RELOAD REPORT

Catawba Unit 1 Cycle 7

Duke Power Company
Nuclear Generation Department
Nuclear Engineering Section

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1. INTRODUCTION AND SUMMARY

This report justifies the operation of the seventh cycle of Catawba Nuclear Station, Unit 1 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 7 for Catawba Unit 1 is the second Catawba cycle for which the reload fuel is supplied by B&W Fuel Company (BWFC). The incoming Batch 9 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-3001P (Reference 12) and DPC-NE-2004PA (Reference 8). The remaining FSAR Chapter 15 non-LOCA system transient analysis methodology is documented in DPC-NE-3002. The FSAR Chapter 15 LOCA system transient analysis methodology is documented in Reference 13. Approval of these topical reports have been completed.

Section 2 of this report is the operating history for fuel in Catawba Unit 1. Section 3 is a general description of the reactor core, and the fuel system design is provided in Section 4. Reactor and system parameters and conditions are summarized in Sections 5, 6, and 7. Changes to the Technical Specifications, Core Operating Limits Report (COLR), and Final Safety Analysis Report (FSAR) are provided in Section 8.

All of the accidents analyzed in the FSAR (Reference 1) have been reviewed for Cycle 7 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. 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-3001P. For the other reanalyzed transients, the approved methodology is documented in DPC-NE-3002. 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 7 are justified in this report. Based on the analyses performed, it has been concluded that Catawba Unit 1 Cycle 7 can be safely operated at a core power level of 3411 MW_{th}.

2. OPERATING HISTORY

The current operating cycle for Catawba, Unit 1 is Cycle 6, which achieved criticality on June 11, 1991 and reached 100% full power on June 25, 1991. Cycle 6 is scheduled to shut down in June 1992 after 350 EFPD. Catawba, Unit 1 Cycle 6 operated with fresh Mark-BW fuel assemblies. Previous to Cycle 6, Unit 1 operated entirely with fuel assemblies of Westinghouse design.

Catawba, Unit 1, Cycle 7 is the second Catawba Unit 1 reload to contain a full reload batch of Mark-BW fuel assemblies (FAs). Cycle 7 is scheduled to startup in August 1992 at a rated power level of 3411 MWt and has a design cycle length of 350 EFPD. No operating anomalies have occurred during previous cycle operations that would adversely affect fuel performance in Cycle 7.

3. GENERAL DESCRIPTION

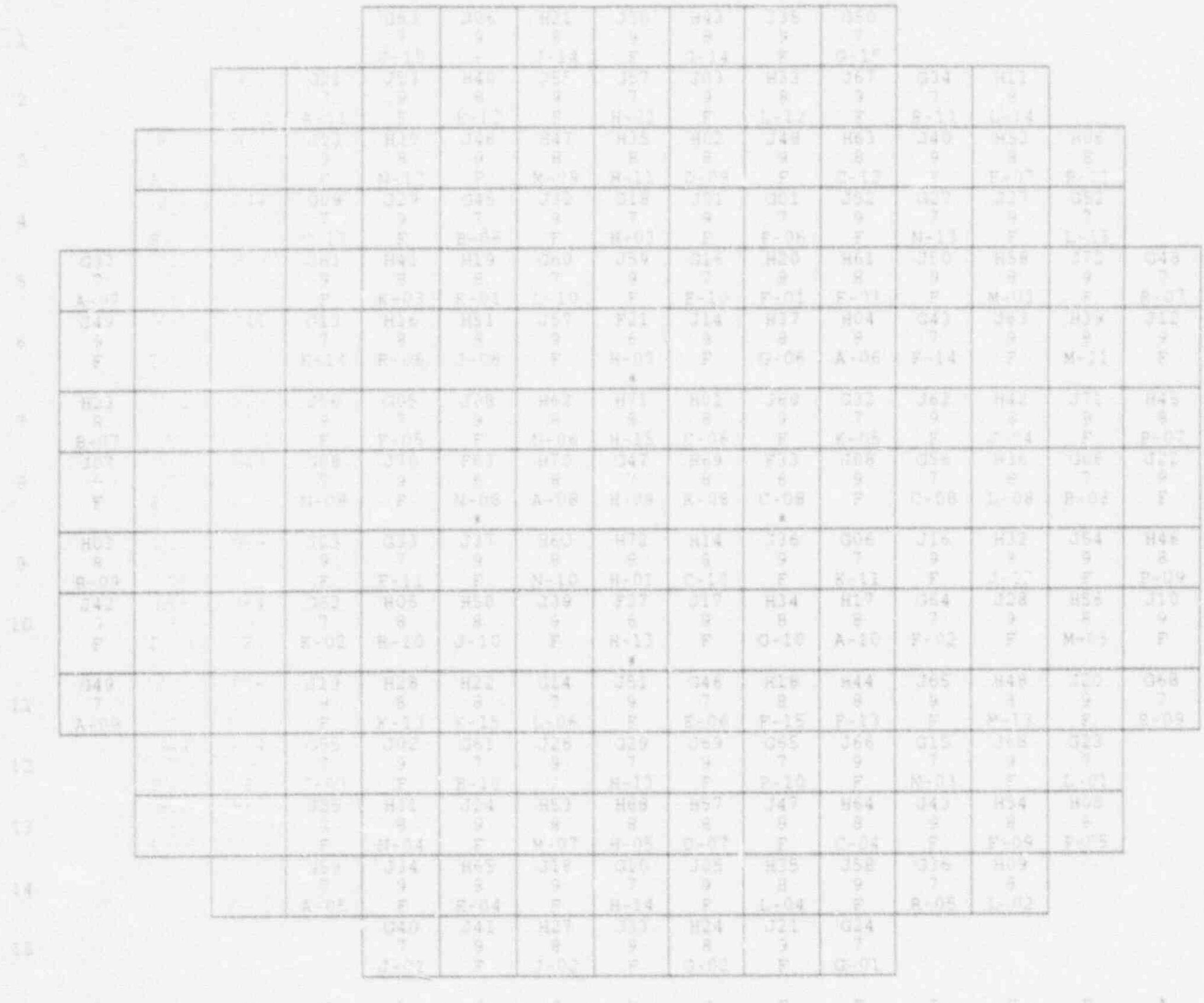
The Catawba Unit 1 reactor core is described in detail in Chapter 4 of the FSAR (Reference 1). The core consists of 193 assemblies, each of which is a 17-by-17 array containing 264 fuel rods, 24 guide tubes, and 1 incore instrument tube. There are 121 burned FAs in the core: 72 of the Mark-BW design (Reference 2) and 49 of the Westinghouse Optimized Fuel Assembly design, and 72 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 average nominal fuel loadings are 424.90, 426.41, 456.68, and 456.20 kg of uranium per fuel assembly in batches 6A, 7A, 8A, and 9A, respectively. The initial enrichments of batches 6A, 7A and 8A were 3.25, 3.40, and 3.55 wt% ^{235}U. The design enrichment of the fresh batches 9A (Mark-BW) is 3.45 wt% ^{235}U.

The 4 batch 6A, 45 batch 7A, and 72 batch 8A assemblies will be shuffled to new locations. One batch 7A FA will be inserted into the center assembly location. The 4 batch 6A FA's are re-inserts from the spent fuel pool discharged from Catawba 1 Cycle 5. The 72 fresh batch 9A assemblies will be loaded into the core in a basically symmetric checkerboard pattern. Figure 3-2 is a quarter core map showing the burnup and region reference number with corresponding initial enrichments of each assembly at the beginning of Cycle 7.

Cycle 7 will be operated in a feed-and-bleed mode. Core reactivity is controlled by 53 rod cluster control assemblies (RCCAs), 528 Mark-BW burnable absorbers, and soluble boron shim. Figure 3-3 shows the Cycle 7 fresh fuel locations with the Mark-BW BPRA clusters and number of pins (enriched to 3.0 wt% B_4C-Al_2O_3) in each location.

FIGURE 3-1
CORE LOADING PATTERN FOR CATAWBA UNIT 1 CYCLE 7

REGION REFERENCE NUMBERS
REGION NUMBERS
PREVIOUS CORE LOCATIONS



XXX	REGION REFERENCE NUMBER
YY	REGION NUMBER
Z-ZZ	CYCLE 6 LOCATION
*	DENOTES CYCLE 1 LOCATION

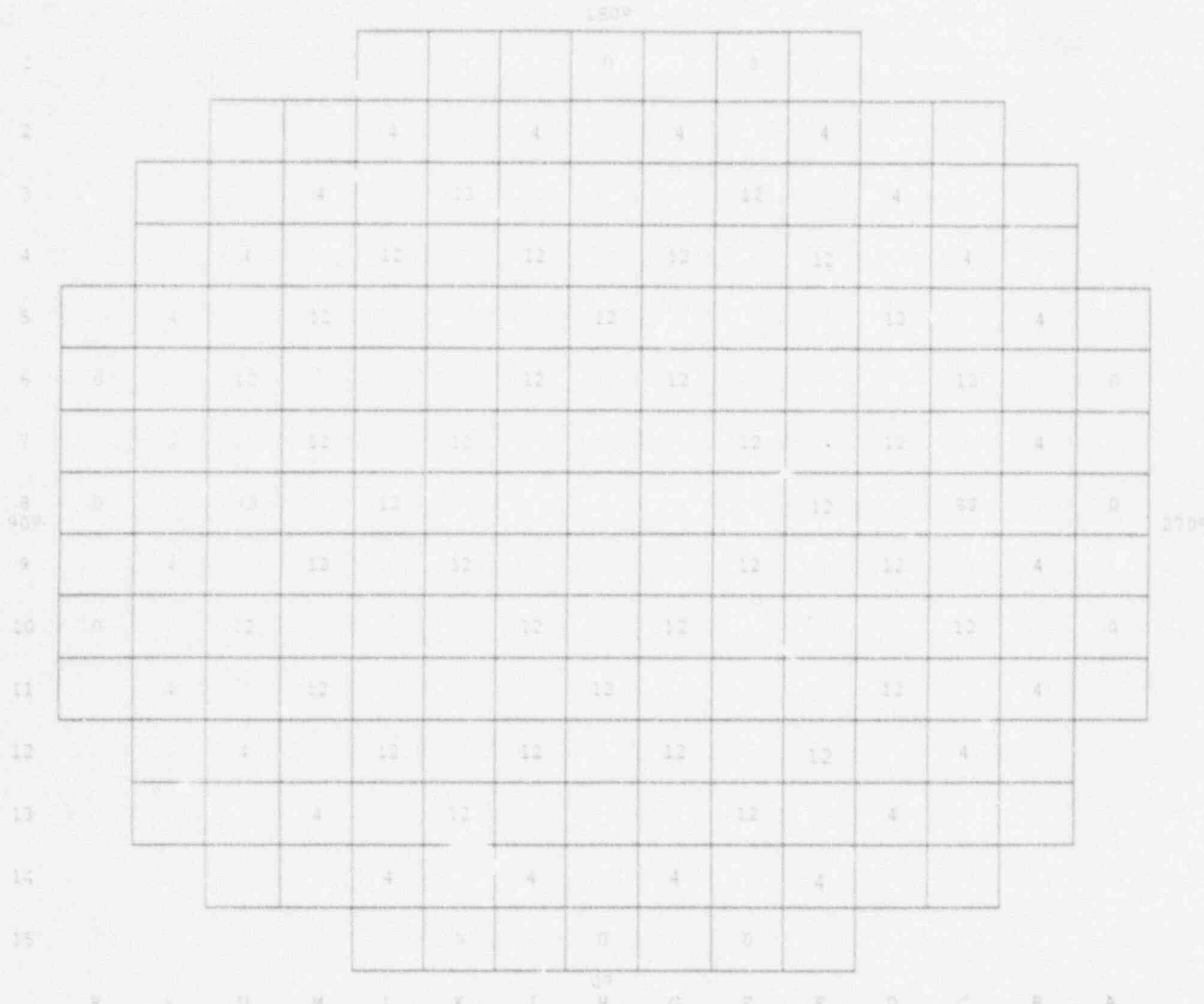
FIGURE 3-2

ENRICHMENT AND BOC BURNUP DISTRIBUTION FOR CATAWBA 1 CYCLE 7

	H	G	F	E	D	C	B	A
8	32589.7 33485.6 7A	11892.0 15859.4 8A	26935.7 30422.3 6A	.0 .0 9A	26228.2 31450.3 7A	17676.4 18709.4 8A	32252.8 33694.8 7A	.0 .0 9A
9	11894.2 15862.5 8A	16757.7 17913.8 8A	.0 .0 9A	26706.2 29975.1 7A	.0 .0 9A	16997.2 18635.5 8A	.0 .0 9A	16602.4 17979.1 8A
10	26932.5 30419.6 6A	.0 .0 9A	17455.5 18698.2 8A	10968.7 15349.7 8A	30412.5 32349.8 7A	.0 .0 9A	17340.3 19066.8 8A	.0 .0 9A
11	.0 .0 9A	26705.3 29974.0 7A	10970.6 15352.8 8A	16760.5 17917.3 8A	.0 .0 9A	15780.1 18062.9 8A	.0 .0 9A	24187.3 27560.1 7A
12	28236.3 31454.8 7A	.0 .0 9A	30417.5 32353.4 7A	.0 .0 9A	26910.4 30360.0 7A	.0 .0 9A	22154.7 26220.2 7A	
13	17677.9 18709.5 8A	16990.4 18617.8 8A	.0 .0 9A	15780.7 18062.5 8A	.0 .0 9A	17458.2 18717.4 8A	14931.8 17796.9 8A	
14	32253.6 33696.8 7A	.0 .0 9A	17342.0 19067.2 8A	.0 .0 9A	22156.3 26220.0 7A	14931.0 17798.6 8A	Average Maximum Region #	
15	.0 .0 9A	16605.9 17984.0 8A	.0 .0 9A	24186.5 27560.5 7A				

REGION	ENRICHMENT w/o U-235	CYCLES BURNED	NUMBER OF ASSEMBLIES	BOC BURNUP MWD/MTU
6A	3.25	2	4	26935
7A	3.40	2	45	26887
8A	3.55	1	72	15736
9A	3.45	0	72	0
CORE			193	12698

FIGURE 3-3
CATAWBA UNIT 1 CYCLE 7
BURNABLE ABSORBER AND SOURCE ASSEMBLY LOCATIONS



NUMBER OF MkBW-BP PINS/ASSEMBLY	NUMBER OF BACKPLATE ASSEMBLIES
4	24
12	36

All locations use 3.0 wt% B_4C-Al_2O_3 MkBW BP's.

4. FUEL SYSTEM DESIGN

4.1 Fuel Assembly Mechanical Design

The Catawba 1 Cycle 7 core will include 72 fresh Mark-BW fuel assemblies. The re-inserted fuel assemblies in Cycle 7 will be Westinghouse Optimized fuel assemblies (49) and Mark-BW fuel assemblies (72). 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 1 Cycle 7 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 7 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 1 Cycle 7 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 7 (in Batch 8) is 33,868 MWD/MTU and the maximum predicted fuel rod burnup (in Batch 8) is 35,502 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. The reinserted OFA fuel design criteria was evaluated with acceptable results for the Cycle 7 predicted operation.

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 cycles 5 and 6 and fuel assembly bow, twist, growth, and holddown spring set were all within nominal bounds. A poolside examination of the Mark-BW fuel is scheduled for late 1992. Four other Mark-BW lead assemblies underwent their first cycle of irradiation in Trojan Cycle 13.

Catawba 1 Cycle 7 will be the fourth complete reload batch of Mark-BW 17 x 17 fuel. The first complete reload batch began operating in Catawba 1 Cycle 6 in June 1991. The second batch, McGuire 1 Cycle 8, began operation in December 1991 and the third batch, McGuire 2 Cycle 8, in March 1992.

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

	Batch 8	Batch 9
Nominal fuel rod OD, in.	0.374	0.374
Nominal fuel rod ID, in.	0.326	0.326
Nominal active fuel length, in.	144.0	144.0
Nominal fuel pellet OD, in.	0.3195	0.3195
Fuel pellet initial density, % TD	96.0	96.0
Initial fuel enrichment, wt. % U_{235}	3.55	3.45
Estimated residence time EOC 7, EFPH	16,800	8400
Cladding collapse time, EFPH	>18,100	>18,100
Nominal linear heat rate (LHR), kW/ft	5.43	5.43
Ave. fuel temperature @ nom. LHR, deg F	1360	1360
Minimum LHR to melt, kW/ft		
0-1000 MWD/MTU	21.5	21.5
> 1000 MWD/MTU	21.8	21.8

5. NUCLEAR DESIGN

5.1 Physics Characteristics

Table 5-1 provides the core physics parameters for Cycles 6 and 7. The values were generated using the methodology described in DPC-NF-2010A (Reference 6) and are valid for the design cycle length (350 EFPD ± 10 EFPD). Figure 5-1 illustrates a representative relative power distribution for the beginning of Cycle 7 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 7 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 7 was analyzed at 360 EFPD.

5.2. Changes in Nuclear Design

No core design changes have been implemented in Cycle 7 which will impact the Nuclear Design parameters. The Cycle 7 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 7 are presented in Section 8.

Table 5.1 Physics Parameters[a] Catawba 1 Cycles 6 and 7

	Cycle 6	Cycle 7
Design cycle length, EFPD	350	350
Design cycle burnup, MWD/MTU	14234	13843
Design average core burnup - EOC, MWD/MTU	26371	26541
Design initial core loading, MTU	84.0899	86.4443
Critical boron - BOC,ppmb, no Xe[b]		
HZP, ARO	1664	1726
HFP, ARO	1529	1565
Critical boron - EOC,ppmb		
HZP, No Xe, ARO	567	622
HFP, Eq Xe, ARO	0	2
Total Control Rod Worths - HZP, pcm		
BOC	6915	6585
EOC[c]	7221	7238
Max ejected rod worth(d) - HZP, pcm		
BOC (D12)	355(e)	449
EOC[c] (D12)	473(e)	[illegible]08
Max stuck rod worth - HZP, pcm		
BOC (F-10)	992	1436
EOC[c] (F-10)	983	1386
Power deficit - HZP to HFP, pcm		
BOC	-1668	-1752
EOC[c]	-3117	-3049
Doppler coeff - HFP, pcm/°F		
BOC, no Xe	-1.16	-1.19
EOC[c], eq Xe	-1.46	-1.48
Moderator coeff - HFP, pcm/°F		
BOC, no Xe	-2.15	-3.73
EOC[c], eq Xe, 0 PPMB	-32.27(e)	-33.85
Boron worth - HFP, pcm/ppmb		
BOC	-8.27	-7.76
EOC[c]	-9.54	-8.86

Table 5.1 Physics Parameters[a] Catawba 1 Cycles 6 and 7 (cont)

	Cycle 6	Cycle 7
Equilibrium Xenon worth - HFP, pcm		
BOC (4 EFPD)	2677	2642
EOC	2993	2814
Effective delayed neutron fraction - HFP		
BOC	0.00626(e)	0.006191
EOC(c)	0.00525(e)	0.005229

(a) Cycle 6 and 7 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.

(e) These values were generated by B&W Fuel Company. All other Cycle 6 values in this table were generated by Duke Power Company. Differences between Cycle 6 and Cycle 7 values for these parameters are more likely due to methodology differences than to differences between the cycles.

Table 5-2. Shutdown Margin Calculation for Catawba 1 Cycle 7

Control Rod Worth	BOC (PCM)	EOC[a] (PCM)
1. All rods inserted (ARI), HZP	6585	7238
2. ARI less most reactive stuck rod, HZP	5149	5852
3. Less 10% uncertainty	4634	5267
Required Rod Worth		
4. Rod Insertion Allowance (RIA)[b]	273	356
5. Power Defect, HFP to HZP[b]	2057	3354
6. Shutdown Margin (total available worth minus total required worth)	2304	1557

NOTE: Required shutdown Margin is 1300 PCM.

(a) EOC physics parameters calculated at 360 EFPD, i.e., design EOC plus 10 EFPD.

