 10.5 psig	$\geq$ 9.5 psig
2) CAPS 5230, 5231, 5232	$\geq$ 6.2 psig	$\geq$ 5.2 psig
a. Unit 1	$\geq$ 6.2 psig	$\geq$ 5.2 psig
b. Unit 2	$\geq$ 6.0 psig	$\geq$ 5.0 psig
9. Containment Sump Recirculation		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Refueling Water Storage Tank Level-Low	$\geq$ 177.15 inches	$\geq$ 162.4 inches
Coincident With Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.	

TABLE 3.3-4 ~~For Unit 1~~ (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
10. Loss of Power		
a. 4 kV Bus Undervoltage-Loss of Voltage	≥ 3500 V	≥ 3200 V
b. 4 kV Bus Undervoltage-Grid Degraded Voltage	≥ 3685 V	≥ 3611 V
11. Control Room Area Ventilation Operation		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Loss-of-Offsite Power	≥ 3500 V	≥ 3200 V
c. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.	
12. Containment Air Return and Hydrogen Skimmer Operation		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure-High-High	≤ 3 psig	≤ 3.2 psig

TABLE 3.3-4 ~~For Unit 1~~ (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
13. Annulus Ventilation Operation		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.	
14. Nuclear Service Water Operation		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Loss-of-Offsite Power	$\geq$ 3500 V	$\geq$ 3200 V
d. Containment Spray	See Item 2. above for all Containment Spray Setpoints and Allowable Values.	
e. Phase "B" Isolation	See Item 3.b. above for all Phase "B" Isolation Setpoints and Allowable Values.	
f. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.	
g. Suction Transfer-Low Pit Level	$\geq$El. 554.4 ft.	$\geq$El. 552.9 ft.
15. Emergency Diesel Gene[illegible]ator Operation (Diesel Bu[illegible]ding Ventilation Operation, Nuclear Service Water Operation)		
a. Manual Initiation	N.A.	N.A.

TABLE 3.3-4 For ~~Unit~~ 1 ~~2~~ (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
15. Emergency Diesel Generator Operation (Diesel Building Ventilation Operation, Nuclear Service Water Operation) (Continued)		
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Loss-of-Offsite Power	$\geq$ 3500 V	$\geq$ 3200 V
d. Safety Injection	See Item. 1 above for all Safety Injection Setpoints and Allowable Values.	
16. Auxiliary Building Filtered Exhaust Operation		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.	
17. Diesel Building Ventilation Operation		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Emergency Diesel Generator Operation	See Item 15. above for all Emergency Diesel Generator Operation Setpoints and Allowable Values.	

TABLE 3.3-4 ~~For Unit 1~~ (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
18. Engineered Safety Features Actuation System Interlocks		
a. Pressurizer Pressure, P-11	1955 psig	$\geq$1944 psig
b. Pressurizer Pressure, not P-11	1955 psig	$\leq$1966 psig
c. Low-Low T_{avg}, P-12	$\geq$553°F	$\geq$550°F
d. Reactor Trip, P-4	N.A.	N.A.
e. Steam Generator Level, P-14	See Item 5. above for all Steam Generator Water Level Trip Setpoints and Allowable Values.	

TABLE 3.3-4 ~~For Unit 1~~ (Continued)

TABLE NOTATIONS

**The time constant utilized in the rate-lag controller for Steam Line Pressure-Negative Rate-High is greater than or equal to 50 seconds. Channel calibration shall ensure that this time constant is adjusted to this value.

TABLE 3.3-4 For Unit 2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
1. Safety Injection (Reactor Trip, Phase "A" Isolation, Feedwater Isolation, Control Room Area Ventilation Operation, Auxiliary Feedwater-Motor-Driven Pump, Purge & Exhaust Isolation, Annulus Ventilation Operation, Auxiliary Building Filtered Exhaust Operation, Emergency Diesel Generator Operation, Component Cooling Water, Turbine Trip, and Nuclear Service Water Operation)		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure-High	$\leq$ 1.2 psig	$\leq$ 1.4 psig
d. Pressurizer Pressure-Low	$\geq$ 1845 psig	$\geq$ 1839 psig
e. Steam Line Pressure-Low	$\geq$ 725 psig	$\geq$ 694 psig*
2. Containment Spray		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure-High-High	$\leq$ 3 psig	$\leq$ 3.2 psig

TABLE 3.3-4 For Unit 2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
3. Containment Isolation		
a. Phase "A" Isolation		
1) Manual Initiation	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.	
b. Phase "B" Isolation (Nuclear Service Water Operation)		
1) Manual Initiation	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Containment Pressure-High-High	$\leq$ 3 psig	$\leq$ 3.2 psig
c. Purge and Exhaust Isolation		
1) Manual Initiation	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.	

TABLE 3.3-4 For Unit 2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
4. Steam Line Isolation		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure-High-High	$\leq$ 3 psig	$\leq$ 3.2 psig
d. Steam Line Pressure - Low	$\geq$ 725 psig	$\geq$ 694 psig*
e. Steam Line Pressure-Negative Rate - High	$\leq$ 100 psi	$\leq$ 122.8 psi**
5. Feedwater Isolation		
a. Automatic Actuation Logic Actuation Relays	N.A.	N.A.
b. Steam Generator Water Level-High-High (P-14)		
1. Unit 1	$\leq$ 82.4% of narrow range instrument span	$\leq$ 84.2% of narrow range instrument span
2. Unit 2	$\leq$ 77.1% of narrow range instrument span	$\leq$ 78.9% of narrow range instrument span
c. T_{avg}-Low	$\geq$ 564°F	$\geq$ 561°F
d. Doghouse Water Level-High	11 inches above 577' floor level	12 inches above 577' floor level
e. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.	

TABLE 3.3-4 For Unit 2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
6. Turbine Trip		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Steam Generator Water Level-High-High (P-14)		
1. Unit 1	$\leq$ 82.4% of narrow range instrument span	$\leq$ 84.2% of narrow range instrument span
2. Unit 2	$\leq$ 77.1% of narrow range instrument span	$\leq$ 78.9% of narrow range instrument span
d. Trip of All Main Feedwater Pumps	N.A.	N.A.
e. Reactor Trip (P-4)	N.A.	N.A.
f. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.	
7. Containment Pressure Control System		
a. Start Permissive	$\leq$ 0.4 psid	$\leq$ 0.45 psid
b. Termination	$\geq$ 0.3 psid	$\geq$ 0.25 psid
8. Auxiliary Feedwater		
a. Manual Initiation	N.A	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.