(b) The Rod Insertion Allowance and Power Defect penalties in the Shutdown Margin Calculation account for the effects of transient xenon conditions.

Figure 5-1: BOC (4 EFPD), Cycle 7 Two-Dimensional Relative Power Distribution - HFP, Equilibrioum Xenon

	H	G	F	E	D	C	B	A
8	1.0110 1.0470	1.3007 1.4063	0.9888 1.0707	1.1452 1.2593	0.9177 0.9755	1.0654 1.1317	0.8083 0.8763	0.7284 0.9811
9	1.3013 1.4066	1.2806 1.3693	1.2722 1.3893	0.9845 1.0310	1.1618 1.2696	1.1603 1.2623	1.1976 1.3655	0.6461 0.9435
10	0.9890 1.0712	1.2724 1.3894	1.2642 1.3508	1.2870 1.4041	0.9854 1.0398	1.2381 1.3178	1.0753 1.2318	0.7205 1.0438
11	1.1451 1.2593	0.9843 1.0309	1.2866 1.4034	1.2775 1.3599	1.2752 1.3722	1.1945 1.3305	1.0397 1.2768	0.3713 0.7534
12	0.9174 0.9751	1.1616 1.2695	0.9851 1.0395	1.2749 1.3712	1.0306 1.1372	1.1765 1.3482	0.5797 0.9446	
13	1.0653 1.1315	1.1603 1.2624	1.2380 1.3177	1.1945 1.3305	1.1765 1.3483	0.7564 1.0305	0.3518 0.7645	
14	0.8082 0.8762	1.1975 1.3654	1.0752 1.2317	1.0397 1.2768	0.5797 0.9447	0.3518 0.7644		
15	0.7283 0.9810	0.6461 0.9434	0.7204 1.0437	0.3713 0.7535				

XXXX YYYY	P(AVG) PEAK PIN

6. THERMAL-HYDRAULIC DESIGN

The generic and cycle-specific analyses supporting Cycle 7 operation were performed by Duke Power Company using the methodology described in Reference 8. Cycle 7 is the second Mark-BW transition cycle and the first cycle to be 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 7 are based on a full Mark-BW core and a design FΔH of 1.50. The Cycle 7 nominal thermal-hydraulic design conditions are given in Table 6-2.

To provide design flexibility, margin is added to the SDL to determine a design DNBR limit (DDL). For the generic Mark-BW and Catawba 1 Cycle 7 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.

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. The peaking penalty is given in Table 6-4.

The two remaining DNBR peanalties assessed against the DNB margin account for fuel rod bow (OFA fuel only), plant instrumentation biases, and clipped mixing vanes (Mark-BW fuel only).

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 1 Cycle 7

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 $F_{\Delta H}$	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 system piping failure
- Turbine trip
- Feedwater system pipe break
- Partial loss of forced reactor coolant flow
- Complete loss of forced reactor coolant flow
- Reactor coolant pump shaft seizure (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, DNBR

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 2 Cycle 8 reload report, Reference 17. The Catawba 1 Cycle 7 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 system piping failure, 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 system piping failure, 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 6 and Chapter 15, respectively, of the Catawba FSAR, Reference 1. The Catawba 1 Cycle 7 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, as well as changes in the dose analysis methodology.

- Reactor coolant pump shaft seizure (locked rotor)
- Single rod withdrawal
- Rod ejection

All of the above dose analyses are described in Section 8 of this report.

Catawba 1 Cycle 7 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 1 Cycle 7.

8. PROPOSED MODIFICATIONS TO LICENSING BASIS DOCUMENTS

Revisions to the Technical Specifications and Core Operating Limits Report (COLR) have been proposed for Cycle 7 operation to accommodate the influence of the Cycle 7 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 7 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 f(ΔI) 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 the Core Operating Limits Report, 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 7, and these changes are identical to those submitted in the approved McGuire 2 Cycle 8 reload report (Reference 17), 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 cover. 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 and COLR described in this report, Cycle 7 of Catawba Unit 1 will operate within the 10CFR 50.46 ECCS acceptance criteria. The following pages contain the required Technical Specification revisions and the revisions to the COLR.

Table 8-1 Techaical Specification Changes

Specification	Description of Change
2.1.1	decreased $F_{\Delta H}$ for Mark-BW fuel
2.2.1	decreased $F_{\Delta H}$ for Mark-BW fuel removed power range neutron flux negative rate reactor trip * removed Total Allowance, Z value, and Sensor Error terms
3.1.3.1	included all accident analyses that would require reevaluation in the event that one full length RCCA is inoperable
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
3/4.2.5	corrected action item requirement
3/4.3.3.1	removed power range neutron flux negative rate reactor trip removed items associated with RTD Bypass System
3/4.3.3.2	increased low steam line pressure setpoint increased feedwater isolation response time increased steam line isolation response time * removed Total Allowance, Z value, and Sensor Error terms removed items associated with RTD Bypass System * removed steam line pressure dynamic compensation
3/4.4.1.2	changed reactor coolant loop operation requirement
3/4.4.2.1	* increased pressurizer safety valve lift setpoint tolerance
3/4.4.2.2	* increased pressurizer safety valve lift setpoint tolerance
3/4.5.1.[illegible]	changed required cold leg accumulator boron concentration
3/4.5.2	changed ECCS pump surveillance requirements
3/4.6.2	* reduced allowable primary to secondary leakage rate
3/4.6.3	* changed feedwater isolation valve, main steam isolation valve, and main steam isolation bypass 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	reflected change to DPC core operating limit methodology

* The proposed Technical Specification change was not included in the approved McGuire 2 Cycle 8 reload report, Reference 17.

Table 8-2 Core Operating Limits Report Changes

Specification	Description of Change
3/4.1.3.5	revised safety 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 7 operation
3/4.2.2	revised for Cycle 7 operation to reflect a change in the heat flux hot channel factor F_Q methodology
3/4.2.3	revised for Cycle 7 operation to reflect a change in the nuclear enthalpy rise hot channel factor $F_{\Delta H}$ methodology

8.1 Changes to Technical Specifications

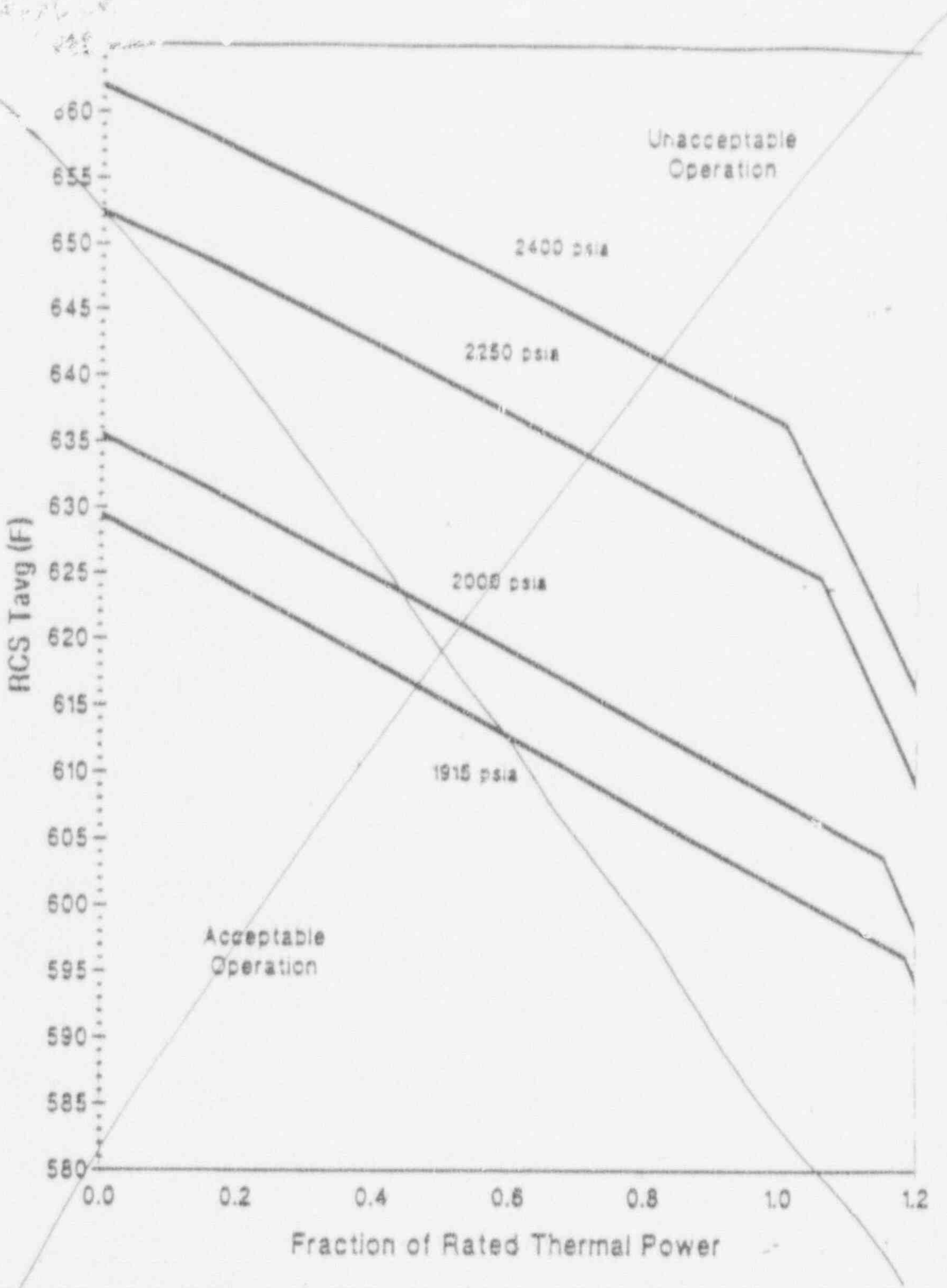


FIGURE 2.1-1a

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

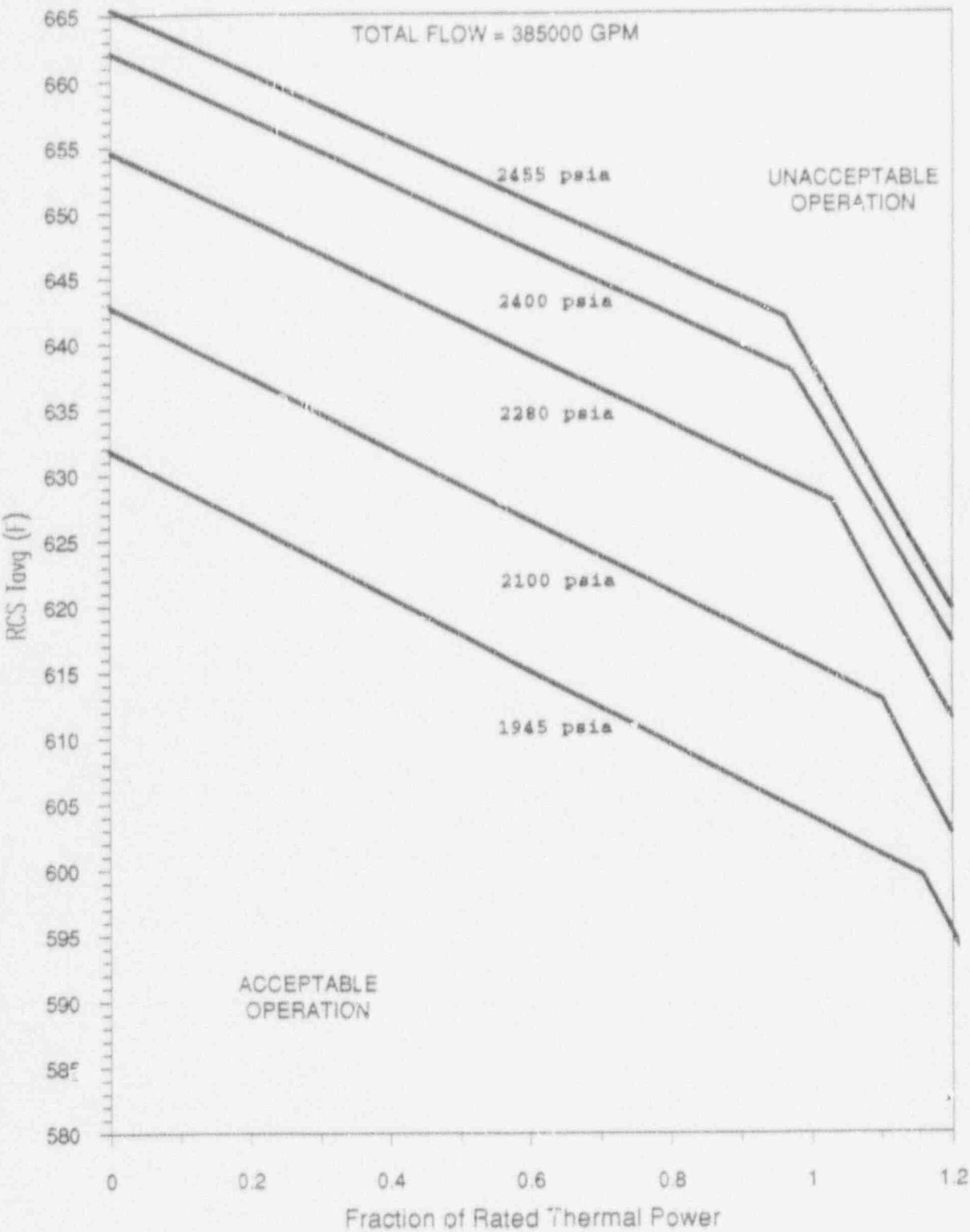


FIGURE 2.1-1a

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

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlocks Setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

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

ACTION:

a. With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value Column of Table 2.2-1, adjust the Setpoint consistent with the Trip Setpoint value.

b. With the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, ~~either:~~

Delete

1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 2.2-1 and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel, or

~~2.~~ ~~D~~declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

Delete

Equation 2.2-1 $Z + R + S \leq TA$

Where:

Z = The value from Column Z of Table 2.2-1 for the affected channel,

R = The "as measured" value (in percent span) of rack error for the affected channel,

S = Either the "as measured" value (in percent span) of the sensor error, or the value from Column S (Sensor Error) of Table 2.2-1 for the affected channel, and

TA = The value from Column TA (Total Allowance) of Table 2.2-1 for the affected channel.

TABLE 2.2.-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

Delete

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux					
a. High Setpoint	7.5	5.92	0	$\leq$109% of RTP*	$\leq$110.9% of RTP*
b. Low Setpoint	8.3	5.92	0	$\leq$25% of RTP*	$\leq$27.1% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	1.6	0.5	0	$\leq$5% of RTP* with a time constant $\geq$ 2 seconds	$\leq$6.3% of RTP* with a time constant $\geq$ 2 seconds
Delete 4. Power Range, Neutron Flux, High Negative Rate	1.6	0.5	0	$\leq$5% of RTP* with a time constant $\geq$2 seconds	$\leq$6.3% of RTP* with a time constant $\geq$2 seconds
~~5~~ 4. Intermediate Range, Neutron Flux	17.0	8.4	0	$\leq$25% of RTP*	$\leq$31% of RTP*
~~6~~ 5. Source Range, Neutron Flux	17.0	10	0	$\leq 10^5$ cps	$\leq 1.4 \times 10^5$ cps
~~7~~ 6. Overtemperature ΔT	6.98 for Unit 1 and 8.9 for Unit 2	3.6 for Unit 1 and 7.3 for Unit 2	2.12 for Unit 1 and 2.7 for Unit 2	See Note 1	See Note 2
~~8~~ 7. Overpower ΔT	4.9	1.24	1.7	See Note 3	See Note 4
~~9~~ 8. Pressurizer Pressure-Low	4.0	2.21	1.5	$\geq$1945 psig	$\geq$1938 psig***
~~10~~ 9. Pressurizer Pressure-High	7.5	0.71	0.5	$\leq$2385 psig	$\leq$2399 psig
~~11~~ 10. Pressurizer Water Level-High	5.0	2.18	1.5	$\leq$92% of instrument span	$\leq$93.8% of instrument span
~~12~~ 11. Reactor Coolant Flow-Low	2.92	1.48	0.6	$\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

Delete

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
12 ~~13~~. Steam Generator Water Level Low-Low					
a. Unit 1	17	14.2	1.5	$\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*
b. Unit 2	11.8	1.7	2.0	$\geq$36.8% of narrow range span	$\geq$35.1% of narrow range span
13 ~~14~~. Undervoltage - Reactor Coolant Pumps	8.57	0	1.0	$\geq$77% of bus voltage (5082 volts) with a 0.7s response time	$\geq$76% (5016 volts)
14 ~~15~~. Underfrequency - Reactor Coolant Pumps	4.0	0	1.0	$\geq$56.4 Hz with a 0.2s response time	$\geq$55.9 Hz
15 ~~16~~. Turbine Trip					
a. Stop Valve EH Pressure Low	N.A.	N.A.	N.A.	$\geq$550 psig	$\geq$500 psig
b. Turbine Stop Valve Closure	N.A.	N.A.	N.A.	$\geq$1% open	$\geq$1% open
16 ~~17~~. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.

*RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

Delete

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

*RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} [T (\frac{1}{1 + \tau_6 S}) - T'] + K_3(P - P') - f_1(\Delta I)\}$$

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_0 = Indicated ΔT at RATED THERMAL POWER;

K_1 = ~~1.38~~; 1.1953

K_2 = ~~0.02401~~/°F; 0.03163

$\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = 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;

UNIT 1

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~~; 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 ~~-22.5%~~ -39.9% and ~~-6.5%~~ +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 ~~-22.5%~~ -39.9%, the ΔI Trip Setpoint shall be automatically reduced by ~~3.151%~~ 3.910% of ~~its value at RATED THERMAL POWER~~ ΔT_0; and

(iii) For each percent ΔI that the magnitude of $q_t - q_b$ is more positive than ~~-6.5%~~ +3.0%, the ΔI Trip Setpoint shall be automatically reduced by ~~1.641% for Unit 1 and 2.414% for Unit 2~~ 2.316% of ~~its value at RATED THERMAL POWER~~ ΔT_0.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3.0% ~~for Unit 1 and 1.3% for Unit 2~~.

UNIT 1

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~~, 1.0819

K_5 = $0.02/^\circ F$ 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,

UNIT 1

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

NOTE 3: (Continued)

0.001291

K_6 = ~~0.001707~~/°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~~

Insert Attachment 1

~~$f_2(\Delta I)$ = 0 for all ΔI.~~

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

and $f_2(\Delta I)$ 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 -35% and +35% ΔI; $f_2(\Delta I) = 0$, 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 -35% ΔI, the ΔT Trip Setpoint shall be automatically reduced by 7.0% of ΔT_0; and

(iii) for each percent ΔI that the 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

UNIT 2

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

Delete

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux					
a. High Setpoint	7.5	5.92	0	≤109% of RTP*	≤110.9% of RTP*
b. Low Setpoint	8.3	5.92	0	≤25% of RTP*	≤27.1% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	1.6	0.5	0	≤5% of RTP* with a time constant ≥ 2 seconds	≤6.3% of RTP* with a time constant ≥ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	1.6	0.5	0	≤5% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.4	0	≤25% of RTP*	≤31% of RTP*
6. Source Range, Neutron Flux	17.0	10	0	≤10^5 cps	≤1.4 x 10^5 cps
7. Overtemperature ΔT	6.98 for Unit 1 and 8.9 for Unit 2	3.0 for Unit 1 and 7.3 for Unit 2	2.12 for Unit 1 and 2.7 for Unit 2	See Note 1	See Note 2
8. Overpower ΔT	4.9	1.24	1.7	See Note 3	See Note 4
9. Pressurizer Pressure-Low	4.0	2.21	1.5	≥1945 psig	≥1938 psig***
10. Pressurizer Pressure-High	7.5	0.71	0.5	≤2385 psig	≤2399 psig
11. Pressurizer Water Level-High	5.0	2.18	1.5	≤92% of instrument span	≤93.8% of instrument span
12. Reactor Coolant Flow-Low	2.92	1.48	0.6	≥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.

UNIT 2

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

Delete

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
13. Steam Generator Water Level Low-Low					
a. Unit 1	17	14.2	1.5	≥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	11.8	1.7	2.0	≥36.8% of narrow range span	≥35.1% of narrow range span
14. Undervoltage - Reactor Coolant Pumps	8.57	0	1.0	≥77% of bus voltage (5082 volts) with a 0.7s response time	≥76% (5016 volts)
15. Underfrequency - Reactor Coolant Pumps	4.0	0	1.0	≥56.4 Hz with a 0.2s response time	≥55.9 Hz
16. Turbine Trip					
a. Stop Valve EH Pressure Low	N.A.	N.A.	N.A.	≥550 psig	≥500 psig
b. Turbine Stop Valve Closure	N.A.	N.A.	N.A.	≥1% open	≥1% open
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.

*RTP = RATED THERMAL POWER

UNIT 2

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

Delete

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	N.A.	N.A.	N.A.	$\geq 1 \times 10^{-10}$ amps	$\geq 6 \times 10^{-11}$ amps
b. Low Power Reactor Trips Block, P-7					
1) P-10 input	N.A.	N.A.	N.A.	$\leq$10% of RTP*	$\leq$12.2% of RTP*
2) P-13 input	N.A.	N.A.	N.A.	<10% RTP* Turbine Impulse Pressure Equivalent	<12.2% RTP* Turbine Impulse Pressure Equivalent
c. Power Range Neutron Flux, P-8	N.A.	N.A.	N.A.	$\leq$48% of RTP*	$\leq$50.2% of RTP*
d. Power Range Neutron Flux, P-9	N.A.	N.A.	N.A.	$\leq$69% of RTP*	$\leq$70% of RTP*
e. Power Range Neutron Flux, P-10	N.A.	N.A.	N.A.	$\geq$10% of RTP*	$\geq$7.8% of RTP*
f. Power Range Neutron Flux, Not P-10	N.A.	N.A.	N.A.	$\leq$10% of RTP*	$\leq$12.2% of RTP*
g. Turbine Impulse Chamber Pressure, P-13	N.A.	N.A.	N.A.	<10% RTP* Turbine Impulse Pressure Equivalent	<12.2% RTP* Turbine Impulse Pressure Equivalent
19. Reactor Trip Breakers	N.A.	N.A.	N.A.	N.A.	N.A.
20. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	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	=	1.38;
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, τ_6 = 0;

UNIT 2

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 ΔI Trip Setpoint shall be automatically reduced by 3.151% of ~~its value at RATED THERMAL POWER~~ ΔT_0; and

(iii) For each percent ΔI that the magnitude of $q_t - q_b$ is more positive than -6.5%, the ΔI Trip Setpoint shall be automatically reduced by ~~1.641% for Unit 1 and~~ 2.414% ~~for Unit 2~~ of ~~its value at RATED THERMAL POWER~~ ΔT_0.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than ~~3.0% for Unit 1 and~~ 1.3% ~~for Unit 2~~.

UNIT 2

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

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

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,

UNIT 2

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

NOTE 3: (Continued)

K_6 = 0.001707/°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

$f_2(\Delta I)$ = 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, fuel fabrication parameters, and in the BWCMV DNB correlation 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 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 ~~1.49~~ 1.50 for Westinghouse Optimized Fuel Assemblies (OFA's) and 1.55 for the BWFC Mark-BW Fuel Assemblies 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:

delete

$$F^N_{\Delta H} = \cancel{1.49}\ 1.50\ [1 + \cancel{0.3}\ (1/RRH)\ (1-P)]$$ ~~For the Westinghouse OFA's~~

delete

$$F^N_{\Delta H} = 1.55\ [1 + 0.3\ (1-P)] \text{ For the BWFC Mark-BW's}$$

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.

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the nominal values at which the Reactor trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the core and Reactor Coolant System are prevented from exceeding their safety limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. The Setpoint for a Reactor Trip System or interlock function is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Reactor Trip Setpoints have been specified in Table 2.2-1. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its Trip Setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 2.2-1, $Z + R + S \leq TA$, the interactive effects of the errors in the rack and the senso , and the "as measured" values of the errors are considered. Z, as specified in Table 2.2-1, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the Trip Setpoint and the value used in the analysis for Reactor trip. R or Rack Error is the "as measured" deviation, in percent span, for the affected channel from the specified Trip Setpoint. S or Sensor Error is either the "as measured" deviation of the sensor from its calibration point or the value specified in Table 2.2-1, in percent span, from the analysis assumptions. Use of Equation 2.2-1 allows for a sensor drift factor, an increased rack drift factor, and provides a threshold value for REPORTABLE EVENTS.

Delete

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

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.

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 Unit 1

TABLE 3.1-1

ACCIDENT ANALYSES REQUIRING REEVALUATION IN THE EVENT OF AN INOPERABLE FULL-LENGTH ROD

Rod Cluster Control Assembly Insertion Characteristi -

Rod Cluster Control Assembly ~~Misalignment~~ Misoperation

Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes Which Actuates the Emergency Core Cooling System

~~Single Rod Cluster Control Assembly Withdrawal at Full Power~~

~~Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident)~~

Major Secondary Coolant System Pipe Rupture

Rupture of a Control Rod Drive Mechanism Housing (Rod Clus   Control Assembly Ejection)

8-28

(a) Within 15 minutes:

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

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

$$+ [\cancel{\text{NSLOPE}_1^{(3)}} \times \text{Margin}^{\text{min}}_{\text{OP}}] \text{ absolute value}$$

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

$$-[\cancel{\text{PSLOPE}_1^{(3)}} \times \text{Margin}^{\text{min}}_{\text{OP}}] \text{ absolute value}$$

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^{AAA} 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.

(5) at least two

d. Extrapolating ~~the two most recent~~ measurements to 31 Effective Full Power Days beyond the most recent measurement and if:

Replace w/ Attachment 1

$[F_Q^M(X,Y,Z)$ (extrapolated) $\geq [F_Q^L(X,Y,Z)]^{OP}$ (extrapolated), or

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

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.

Include

(5) Extrapolation of F_Q^M 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. 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.

Attachment 1:

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

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

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:

$$F\Delta H^{M}_{\cancel{R}}(X,Y) \le \left[F\Delta H^{L}_{\cancel{R}}(X,Y)\right]^{LCO}$$

Where: $F\Delta H^{M}_{\cancel{R}}(X,Y)$ = the ~~maximum~~ measured radial peak. ~~ratio as defined in the CORE OPERATING LIMITS REPORT (COLR).~~

$\left[F\Delta H^{L}_{\cancel{R}}(X,Y)\right]^{LCO}$ = the maximum allowable radial peak ~~ratio~~ 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 $F\Delta H^{M}_{\cancel{R}}(X,Y)$ exceeds the limit, and

b. Within 6 hours either:

1. Restore $F\Delta H^{M}_{\cancel{R}}(X,Y)$ 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 $F\Delta H^{M}_{\cancel{R}}(X,Y)$ exceeds that limit, and

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

1. Restore $F\Delta H^{M}_{\cancel{R}}(X,Y)$ to within the limit of Specification 3.2.3 for RATED THERMAL POWER, or
2. Perform the following actions:

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

   (b) Verify through incore mapping that $F\Delta H^{M}_{\cancel{R}}(X,Y)$ 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~~ reduction required to compensate for each 1% that $F\Delta H^{M}_{\cancel{R}}(X,Y)$ exceeds ~~$F\Delta HR^{L}(X,Y)$~~ provided in the COLR per Specification 6.9.1.9.
the limits of Specification 3.2.3

(2) TRH is the amount of $OT\Delta T$ K_1 setpoint reduction required to compensate for each 1% that $F\Delta H^{M}_{\cancel{R}}(X,Y)$ 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 b.2., above; subsequent POWER OPERATION may proceed provided that $F\Delta HR^{M}(X,Y)$ 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 $F\Delta HR^{M}(X,Y)$ 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 $F\Delta HR^{M}(X,Y)$ according to the following schedule:

Replace w/ Attachment 1

1. Prior to operation above 75% of RATED THERMAL POWER at the beginning of each fuel cycle, and the earlier of:

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:

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

No additional uncertainities are required for $F\Delta HR^{M}(X,Y)$, because $[F\Delta HR(X,Y)]_{SUR}$ includes uncertainties.

for Specification 4.2.3.3

Attachment 1:

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

(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$_x$ as if $[F_{\Delta H}^{L}(X,Y)]^{SURV}$ is the same as $[F_{\Delta H}^{L}(X,Y)]^{LCO}$.

d. Extrapolating the two most recent measurements to 31 Effective Full Power Days beyond the most recent measurement and if:

$F\Delta HR^{M}$ (extrapolated) $\geq$ $F\Delta HR^{L}$ (extrapolated)

either of the following actions shall be taken:

1. $F\Delta HR^{M}(X,Y)$ 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.

Replace with Attachment 1

Attachment 1:

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)

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

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.

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},

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 ~~a.~~ 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.

UNIT 1

TABLE 3.3-1

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
Delete 4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2
~~5.~~ 4 Intermediate Range, Neutron Flux	2	1	2	1###, 2	3
~~6.~~ 5 Source Range, Neutron Flux					
a. Startup	2	1	2	2##	4
b. Shutdown	2	1	2	3*, 4*, 5*	10
~~7.~~ 6 Overtemperature ΔT Four Loop Operation	4	2	3	1, 2	6
~~8.~~ 7 Overpower ΔT Four Loop Operation	4	2	3	1, 2	6
~~9.~~ 8 Pressurizer Pressure-Low	4	2	3	1	6**

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
~~10.~~ 9 Pressurizer Pressure-High	4	2	3	1, 2	6**
~~11.~~ 10 Pressurizer Water Level-High	3	2	2	1	6
~~12.~~ 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
~~13.~~ 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**
~~14.~~ 13 Undervoltage-Reactor Coolant Pumps (Above P-7)	4-1/bus	2	3	1	6
~~15.~~ 14 Underfrequency-Reactor Coolant Pumps (Above P-7)	4-1/bus	2	3	1	6
~~16.~~ 15 Turbine Trip					
a. Stop Valve EH Pressure - Low	4	2	3	1####	6
b. Turbine Stop Valve Closure	4	4	1	1####	11
~~17.~~ 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 ~~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
18 ~~19~~. Reactor Trip Breakers	2	1	2	1, 2	9
	2	1	2	3*, 4*, 5*	10
19 ~~20~~. Automatic Trip and Interlock Logic	2	1	2	1, 2	9
	2	1	2	3*, 4*, 5*	10

UNIT 1

TABLE 3.3-1 (Continued)

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.

UNIT 1

TABLE 3.3-2

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. Power Range, Neutron Flux, High Negative Rate	≤ 0.5 second*
~~5.~~ 4 Intermediate Range, Neutron Flux	N.A.
~~6.~~ 5 Source Range, Neutron Flux	N.A.
~~7.~~ 6 Overtemperature ΔT	≤ ~~4(8#)~~ seconds*
~~8.~~ 7 Overpower ΔT	≤ ~~4(8#)~~ seconds*
~~9.~~ 8 Pressurizer Pressure-Low	≤ 2 seconds
~~10.~~ 9 Pressurizer Pressure-High	≤ 2 seconds
~~11.~~ 10 Pressurizer Water Level-High	N.A.

Delete

*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.
~~#Applicable upon deletion of RTD Bypass System.~~

UNIT 1

TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
~~12.~~ 11 Low Reactor Coolant Flow	
a. Single Loop (Above P-8)	≤ 1 second
b. Two Loops (Above P-7 and below P-8)	≤ 1 second
~~13.~~ 12 Steam Generator Water Level-Low-Low	
a. Unit 1	≤ 3.5 seconds
b. Unit 2	≤ 2.0 seconds
~~14.~~ 13 Undervoltage-Reactor Coolant Pumps	≤ 1.5 seconds
~~15.~~ 14 Underfrequency-Reactor Coolant Pumps	≤ 0.6 second
~~16.~~ 15 Turbine Trip	
a. Stop Valve EH Pressure-Low	N.A.
b. Turbine Stop Valve Closure	N.A.
~~17.~~ 16 Safety Injection Input from ESF	N.A.
~~18.~~ 17 Reactor Trip System Interlocks	N.A.
~~19.~~ 18 Reactor Trip Breakers	N.A.
~~20.~~ 19 Automatic Trip and Interlock Logic	N.A.

UNIT 1

TABLE 4.3-1

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	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
Delete 4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(4)	M	N.A.	N.A.	1, 2
~~5.~~ 4 Intermediate Range, Neutron Flux	S	R(4, 5)	S/U(1),M	N.A.	N.A.	1###, 2
~~6.~~ 5 Source Range, Neutron Flux	S	R(4, 5)	S/U(1),M(9)	N.A.	N.A.	2##, 3, 4, 5
~~7.~~ 6 Overtemperature ΔT	S	R	M	N.A.	N.A.	1, 2
~~8.~~ 7 Overpower ΔT	S	R	M	N.A.	N.A.	1, 2
~~9.~~ 8 Pressurizer Pressure-Low	S	R	M	N.A.	N.A.	1
~~10.~~ 9 Pressurizer Pressure-High	S	R	M	N.A.	N.A.	1, 2
~~11.~~ 10 Pressurizer Water Level-High	S	R	M	N.A.	N.A.	1
~~12.~~ 11 Reactor Coolant Flow-Low	S	R	M	N.A.	N.A.	1

UNIT 1

TABLE 4.3-1 (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
~~13~~. 12 Steam Generator Water Level-Low-Low	S	R(13)	M	N.A.	N.A.	1, 2
~~14~~. 13 Undervoltage - Reactor Coolant Pumps	N.A.	R	N.A.	M	N.A.	1
~~15~~. 14 Underfrequency - Reactor Coolant Pumps	N.A.	R	N.A.	M	N.A.	1
~~16~~. 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#
~~17~~. 16 Safety Injection Input from ESF	N.A.	N.A.	N.A.	R**	N.A.	1, 2
~~18~~. 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.

UNIT 1

TABLE 4.3-1 (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
~~18.~~ 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
~~19.~~ 18 Reactor Trip Breaker	N.A.	N.A.	N.A.	M(7, 11)	N.A.	1, 2, 3*, 4*, 5*
~~20.~~ 19 Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M(7)	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (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) At least once per 18 months and following maintenance or adjustment of the Reactor trip breakers, 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.

UNIT 2

TABLE 3.3-1

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. 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**

UNIT 2

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

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	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
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, [illegible]	9

UNIT 2

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

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
	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

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TABLE 3.3-1 (Continued)

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.

UNIT 2

TABLE 3.3-2

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. Power Range, Neutron Flux, High Negative Rate	≤ 0.5 second*
5. Intermediate Range, Neutron Flux	N.A.
6. Source Range, Neutron Flux	N.A.
7. Overtemperature ΔT	≤ ~~4(~~8~~#)~~ seconds*
8. Overpower ΔT	≤ ~~4(~~8~~#)~~ seconds*
9. Pressurizer Pressure-Low	≤ 2 seconds
10. Pressurizer Pressure-High	≤ 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.
~~#Applicable upon deletion of RTD Bypass System.~~

UNIT 2

TABLE 3.3-2 (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.

UNIT 2

TABLE 4.3-1

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	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, [illegible]
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

UNIT 2

TABLE 4.3-1 (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
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 (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*

TABLE 4.3-1 (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) At least once per 18 months and following maintenance or adjustment of the Reactor trip breakers, 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.

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Features Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their Trip Setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3

ACTION:

a. With an ESFAS Instrumentation or Interlock Trip Setpoint trip less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 3.3-4, adjust the Setpoint consistent with the Trip Setpoint value.

b. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Allowable Values Column of Table 3.3-4, ~~either~~:

Delete

1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 3.3-4, and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel, or

~~2.~~ ~~D~~declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3.3 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

Delete

Equation 2.2-1 $Z + R + S \leq TA$

Where:

Z = The value from Column Z of Table 3.3-4 for the affected channel,

R = The "as measured" value (in percent span) of rack error for the affected channel,

S = Either the "as measured" value (in percent span) of the sensor error, or the value from Column S (Sensor Error) of Table 3.3-4 for the affected channel, and

TA = The value from Column TA (Total Allowance) of Table 3.3-4 for the affected channel.

c. With an ESFAS instrumentation channel or interlock inoperable, take the ACTION shown in Table 3.3-3.

UNIT 1

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

Delete

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	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.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure-High	8.2	0.71	1.5	≤ 1.2 psig	≤ 1.4 psig
d. Pressurizer Pressure-Low	16.1	14.4	1.5	≥ 1845 psig	≥ 1839 psig
e. Steam Line Pressure-Low	4.6	1.31	1.5	≥ ~~725~~ psig 775	≥ ~~694~~ psig 744
2. Containment Spray					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure-High-High	12.7	0.71	1.5	≤ 3 psig	≤ 3.2 psig

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

Delete

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	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)	Delete				
1) Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
3) Containment Pressure-High-High	12.7	0.71	1.5	$\leq$ 3 psig	$\leq$ 3.2 psig
c. Purge and Exhaust Isolation					
1) Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.				

UNIT 1

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

Delete

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
4. Steam Line Isolation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure-High-High	12.7	0.71	1.5	≤ 3 psig	≤ 3.2 psig
d. Steam Line Pressure - Low	4.6	1.31	1.5	≥ ~~725~~ 775 psig	≥ ~~694~~ 744 psig*
e. Steam Line Pressure-Negative Rate - High	8.0	0.5	0	≤ 100 psi	≤ 122.8 psi**
5. Feedwater Isolation					
a. Automatic Actuation Logic Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. Steam Generator Water Level-High-High (P-14)					
1. Unit 1	5.4	2.18	1.5	≤ 82.4% of narrow range instrument span	≤ 84.2% of narrow range instrument span
2. Unit 2	21.9	1.7	2.0	≤ 77.1% of narrow range instrument span	≤ 78.9% of narrow range instrument span
c. T_{avg}-Low	4.0(6.0#)	1.12(0.71#)	1.2(0.8#)	≥ 564°F	≥ ~~562°F~~ (561°F*)
d. Doghouse Water Level-High	1.0	0	0.5	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 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

Delete

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
6. Turbine Trip					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Steam Generator Water Level-High-High (P-14)					
1. Unit 1	5.4	2.18	1.5	≤ 82.4% of narrow range instrument span	≤ 84.2% of narrow range instrument span
2. Unit 2	21.9	1.7	2.0	≤ 77.1% of narrow range instrument span	≤ 78.9% of narrow range instrument span
d. Trip of All Main Feedwater Pumps	N.A.	N.A.	N.A.	N.A.	N.A.
e. Reactor Trip (P-4)	N.A.	N.A.	N.A.	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	N.A.	N.A.	N.A.	≤ 0.4 psid	≤ 0.45 psid
b. Termination	N.A.	N.A.	N.A.	≥ 0.3 psid	≥ 0.25 psid
8. Auxiliary Feedwater					
a. Manual Initiation	N.A	N.A.	N.A.	N.A	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.

Delete

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

Delete

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
8. Auxiliary Feedwater (Continued)					
c. Steam Generator Water Level - Low-Low					
1) Unit 1	17	14.2	1.5	≥ 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	11.8	1.7	2.0	≥ 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	N.A.	N.A.	N.A.	≥ 3500 V	≥ 3200 V
f. Trip of All Main Feedwater Pumps	N.A.	N.A.	N.A.	N.A.	N.A.
g. Auxiliary Feedwater Suction Pressure-Low	Delete				
1) CAPS 5220, 5221, 5222	N.A.	N.A.	N.A.	≥ 10.5 psig	≥ 9.5 psig
2) CAPS 5230, 5231, 5232	N.A.	N.A.	N.A.	≥ 6.2 psig	≥ 5.2 psig
a. Unit 1	N.A.	N.A.	N.A.	≥ 6.2 psig	≥ 5.2 psig
b. Unit 2	N.A.	N.A.	N.A.	≥ 6.0 psig	≥ 5.0 psig
9. Containment Sump Recirculation					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. Refueling Water Storage Tank Level-Low	N.A.	N.A.	N.A.	≥ 177.15 inches	≥ 162.4 inches
Coincident With Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.				