TABLE 3.3-4 For Unit 2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
8. Auxiliary Feedwater (Continued)		
c. Steam Generator Water Level - Low-Low		
1) Unit 1	≥ 17% of span from 0% to 30% RTP increasing linearly to ≥ 40.0% of span from 30% to 100% RTP	≥ 15.3% of span from 0% to 30% RTP increasing linearly to ≥ 38.3% of span from 30% to 100% RTP
2) Unit 2	≥ 36.8% of narrow range span	≥ 35.1% of narrow range instrument span
d. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.	
e. Loss-of-Offsite Power	≥ 3500 V	≥ 3200 V
f. Trip of All Main Feedwater Pumps	N.A.	N.A.
g. Auxiliary Feedwater Suction Pressure-Low		
1) CAPS 5220, 5221, 5222	≥ 10.5 psig	≥ 9.5 psig
2) CAPS 5230, 5231, 5232	≥ 6.2 psig	≥ 5.2 psig
a. Unit 1	≥ 6.2 psig	≥ 5.2 psig
b. Unit 2	≥ 6.0 psig	≥ 5.0 psig
9. Containment Sump Recirculation		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Refueling Water Storage Tank Level-Low	≥ 177.15 inches	≥ 162.4 inches
Coincident With Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.	

TABLE 3.3-4 For Unit 2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
10. Loss of Power		
a. 4 kV Bus Undervoltage-Loss of Voltage	$\geq$ 3500 V	$\geq$ 3200 V
b. 4 kV Bus Undervoltage-Grid Degraded Voltage	$\geq$ 3685 V	$\geq$ 3611 V
11. Control Room Area Ventilation Operation		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Loss-of-Offsite Power	$\geq$ 3500 V	$\geq$ 3200 V
c. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.	
12. Containment Air Return and Hydrogen Skimmer Operation		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure-High-High	$\leq$ 3 psig	$\leq$ 3.2 psig

TABLE 3.3-4 For Unit 2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT		TRIP SETPOINT	ALLOWABLE VALUE
13.	Annulus Ventilation Operation		
	a. Manual Initiation	N.A.	N.A.
	b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
	c. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.	
14.	Nuclear Service Water Operation		
	a. Manual Initiation	N.A.	N.A.
	b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
	c. Loss-of-Offsite Power	$\geq$ 3500 V	$\geq$ 3200 V
	d. Containment Spray	See Item 2. above for all Containment Spray Setpoints and Allowable Values.	
	e. Phase "B" Isolation	See Item 3.b. above for all Phase "B" Isolation Setpoints and Allowable Values.	
	f. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.	
	g. Suction Transfer-Low Pit Level	$\geq$El. 554.4 ft.	$\geq$El. 552.9 ft.
15.	Emergency Diesel Generator Operation (Diesel Building Ventilation Operation, Nuclear Service Water Operation)		
	a. Manual Initiation	N.A.	N.A.

TABLE 3.3-4 For Unit 2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
15. Emergency Diesel Generator Operation (Diesel Building Ventilation Operation, Nuclear Service Water Operation) (Continued)		
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Loss-of-Offsite Power	$\geq$ 3500 V	$\geq$ 3200 V
d. Safety Injection	See Item. 1 above for all Safety Injection Setpoints and Allowable Values.	
16. Auxiliary Building Filtered Exhaust Operation		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.	
17. Diesel Building Ventilation Operation		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Emergency Diesel Generator Operation	See Item 15. above for all Emergency Diesel Generator Operation Setpoints and Allowable Values.	

TABLE 3.3-4 For Unit 2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
18. Engineered Safety Features Actuation System Interlocks		
a. Pressurizer Pressure, P-11	1955 psig	$\geq$1944 psig
b. Pressurizer Pressure, not P-11	1955 psig	$\leq$1966 psig
c. Low-Low T_{avg}, P-12	$\geq$553°F	$\geq$550°F
d Reactor Trip, P-4	N.A.	N.A.
e. Steam Generator Level, P-14	See Item 5. above for all Steam Generator Water Level Trip Setpoints and Allowable Values.	

TABLE 3.3-4 For Unit 2 (C. tinued)

TABLE NOTATIONS

*Time constants utilized in the lead-lag controller for Steam Line Pressure-Low are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds. Channel calibration shall ensure that these time constants are adjusted to these values.

**The time constant utilized in the rate-lag controller for Steam Line Pressure-Negative Rate-High is greater than or equal to 50 seconds. Channel calibration shall ensure that this time constant is adjusted to this value.

TABLE 3.3-5 ~~FOR UNIT 1~~

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATION SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual Initiation	
a. Safety Injection (ECCS)	N.A.
b. Containment Spray	N.A.
c. Phase "A" Isolation	N.A.
d. Phase "B" Isolation	N.A.
e. Purge and Exhaust Isolation	N.A.
f. Steam Line Isolation	N.A.
g. Diesel Building Ventilation Operation	N.A.
h. Nuclear Service Water Operation	N.A.
i. Turbine Trip	N.A.
j. Component Cooling Water	N.A.
k. Annulus Ventilation Operation	N.A.
l. Auxiliary Building Filtered Exhaust Operation	N.A.
m. Reactor Trip	N.A.
n. Emergency Diesel Generator Operation	N.A.
o. Containment Air Return and Hydrogen Skimmer Operation	N.A.
p. Auxiliary Feedwater	N.A.
2. Containment Pressure-High	
a. Safety Injection (ECCS)	$\leq 27^{(1)}/12^{(3)}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	≤ 12
3) Phase "A" Isolation(2)	$\leq 18^{(3)}/28^{(4)}$
4) Purge and Exhaust Isolation	≤ 6
5) Auxiliary Feedwater(5)	N.A.
6) Nuclear Service Water Operation	$\leq 65^{(3)}/76^{(4)}$
7) Turbine Trip	N.A.
8) Component Cooling Water	$\leq 65^{(3)}/76^{(4)}$
9) Emergency Diesel Generator Operation	≤ 11
10) Control Room Area Ventilation Operation	N.A.