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UNIT 1

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

Delete

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
10. Loss of Power					
a. 4 kV Bus Undervoltage-Loss of Voltage	N.A.	N.A.	N.A.	≥ 3500 V	≥ 3200 V
b. 4 kV Bus Undervoltage-Grid Degraded Voltage	N.A.	N.A.	N.A.	≥ 3685 V	≥ 3611 V
11. Control Room Area Ventilation Operation					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. Loss-of-Offsite Power	N.A.	N.A.	N.A.	≥ 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	Delete				
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure-High-High	12.7	0.71	1.5	≤ 3 psig	≤ 3.2 psig

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

Delete

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
13. Annulus Ventilation Operation					
a. Manual Initiation	N.A	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A
c. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.				
14. Nuclear Service Water Operation	Delete				
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Loss-of-Offsite Power	N.A.	N.A.	N.A.	$\geq$ 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	NA	NA	NA	$\geq$El. 554.4 ft.	$\geq$El. 552.9 ft.
15. Emergency Diesel Generator Operation (Diesel Building Ventilation Operation, Nuclear Service Water Operation)	Delete				
a. Manual Initiation	N.A	N.A.	N.A.	N.A.	N.A.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

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

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UNIT 1

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

Delete

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
18. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	N.A.	N.A.	N.A.	1955 psig	$\geq$1944 psig
b. Pressurizer Pressure, not P-11	N.A.	N.A.	N.A.	1955 psig	$\leq$1966 psig
c. Low-Low T_{avg}, P-12	N.A.	N.A.	N.A.	$\geq$553°F	~~$\geq$551°F~~ (550°F*)
d. Reactor Trip, P-4	N.A.	N.A.	N.A.	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 (Continued)

TABLE NOTATIONS

Delete

*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.

~~#Applicable upon deletion of RTD Bypass System.~~

UNIT 2

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

Delete

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	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.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure-High	8.2	0.71	1.5	≤ 1.2 psig	≤ 1.4 psig
d. Pressurizer Pressure-Low	16.1	14.4	1.5	≥ 1845 psig	≥ 1839 psig
e. Steam Line Pressure-Low	4.6	1.31	1.5	≥ 725 psig	≥ 694 psig*
2. Containment Spray					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure-High-High	12.7	0.71	1.5	≤ 3 psig	≤ 3.2 psig

UNIT 2

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

Delete

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	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)	Delete				
1) Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
3) Containment Pressure-High-High	12.7	0.71	1.5	$\leq$ 3 psig	$\leq$ 3.2 psig
c. Purge and Exhaust Isolation					
1) Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.				

UNIT 2

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

Delete

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
4. Steam Line Isolation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure-High-High	12.7	0.71	1.5	≤ 3 psig	≤ 3.2 psig
d. Steam Line Pressure - Low	4.6	1.31	1.5	≥ 725 psig	≥ 694 psig*
e. Steam Line Pressure-Negative Rate - High	8.0	0.5	0	≤ 100 psi	≤ 122.8 psi**
5. Feedwater Isolation					
a. Automatic Actuation Logic Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. Steam Generator Water Level-High-High (P-14)					
1. Unit 1	5.4	2.18	1.5	≤ 82.4% of narrow range instrument span	≤ 84.2% of narrow range instrument span
2. Unit 2	21.9	1.7	2.0	≤ 77.1% of narrow range instrument span	≤ 78.9% of narrow range instrument span
c. T_{avg}-Low	4.0(6.0#)	1.12(0.71#)	1.2(0.8#)	≥ 564°F	≥ ~~562°F~~(561°F*)
d. Doghouse Water Level-High	1.0	0	0.5	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 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

Delete

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
6. Turbine Trip					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Steam Generator Water Level-High-High (P-14)					
1. Unit 1	5.4	2.18	1.5	≤ 82.4% of narrow range instrument span	≤ 84.2% of narrow range instrument span
2. Unit 2	21.9	1.7	2.0	≤ 77.1% of narrow range instrument span	≤ 78.9% of narrow range instrument span
d. Trip of All Main Feedwater Pumps	N.A.	N.A.	N.A.	N.A.	N.A.
e. Reactor Trip (P-4)	N.A.	N.A.	N.A.	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	N.A.	N.A.	N.A.	≤ 0.4 psid	≤ 0.45 psid
b. Termination	N.A.	N.A.	N.A.	≥ 0.3 psid	≥ 0.25 psid
8. Auxiliary Feedwater					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.

Delete

UNIT 2

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

Delete

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
8. Auxiliary Feedwater (Continued)					
c. Steam Generator Water Level - Low-Low					
1) Unit 1	17	14.2	1.5	≥ 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	11.8	1.7	2.0	≥ 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	N.A.	N.A.	N.A.	≥ 3500 V	≥ 3200 V
f. Trip of All Main Feedwater Pumps	N.A.	N.A.	N.A.	N.A.	N.A.
g. Auxiliary Feedwater Suction Pressure-Low	Delete				
1) CAPS 5220, 5221, 5222	N.A.	N.A.	N.A.	≥ 10.5 psig	≥ 9.5 psig
2) CAPS 5230, 5231, 5232	N.A.	N.A.	N.A.	≥ 6.2 psig	≥ 5.2 psig
a. Unit 1	N.A.	N.A.	N.A.	≥ 6.2 psig	≥ 5.2 psig
b. Unit 2	N.A.	N.A.	N.A.	≥ 6.0 psig	≥ 5.0 psig
9. Containment Sump Recirculation					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. Refueling Water Storage Tank Level-Low Coincident With Safety Injection	N.A.	N.A.	N.A.	≥ 177.15 inches	≥ 162.4 inches
	See Item 1. above for all Safety Injection Setpoints and Allowable Values.				

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UNIT 2

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

Delete

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
10. Loss of Power					
a. 4 kV Bus Undervoltage-Loss of Voltage	N.A.	N.A.	N.A.	$\geq$ 3500 V	$\geq$ 3200 V
b. 4 kV Bus Undervoltage-Grid Degraded Voltage	N.A.	N.A.	N.A.	$\geq$ 3685 V	$\geq$ 3611 V
11. Control Room Area Ventilation Operation					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. Loss-of-Offsite Power	N.A.	N.A.	N.A.	$\geq$ 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	Delete				
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure-High-High	12.7	0.71	1.5	$\leq$ 3 psig	$\leq$ 3.2 psig

UNIT 2

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

Delete

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
13. Annulus Ventilation Operation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.				
14. Nuclear Service Water Operation	Delete				
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Loss-of-Offsite Power	N.A.	N.A.	N.A.	≥ 3500 V	≥ 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	NA	NA	NA	≥El. 554.4 ft.	≥El. 552.9 ft.
15. Emergency Diesel Generator Operation (Diesel Building Ventilation Operation, Nuclear Service Water Operation)	Delete				
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.

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UNIT 2

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	Delete TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
Emergency Diesel Generator Operation (Diesel Building Ventilation Operation, Nuclear Service Water Operation) (Continued)					
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Loss-of-Offsite Power	N.A.	N.A.	N.A.	$\geq$ 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	Delete				
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.				
17. Diesel Building Ventilation Operation	Delete				
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Emergency Diesel Generator Operation	See Item 15. above for all Emergency Diesel Generator Operation Setpoints and Allowable Values.				

UNIT 2

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

Delete

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
18. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	N.A.	N.A.	N.A.	1955 psig	$\geq$1944 psig
b. Pressurizer Pressure, not P-11	N.A.	N.A.	N.A.	1955 psig	$\leq$1966 psig
c. Low-Low T_{avg}, P-12	N.A.	N.A.	N.A.	$\geq$553°F	$\geq$~~551°F~~(550°F*)
d. Reactor Trip, P-4	N.A.	N.A.	N.A.	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.				

UNIT 2

TABLE 3.3-4 (Continued)

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.

~~#Applicable upon deletion of RTD Bypass System.~~

UNIT 1

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATION SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual Initiation	
a. Safety jection (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	$\leq$ ~~7~~ 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.

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UNIT 1

TABLE 3.3-5 (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~~ 12
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~~ 12
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

UNIT 1

TABLE 3.3-5 (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	≤ 23
12) Auxiliary Building Filtered Exhaust Isolation	N.A.
13) Containment Sump Recirculation	N.A.
b. Steam Line Isolation	≤ 10
5. Containment Pressure-High-High	
a. Containment Spray	≤ 45
b. Phase "B" Isolation	≤ $65^{(3)}/76^{(4)}$
Nuclear Service Water Operation	N.A.
c. Steam Line Isolation	≤ 10
d. Containment Air Return and Hydrogen Skimmer Operation	≤ 600
6. Steam Line Pressure - Negative Rate-High	
Steam Line Isolation	≤ 10
7. Steam Generator Water Level-High-High	
a. Turbine Trip	≤ 3
b. Feedwater Isolation	≤ 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.

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UNIT 1

TABLE 3.3-5 (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.

UNIT 2

TABLE 3.3-5

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)	≤ 27(1)/12(3)
1) Reactor Trip	≤ 2
2) Feedwater Isolation	≤ 7
3) Phase "A" Isolation(2)	≤ 18(3)/28(4)
4) Purge and Exhaust Isolation	≤ 6
5) Auxiliary Feedwater(5)	N.A.
6) Nuclear Service Water Operation	≤ 65(3)/76(4)
7) Turbine Trip	N.A.
8) Component Cooling Water	≤ 65(3)/76(4)
9) Emergency Diesel Generator Operation	≤ 11
10) Control Room Area Ventilation Operation	N.A.

UNIT 2

TABLE 3.3-5 (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

UNIT 2

TABLE 3.3-5 (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 (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.

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least three of the reactor coolant loops listed below shall be OPERABLE and ~~at least two of these reactor coolant loops shall be~~ in operation:*

a. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,

b. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,

c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump, and

d. Reactor Coolant Loop D and its associated steam generator and reactor coolant pump.

APPLICABILITY: MODE 3.

ACTION:

a. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With less than the above required ~~only one~~ reactor coolant loop in operation, restore the required ~~at least two~~ loops to operation within 72 hours or open the Reactor Trip System breakers.

c. With no reactor coolant loops in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required reactor coolant loops to operation

SURVEILLANCE REQUIREMENTS

Delete 4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.~~2~~1 The required steam generators shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 12% at least once per 12 hours.

4.4.1.2.~~3~~2 At least ~~two~~ 3 reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

---

*All reactor coolant pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

for unit 2

3.4.2.1 A minimum of one pressurizer Code safety valve shall be OPERABLE with a lift setting of 2485 psig ± 1%.*

+3%, -2% for unit 1 and

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

for unit 2

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

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.

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. ~~1~~ 0.5 gpm total reactor-to-secondary leakage through all steam generators and ~~500~~ 200 gallons per day through any one steam generator,

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. 40 gpm CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig, and

f. 1 gpm leakage at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.

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

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3/4.5.1 ACCUMULATORS

COLD LEG INJECTION

LIMITING CONDITION FOR OPERATION

3.5.1 Each cold leg injection accumulator shall be OPERABLE with:

a. The discharge isolation valve open,

b. A contained borated water volume of between 7704 and 8004 gallons,

c. A boron concentration of between 1900 and 2100 ppm,

d. A nitrogen cover-pressure of between 585 and 678 psig, and

e. A water level and pressure channel OPERABLE.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one cold leg injection accumulator inoperable, except as a result of a closed isolation valve or boron concentration less than 1900 ppm, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one cold leg injection accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With one accumulator inoperable due to boron concentration less than 1900 ppm and:

   1) The volume weighted average boron concentration of the ~~three limiting~~ accumulators 1900 ppm or greater, restore the inoperable accumulator to OPERABLE status within 24 hours of the low boron determination or be in at least HOT STANDBY within the next 6 hours and reduce Reactor Coolant System pressure to less than 1000 psig within the following 6 hours.

   2) The volume weighted average boron concentration of the ~~three limiting~~ accumulators less than 1900 ppm but greater than 1800 ~~1500~~ ppm, restore the inoperable accumulator to OPERABLE status or return the volume weighted average boron concentration of the three limiting accumulators to greater than 1900 ppm and

*Reactor Coolant System pressure above 1000 psig.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

enter ACTION c.1 within 6 hours of the low boron determination or be in HOT STANDBY within the next 6 hours and reduce Reactor Coolant System pressure to less than 1000 psig within the following 6 hours.

3) The volume weighted average boron concentration of the ~~three limiting~~ accumulators ~~1500~~ 1800 ppm or less, return the volume weighted average boron concentration of the three limiting accumulators to greater than ~~1500~~ ppm and enter ACTION c.2 within 1 hour of the low boron determination or be in HOT STANDBY within the next 6 hours and reduce Reactor Coolant System pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each cold leg injection accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying, by the absence of alarms, the contained borated water volume and nitrogen cover-pressure in the tanks, and

2) Verifying that each cold leg injection accumulator isolation valve is open.

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 75 gallons by verifying the boron concentration of the accumulator solution;

c. At least once per 31 days when the Reactor Coolant System pressure is above 2000 psig by verifying that power is removed from the isolation valve operators on Valves NI54A, NI65B, NI76A, and NI88B and that the respective circuit breakers are padlocked; and

d. At least once per 18 months by verifying that each cold leg injection accumulator isolation valve opens automatically under each of the following conditions:**

1) When an actual or a simulated Reactor Coolant System pressure signal exceeds the P-11 (Pressurizer Pressure Block of Safety Injection) Setpoint, and

2) Upon receipt of a Safety Injection test signal.

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

b) With a simulated or actual Reactor Coolant System pressure signal less than or equal to 660 psig the interlocks will cause the valves to automatically close.

2) A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or abnormal corrosion.

e. At least once per 18 months, during shutdown, by:**

1) Verifying that each automatic valve in the flow path actuates to its correct position on Safety Injection and Containment Sump Recirculation test signals, and

2) Verifying that each of the following pumps start automatically upon receipt of a Safety Injection test signal:

a) Centrifugal charging pump,

b) Safety Injection pump, and

c) Residual heat removal pump.

f. By verifying that each of the following pumps develops the indicated differential pressure when tested pursuant to Specification 4.0.5:

1) Centrifugal charging pump $\geq$ ~~2380~~ 2223 psid,

2) Safety Injection pump $\geq$ ~~1430~~ 1341 psid, and

3) Residual heat removal pump $\geq$ 165 psid.

g. By verifying the correct position of each electrical and/or mechanical stop for the following ECCS throttle valves:

1) Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE, and

2) At least once per 18 months.

Centrifugal Charging Pump Injection Throttle Valve Number	Safety Injection Throttle Valve Number
NI-14	NI-164
NI-16	NI-166
NI-18	NI-168
NI-20	NI-170

** This surveillance need not be performed until prior to entering HOT SHUTDOWN following the Unit One first refueling.

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying that:

1) For centrifugal charging pump lines, with a single pump running:

a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 345 gpm, and

b) The total pump flow rate is less than or equal to ~~565~~ 560 gpm.

2) For Safety Injection pump lines, with a single pump running:

a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 450 gpm, and

b) The total pump flow rate is less than or equal to ~~660~~ 675 gpm.

3) For residual heat removal pump lines, with a single pump running, the sum of the injection line flow rates is greater than or equal to 3648 gpm.

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
CF-89#	Steam Generator 1B Feedwater Purge Valve	≤5
CF-88#	Steam Generator 1C Feedwater Purge Valve	≤5
CF-87#	Steam Generator 1D Feedwater Purge Valve	≤5

TABLE 3.6-2a (Continued)

UNIT 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

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

8 seconds for unit 1 and

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

for unit 2

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;

$F^N_{\Delta H}(X,Y)$ 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.

$F\Delta H(X,Y)$ will be maintained within its limits provided Conditions a. through d. above are maintained.

The limits on the nuclear enthalpy rise hot channel factor, $F\Delta H(X,Y)$, 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 $F\Delta H(X,Y)$ value of 1.50 1.55 and a limiting reference axial power shape. The Mark-BW MAP limits may be applied to OFA fuel, provided an appropriate adjustment factor is applied to provide equivalence to a 1.49 design $F\Delta H(X,Y)$, for the OFA. This is reflected in the MAP limits specified in the COLR. Replace w ATTACHED The relaxation of $F\Delta H(X,Y)$, 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 $f(\Delta I)$ limits and for reducing THERMAL POWER.

for Power Distribution Limits Bases

Attachment 1:

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.

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

from a full core map

When an $FQ^M(X,Y,Z)$ measurement is obtained in accordance with the surveillance requirements of Specification 4.2.2, no uncertainties are applied to the measured peak the required uncertainties are included in the peaking limit.

delete space Since

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 ~~to convert it to a local peak.~~ and allowances of 5% for measurement uncertainty and 3% for manufacturing tolerances. ~~are then applied to the measured peak.~~

from a full core map

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

Since

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. The two sets of four symmetric thimbles is a unique set of eight detector locations.

delete

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

BASES

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, (X,Y) 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.

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Trip System and the Engineered Safety Features Actuation System instrumentation and interlocks ensures that: (1) the associated ACTION and/or Reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic and sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance, consistent with maintaining an appropriate level of reliability of the Reactor Protection and Engineered Safety Features instrumentation and (3) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses. The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System", and supplements to that report. Surveillance intervals and out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation. (Implementation of quarterly testing of RTS is being postponed until after approval of a similar testing interval for ESFAS). The NRC Safety Evaluation Report for WCAP-10271 was provided in a letter dated February 21, 1985 from C. O. Thomas (NRC) to J. J. Sheppard (WOG-CP&L).

The Engineered Safety Features Actuation System Instrumentation Trip Setpoints specified in Table 3.3-4 are the nominal values at which the bistables are set for each functional unit. A Setpoint is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Setpoints have been specified in Table 3.3-4. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its Trip Setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and

Delete

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

Delete

sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 3.3-1, $Z + R + S \le TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 3.3-4, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, R or Rack Error is the "as measured" deviation, in the percent span, for the affected channel from the specified Trip Setpoint. S or Sensor Error is either the "as measured" deviation of the sensor from its calibration point or the value specified in Table 3.3-4, in percent span, from the analysis assumptions. Use of Equation 3.3-1 allows for a sensor drift factor, an increased rack drift factor, and provides a threshold value for REPORTABLE EVENTS.

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensor and rack instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

The measurement of response time at the specified frequencies provides assurance that the Reactor trip and the Engineered Safety Features actuation associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either: (1) in place, onsite, or offsite test measurements, or (2) utilizing replacement sensors with certified response time.

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents, events, and transients. Once the required logic combination is completed, the system sends actuation signals to those Engineered Safety Features components whose aggregate function best serves the requirements of the condition. As an example, the following actions may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss-of-coolant accident: (1) Safety Injection pumps start and automatic valves position, (2) Reactor trip, (3) feedwater isolation, (4) startup of the emergency diesel generators, (5) containment

3/4.4.6.2 OPERATIONAL LEAKAGE

1

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

Industry experience has shown that while a limited amount of leakage is expected from the Reactor Coolant System, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

Replace with Attachment 1

The total steam generator tube leakage limit of 1 gpm for all steam generators not isolated from the Reactor Coolant System ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the Leakage Detection Systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 40 gpm with the modulating valve in the supply line fully open at a nominal Reactor Coolant System pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the safety analyses.

The 1 gpm leakage from any Reactor Coolant System pressure isolation valve is sufficiently low to ensure early detection of possible in-series check valve failure. It is apparent that when pressure isolation is provided by two in-series check valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA that bypasses containment, these valves should be tested periodically to ensure low probability of gross failure.

The Surveillance Requirements for Reactor Coolant System pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the pressure isolation valve is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

Attachment 1:

The total steam generator tube leakage limits of 0.5 gpm for all steam generators and 200 gpd maximum leakage into one steam generator ensures that the dosage contribution from the tube leakage will be limited to the applicable fraction of 10 CFR Part 100 dose guideline values in the event of any FSAR Chapter 15 transient or accident. The 0.5 gpm and 200 gpd limits bound the assumptions used in the analysis of these accidents. The 200 gpd leakage limit per steam generator also ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

MONTHLY OPERATING REPORTS

6.9.1.8 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or safety valves, shall be submitted on a monthly basis to the NRC in accordance with 10 CFR 50.4, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.9.1.9 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,
2. Shutdown Bank Insertion Limit for Specification 3/4.1.3.5,
3. Control Bank Insertion Limits for Specification 3/4.1.3.6,
4. Axial Flux Difference Limits, target band*, and APL^{ND*} for Specification 3/4.2.1,
5. Heat ~~Flux~~ Hot Channel Factor, F_Q^{RTP}, K(Z), W(Z)**, APL^{ND**}, ~~and~~ $F_Q^L(X,Y,Z)$ and $W(Z)_{BL}$** for Specification 3/4.2.2, and
6. Nuclear Enthalpy Rise Hot Channel Factor, $F\Delta H^L(X,Y)$***, $F_{\Delta H}^{RTP}$****, and Power Factor Multiplier, $MF\Delta H$****, limits for Specification 3/4.2.3.

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC in:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary).

 (Methodology for Specifications 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux

*Reference ~~5 is not~~ applicable to target band and APL^{ND}.

**References ~~4~~ 4 and ~~5~~ 5 are not applicable to W(Z), and APL^{ND}, and $W(Z)_{BL}$.

***Reference 1 is not applicable to $F\Delta H^L$.

****Reference 5 is not applicable to $F_{\Delta H}^{RTP}$ and $MF_{\Delta H}$.

CORE OPERATING LIMITS REPORT (Continued)

Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

2. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION," June 1983 (W Proprietary).

   (Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor W(Z) surveillance requirements for F_Q Methodology.)

3. WCAP-10266-P-A Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE," March 1987, (W Proprietary).

   (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

4. BAW-10152-A, "NOODLE - A Multi-Dimensional Two-Group Reactor Simulator," June 1985.

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient.)

delete replace w/ ATTACHMENT 1

5. BAW-10163P-A, "Core Operating Limit Methodology for Westinghouse-Designed PWR's," June 1989.

   (Methodology for Specifications 3.1.3.5 - Shutdown Rod Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

6. BAW-10168P, Rev. 1, "B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants." September, 1989.
   (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

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.

Attachment 1:

4. BAW-10168PA, Rev. 1, "B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," January 1991 (B&W Proprietary).

   (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

5. DPC-NE-2011P-A, "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors," March, 1990 (DPC Proprietary).

   (Methodology for Specification 3.1.3.5 - Shutdown Rod Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

6. DPC-NE-3001P-A, "Multidimensional Reactor Transients and Safety Analysis Physics Parameter Methodology," November 1991 (DPC Proprietary).

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient , 3.1.3.5 - Shutdown Rod Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

7. DPC-NF-2010P-A, "Duke Power Company McGuire Nuclear Station Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," June 1985 (DPC Proprietary).

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient.)

8. DPC-NE-3002A,"FSAR Chapter 15 System Transient Analysis Methodology," November 1991.

   (Methodology used in the system thermal-hydraulic analyses which determine the core operating limits)

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

   (Modeling used in the system thermal-hydraulic analyses)

8.2 Changes to Core Operating Limits Report

Catawba 1 Cycle 7 Core Operating Limits Report

1.0 Core Operating Limits Report

This Core Operating Limits Report (COLR) for Catawba Unit 1 Cycle 7 has been prepared in accordance with the requirements of Technical Specification 6.9.1.9.

The technical specifications affected by this report are listed below:

3/4.1.1.3	Moderator Temperature Coefficient
3/4.1.3.5	Shutdown Rod Insertion Limit
3/4.1.3.6	Control Rod Insertion Limit
3/4.2.1	Axial Flux Difference
3/4.2.2	Heat Flux Hot Channel Factor
3/4.2.3	Nuclear Enthalpy Rise Hot Channel Factor

2.2 Shutdown Rod Insertion Limit (Specification 3/4.1.3.5)

2.2.1 The shutdown rods shall be withdrawn to at least 222 steps.

2.3 Control Rod Insertion Limits (Specification 3/4.1.3.6)

2.3.1 The control rod banks shall be limited in physical insertion as shown in Figure 2.

2.4 Axial Flux Difference (Specification 3/4.2.1)

2.4.1 The AXIAL FLUX DIFFERENCE (AFD) Limits are provided in Figure 3.

$(\text{AFD Limit})^{\text{COLR}}_{\text{Negative}}$ is the negative AFD limit from Figure 3.

$(\text{AFD Limit})^{\text{COLR}}_{\text{Positive}}$ is the positive AFD limit from Figure 3.

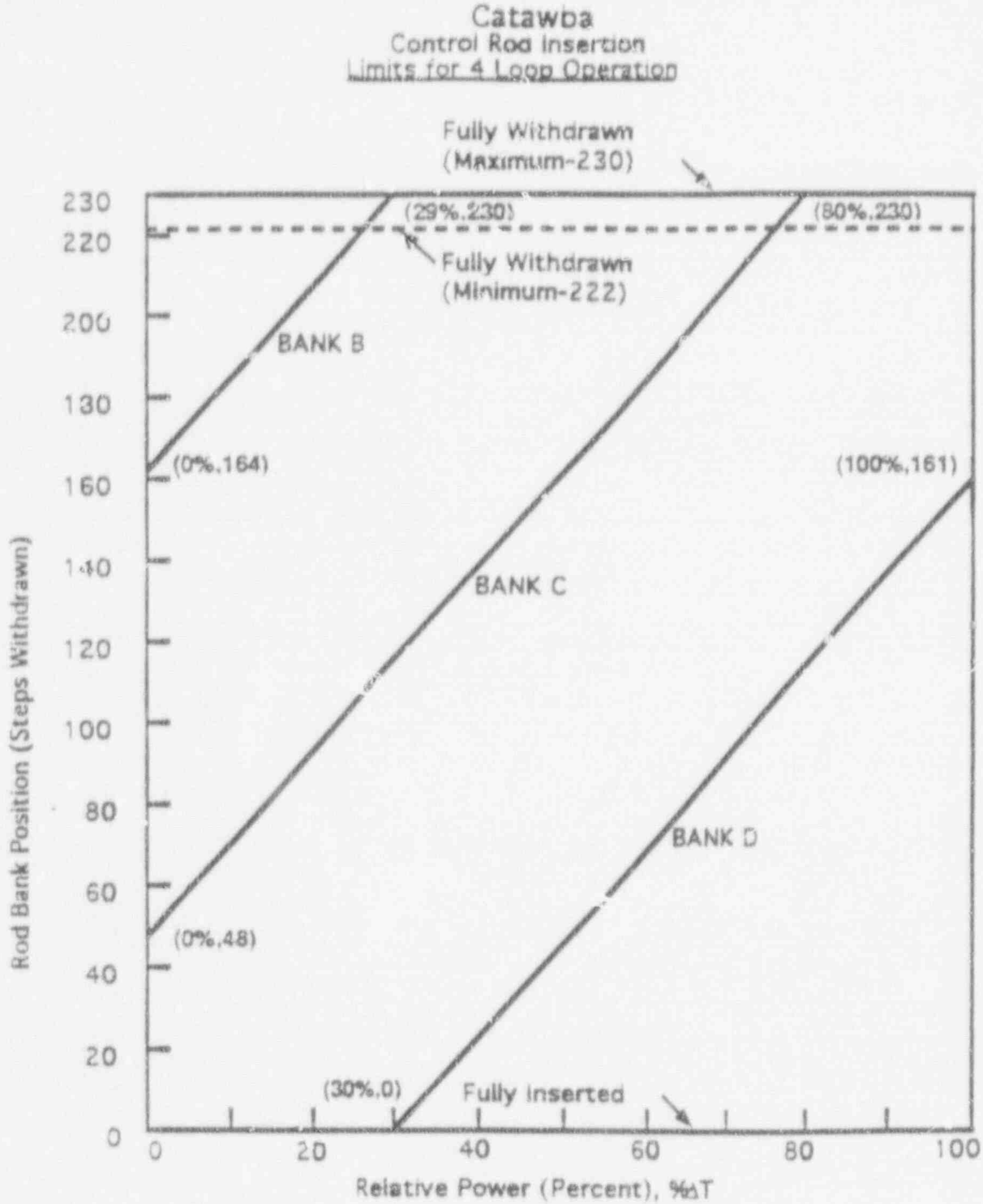

Figure 2

Control Rod Bank Insertion Limits Versus Percent Rated Thermal Power

Bank insertion limits are to be set as a function of reactor power, as measured by delta T.

*100%ΔT = indicated ΔT Normal Full Power Conditions

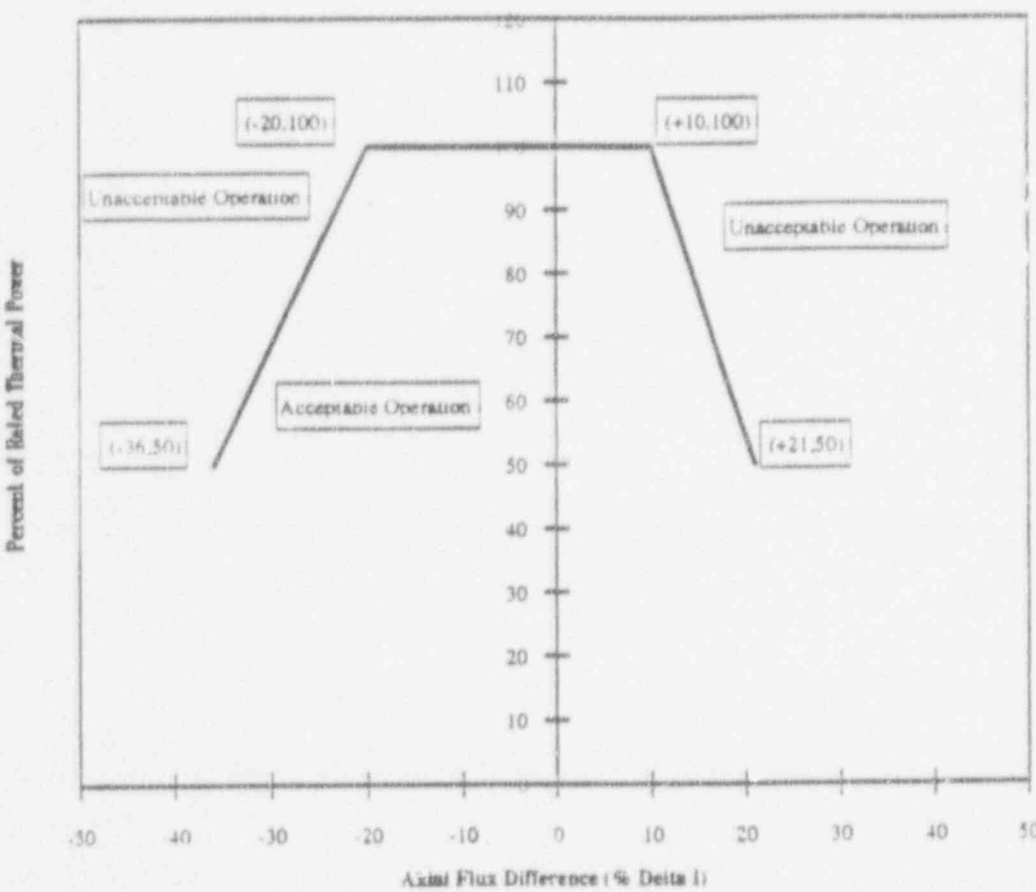

Figure 3

Percent of Rated Thermal Power Versus Axial Flux Difference Limits

2.5 Heat Flux Hot Channel Factor - $F_Q(X,Y,Z)$ (Specification 3/4.2.2)

2.5.1 F_Q^{RTP} = 2.32

2.5.2 K(Z) is provided in Figure 4 for Mark-BW fuel.

2.5.3 K(Z) is provided in Figure 5 for OFA fuel.

The following parameters are required for core monitoring per the Surveillance Requirements of Specification 3/4.2.2:

2.5.4 $(F_Q^L(X,Y,Z))^{OP} = F_Q^D(X,Y,Z) * M_Q(X,Y,Z)/(UMT*MT*TILT)$

where $(F_Q^L(X,Y,Z))^{OP}$ = cycle dependent maximum allowable design peaking factor which ensures that the $(F_Q^L(X,Y,Z))$ limit will be preserved for operation within the LCO limits. $(F_Q^L(X,Y,Z))^{OP}$ includes allowances for calculational and measurement uncertainties.

$F_Q^D(X,Y,Z)$ = the design power distribution for F_Q. $F_Q^D(X,Y,Z)$ is provided in Table 1.

$M_Q(X,Y,Z)$ = the margin remaining in core location X,Y,Z to the LOCA limit in the transient power distribution. $M_Q(X,Y,Z)$ is provided in Table 2.

Note: $(F_Q^L(X,Y,Z))^{OP}$ is the parameter identified as $F_Q^{MAX}(X,Y,Z)$ in DPC-NE-2011PA.

2.5.5 $(F_Q^L(X,Y,Z))^{RPS} = F_Q^D(X,Y,Z) * (M_C(X,Y,Z)/(UMT*MT*TILT))$

where $(F_Q^L(X,Y,Z))^{RPS}$ = cycle dependent maximum allowable design peaking factor which ensures that the centerline fuel melt limit will be preserved for operation within the LCO limits. $(F_Q^L(X,Y,Z))^{RPS}$ includes allowances for calculational and measurement uncertainties.

$F_Q^D(X,Y,Z)$ = the design power distribution for F_Q. $F_Q^D(X,Y,Z)$ is provided in Table 1.

$M_C(X,Y,Z)$ = the margin remaining in core location X,Y,Z to the CFM limit in the transient power distribution. $M_C(X,Y,Z)$ calculations parallel the $M_Q(X,Y,Z)$ calculation described in DPC-NE-2011PA, except that the LOCA limit is replaced with the CFM limit. $M_C(X,Y,Z)$ is provided in Table 3.

UMT = Measurement Uncertainty (UMT = 1.05).

MT = Engineering hot channel factor (MT = 1.03).

TILT = Peaking penalty that accounts for allowable quadrant power tilt ratio of 1.02.

2.5.6 KSLOPE = 0.078*

where KSLOPE = Adjustm[illegible] to the K_1 value from OTΔT requir[illegible]d to [illegible] for each 1% that $F_Q(X,Y,$[illegible]) [illegible] its limit.

* typical value; actual values will be [illegible]ed when monitoring inputs are computed.

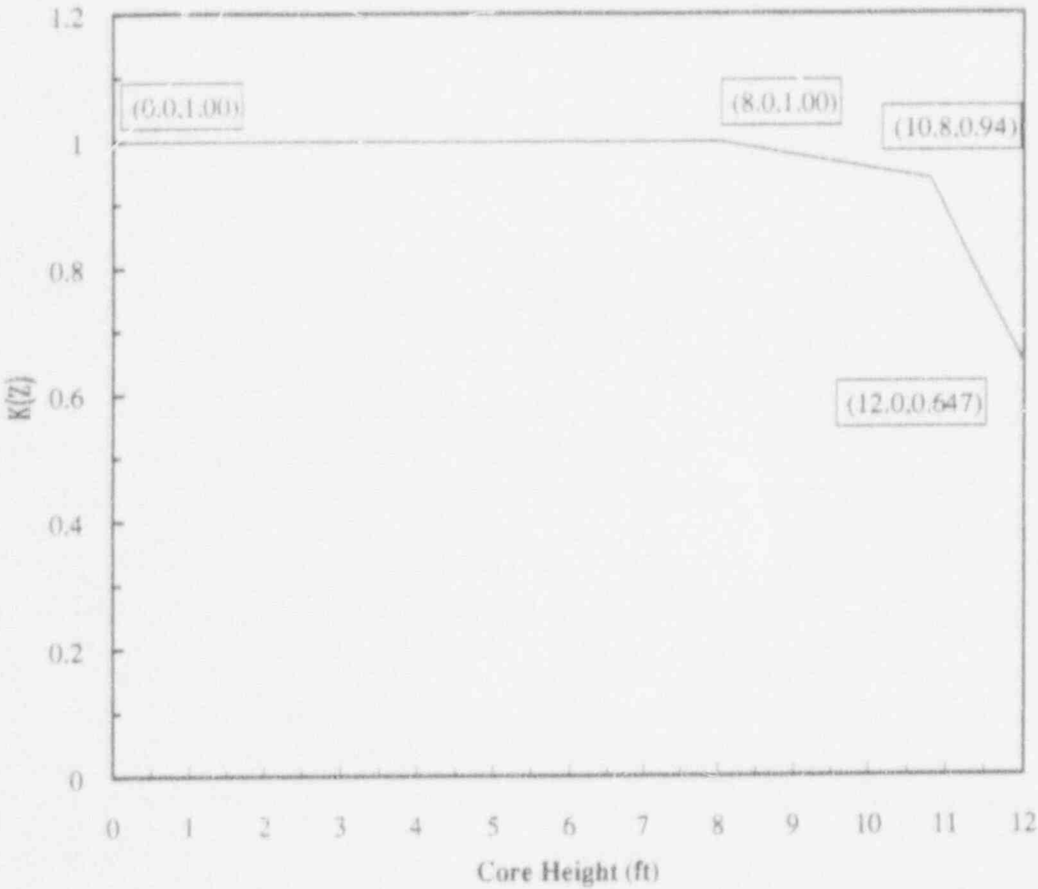

Figure 4

K(Z), Normalized $F_Q(X,Y,Z)$ as a Function of Core Height for MkBW Fuel

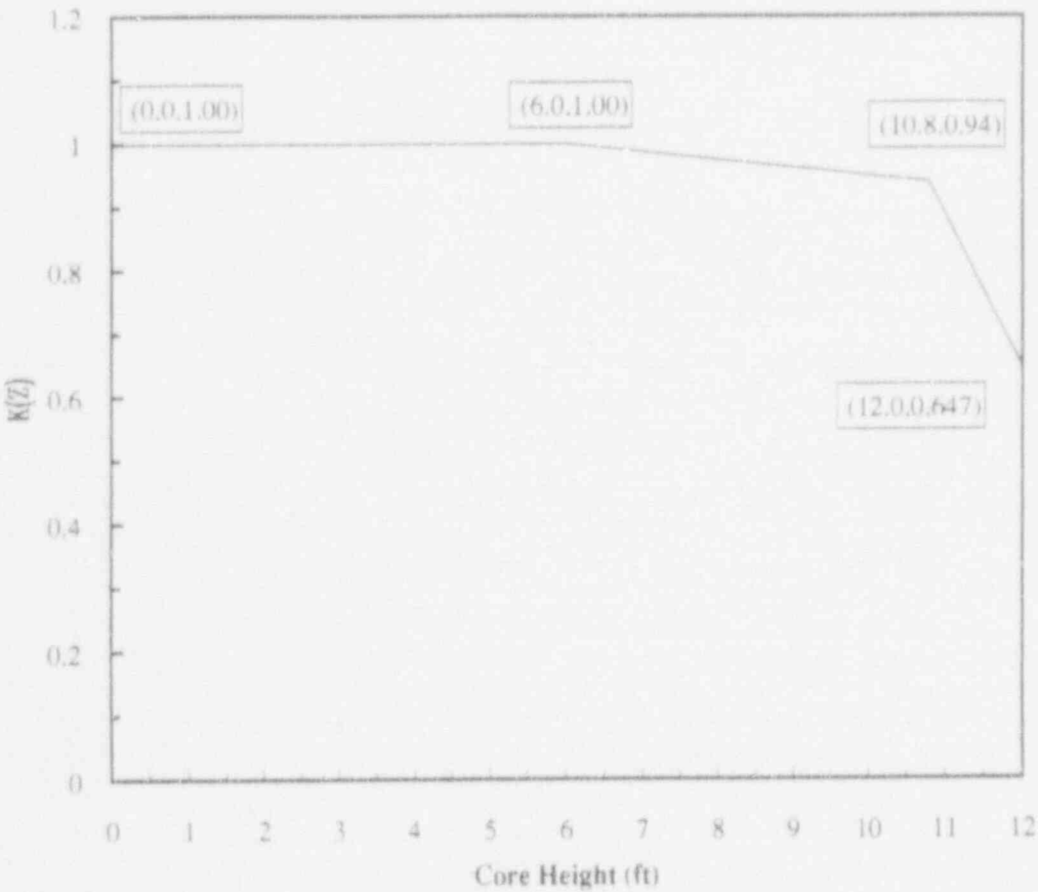

Figure 5

K(Z), Normalized $F_Q(X,Y,Z)$ as a Function of Core Height for OFA Fuel

2.6 <u>Nuclear Enthalpy Rise Hot Channel Factor</u> - $F_{\Delta H}(X,Y)$

$(F^L_{\Delta H}(X,Y))^{LCO}$ = MARP(X,Y)*[1.0 + (1/RRH)*(1.0 - P)]

2.6.1 Catawba 1 Cycle 7 Operating Limit Maximum Allowable Radial Peaks (MARP(X,Y)) are provided in Table 4.

The following parameters are required for core monitoring per the surveillance requirements of 3/4.2.3:

$(F^L_{\Delta H}(X,Y))^{SURV}$ = $F^D_{\Delta H}(X,Y)$ * $M_{\Delta H}(X,Y)$/(UMR*TILT)

Note: $(F^L_{\Delta H}(X,Y))^{SURV}$ is the parameter identified as $F^{MAX}_{\Delta H}(X,Y)$ in DPC-NE-2011PA.

where

UMR = Uncertainty value for measured radial peaks, (UMR=1.04).