TABLE 3.3-5 ~~FOR UNIT 1~~ (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
2. Containment Pressure-High (Continued)	
11) Annulus Ventilation Operation	≤ 23
12) Auxiliary Building Filtered Exhaust Operation	N.A.
13) Containment Sump Recirculation	N.A.
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	$\le 27^{(1)}/12^{(3)}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	≤ 12
3) Phase "A" Isolation(2)	$\le 18^{(3)}/28^{(4)}$
4) Purge and Exhaust Isolation	≤ 6
5) Auxiliary Feedwater(5)	N.A.
6) Nuclear Service Water Operation	$\le 65^{(3)}/76^{(4)}$
7) Turbine Trip	N.A.
8) Component Cooling Water	$\le 65^{(3)}/76^{(4)}$
9) Emergency Diesel Generator Operation	≤ 11
10) Control Room Area Ventilation Operation	N.A.
11) Annulus Ventilation Operation	≤ 23
12) Auxiliary Building Filtered Exhaust Operation	N.A.
13) Containment Sump Recirculation	N.A.
4. Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\le 12^{(3)}/22^{(4)}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	≤ 12
3) Phase "A" Isolation(2)	$\le 18^{(3)}/28^{(4)}$
4) Purge and Exhaust Isolation	≤ 6
5) Auxiliary Feedwater(5)	≤ 60
6) Nuclear Service Water Operation	$\le 65^{(3)}/76^{(4)}$
7) Turbine Trip	N.A.
8) Component Cooling Water	$\le 65^{(3)}/76^{(4)}$
9) Emergency Diesel Generator Operation	≤ 11

TABLE 3.3-5 ~~FOR UNIT 1~~ (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
4. Steam Line Pressure-Low (Continued)	
10) Control Room Area Ventilation Operation	N.A.
11) Annulus Ventilation Operation	$\leq$ 23
12) Auxiliary Building Filtered Exhaust Isolation	N.A.
13) Containment Sump Recirculation	N.A.
b. Steam Line Isolation	$\leq$ 10
5. Containment Pressure-High-High	
a. Containment Spray	$\leq$ 45
b. Phase "B" Isolation	$\leq 65^{(3)}/76^{(4)}$
Nuclear Service Water Operation	N.A.
c. Steam Line Isolation	$\leq$ 10
d. Containment Air Return and Hydrogen Skimmer Operation	$\leq$ 600
6. Steam Line Pressure - Negative Rate-High	
Steam Line Isolation	$\leq$ 10
7. Steam Generator Water Level-High-High	
a. Turbine Trip	$\leq$ 3
b. Feedwater Isolation	$\leq$ 12
8. T_{avg}-Low	
Feedwater Isolation	N.A.
9. Doghouse Water Level-High	
Feedwater Isolation	N.A.
10. Start Permissive	
Containment Pressure Control System	N.A.
11. Termination	
Containment Pressure Control System	N.A.

TABLE 3.3-5 ~~FOR UNIT 1~~ (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
12. Steam Generator Water Level-Low-Low	
a. Motor-Driven Auxiliary Feedwater Pumps	$\leq$ 60
b. Turbine-Driven Auxiliary Feedwater Pump	$\leq$ 60
13. Loss-of-Offsite Power	
a. Motor-Driven Auxiliary Feedwater Pumps	$\leq$ 60
b. Turbine-Driven Auxiliary Feedwater Pumps	$\leq$ 60
c. Control Room Area Ventilation Operation	N.A.
d. Emergency Diesel Generator Operation	$\leq$ 11
1) Diesel Building Ventilation Operation	N.A.
2) Nuclear Service Water Operation	$\leq 65^{(3)}/76^{(4)}$
14. Trip of All Main Feedwater Pumps	
a. Motor-Driven Auxiliary Feedwater Pumps	$\leq$ 60
b. Turbine Trip	N.A.
15. Auxiliary Feedwater Suction Pressure-Low	
Auxiliary Feedwater (Suction Supply Automatic Realignment)	$\leq 16^{(6)}$
16. Refueling Water Storage Tank Level-Low	
Coincident with Safety Injection Signal (Automatic Switchover to Containment Sump)	$\leq$ 60
17. Loss of Power	
a. 4 kV Bus Undervoltage - Loss of Voltage	$\leq$ 8.5
b. 4 kV Bus Undervoltage-Grid Degraded Voltage	$\leq$ 600
18. Suction Transfer-Low Pit Level	
Nuclear Service Water Operation	N.A.

TABLE 3.3-5 FOR UNIT 2

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATION SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual Initiation	
a. Safety Injection (ECCS)	N.A.
b. Containment Spray	N.A.
c. Phase "A" Isolation	N.A.
d. Phase "B" Isolation	N.A.
e. Purge and Exhaust Isolation	N.A.
f. Steam Line Isolation	N.A.
g. Diesel Building Ventilation Operation	N.A.
h. Nuclear Service Water Operation	N.A.
i. Turbine Trip	N.A.
j. Component Cooling Water	N.A.
k. Annulus Ventilation Operation	N.A.
l. Auxiliary Building Filtered Exhaust Operation	N.A.
m. Reactor Trip	N.A.
n. Emergency Diesel Generator Operation	N.A.
o. Containment Air Return and Hydrogen Skimmer Operation	N.A.
p. Auxiliary Feedwater	N.A.
2. Containment Pressure-High	
a. Safety Injection (ECCS)	$\leq 27^{(1)}/12^{(3)}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	≤ 7
3) Phase "A" Isolation(2)	$\leq 18^{(3)}/28^{(4)}$
4) Purge and Exhaust Isolation	≤ 6
5) Auxiliary Feedwater(5)	N.A.
6) Nuclear Service Water Operation	$\leq 65^{(3)}/76^{(4)}$
7) Turbine Trip	N.A.
8) Component Cooling Water	$\leq 65^{(3)}/76^{(4)}$
9) Emergency Diesel Generator Operation	≤ 11
10) Control Room Area Ventilation Operation	N.A.