TILT = Peaking penalty that accounts for allowable quadrant power tilt ratio of 1.02.

2.6.2 $F^D_{\Delta H}(X,Y)$ is provided in Table 5.

2.6.3 $M_{\Delta H}(X,Y)$ is provided in Table 6.

2.6.4 RRH = 3.34

where RRH = Thermal Power reduction required to compensate for each 1% that $F_{\Delta H}(X,Y)$ exceeds its limit.

$$P = \frac{\text{Thermal Power}}{\text{Rated Thermal Power}}$$

2.6.5 TRH = 0.04

where TRH = Reduction in OTΔT K_1 setpoint required to compensate for each 1% that $F_{\Delta H}(X,Y)$ exceeds its limit.

NOTE: Tables 5 and 6 will be supplied when monitoring inputs are computed.

Table 4. Maximum Allowable Radial Peak (MARP) Values*

Elevation (ft)	1.1 Axial Peak MARP	1.2 Axial Peak MARP	1.2 Axial Peak MARP	1.4 Axial Peak MARP
.13	1.5809	1.6266	1.6722	1.7113
1.2	1.5806	1.6259	1.6677	1.7085
2.4	1.5836	1.6265	1.6663	1.7025
3.6	1.5859	1.6263	1.6635	1.6960
4.8	1.5871	1.6240	1.6571	1.6751
6.0	1.5878	1.6196	1.6470	1.6303
7.2	1.5864	1.6130	1.6265	1.5848
8.4	1.5781	1.5956	1.5773	1.5327
9.6	1.5655	1.5612	1.5208	1.4815
10.8	1.5459	1.5152	1.4717	1.4292
12.0	1.5133	1.4693	1.4274	1.3878

Elevation (ft)	1.5 Axial Peak MARP	1.6 Axial Peak MARP	1.7 Axial Peak MARP	1.8 Axial Peak MARP
.12	1.7477	1.7331	1.7054	1.6438
1.2	1.7433	1.7029	1.6789	1.6193
2.4	1.7126	1.6616	1.6433	1.5869
3.6	1.6735	1.6211	1.6011	1.5504
4.8	1.6313	1.5811	1.5622	1.5121
6.0	1.5868	1.5415	1.5238	1.4763
7.2	1.5379	1.4913	1.4766	1.4344
8.4	1.4886	1.4450	1.4296	1.3880
9.6	1.4399	1.4013	1.3882	1.3490
10.8	1.3887	1.3526	1.3433	1.3081
12.0	1.3500	1.3140	1.3078	1.2749

Elevation (ft)	1.9 Axial Peak MARP	2.1 Axial Peak MARP
.12	1.5839	1.5401
1.2	1.5624	1.5154
2.4	1.5328	1.4801
3.6	1.5013	1.4395
4.8	1.4626	1.4030
6.0	1.4291	1.3619
7.2	1.3920	1.3271
8.4	1.3485	1.2824
9.6	1.3126	1.2501
10.8	1.2726	1.2091
12.0	1.2443	1.1890

* Typical values, to be completed when C1C07 maneuvering analysis is complete.

8.3 Changes to the Final Safety Analysis Report

0 Table 15-22 (Page 1 of 2). Parameters for Postulated Locked Rotor Offsite Dose Analysis

				Conservative
	1.	Data and and assumptions used to estimate radioactive source from postulated accident		
		a.	Power Level (MWt)	3565
		b.	Percent of fuel defected	~~1~~ 15
		c.	Total steam generator tube leak rate during accident and initial 8 hours	~~1~~ 0.5 gpm
0		d.	Activity released to reactor coolant from failed fuel	10% of gap inventory
		e.	Offsite Power	Not available
		f.	Reactor coolant activity prior to accident	Primary and Secondary Activity During Normal Operations (Table Table 11-4 on page 11-11)
	2.	Data and assumptions used to estimate activity released		
0		a.	Iodine partition factor	0.01
		b.	~~Initial~~ Steam release ~~from 4 steam generators~~ - ISOLATED S/G - UNISOLATED S/Gs	~~615,247 lb (0-2 hr)~~ 100,000 LBM (0-15 MINUTES) ~~1,040,910 lb (2-8 hr)~~ 1,200,000 LBM (0 - 4.58 HRS)
		~~c.~~	~~Duration of plant cooldown by secondary system after accident, (hrs)~~	~~8~~
	3.	Dispersion data		
		a.	Distance to exclusion area boundary (m)	762
		b.	Distance to low population zone (m)	6096
		c.	χ/Q at exclusion area boundary (sec/m^3)	~~5.5E-04~~ 4.78E-4
		d.	χ/Q at low population zone (sec/m^3)	~~1.8E-05~~ 6.85E-5
	4.	Dose data		
		a.	Method of dose calculations	SRP 15.3.3-15.3.4 AND Regulatory Guide 1.4
		b.	Dose conversion assumptions	ICRP-30 AND Regulatory Guide ~~and~~ 1.109
		Case 1 (No iodine spike)		
		Exclusion area boundary		
0		Whole body		~~4.4E-01~~ 2.53E-1
0		Thyroid		~~3.6~~ 25.5
		Low population zone		
0		Whole body		~~3.2E-02~~ 6.1E-2
0		Thyroid		~~1.2~~ 6.74

0 **Table 15-22 (Page 2 of 2). Parameters for Postulated Locked Rotor Offsite Dose Analysis**

		Conservative
	Case 2 (With preexisting iodine spike)	
	Exclusion area boundary	
0	Whole body	~~4.4E-01~~ 2.53E-1
0	Thyroid	~~3.7~~ 25.71
	Low population zone	
0	Whole body	~~2.2E-02~~ 6.1E-2
0	Thyroid	~~1.2~~ 6.74
0		
0		

0 Table 15-26 (Page 1 of 2). Parameters for Postulated Rod Ejection Offsite Dose Analysis

			Conservative	Realistic
	1.	Data and assumptions used to estimate radioactive source from postulated accidents		
	a.	Power level (MWt)	3565.	3565.
	~~b.~~	~~Percent of fuel defected~~	~~1.~~	~~0.12~~
	c.	Steam generator tube leak rate prior to ~~and during steam dump (gpm)~~ ACCIDENT	~~1.~~ 0.5	~~0.000~~
	d.	Failed fuel	~~10~~ 50 percent of fuel rods in core	same
	e.	Activity released to reactor coolant from failed fuel and available for release		
		Noble gases	10 percent of core gap inventory	same
		Iodines	10 percent of core gap inventory	same
	~~f.~~	~~Melted fuel~~	~~0.25 percent of core~~	~~0.~~
	~~g.~~	~~Activity released to reactor coolant from melted fuel and available for release to containment~~		
		~~Noble gases~~	~~0.25 percent of core inventory~~	~~0.~~
		~~Iodines~~	~~0.125 percent of core inventory~~	~~0.~~
	h.	Iodine Fractions (organic, elemental, and particulate)	Regulatory Guide 1.4	same
	2.	Data and assumptions used to estimate activity released		
	a.	Containment Free volume (ft^3)	1.015E + 06	same
	b.	Containment leak rate	0.3 percent of containment volume per day, 0≤t≤24 hr	0.05 percent of containment volume per day, 0≤t≤24 hr
			0.15 percent of containment volume per day, t > 24 hr	0.025 percent of containment volume per day, t > 24 hr
	c.	Bypass leakage fraction	0.07	0.07
0	~~d.~~	~~Iodine partition factor for steam release~~	[illegible]	
	e.	Offsite power	Lost	.

0 Table 15-26 (Page 2 of 2). Parameters for Postulated Rod Ejection Offsite Dose Analysis

INSERT 15]

Rev	No.	Parameter	Conservative	Realistic
	f.	~~Steam dump from relief valves (lb)~~	~~44500.~~	~~-~~
	g.	~~Duration of dump from relief valves (sec)~~	~~120.~~	~~-~~
	3.	Dispersion data		
	a.	Distance to exclusion area boundary (m)	762.	762.
	b.	Distance to low population zone (m)	6096.	6096.
	c.	χ/Q at exclusion area boundary (sec/m^3)		
		0-2 hrs	~~5.5E-04~~ 4.78E-4	~~1.3E-04~~
	d.	χ/Q at low population zone (sec/m^3)		
		0-8 hrs	~~1.0E-05~~ 6.85E-5	~~6.2E-06~~
		~~8-24 hrs~~	~~1.2E-05~~	~~5.4E-06~~
		~~1-4 days~~	~~4.3E-06~~	~~2.5E-06~~
		~~4+ days~~	~~1.2E-06~~	~~9.7E-07~~
	4.	Dose data		
	a.	Method of dose calculation	Regulatory Guide 1.77	same
	b.	Dose conversion assumptions	~~Regulatory Guides 1.4~~ ICRP-30 and 1.109	same
	c.	Doses (Rem)		
		Primary side		
		Exclusive area boundary		
0		Whole body	~~6.9E-2~~ 2.4E-1	
0		Thyroid	~~1.5~~ 3.6	
		Low population zone		
0		Whole body	~~8.1E-3~~ 1.08E-1	
0		Thyroid	~~1.7E-1~~ 1.98	
		Secondary side		
		Exclusion area boundary		
0		Whole body	~~2.5E-2~~ 3.99E-2	
0		Thyroid	~~3.5E-2~~ 14.8	
		Low population zone		
0		Whole body	~~7.8E-4~~ 5.68E-3	
0		Thyroid	~~3.9E-4~~ 2.12	

NEXT PAGE

Form 00184 (R4-88)

DUKE POWER COMPANY

Station ______ Unit ______ Rev. ______ File No. ______ Sheet ______ Of ______

Subject ______

______ By ______ Date ______

Prob No. ______ Checked By ______ Date ______

f. DURATION OF TUBE BUNDLE UNCOVERY	15 MIN (1 S/G) 10 MIN (OTHER 3 S/Gs)
g. IODINE PARTITION FACTOR IN S/G	1.0
h. NOBLE GAS PARTITION FACTOR IN S/G	1.0
i LEAKAGE MASS (RCS INVENTORY LEAKED TO SECONDARY SIDE)	2.4 LBM (MAXIMUM LEAKING S/G WITH TUBE BUNDLE UNCOVERED FOR 15 MIN.) 3.0 LBM (OTHER 3 S/Gs)

9. REFERENCES

1. Catawba Nuclear Station, Final Safety Analysis Report, Docket Nos. 50 - 413/414.

2. BAW-10172P-A, Mark-BW Mechanical Design Report, Babcock & Wilcox, Lynchburg, Virginia, December 19, 1989.

3. DPC-NE-2001P-A, Rev. 1, Fuel Mechanical Reload Analysis Methodology for Mark-BW Fuel, Duke Power Company, October 1990.

4. BAW-10084A, Rev. 2, Program to Determine In-Reactor Performance of B&W Fuels - Cladding Creep Collapse, Babcock & Wilcox, October 1978.

5. BAW-10141P-A, Rev. 1, TACO2 - Fuel Performance Analysis, Babcock & Wilcox, June 1983.

6. DPC-NF-2010A, McGuire Nuclear Station/Catawba Nuclear Station Nuclear Physics Methodology for Reload Design, Duke Power Company, June 1985.

7. DPC-NE-2011P-A, Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors, Duke Power Company, March 1990.

8. DPC-NE-2004P-A, McGuire and Catawba Nuclear Stations Core Thermal- Hydraulic Methodology using VIPRE-01, Duke Power Company, December 1991.

9. BAW-10159P-A, BWCMV Correlation of Critical Heat Flux in Mixing Vane Grid Fuel Assemblies, Babcock & Wilcox, July 1990.

10. BAW-10173P-A, Mark-BW Reload Safety Analysis for Catawba and McGuire, Babcock & Wilcox, Revision 2, February 20, 1991.

11. DPC-NE-3000P, Duke Power Company, Thermal-Hydraulic Transient Analysis Methodology, Revision 2, February 20, 1990.

12. DPC-NE-3001P, Duke Power Company, Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology, Revision 2, November 1991.

13. BAW-10174-A, Mark-BW Reload LOCA Analysis for the Catawba and McGuire Units, Babcock & Wilcox, Revision 1, November 1990.

14. BAW-10168-A, B&W Loss-of-Coolant Accident Evaluation Model For Recirculating Steam Generator Plants, Babcock & Wilcox, Lynchburgh, Virginia, January 1991.

15. DPC-NE-1003A, Revision 1, McGuire Nuclear Station/Catawba Nuclear Station Rod Swap Methodology Report for Startup Physics Testing, December 1986.

16. DPC-NE-3002, McGuire Nuclear Station/Catawba Nuclear Station FSAR Chapter 15 System Transient Analysis Methodology, Revision 1, November 1991.

17. McGuire Nuclear Station Unit 2, Docket Number 50-370, Cycle 8 Reload Submittal, Duke Power Company, December 18,1991.

Attachment 2

Technical Justification

Proposed Revision to Technical Specification Figure 2.1-1a

This change is the same as that approved for McGuire Units 1 and 2.

This proposed Technical Specification (TS) revision changes Figure 2.1-1a to reflect use of the BWCMV CHF correlation and Duke Power Company's Statistical Core Design (SCD) methodology with a 1.55 thermal design DNBR limit.

Technical Justification

Duke Power Company (DPC) has recalculated the Catawba reactor core safety limits using the BWCMV CHF correlation (Reference 1) along with its Statistical Core Design (SCD) methodology, Reference 5. With the implementation of these design methodologies, the nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, was decreased from 1.55 to 1.50. The proposed changes to Figure 2.1-1 reflect the use of this new design limit as well as the use of BWCMV and the Duke Power Company SCD.

The reactor core safety limits provided on Figure 2.1-1 depict the combinations of thermal power, Reactor Coolant System pressure, and average temperature within which the calculated DNBR is no less than the design limit DNBR value, or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid. The analysis which defines these limits is based on a full core of Mark-BW assemblies with a thermal design flow rate that bounds the minimum measured flow at Catawba. The DNB limited portions of these curves are defined using the BWCMV CHF correlation with a design DNBR limit of 1.55. This design DNBR limit provides 10.7 percent thermal margin to the 1.40 BWCMV statistical design limit which is defined for the Catawba core using the DPC SCD methodology. The safety limits are based on a design peaking distribution with a nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, of 1.50 and a reference chopped cosine axial power shape with a peak of 1.55. To verify that this design peaking distribution is conservative on a cycle-specific basis, maximum allowable peaking (MAP) limits, which provide DNB equivalence to the design distribution at various safety limit statepoints, are defined. To verify that margin was available, these MAP limits were compared to predicted cycle-specific peaking distributions, Reference 2. As part of the safety limit/MAP limit analyses, an evaluation was performed which showed that, if power were reduced below 100 percent, peaking could be increased according to the following relationship:

$$k = 1 + 0.3(1 - P)$$

where k = the factor by which the MAP limits are adjusted to define reduced power limits

P = the fraction of rated power

while maintaining the core within the aforementioned thermal limits.

Comparison of the Mark-BW safety limits and the Westinghouse OFA safety limits shows that at all points the Mark-BW limits are outside the OFA limits. Mixed core studies have shown that the Mark-BW safety limits are applicable to the Westinghouse OFA fuel if a DNBR penalty is included for those assemblies. The DNBR penalty for OFA fuel is applied against the 10.7 % margin included in the design DNBR limit.

Proposed Revision to Technical Specification 2.2.1

This proposed Technical Specification (TS) change deletes ACTION 2.2.1.b.1 and equation 2.2-1. This change is consistent with the proposed change to Technical Specification 3.3.2, and the removal of the Total Allowance (TA) column, the Z column, and the Sensor Error (S) column from Tables 2.2-1 and 3.3-4.

Technical Justification

ACTION 2.2.1.b.1 provides the option of declaring an instrumentation channel operable by use of equation 2.2-1 when the Reactor Trip System Instrumentation or Interlock Setpoint is less conservative than the Allowable Value. Use of this option necessitates the inclusion of the Total Allowance (TA), the Value Z, and the Sensor Error (S) terms in TS 2.2.1 and Table 2.2-1. Both ACTION 2.2.1.b.1 and 2.2.1.b.2 require that the Setpoint be adjusted consistent with the Setpoint value given in Table 2.2-1; however, deletion of ACTION 2.2.1.b.1 and equation 2.2-1 makes TS 2.2.1 more restrictive, in that the channel must be declared inoperable with the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the Allowable Value. The deletion of ACTION 2.2.1.b.1 and equation 2.2-1 improves the safe operation of the plant by reducing the complexity of the Technical Specifications. This change also provides increased conformity with the comparable Technical Specification for McGuire Nuclear Station.

Proposed Revision to Technical Specification Table 2.2-1

This proposed Technical Specification (TS) revision changes the K values for the overtemperature and overpower ΔT trip functions to reflect the use of the BWCMV CHF correlation and Duke Power Company's Statistical Core Design (SCD) methodology with a 1.55 thermal design DNBR limit. In addition, an axial imbalance penalty, $f_2(\Delta I)$, is applied to the OPΔT reactor trip. The power range neutron flux negative rate reactor trip is deleted from the Reactor Protection System.

In addition, the table is revised to delete the Total Allowance (TA) column, the Z column, and the Sensor Error (S) column from Table 2.2-1. This change is consistent with the proposed change to Technical Specifications 2.2.1, 3.3-2 and Table 3.3-4.

Technical Justification

The overtemperature and overpower ΔT reactor trips are designed to protect the reactor from DNBR and centerline fuel melt (CFM). Due to the new DNBR methodology, the allowable operating region is modified. Therefore, new K values are calculated to conservatively bound this operating region.

The purpose of the OPΔT trip function is to prevent center-line fuel melt (CFM) during normal operation and condition II transients. The OPΔT trip function is designed to trip the reactor when the measured ΔT exceeds 118% of normal full power ΔT. The $f_2(\Delta I)$ portion of the trip function is designed to lower the trip setpoint when axial flux differences (AFDs) exceed predetermined limits. Since the limiting margins to CFM occur as the result of highly skewed power distributions, a $f_2(\Delta I)$ trip reset function can be developed to prevent

CFM limits from being exceeded, or to increase the available margin to the CFM limit.

Analysis of the C1C7 core indicates that no $f_2(\Delta I)$ trip function is required to protect the core from CFM. However, from an operational and design standpoint, it is desirable to prevent power distributions corresponding to high AFDs. For the C1C7 core, the $f_2(\Delta I)$ trip reset function was developed to produce a reactor trip on high AFDs for credible overpower events protected by the OPΔT trip function. The implementation of this function prevents power distributions for highly skewed conditions and affords increased margin to the CFM limit. This margin is important in that it will reduce the probability that adjustments to the OTΔT trip function will be required as the result of CFM surveillance requirements.

A reactor trip on negative flux rate is not assumed in any of the licensing basis accident analyses. The analysis of the dropped rod accident (Reference 4), for which the negative flux rate trip was designed to provide protection, assumes that reactor trip occurs on low pressurizer pressure, if at all. For cases in which no trip occurs, i.e. for low dropped rod worths, this analysis shows that none is needed. The removal of the negative flux rate trip from the Reactor Protection System will eliminate both unnecessary reactor trips resulting from such low worth rod drop events and spu ous trips from maintenance/surveillance activities.

In addition to the technical justification provided for the proposed revision to Technical Specification 2.2-1, the removal of the TA, Z, and S columns from Table 2.2-1 improves the safe operation of the plant by improving the readability of the table. Furthermore, retention of these terms necessitates a change in many of the values in order to be consistent with the licensing basis accident analyses performed by Duke Power Company. It should be noted that the Trip Setpoints and Allowable Values remain unchanged unless specifically addressed as a proposed revision.

Proposed Revision to Technical Specification Table 3.1-1

The table is revised to include all accident analyses that would require reevaluation in the event that one full-length control rod is inoperable.

Technical Justification

This change is the same as that approved for McGuire Units 1 and 2.

The existing Table 3.1-1 listed the rod cluster control assembly misalignment event when, in fact, the broader-scoped event of rod misoperation should have been given. This accident includes the static misalignment as well as single rod withdrawal, dropped rod and dropped bank events, which might all be impacted by the inoperable rod. The large break LOCA analysis, which was listed in the table, does not take credit for any control rod insertion and should therefore be removed.

Proposed Revision to Technical Specification 3/4.2.2

The change to SR 4.2.2.2.c.2.a and SR 4.2.2.2.c.2.b is to reflect Duke Power vs. vendor (B&W) methods for surveillance (i.e. remove NSLOPE and PSLOPE from SR 4.2.2.2.c.2.a) and provide clarification of SR 4.2.2.2.c.2.b action requirements. The change to SR 4.2.2.2.d surveillance method is to provide clarification of 4.2.2.2.d.

Technical Justification

The changes were made to provide surveillance requirements consistent with Duke Power Company methodology for core power distribution control and surveillance of the heat flux hot channel factor, as discussed in DPC-NE-2011PA (Reference 2). The nomenclature in SR 4.2.2.2.c.2.a is changed to reflect Duke Power vs. vendor (B&W) nomenclature. SR 4.2.2.2.c.2.b provides clarification of which margin is to be used for this action. SR 4.2.2.2.d requires projections of the measurements to be made to determine at what point $F_Q(X,Y,Z)$ would exceed the allowable limit if the current trend continues. However, for the case when the projection of the current trend would indicate that the margin is increasing, the actions for the current negative margin (from SR 4.2.2.2.c.2 and 4.2.2.2.c.3), would be sufficient since this margin is less than the extrapolated margin. In this case it would not be necessary to penalize the current margin based on the results of the extrapolation, since the extrapolation indicates an improvement over time.

Proposed Revision to Technical Specification 3/4.2.3

Specification 3/4.2.3 was revised to reflect the power peaking surveillance method described in DPC-NE-2011PA. These revisions are summarized as follows:

1. The statement Specification 3.2.3 LCO was revised to reflect Duke's nomenclature for the nuclear enthalpy rise hot channel factor [$F^M_{DH}(X,Y)$], and related parameters required by the methodology of DPC-NE-2011PA vs. vendor (B&W) nomenclature and used throughout the Reload Report.

2. There are no changes to Specification 3.2.3 Actions a, b, c and d other than nomenclature as mentioned above.

3. SR 4.2.3.2.b addresses the frequency for confirming that $F_{DH}(X,Y)$ is within its limit. In addition to performing the surveillance at least once per 31 EFPD, the revised surveillance requires measurement of the peaking factor whenever the excore quadrant power tilt ratio is normalized using incore detector measurements. This ensures that the impact of any core tilt on $F_{DH}(X,Y)$ will be determined and reflected in the margin calculation. This is comparable to SR 4.2.2.2.b in the $F_Q(X,Y,Z)$ specification. SR 4.2.3.2.b.1 requires a surveillance to be performed upon reaching equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER at which $F^M_{DH}(X,Y)$ was last determined.

4. The purpose of SR 4.2.3.2.c.1 is to perform margin calculations based on the measured radial peak. The limit $[F^M_{DH}(X,Y)]^{SURV}$ to which the measurement is compared is based on the allowable design MARP limit, increased by a factor that represents the maximum amount that the power at the given assembly location can

increase above the design value before the measured value may become limiting. Part c.2 uses the amount of margin determined by this procedure to form the basis for the amount of power level reduction and the reduction in the high flux and OTDT K_1 trip setpoints required in the ACTION statements for the specification.

5. SR 4.2.3.2.d has been changed to reflect Duke's nomenclature and to allow for the case when margin may be increasing over time. This surveillance requires projections of the measurements to be made to determine at what point $F_{DH}(X,Y)$ would exceed the allowable limit if the current trend continues. In part d.1 a penalty is applied to $F^{M}_{DH}(X,Y)$ if the trend indicates that the measured peak would exceed the limiting peak within the 31 EFPD surveillance period, and the margin calculations are repeated. In part d.2, the measurement is obtained and the margin calculations are repeated so that appropriate actions can be taken before zero margin is reached.

Technical Justification

Specification 3/4.2.3 was revised to provide required actions and surveillance requirements consistent with Duke Power Company methodology for core power distribution control and surveillance of the nuclear enthalpy rise hot channel factor, as discussed in DPC-NE-2011PA (Reference 2). The nomenclature is changed to reflect Duke Power nomenclature vs. vendor (B&W) nomenclature and clarification of surveillance requirements is provided to be consistent with Duke's methods described in DPC-NE-2011PA.

SR 4.2.3.2.b.1 is changed to better clarify the surveillance requirement and to ensure that the plant is at equilibrium conditions prior to a measurement. The surveillance requirement also has a provision similar to the requirement it replaced stating that 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.

SR 4.2.3.2.c is changed to reflect Duke Power vs. vendor (B&W) nomenclature. This surveillance requirement is comparable to the SR 4.2.2.2.c on $F_Q(X,Y,Z)$.

SR 4.2.3.2.d is changed to reflect Duke Power vs. vendor (B&W) nomenclature and to provide clarification of 4.2.3.2.d. An additional check has been added to determine if the margin is increasing. This surveillance requirement is comparable to the new SR 4.2.2.2.d on $F_Q(X,Y,Z)$.

Proposed Revision to Technical Specification 3/4.2.5

This revision is intended to provide consistency between Duke's Westinghouse plants, and correct an action item.

Technical Justification

This change is administrative in nature because it corrects a typograghpical error. It does not represent an actual change to the Requirement of 3/4.2.5.

Proposed Revision to Technical Specification Table 3.3-1

This change is to delete the reactor trip on power range neutron flux negative rate from the Reactor Protection System.

Technical Justification

Refer to the technical justification for the proposed revision to TS Table 2.2-1.

Proposed Revision to Technical Specification Table 3.3-2

The reactor trip on power range neutron flux negative rate is deleted. The response times associated with the RTD Bypass System for the OTΔT and OPΔT are deleted, as is footnote regarding RTD Bypass System. Neutron detector response time exemption is added to OPΔT trip.

Technical Justification

For the deletion of the power range neutron flux negative rate, refer to the technical justification for the proposed revision to TS Table 2.2-1.

For the response times and footnote regarding RTD Bypass System, the RTD Bypass System has been deleted.

The neutron detector response time exemption which is applicable to the OTΔT trip is now applicable to the OPΔT trip due to the addition of the $f_2(\Delta I)$ function to the OPΔT trip. Refer to the technical justification for the proposed revision to TS Table 2.2-1.

Proposed Revision to Technical Specification Table 4.3-1

This change is to delete the reactor trip on power range neutron flux negative rate from the Reactor Protection System.

Technical Justification

Refer to the technical justification for the proposed revision to TS Table 2.2-1.

Proposed Revision to Technical Specification 3.3.2

This proposed Technical Specification (TS) change deletes ACTION 3.3.2.b.1 and equation 2.2-1. This change is consistent with the proposed change to Technical Specification 2.2.1, and the removal of the Total Allowance (TA) column, the Z column, and the Sensor Error (S) column from Tables 2.2-1 and 3.3-4.

Technical Justification

ACTION 3.3.2.b.1 provides the option of declaring an instrumentation channel operable by use of equation 2.2-1 when the ESFAS Instrumentation or Interlock Setpoint is less conservative than the Allowable Value. Use of this option necessitates the inclusion of the Total Allowance (TA), the value Z, and the Sensor Error (S) terms in TS

3.3.2 and Table 3.3-4. Both ACTION 3.3.2.b.1 and 3.3.2.b.2 require that the Setpoint be adjusted consistent with the Setpoint value given in Table 3.3-4; however, deletion of ACTION 3.3.2.b.1 and equation 2.2-1 makes TS 3.3.2 more restrictive, in that the channel must be declared inoperable with the ESFAS Instrumentation or Interlock Setpoint less conservative than the Allowable Value. The deletion of ACTION 3.3.2.b.1 and equation 2.2-1 improves the safe operation of the plant by reducing the complexity of the Technical Specifications. This change also provides increased conformity with the comparable Technical Specification for McGuire Nuclear Station

Proposed Revision to Technical Specification Table 3.3-4

This proposed revision changes the low steam line pressure setpoint for safety injection and main steam line isolation from 725 psig to 775 psig. The allowable value for this trip function is changed from 694 psig to 744 psig, maintaining the same 31 psig allowance for rack uncertainties, and the lead-lag controller for steam line pressure-low is deleted. In addition the table is revised to delete the Total Allowance (TA) column, the Z column, and the Sensor Error (S) column from Table 3.3-4. This change is consistent with the proposed change to Technical Specifications 2.2.1, 3.3.2 and Table 2.2-1. Lastly, the Allowable Values associated with RTD Bypass System for the Feedwater isolation on Tavg-Low and ESFAS P-12 interlock on Low-Low Tavg are deleted.

Technical Justification

For TS Table 3.3-4 items 1.e., 4.d, and lead-lag removal:

The higher steam line pressure setpoint is consistent with all licensing basis safety analyses. The steam line break event was reanalyzed (Reference 4) assuming an uncompensated low steam line pressure setpoint of 700 psig, allowing 75 psig for instrument uncertainty and margin. This reanalysis shows this Condition IV transient does not exceed the imposed Condition II acceptance criterion of no DNB.

The inadvertent opening of a steam generator relief or safety valve, in terms of primary system overcooling, is essentially a small steam line break. This Condition II event exhibits steam releases markedly less than the steam line break event. Therefore, this event is bounded by the steam line break event and does not require reanalysis.

This change does not necessitate reanalysis of the peak containment temperature analysis. The removal of the dynamic compensation of the steam line pressure signal, even with the increased setpoint, will delay main steam line isolation following certain breaks in the main steam line. Since the effect of main steam isolation is to terminate the blowdown of the intact steam generators, postponing MSIV closure will tend to retard the blowdown of the faulted generator and delay tube uncovery. Also, the extended blowdown of the intact steam generators will reduce the Reactor Coolant System temperature, which will further delay tube uncovery. Once uncovery does occur, this lower primary system temperature will effectively reduce the enthalpy of the break flow.

The removal of the dynamic compensation of the steam pressure signal, which accompanies the change in the low pressure setpoint, will essentially eliminate the spurious ESF actuation on minor (but rapid) pressure decreases in the secondary system. In addition, the increase in containment temperature following a steam line break is caused by the release of steam from the steam generators to the containment atmosphere via the break. This steam initially comes from two sources, the steam generator connected to the steam line with the break and the other three steam generators. In the subsequent discussion, the former is referred to as the faulted SG, while the latter are referred to as the intact SGs. Since the most limiting break locations are between the faulted SG and its MSIV, the steam line isolation function does not prevent blowdown from that SG. However, even for a single MSIV failure, steam line isolation terminates the blowdown from the intact SGs.

Because of the energy content of the RCS, steam line breaks occurring from an at-power initial condition are limiting for peak containment temperature. During power operation, steam line isolation is actuated by low steam line pressure or high-high containment pressure. The current configuration for the steam line pressure actuation has lead/lag compensation on the pressure signal. For a decreasing measured pressure in the steam line, this compensated signal will fall faster than the measured pressure signal. Therefore, for a given transient in measured pressure and a given isolation setpoint, isolation on compensated pressure will occur faster than isolation on measured pressure. As the rate of decrease of the measured pressure slows (smaller breaks), the compensation becomes less important, resulting in compensated pressure and measured pressure becoming more nearly identical. For this situation the isolation setpoint is a more important factor, in determining isolation time, than is the presence of the lead/lag compensation. The proposed Catawba 1 Cycle 7 Technical Specification change removes the lead/lag compensation and increases the isolation setpoint from 725 psig to 775 psig. This change, as explained above, would result in later steam line isolation for larger breaks and earlier isolation for smaller breaks. The proposed Technical Specification change results in a faster, and therefore more conservative, isolation time for these smaller breaks.

For the three cases for which mass and energy releases are presented in the Catawba FSAR, credit for steam line isolation on low steam line pressure is taken only for the 1.4 ft^2 double ended break case. Steam line isolation for this case occurs at 11 seconds. However, high-high containment pressure is actuated at 3 psig per the Technical Specifications. A safety analysis setpoint of 3.75 psig bounds the instrument uncertainty associated with the measurement of containment pressure, and the time to reach containment pressure of 3.75 psig is 0.85 seconds. Adding 7 seconds for the maximum allowable Technical Specification response time for steam line isolation on high-high containment pressure gives 7.85 seconds for completion of steam line isolation, compared to the 11 second time given in the FSAR. Therefore, even if there were a delay, due to removal of the lead/lag compensation, in reaching th[illegible]w steam line pressure isolation setpoint for larger breaks [illegible]SAR mass and energy releases would remain conservative becau[illegible]ation would be accomplished on high-high containment pressure [illegible]ior to the time that isolation is credited in the FSAR.

For the removal of the Total Allowance (TA) column, the Z column, and the Sensor Error (S) column from Table 3.3-4:

In addition to the technical justification provided for the proposed revision to Technical Specification 3.3.2, the removal of these terms from Table 2.2-1 improves the safe operation of the plant by improving the readability of the table. It should be noted that the Trip Setpoints and Allowable Values remain unchanged unless specifically addressed as a proposed revision.

For TS Table 3.3-4 items 5.c, 18.c, and Table Notations:

The RTD Bypass System has been deleted.

Proposed Revision to Technical Specification Table 3.3-5

Two response times are modified in this proposed change, the feedwater isolation response time is changed from 7 seconds to 12 seconds and the steam line isolation time is changed from 7 seconds to 10 seconds.

Technical Justification

The extended response times are consistent with or conservative for all licensing basis safety analyses. These two response times are assumptions in the steam line break analysis. The 12 second feedwater isolation time and the 10 second steam line isolation times have been employed in the steam line break analysis approved in Reference 4. Increasing these response times, from the current Technical Specification values, causes the primary system overcooling to worsen due to the extended blowdown of the intact generators and the additional mass of main feedwater delivered to the faulted generator. Reanalysis (Reference 4) shows this Condition IV transient does not exceed the imposed Condition II acceptance criterion of no DNB.

The inadvertent opening of a steam generator relief or safety valve, in terms of primary system overcooling, is essentially a small steam line break. This Condition II event exhibits steam releases markedly less than the steam line break event. Therefore, this event is bounded by the steam line break event and does not require reanalysis.

The increased feedwater isolation response time also impacts the analysis of the excessive feedwater flow event. The effect is negligible, however, since the DNBR decrease is terminated by the turbine trip, which occurs 3 seconds after the high-high steam generator level setpoint is reached. The fact that the feedwater isolation valves close 12 seconds after this setpoint is reached rather than 7 seconds does not affect the minimum DNBR achieved.

For the peak containment temperature analysis (Reference 6), a lower liquid mass in the faulted generator yields conservative results. This leads to an earlier uncovery of the steam generator tubes and, thus, the advent of superheated steam exiting the break. Lengthening the feedwater isolation response time will increase the amount of feedwater delivered to the faulted generator and delay tube uncovery. Increasing the main steam line isolation response time has a similar effect as the removal of the dynamic compensation in the steam line pressure signal discussed in the technical justification for the proposed change to TS Table 3.3-4. In addition, as explained in Item 7 of Section 6.2.1.4.1 of the Catawba FSAR, the mass and energy releases were calculated beyond the point of peak containment temperature. As shown in Tables 6-47, 6-49, and 6-51 of the Catawba FSAR, the mass and energy releases from the faulted SG are not zero at the end of the release calculation, indicating that inventory remains in the faulted SG at that time, which is after the time of peak containment temperature. Therefore, the addition of feedwater beyond the amount assumed in the FSAR analysis, due to an increase of feedwater isolation time of 5 seconds, would simply increase the amount of inventory remaining in the faulted SG at the end of the mass and energy release calculation. It would not increase the calculated containment temperature.

Significant difficulty has been experienced in meeting the current specification response times for both of these ESF functions. Increasing the allowable response times should eliminate this difficulty.

Proposed Revision to Technical Specification 3.4.1.2

The specification is being changed to require that the three operable reactor coolant loops be in operation in Mode 3.

Technical Justification

This restriction is imposed in order to make the specifications consistent with the reanalysis of the uncontrolled bank withdrawal from subcritical or low power startup condition.

Proposed Revision to Technical Specification 3.4.2.1 & 3.4.2.2

This modification changes the tolerances on the pressurizer safety valve lift setpoint from ±1% to +3%,-2% in all modes of operation.

Technical Justification

The pressurizer code safeties are not tested in place but are removed and shipped to a testing facility. The safety concerns for removal and replacement of these valves are difficult access to the work area, difficulty in lifting device rigging for valve removal/replacement, and valve transport to/from the pressurizer. Since this work is performed in a radiological environment, work activities are further complicated by anti-contamination clothing. For a conservative approach, all three valves are removed each outage for testing. The setpoint drift seen during testing would again fall under the proposed setpoint variance change. The change would possibly reduce work in the pressurizer by 66% by requiring only one valve to be tested per outage. In summary, safety benefits would be gained by less work in a dangerous environment and less radiation exposure.

The larger allowable deviation from the nominal lift setting is consistent with the licensing basis analyses. An increased pressurizer safety valve lift setpoint impacts the peak Reactor Coolant System pressure calculated for pressure increase transients. A pressure increase is the result of a heatup in the Reactor Coolant System due a mismatch between the heat generated in the reactor core and the heat removed by the secondary system. The three accident categories involving such heat transfer mismatches are the decrease in secondary heat removal, decrease in Reactor Coolant System flow rate, and reactivity and power distribution anomaly transients. The feedline break, locked rotor and rod ejection events are the limiting pressure increase transients in these three accident categories, respectively. These events have all been analyzed assuming a lift setpoint 3 percent above the nominal value. These analyses show that the peak Reactor Coolant System pressure criterion, 110% of design pressure, is met for the feedline break and locked rotor events. The peak Reactor Coolant System pressure criterion of 120% of design pressure is met in the rod ejection accident analysis.