TABLE 3.3-5 FOR UNIT 2 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
2. Containment Pressure-High (Continued)	
11) Annulus Ventilation Operation	$\leq$ 23
12) Auxiliary Building Filtered Exhaust Operation	N.A.
13) Containment Sump Recirculation	N.A.
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	$\leq 27^{(1)}/12^{(3)}$
1) Reactor Trip	$\leq$ 2
2) Feedwater Isolation	$\leq$ 7
3) Phase "A" Isolation(2)	$\leq 18^{(3)}/28^{(4)}$
4) Purge and Exhaust Isolation	$\leq$ 6
5) Auxiliary Feedwater(5)	N.A.
6) Nuclear Service Water Operation	$\leq 65^{(3)}/76^{(4)}$
7) Turbine Trip	N.A.
8) Component Cooling Water	$\leq 65^{(3)}/76^{(4)}$
9) Emergency Diesel Generator Operation	$\leq$ 11
10) Control Room Area Ventilation Operation	N.A.
11) Annulus Ventilation Operation	$\leq$ 23
12) Auxiliary Building Filtered Exhaust Operation	N.A.
13) Containment Sump Recirculation	N.A.
4. Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq 12^{(3)}/22^{(4)}$
1) Reactor Trip	$\leq$ 2
2) Feedwater Isolation	$\leq$ 7
3) Phase "A" Isolation(2)	$\leq 18^{(3)}/28^{(4)}$
4) Purge and Exhaust Isolation	$\leq$ 6
5) Auxiliary Feedwater(5)	$\leq$ 60
6) Nuclear Service Water Operation	$\leq 65^{(3)}/76^{(4)}$
7) Turbine Trip	N.A.
8) Component Cooling Water	$\leq 65^{(3)}/76^{(4)}$
9) Emergency Diesel Generator Operation	$\leq$ 11

TABLE 3.3-5 FOR UNIT 2 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
4. Steam Line Pressure-Low (Continued)	
10) Control Room Area Ventilation Operation	N.A.
11) Annulus Ventilation Operation	$\leq$ 23
12) Auxiliary Building Filtered Exhaust Isolation	N.A.
13) Containment Sump Recirculation	N.A.
b. Steam Line Isolation	$\leq$ 7
5. Containment Pressure-High-High	
a. Containment Spray	$\leq$ 45
b. Phase "B" Isolation	$\leq 65^{(3)}/76^{(4)}$
Nuclear Service Water Operation	N.A.
c. Steam Line Isolation	$\leq$ 7
d. Containment Air Return and Hydrogen Skimmer Operation	$\leq$ 600
6. Steam Line Pressure - Negative Rate-High	
Steam Line Isolation	$\leq$ 7
7. Steam Generator Water Level-High-High	
a. Turbine Trip	$\leq$ 3
b. Feedwater Isolation	$\leq$ 7
8. T_{avg}-Low	
Feedwater Isolation	N.A.
9. Doghouse Water Level-High	
Feedwater Isolation	N.A.
10. Start Permissive	
Containment Pressure Control System	N.A.
11. Termination	
Containment Pressure Control System	N.A.

TABLE 3.3-5 FOR UNIT 2 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
12. Steam Generator Water Level-Low-Low	
a. Motor-Driven Auxiliary Feedwater Pumps	$\leq$ 60
b. Turbine-Driven Auxiliary Feedwater Pump	$\leq$ 60
13. Loss-of-Offsite Power	
a. Motor-Driven Auxiliary Feedwater Pumps	$\leq$ 60
b. Turbine-Driven Auxiliary Feedwater Pumps	$\leq$ 60
c. Control Room Area Ventilation Operation	N.A.
d. Emergency Diesel Generator Operation	$\leq$ 11
1) Diesel Building Ventilation Operation	N.A.
2) Nuclear Service Water Operation	$\leq 65^{(3)}/76^{(4)}$
14. Trip of All Main Feedwater Pumps	
a. Motor-Driven Auxiliary Feedwater Pumps	$\leq$ 60
b. Turbine Trip	N.A.
15. Auxiliary Feedwater Suction Pressure-Low	
Auxiliary Feedwater (Suction Supply Automatic Realignment)	$\leq 16^{(6)}$
16. Refueling Water Storage Tank Level-Low	
Coincident with Safety Injection Signal (Automatic Switchover to Containment Sump)	$\leq$ 60
17. Loss of Power	
a. 4 kV Bus Undervoltage - Loss of Voltage	$\leq$ 8.5
b. 4 kV Bus Undervoltage-Grid Degraded Voltage	$\leq$ 600
18. Suction Transfer-Low Pit Level	
Nuclear Service Water Operation	N.A.

BORON DILUTION MITIGATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.11 As a minimum, two trains of the Boron Dilution Mitigation System shall be OPERABLE and operating with Shutdown Margin Alarm ratios set at less than or equal to 4 times the steady-state count rate.

APPLICABILITY: MODES 3, 4, AND 5

ACTION:

(a) With one train of the Boron Dilution Mitigation System inoperable or not operating, restore the inoperable train to OPERABLE status within 48 hours, or

(1) suspend all operations involving positive reactivity changes and verify that valve NV-230 is closed and secured within the next hour, or

(2) verify two Source Range Neutron Flux Monitors are OPERABLE with Alarm Setpoints less than or equal to one-half decade (square root of 10) above the steady-state count rate and verify that the combined flowrate from both Reactor Makeup Water Pumps is less than or equal to 150 gpm (Mode 3 or 4) or ~~75~~ 70 gpm (Mode 5) within the next hour.

(b) With both trains of the Boron Dilution Mitigation System inoperable or not operating, restore the inoperable trains to OPERABLE status within 12 hours, or

(1) suspend all operations involving positive reactivity changes and verify that valve NV-230 is closed and secured within the next hour, or

(2) verify two Source Range Neutron Flux Monitors are OPERABLE with Alarm Setpoints less than or equal to one-half decade (square root of 10) above the steady-state count rate and verify that the combined flow rate from both Reactor Makeup Water Pumps is less than or equal to 150 gpm (Mode 3 or 4) or ~~75~~ 70 gpm (Mode 5) within the next hour.

SURVEILLANCE REQUIREMENTS

4.3.3.11.1 Each train of the Boron Dilution Mitigation System shall be demonstrated OPERABLE by performance of:

(a) A CHANNEL CHECK at least once per 12 hours,

SURVEILLANCE REQUIREMENTS (Continued)

(b) An ANALOG CHANNEL OPERATIONAL TEST at least once per 31 days, and

(c) At least once per 18 months the BDMS shall be demonstrated OPERABLE by:

(1) Verifying that each automatic valve actuated by the BDMS moves to its correct position upon receipt of a trip signal, and

(2) Verifying each reactor makeup water pump stops, as designed, upon receipt of a trip signal.

4.3.3.11.2 If using the Source Range Neutron Flux Monitors to meet the requirements of Technical Specification 3.3.3.11,

(a) The monthly surveillance requirements of Table 4.3-1 for the Source Range Neutron Flux Monitors shall include verification that the Alarm Setpoint is less than or equal to one-half decade (square root of 10) above the steady-state count rate.

(b) The combined flow rate from both Reactor Makeup Water Pumps shall be verified as less than or equal to 150 gpm (Mode 3 or 4) or ~~75~~ 70 gpm (Mode 5) at least once per 31 days.

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer Code safety valve shall be OPERABLE with a lift setting of 2485 psig + 3%, -2% ~~for Unit 1 and ± 1% for Unit 2~~.*

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer Code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE residual heat removal loop into operation in the shutdown cooling mode.

SURVEILLANCE REQUIREMENTS

4.4.2.1 No additional requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting of 2485 psig + 3%, -2% ~~for Unit 1 and ± 1% for Unit 2~~.*

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With one pressurizer Code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

TABLE 3.6-2a

UNIT 1 CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (s)
1. Phase "A" Isolation		
BB-57B#	Steam Generator 1A Blowdown Containment Outside Isolation	≤10
BB-21B#	Steam Generator 1B Blowdown Containment Outside Isolation	≤10
BB-61B#	Steam Generator 1C Blowdown Containment Outside Isolation	≤10
BB-10B#	Steam Generator 1D Blowdown Containment Outside Isolation	≤10
BB-56A#	Steam Generator 1A Blowdown Containment Inside Isolation	≤10
BB-19A#	Steam Generator 1B Blowdown Containment Inside Isolation	≤10
BB-60A#	Steam Generator 1C Blowdown Containment Inside Isolation	≤10
BB-8A#	Steam Generator 1D Blowdown Containment Inside Isolation	≤10
BB-148B#	Steam Generator 1A Blowdown Containment Isolation Bypass	≤10
BB-150B#	Steam Generator 1B Blowdown Containment Isolation Bypass	≤10
BB-149B#	Steam Generator 1C Blowdown Containment Isolation Bypass	≤10
BB-147B#	Steam Generator 1D Blowdown Containment Isolation Bypass	≤10
CA-149#	Steam Generator 1A Main Feedwater to Auxiliary Feedwater Nozzle Isolation	~~≤5~~ NA
CA-150#	Steam Generator 1B Main Feedwater to Auxiliary Feedwater Nozzle Isolation	~~≤5~~ NA
CA-151#	Steam Generator 1C Main Feedwater to Auxiliary Feedwater Nozzle Isolation	~~≤5~~ NA
CA-152#	Steam Generator 1D Main Feedwater to Auxiliary Feedwater Nozzle Isolation	~~≤5~~ NA
CA-185#	Auxiliary Nozzle Temper SG1A	~~≤5~~ NA
CA-186#	Auxiliary Nozzle Temper SG1B	~~≤5~~ NA
CA-187#	Auxiliary Nozzle Temper SG1C	~~≤5~~ NA
CA-188#	Auxiliary Nozzle Temper SG1D	~~≤5~~ NA
CF-60#	Steam Generator 1D Feedwater Containment Isolation	~~≤5~~ NA
CF-51#	Steam Generator 1C Feedwater Containment Isolation	~~≤5~~ NA
CF-42#	Steam Generator 1B Feedwater Containment Isolation	~~≤5~~ NA
CF-33#	Steam Generator 1A Feedwater Containment Isolation	~~≤5~~ NA
CF-90#	Steam Generator 1A Feedwater Purge Valve	~~≤5~~ NA
CF-89#	Steam Generator 1B Feedwater Purge Valve	~~≤5~~ NA
CF-88#	Steam Generator 1C Feedwater Purge Valve	~~≤5~~ NA
CF-87#	Steam Generator 1D Feedwater Purge Valve	~~≤5~~ NA

TABLE 3.6-2a (Continued)

UNTI 1 CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (s)
2. Phase "B" Isolation (Continued)		
RN-437B	Supply to NC Pumps and LCVU Supply Outside Containment Isolation	≤60
RN-484A	Return from NC Pumps and LCVU Return Inside Containment Isolation	≤60
RN-487B	Return from NC Pumps and LCVU Return Outside Containment Isolation	≤60
RN-404B	Supply to Upper Containment Supply Ventilation Units Containment Isolation (Outside)	≤10
RN-429A	Return from Upper Containment Ventilation Units Containment Isolation (Inside)	≤10
RN-432B	Return from Upper Containment Ventilation Units Containment Isolation (Outside)	≤10
VI-77B	Instrument Air Containment Outside Isolation	≤10
SM-1 #	Main Steam 1D Isolation	~~≤5~~ NA
SM-3 #	Main Steam 1C Isolation	~~≤5~~ NA
SM-5 #	Main Steam 1B Isolation	~~≤5~~ NA
SM-7 #	Main Steam 1A Isolation	~~≤5~~ NA
SM-9 #	Main Steam 1D Isolation Bypass Ctrl.	~~≤5~~ NA
SM-10 #	Main Steam 1C Isolation Bypass Ctrl.	~~≤5~~ NA
SM-11 #	Main Steam 1B Isolation Bypass Ctrl.	~~≤5~~ NA
SM-12 #	Main Steam 1A Isolation Bypass Ctrl.	~~≤5~~ NA
SV-19 #	Main Steam 1A PORV	≤5
SV-13 #	Main Steam 1B PORV	≤5
SV-7 #	Main Steam 1C PORV	≤5
SV-1 #	Main Steam 1D PORV	≤5
WL-867A**	Containment Vent Unit Drains Inside Containment Isolation	≤10
WL-869B**	Containment Vent Unit Drains Outside Containment Isolation	≤10

TABLE 3.6-2b

UNIT 2 CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (s)
1. Phase "A" Isolation		
BB-57B#	Steam Generator 2A Blowdown Containment Outside Isolation	≤10
BB-21B#	Steam Generator 2B Blowdown Containment Outside Isolation	≤10
BB-61B#	Steam Generator 2C Blowdown Containment Outside Isolation	≤10
BB-10B#	Steam Generator 2D Blowdown Containment Outside Isolation	≤10
BB-56A#	Steam Generator 2A Blowdown Containment Inside Isolation	≤10
BB-19A#	Steam Generator 2B Blowdown Containment Inside Isolation	≤10
BB-60A#	Steam Generator 2C Blowdown Containment Inside Isolation	≤10
BB-8A#	Steam Generator 2D Blowdown Containment Inside Isolation	≤10
BB-148B#	Steam Generator 2A Blowdown Containment Isolation Bypass	≤10
BB-150B#	Steam Generator 2B Blowdown Containment Isolation Bypass	≤10
BB-149B#	Steam Generator 2C Blowdown Containment Isolation Bypass	≤10
BB-147B#	Steam Generator 2D Blowdown Containment Isolation Bypass	≤10
CA-149#	Steam Generator 2A Main Feedwater to Auxiliary Feedwater Nozzle Isolation	~~≤5~~ NA
CA-150#	Steam Generator 2B Main Feedwater to Auxiliary Feedwater Nozzle Isolation	~~≤5~~ NA
CA-151#	Steam Generator 2C Main Feedwater to Auxiliary Feedwater Nozzle Isolation	~~≤5~~ NA
CA-152#	Steam Generator 2D Main Feedwater to Auxiliary Feedwater Nozzle Isolation	~~≤5~~ NA
CA-185#	Auxiliary Nozzle Temper SG2A	~~≤5~~ NA
CA-186#	Auxiliary Nozzle Temper SG2B	~~≤5~~ NA
CA-187#	Auxiliary Nozzle Temper SG2C	~~≤5~~ NA
CA-188#	Auxiliary Nozzle Temper SG2D	~~≤5~~ NA
CF-60#	Steam Generator 2D Feedwater Containment Isolation	~~≤5~~ NA
CF-51#	Steam Generator 2C Feedwater Containment Isolation	~~≤5~~ NA
CF-42#	Steam Generator 2B Feedwater Containment Isolation	~~≤5~~ NA
CF-33#	Steam Generator 2A Feedwater Containment Isolation	~~≤5~~ NA
CF-90#	Steam Generator 2A Feedwater Purge Valve	~~≤5~~ NA
CF-89#	Steam Generator 2B Feedwater Purge Valve	~~≤5~~ NA
CF-88#	Steam Generator 2C Feedwater Purge Valve	~~≤5~~ NA
CF-87#	Steam Generator 2D Feedwater Purge Valve	~~≤5~~ NA

TABLE 3.6-2b (Continued)

UNTI 2 CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (s)
2. Phase "B" Isolation (Continued)		
RN-437B	Supply to NC Pumps and LCVU Supply Outside Containment Isolation	≤60
RN-484A	Return from NC Pumps and LCVU Return Inside Containment Isolation	≤60
RN-487B	Return from NC Pumps and LCVU Return Outside Containment Isolation	≤60
RN-404B	Supply to Upper Containment Supply Ventilation Units Containment Isolation (Outside)	≤10
RN-429A	Return from Upper Containment Ventilation Units Containment Isolation (Inside)	≤10
RN-432B	Return from Upper Containment Ventilation Units Containment Isolation (Outside)	≤10
VI-77B	Instrument Air Containment Outside Isolation	≤10
SM-1 #	Main Steam 2D Isolation	~~≤5~~ NA
SM-3 #	Main Steam 2C Isolation	~~≤5~~ NA
SM-5 #	Main Steam 2B Isolation	~~≤5~~ NA
SM-7 #	Main Steam 2A Isolation	~~≤5~~ NA
SM-9 #	Main Steam 2D Isolation Bypass Ctrl.	~~≤5~~ NA
SM-10 #	Main Steam 2C Isolation Bypass Ctrl.	~~≤5~~ NA
SM-11 #	Main Steam 2B Isolation Bypass Ctrl.	~~≤5~~ NA
SM-12 #	Main Steam 2A Isolation Bypass Ctrl.	~~≤5~~ NA
SV-19 #	Main Steam 2A PORV	≤5
SV-13 #	Main Steam 2B PORV	≤5
SV-7 #	Main Steam 2C PORV	≤5
SV-1 #	Main Steam 2D PORV	≤5
WL-867A**	Containment Vent Unit Drains Inside Containment Isolation	≤10
WL-869B**	Containment Vent Unit Drains Outside Containment Isolation	≤10

PLANT SYSTEMS

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.4 Each main steam line isolation valve (MSLIV) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

MODE 1:

With one MSLIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

MODES 2 and 3:

With one MSLIV inoperable, subsequent operation in MODE 2 or 3 may proceed provided the isolation valve is maintained closed. The provisions of Specification 3.0.4 are not applicable. Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 Each MSLIV shall be demonstrated OPERABLE by verifying full closure within 8 seconds ~~for Unit 1 and 5 seconds for Unit~~ 2 when tested pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) maintaining the calculated DNBR in the core greater than or equal to design limit DNBR during normal operation and in short-term transients, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria are not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(X,Y,Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods;

$F\Delta H(X,Y)$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

K(z) is defined as the normalized $F_Q(X,Y,Z)$ limit for a given core height.

3/4.2.1 AXIAL FLUX DIFFERENCE ~~Unit 1~~

The limits on AXIAL FLUX DIFFERENCE (AFD) specified in the CORE OPERATING LIMITS REPORT (COLR) ensure that the $F_Q(X,Y,Z)$ and the $F\Delta H(X,Y)$ limits are not exceeded during either normal operation or in the event of xenon redistribution following power changes. The AFD envelope specified in the COLR has been adjusted for measurement uncertainty.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ~~(Unit 1)~~

The limits on heat flux hot channel factor, and nuclear enthalpy rise hot channel factor ensure that: (1) the design limits on peak local power density and minimum DNBR are not exceeded and (2) in the event of a LOCA the ECCS acceptance criteria are not exceeded. The peaking limits are specified in the CORE OPERATING LIMITS REPORT (COLR) per Specification 6.9.1.9.

The heat flux hot channel factor and nuclear enthalpy rise hot channel factor are each measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

HEAT FLUX HOT CHANNEL FACTOR, AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ~~(Unit 1)~~ (Continued)

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position;

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6;

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained; and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

FΔH(X,Y) will be maintained within its limits provided Conditions a. through d. above are maintained.

The limits on the nuclear enthalpy rise hot channel factor, FΔH(X,Y), are specified in the COLR as Maximum Allowable Radial Peaking limits, obtained by dividing the Maximum Allowable Total Peaking (MAP) limit by the axial peak [AXIAL(X,Y)] for location (X,Y). By definition, the Maximum Allowable Radial Peaking limits will, for Mark-BW fuel, result in a DNBR for the limiting transient that is equivalent to the DNBR calculated with a design FΔH(X,Y) value of 1.50 and a limited reference axial power shape. For transition cores, MAP limits may be applied to both Mark-BW and optimized fuel types provided allowances for differences in DNBR are accounted for in the generation of MAP limits. The MAP limits specified in the COLR include allowances for mixed core DNBR effects. The relaxation of FΔH(X,Y), as a function of THERMAL POWER allows changes in the radial power for all permissible control bank insertion limits. This relaxation is implemented by the application of the following factors:

$k = [1 + (1/RRH)(1 - P)]$

where k = power factor multiplier applied to the MAP limits
P = THERMAL POWER / RATED THERMAL POWER
RRH is given in the COLR

$FQ^M(X,Y,Z)$ and $F\Delta H^M(X,Y)$ are measured periodically, and comparisons to the allowable limit are made to provide reasonable assurance that the core is operating as designed and that the limiting criteria will not be exceeded for operation within the Technical Specification limits of Sections 2.2 (Limiting Safety Systems Settings), 3.1.3 (Movable Control Assemblies), 3.2.1 (Axial Flux Difference), and 3.2.4 (Quadrant Power Tilt Ratio). A peaking margin calculation is performed to provide a basis for decreasing the width of the AFD and f(ΔI) limits and for reducing THERMAL POWER.

HEAT FLUX HOT CHANNEL FACTOR, AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ~~(Unit 1)~~ (Continued)

When an $FQ^M(X,Y,Z)$ measurement is obtained from a full core map in accordance with the surveillance requirements of Specification 4.2.2, no uncertainties are applied to the measured peak since the required uncertainties are included in the peaking limit. When $FQ^M(X,Y,Z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2, the measured peak is increased by the radial-local peaking factor and allowances of 5% for measurement uncertainty and 3% for manufacturing tolerances.

When an $F\Delta H^M(X,Y)$ measurement is obtained from a full core map regardless of the reason, no uncertainties are applied to the measured peak since the required uncertainties are included in the peaking limit.

3/4.2.4 QUADRANT POWER TILT RATIO ~~(Unit 1)~~

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A peaking increase that reflects a QUADRANT POWER TILT RATIO of 1.02 is included in the generation of the AFD limits.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on $F_Q(X,Y,Z)$ is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 2%.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the movable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles.

3/4.2.5 DNB PARAMETERS ~~(UNIT 1)~~

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a

3/4.2.5 DNB PARAMETERS ~~(UNIT 1)~~ (Continued)

design limit DNBR throughout each analyzed transient. As noted on Figure 3.2-1, Reactor Coolant System flow rate and THERMAL POWER may be "traded off" against one another (i.e., a low measured Reactor Coolant System flow rate is acceptable if the THERMAL POWER is also low) to ensure that the calculated DNBR will not be below the design DNBR value. The relationship defined on Figure 3.2-1 remains valid as long as the limits placed on the nuclear enthalpy rise hot channel factor, $F\Delta H(X,Y)$ in Specification 3.2.3 are maintained. The indicated T_{avg} value and the indicated pressurizer pressure value correspond to analytical limits of 594.8°F and 2205.3 psig respectively, with allowance for measurement uncertainty. When Reactor Coolant System flow rate is measured, no additional allowances are necessary prior to comparison with the limits of Figure 3.2-1 since a measurement error of 2.1% for Reactor Coolant System total flow rate has been allowed for in determination of the design DNBR value.

The measurement error for Reactor Coolant System total flow rate is based upon performing a precision heat balance and using the result to calibrate the Reactor Coolant System flow rate indicators. Potential fouling of the feedwater venturi which might not be detected could bias the result from the precision heat balance in a nonconservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi is included in Figure 3.2-1. Any fouling which might bias the Reactor Coolant System flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant performance parameters. If detected, action shall be taken before performing subsequent precision heat balance measurements, i.e., either the effect of the fouling shall be quantified and compensated for in the Reactor Coolant System flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. Indication instrumentation measurement uncertainties are accounted for in the limits provided in Table 3.2-1.

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) maintaining the calculated DNBR in the core greater than or equal to design limit DNBR during normal operation and in short-term transients, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods;

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

3/4.2.1 AXIAL FLUX DIFFERENCE (Unit 2)

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of the F_Q^{RTP} limit specified in the CORE OPERATING LIMITS REPORT (COLR) times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full-length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady-state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

At power levels below APL^{ND}, the limits on AFD are defined in the COLR, i.e., that defined by the RAOC operating procedure and limits. These limits were calculated in a manner such that expected operational transients, e.g., load follow operations, would not result in the AFD deviating outside of those limits. However, in the event such a deviation occurs, the short period of time allowed outside of the limits at reduced power levels will not result in significant xenon redistribution such that the envelope of peaking factors would change sufficiently to prevent operation in the vicinity of the APL^{ND} power level.

At power levels greater than APL^{ND}, two modes of operation are permissible; 1) RAOC, the AFD limits of which are defined in the COLR, and 2) Base Load operation, which is defined as the maintenance of the AFD within a COLR specified band about a target value. The RAOC operating procedure above APL^{ND} is the same as that defined for operation below APL^{ND}. However, it is possible when following extended load following maneuvers that the AFD limits may result in restrictions in the maximum allowed power or AFD in order to guarantee operation with $F_Q(Z)$ less than its limiting value. To allow operation at the maximum permissible value, the Base Load operating procedure restricts

the indicated AFD to relatively small target band and power swings (AFD target band as specified in the COLR, $APL^{ND} \leq$ power $\leq APL^{BL}$ or 100% Rated Thermal Power, whichever is lower). For Base Load operation, it is expected that the Units will operate within the target band. Operation outside of the target band for the short time period allowed will not result in significant xenon redistribution such that the envelope of peaking factors would change sufficiently to prohibit continued operation in the power region defined above. To assure there is no residual xenon redistribution impact from past operation on the Base Load operation, a 24 hour waiting period at a power level above APL^{ND} and allowed by RAOC is necessary. During this time period load changes and rod motion are restricted to that allowed by the Base Load procedure. After the waiting period extended Base Load operation is permissible.

The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are: 1) outside the allowed ΔI power operating space (for RAOC operation), or 2) outside the allowed ΔI target band (for Base Load operation). These alarms are active when power is greater than: 1) 50% of RATED THERMAL POWER (for RAOC operation), or 2) APL^{ND} (for Base Load operation). Penalty deviation minutes for Base Load operation are not accumulated based on the short period of time during which operation outside of the target band is allowed.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, and REACTOR COOLANT SYSTEM FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Unit 2)

The limits on heat flux hot channel factor, coolant flow rate, and nuclear enthalpy rise hot channel factor ensure that: (1) the design limits on peak local power density and minimum DNBR are not exceeded and (2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit. These limits are specified in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.9.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position;

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6;

HEAT FLUX HOT CHANNEL FACTOR, and REACTOR COOLANT SYSTEM FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Unit 2) (Continued)

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained; and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. As noted on the figure specified in the CORE OPERATING LIMITS REPORT (COLR), Reactor Coolant System flow rate and $F^N_{\Delta H}$ may be "traded off" against one another (i.e., a low measured Reactor Coolant System flow rate is acceptable if the measured $F^N_{\Delta H}$ is also low) to ensure that the calculated DNBR will not be below the design DNBR value. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

R as calculated in Specification 3.2.3 and used in the figure specified in the COLR, accounts for $F^N_{\Delta H}$ less than or equal to the $F^{RTP}_{\Delta H}$ limit specified in the COLR. This value is used in the various accident analyses where $F^N_{\Delta H}$ influences parameters other than DNBR, e.g., peak clad temperature, and thus is the maximum "as measured" value allowed. The rod bow penalty as a function of burnup applied for $F^N_{\Delta H}$ is calculated with the methods described in WCAP-8691, Revision 1, "Fuel Rod Bow Evaluation," July 1979, and the maximum rod bow penalty is 2.7% DNBR. Since the safety analysis is performed with plant-specific safety DNBR limits compared to the design DNBR limits, there is sufficient thermal margin available to offset the rod bow penalty of 2.7% DNBR.

The hot channel factor $F^M_Q(z)$ is measured periodically and increased by a cycle and height dependent power factor appropriate to either RAOC or Base Load operation, $W(z)$ or $W(z)_{BL}$, to provide assurance that the limit on the hot channel factor, $F_Q(z)$, is met. W(z) accounts for the effects of normal operation transients and was determined from expected power control maneuvers over the full range of burnup conditions in the core. $W(z)_{BL}$ accounts for the more restrictive operating limits allowed by Base Load operation which result in less severe transient values. The W(z) function for normal operation and the $W(Z)_{BL}$ function for Base Load Operation are specified in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.9.

HEAT FLUX HOT CHANNEL FACTOR, and REACTOR COOLANT SYSTEM FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Unit 2) (Continued)

When Reactor Coolant System flow rate and $F^N_{\Delta H}$ are measured, no additional allowances are necessary prior to comparison with the limits of the figure specified in the COLR. Measurement errors of 2.1% for Reactor Coolant System total flow rate and 4% for $F^N_{\Delta H}$ have been allowed for in determination of the design DNBR value.

The measurement error for Reactor Coolant System total flow rate is based upon performing a precision heat balance and using the result to calibrate the Reactor Coolant System flow rate indicators. Potential fouling of the feedwater venturi which might not be detected could bias the result from the precision heat balance in a nonconservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi is included in the figure specified in the COLR. Any fouling which might bias the Reactor Coolant System flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant performance parameters. If detected, action shall be taken before performing subsequent precision heat balance measurements, i.e., either the effect of the fouling shall be quantified and compensated for in the Reactor Coolant System flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

The 12-hour periodic surveillance of indicated Reactor Coolant System flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of operation specified on the figure specified in the COLR.

3/4.2.4 QUADRANT POWER TILT RATIO (Unit 2)

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the movable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore

QUADRANT POWER TILT RATIO (Unit 2) (Continued)

flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. The normal locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8. Alternate locations are available if any of the normal locations are unavailable.

3/4.2.5 DNB PARAMETERS (Unit 2)

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a design limit DNBR throughout each analyzed transient. The indicated T_{avg} value and the indicated pressurizer pressure value correspond to analytical limits of 594.8°F and 2205.3 psig respectively, with allowance for measurement uncertainty.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. Indication instrumentation measurement uncertainties are accounted for in the limits provided in Table 3.2-1.

CORE OPERATING LIMITS REPORT (Continued)

9. DPC-NE-3000P-A, Rev. 1, "Thermal-Hydraulic Transient Analysis Methodology," November 1991.

   (Modeling used in the system thermal-hydraulic analyses)

10. DPC-NE-1004A, "Nuclear Design Methodology Using CASMO-3/SIMULATE-3P," November 1992.

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient.)

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

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC in accordance with 10 CFR 50.4.

8.2 Changes to Core Operating Limits Report