The amount by which the safety valve lift setpoint is allowed to drift downward is restricted to 2 percent of nominal in order to ensure that safety valve lift cannot preclude reactor trip on high pressurizer pressure. For DNB transients in which a high pressurizer pressure reactor trip does not prevent the lifting of the safety valves, the effect of this reduced setpoint on the transient DNBR is evaluated. Since low pressure is conservative for DNBR analyses, it is typically assumed that the pressurizer PORVs and sprays mitigate the pressure increase due to the system heatup and thereby preclude safety valve lift. For the uncontrolled bank withdrawal at power and single rod withdrawal events, however, the operation of the pressurizer pressure control system would tend to yield an earlier reactor trip on overtemperature ΔT due to pressure compensation of the trip setpoint. The reanalysis of these events show that all acceptance criteria are met.

Proposed Revision to Technical Specification 3.4.6.2.c

This change reduces the allowable total reactor-to-secondary leakage rate.

Technical Justification

This revision is required to limit primary-to-secondary leakage to values which ensure that the offsite doses for FSAR Chapter 15 transients and accidents meet the applicable fraction of the 10 CFR 100 limits. The change is necessary to account for an increase in the prediction of the number of fuel pins exceeding the DNBR limit for the locked rotor transient. This transient was reanalyzed for C1C7 using approved Duke Power methods. The revised dose calculation also incorporates the impact of steam generator tube bundle uncovery, which results in less credit for iodine partitioning in the steam generator secondary.

Proposed Revision to Technical Specification 3.5.1.1

This change raises the required average cold leg accumulator boron concentration in ACTIONS c.2 and c.3 from 1500 to 1800 ppm, and bases this average on all four accumulators instead of just the limiting three.

Technical Justification

Calculating the volumetric average boron concentration based on all four cold leg accumulators is valid, since, regardless of the break location, the contents of each accumulator will be emptied (either directly or indirectly) into the containment sump. A volumetric average concentration of 1800 ppm will ensure long-term subcriticality following a LOCA.

Proposed Revision to Technical Specification 4.5.2 (f & h)

T. S. 4.5.2(f) gives the ECCS pump performance requirements. The centrifugal charging pump required developed head is decreased from 2380 to 2223 psid. The safety injection pump required developed head is decreased from 1430 to 1341 psid. T. S. 4.5.2(h) gives the ECCS delivered flow requirements. The centrifugal charging pump total flow

.ate is decreased from 565 to 560 gpm. The safety injection pump total flow rate is increased from 660 to 675 gpm. These revisions are applicable to Unit 1 and Unit 2.

Technical Justification

The ECCS pump required developed head and delivered flow specifications are being revised to provide performance test margin for periodic testing, and to be consistent with revised ECCS pump and system performance requirements. These specifications are consistent with the current safety analyses and with the revised analyses associated with Catawba 1 Cycle 7. The ECCS pump runout flow rates are being adjusted to reflect revised pump vendor information. Performance test margin is required to enable sufficient allowances for instrument errors and to permit reasonable test acceptance criteria. Pump performance at the new specification values is sufficient to meet all acceptance criteria in both the current FSAR analyses, and in the revised analyses submitted with or referenced by the Catawba 1 Cycle 7 reload. The proposed revisions are therefore acceptable.

Proposed Revision to Technical Specification Table 3.6-2a

This change clarifies the required maximum stroke time of the feedwater isolation valves, main steam isolation valves, and main steam isolation bypass valves. The numerical value of the stroke time of these valves is changed to NA.

Technical Justification

The justification for the change in valve stroke time as it relates to system thermal-hydraulic response during a steam line break event is presented for the change to Technical Specification Table 3.3-5. Although these valves are included in Table 3.6-2a, the list of containment isolation valves for Unit 1, these valves do not receive a containment isolation signal. These valves perform a containment isolation function only to the extent that credit for their operation might be taken in the dose analysis. Since these valves receive no containment isolation signal, and credit for the operation of these valves is not taken in the dose analysis, a maximum stroke time is not applicable for these valves.

Proposed Revision to Technical Specification 4.7.1.4

The permissible stroke time for the main steam isolation valves is changed from 5 to 8 seconds.

Technical Justification

The larger allowable isolation valve stroke time is consistent with or conservative for all licensing basis safety analyses. The valve stroke time, when added to the applicable instrumentation delays, yields the overall ESF response time. As stated in the technical justification for the proposed revision to TS Table 3.3-4, this response time is input to the steam line break transient analysis. Increasing the stroke time causes the primary system overcooling to worsen due to the extended blowdown of the intact generators. Analysis using the approved methodology in Reference 4 shows this Condition IV transient does not violate the imposed Condition II acceptance criterion of no DNB.

The inadvertent opening of a steam generator relief or safety valve, in terms of primary system overcooling, is essentially a small steam line break. This Condition II event exhibits steam releases markedly less than the steam line break event. Therefore, this event is bounded by the steam line break event and does not require reanalysis.

For the peak containment temperature analysis (Reference 6), increasing the main steam line isolation response time has a similar effect as the removal of the dynamic compensation in the steam line pressure signal discussed in the technical justification for the proposed change to TS Table 3.3-4. In addition, the issue of peak containment temperature is one of equipment qualification for the post-accident environment. For the following reasons this proposed Technical Specification change is judged to be acceptable:

Since the effect of steam line isolation is to terminate the blowdown of the intact SGs, postponing MSIV closure will tend to retard the blowdown of the faulted SG and delay tube uncovery. Also, the extended blowdown of the intact SGs will reduce the RCS temperature, which will further delay tube uncovery. Once uncovery does occur, this lower RCS temperature will result in a reduced break flow enthalpy. It should also be noted that, whether the MSIV stroke time is 5 seconds or 8 seconds, the isolation function is completed well before tube uncovery.

Except for the non-limiting 1.4 ft² double-ended break, the peak containment temperature occurs at a relatively long time after the completion of isolation, even if the stroke time is extended to 8 seconds:

Break Size (ft²)	0.86	0.4
5 second Isolation Time (sec)	17.5	26
8 second Isolation Time (sec)	20.5	29
Peak Temperature (sec)	127	>150

From reviewing the mass and energy releases in Tables 6-50 and 6-52 of the Catawba FSAR, the following information can be determined:

Break Size (ft²)	0.86	0.4
FSAR Intact SG Mass Release (lbm)	24,780	16,830
Intact SG Mass Release Rate(lbm/sec)	1490	650
Releases During 3 Additional Sec (lbm)	4470	1950
Estimated Intact SG Mass Releases (lbm)	29,250	18,780

It can be seen that, while the intact SG releases during the FSAR 0.86 ft² break are more than 45% larger than during the 0.4 ft² break (24,780 vs. 16,830, a difference of 7950 lbm), the difference in peak containment temperatures given in Figures 6-21 and 6-22 of the Catawba FSAR is barely discernible (327.7°F vs. 326.8°F). This small difference is expected since the containment temperature is primarily sensitive to the higher enthalpy later releases from the faulted SG. The additional intact SG releases due to extending the valve stroke time are, as shown above, conservatively estimated to be 4470 lbm and 1950 lbm. These values are much less than the 7950 lbm difference between the intact SG releases for the two break sizes, which resulted in an insignificant change in peak containment temperature. Therefore, the additional release due to increased valve stroke time during this period of the transient will not cause a significant increase in the peak containment temperature.

There is margin in the current equipment qualification limit of 340°F to accommodate an increase of 12.3°F (above the FSAR result of 327.7°F) due to any effect of increased total mass and energy released from the intact steam generators due to delayed completion of the steam line isolation function.

In summary, the release of saturated steam for 3 additional seconds from the intact SGs, during a period well before the peak containment temperature, would not cause a significant impact on this peak temperature. The magnitude of the additional releases is much less than the difference between intact SG releases for the different FSAR split break cases, which have resulting temperature differences of 0.22°F. Even if such a release were to cause a nonnegligible temperature increase, there is approximately 12°F margin between the current FSAR result and the equipment qualification limit.

Significant difficulty has been experienced in meeting the current specification stroke time for these valves. Increasing the allowable stroke time should eliminate this difficulty.

Proposed Revision to Technical Specification 6.9.1.9

Changes to Specification 6.9.1.9 are as follows. $F_Q^L(X,Y,Z)$ was added to the list of items for Heat Flux Hot Channel Factor, note '**' was added to $W(Z)_{BL}$, $F\Delta HR^L$ was changed to $F\Delta H^L(X,Y)$, reference numbers in note '**' were changed and the reference list was updated.

Technical Justification

The addition of $F_Q^L(X,Y,Z)$ and the change of $F\Delta HR^L$ to $F\Delta H^L(X,Y)$ were changes in nomenclature due to a methodology change. The new nomenclature is consistent with Duke's methods described in DPC-NE-2011PA. The '**' note had previously been excluded from $W(Z)_{BL}$, so it was added. The references for the '**' note were changed since the references changed. The reference list was updated to include references describing the new methodology.

References

1. BAW-10159P-A, BWCMV Correlation of Critical Heat Flux in Mixing Vane Grid Fuel Assemblies, Babcock & Wilcox, July 1990.

2. DPC-NE-2011P-A, Duke Power Company, Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors, March, 1990.

3. BAW-10174-A, Mark-BW Reload LOCA Analysis for the Catawba and McGuire Units, Babcock & Wilcox, May 1991.

4. DPC-NE-3001P, Duke Power Company, Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology, Revision 1, November 1991.

5. DPC-NE-2004P-A, Duke Power Company, McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology using VIPRE-01, December 1991.

6. WCAP-10988, Cobra-NC, Analysis for a Main Steamline Break in the Catawba Unit 1 Ice Condenser Containment, Westinghouse Nuclear Energy Systems, November 1985.

Attachment 3

NO SIGNIFICANT HAZARDS ANALYSIS

The following analysis, required by 10 CFR 50.91, concludes that the proposed amendment will not involve significant hazards consideration as defined by 10 CFR 50.92.

10 CFR 50.92 states that a proposed amendment involves no significant hazards consideration if operation in accordance with the proposed amendment would not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

2) Create the possibility of a new or different kind of accident from any previously evaluated; or

3) Involve a significant reduction in the margin of safety.

POWER DISTRIBUTION AND SAFETY LIMITS

Catawba Unit 1 Cycle 6 was the first Duke Power Nuclear Station for which B&W Fuel Company (BWFC) supplied the reload fuel. The Catawba Unit 1, Cycle 6 Reload Report presented an evaluation that concluded the core reload using Mark-BW fuel would not adversely impact the safety of the plant. This reload report is similar, but reflects that Duke Power performed the analyses in support of the operation of Cycle 7 rather than BWFC.

The Catawba Unit 1, Cycle 7 Reload Safety Evaluation Report (Attachment 1) presents an evaluation which demonstrates that the core reload using Mark-BW fuel will not adversely impact the safety of the plant. During Cycle 7, the core will contain 72 fresh fuel assemblies, 72 burned fuel assemblies supplied by B&W and 49 Westinghouse supplied Optimized Fuel Assemblies (OFA).

A LOCA evaluation for operation of Catawba Nuclear Station with Mark-BW fuel has been completed (BAW 10174, Mark-BW Reload LOCA Analysis for the Catawba and McGuire Units). Operation of the station while in transition from Westinghouse supplied OFA fuel to B&W supplied Mark-BW fuel is also justified in this topical.

BAW-10174 demonstrates that Catawba Nuclear Station continues to meet the criteria of 10 CFR 50.46 when operated with Mark-BW fuel. Large Break LOCA calculations completed consistent with an approved evaluation model (BAW-10168P and revisions) demonstrate compliance with 10 CFR 50.46 for breaks up to and including the double ended severance of the largest primary coolant pipe. The small break LOCA calculations used to license the plant during previous fuel cycles are shown to be bounding with respect to the new fuel design. This demonstrates that the plant meets 10 CFR 50.46 criteria when the core is loaded with Mark-BW fuel.

During the transition from Westinghouse OFA fuel to Mark-BW fuel, both types of fuel assemblies will reside in the core for several fuel cycles. Appendix A to BAW-10174 demonstrates that results presented above apply to the Mark-BW fuel in the transition core, and that insertion of the Mark-BW fuel will not have an adverse impact on the cooling of the Westinghouse fuel assemblies.

Duke Power Company's Topical Reports DPC-NE-3000, DPC-NE-3001, and DPC-NE-2004 provide evaluations and analyses for non-LOCA transients which are applicable to Catawba. The scope of these analyses includes all events specified by sections 15.1-15.6 of Regulatory Guide 1.70 (Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants) and presented in the Final Safety Analysis Report for Catawba. The analysis and evaluations performed for these topicals confirm that operation of Catawba Nuclear Station for reload cycles with Mark-BW fuel will continue to be within the previously reviewed and licensed safety limits.

One of the primary objectives of the Mark-BW replacement fuel is compatibility with the resident Westinghouse fuel assemblies. The description of the Mark-BW fuel design and the thermal-hydraulics and the core physics performance evaluation demonstrate the similarity between the reload fuel and the resident fuel. The extensive testing and analysis summarized in BAW-10173P shows that the Mark-BW fuel design performs, from the standpoint of neutronics and thermal-hydraulics, within the bounds and limiting design criteria applied to the resident Westinghouse fuel for the Catawba plant safety analysis.

Each FSAR accident has been reviewed to determine the effects of Cycle 7 operation and to ensure that the radiological consequences of postulated accidents are within applicable regulatory guidelines, and do not adversely affect the health and safety of the public. The design basis LOCA evaluations assessed the radiological impact of differences between the Mark-BW fuel and Westinghouse OFA fuel fission product core inventories. Also, the dose calculation effects from non-LOCA transients reanalyzed by Duke Power were evaluated using Cycle 7 characteristics. The calculated radiological consequences are all within specified regulatory guidelines and contain significant levels of margin.

The analyses contained in the referenced Topical Reports indicate that the existing design criteria will continue to be met. Therefore, the enclosed TS changes will not increase the probability or consequences of an accident previously evaluated.

As stated in the above discussion, normal operational conditions and all fuel-related transients have been evaluated for the use of Mark-BW fuel at Catawba Nuclear Station. Testing and analysis was also completed to ensure that, from the standpoint of neutronics and thermal-hydraulics, the Mark-BW fuel would perform within the limiting design criteria. Because the Mark-BW fuel performs within the previously licensed safety limits, the possibility of a new or different accident from any previously evaluated is not created.

The reload-related changes to the TSs do not involve a significant reduction in the margin of safety. The calculations and evaluations documented in BAW-10174 show that Catawba will continue to meet the criteria of 10 CFR 50.46 when operated with Mark-BW fuel. The evaluation of non-LOCA transients documented in DPC-NE-3001 also confirms that Catawba will continue to operate within previously reviewed and licensed safety limits. Because of this, the TS changes to support the use of Mark-BW fuel will not involve a significant reduction in the margin of safety.

An administrative change is being made to TS Tables 2.2-1 (Reactor Trip System Instrumentation Trip Setpoints), and Table 3.3-4 (Engineered Safety Features Actuation System Instrumentation Trip Setpoints). Since these tables contain values that are not identical for each unit, a separate table will be provided for each unit. The pages will be labeled "Unit 1" or "Unit 2", and there will be an "A" in the page number for Unit 1 and a "B" in the page number for Unit 2. The TS Tables will be copied on white paper for Unit 1 and on yellow paper for Unit 2 to further distinguish applicability. Table 3.3-4 will also have references to the RTD bypass system deleted, since the RTD bypass system has been removed, and they no longer apply. These changes are administrative in nature, and are being made only to clarify the TS. Since they involve no change in requirements, they involve no significant hazards.

REMOVAL OF TOTAL ALLOWANCE Z AND SENSOR ERROR FROM TABLES 2.2-1 AND 3.3-4

The removal of the Total Allowance, Sensor error, and Z columns from Tables 2.2-1 and 3.3-4, along with the deletion of TS 2.2.1.b.1, 3.3.2.b.1, and equation 2.2-1, which provide for the use of these values, do not involve any significant hazards consideration. These specifications provide the option of declaring instrumentation operable when the setpoint is less conservative than the allowable value. This is done through the use of equation 2.2-1. With the deletion of Specifications 2.2.1.b.1, 3.3.2.b.1, equation 2.2-1, and the Total Allowance, Sensor Error, and Z columns from Tables 2.2-1 and 3.3-4 the channel must be declared inoperable with the setpoint less conservative than the Allowable Value. This change is more conservative than the current requirements, and therefore involves no significant hazards.

DELETION OF NEUTRON FLUX HIGH NEGATIVE RATE TRIP

The removal of the Power Range Neutron Flux High Negative Rate trip will not result in any previously-reviewed accident becoming more probable or more severe. The trip is a response to a pre-existing transient condition and would not initiate any accident. The trip is designed to provide protection from a dropped control rod. However, in the event of a dropped rod, the reactor is assumed to trip on low pressurizer pressure. Therefore, the protection function is retained. The consequences of a dropped rod have been analyzed and found to be within acceptable limits.

Likewise, the removal of this trip will not create a new accident not previously reviewed. The removal of a response to a transient will not initiate a new transient. There are no credible unanalyzed transients which will occur as a result of a dropped rod. The removal of this trip will reduce the potential for spurious or unnecessary trips which may occur as a result of maintenance or the drop of a low-worth rod. There are no other hardware modifications or procedure changes that will be made as a result of this deletion which could create the possibility of a new accident.

No margin of safety will be reduced by this change. As noted above, if a dropped rod necessitates a trip, the trip function will be accomplished as a result of low pressurizer pressure. For those dropped rods for which no trip is necessary, the removal of this trip will provide protection against an unnecessary transient.

REDUCE ALLOWABLE PRIMARY TO SECONDARY LEAKAGE

The allowable primary to secondary leakage has been reduced to limit the offsite radiological dose consequences due to the reanalysis of the locked rotor, rod ejection, and single uncontrolled rod withdrawal FSAR Chapter 15 events. The new limits are more conservative than the current TS requirements. Lowering the allowable primary to secondary leakage will not increase the probability of a previously evaluated accident, it will ensure that the dose consequences of an accident are within allowable limits. The possibility of a new or different accident from any previously evaluated is not created because there will be no physical changes to the plant operating procedures, other than to more conservatively limit leakage. There will not be a significant reduction in the margin of safety due to the fact that the allowable leakage is more conservative.

Based on the above, it is concluded that no significant hazards are associated with this change.

INCREASE IN OPERABLE RCS LOOPS IN MODE 3 and INCREASE COLD LEG ACCUMULATOR REQUIRED BORON CONCENTRATION

These amendments will not involve any significant hazards consideration. The proposed changes will result in the parameter or operating condition involved becoming more conservative than the current TS requirement. The NRC's own guidance, published in the Federal Register (48CFR 14870), states that an amendment which results in conditions becoming more restrictive is not likely to result in significant hazards consideration as defined by 10 CFR 50.92. Therefore, it may be concluded, with no further analysis, that these amendments will not involve a significant hazards consideration.

ECCS PUMP PERFORMANCE REQUIREMENTS

The proposed amendments will not involve a significant increase in the probability or consequences of an accident previously evaluated because the Loss-of-Coolant-Accident (LOCA) analysis, to which the ECCS flowrates are input assumptions, is unchanged and, therefore, continues to meet applicable acceptance criteria.

The proposed amendments will not result in a significant increase in the possibility of a new accident because the new values represent a change in required pump performance. The new values represent no change in the assumptions made in the LOCA analysis, or any a physical change in the plant. Enough margin exists between the flow used in the LOCA analysis and the new required pump flows that a reanalysis was not necessary
The proposed changes will not result in a significant decrease in a margin of safety, because pump performance at the new values is sufficient to meet all acceptance criteria in both the current FSAR analysis and any analysis associated with Catawba 1 Cycle 7.

Based on the above, it is concluded that no significant hazards exist.

INCREASE IN PRESSURIZER CODE SAFETY VALVE SETPOINT TOLERANCES

The proposed amendment will not result in a significant increase in the probability or consequences of any previously analyzed accident. The valve lift setting is challenged only after a transient has been initiated and is not a contributor to the probability of any transient or accident. The transients which involve pressure increases which would potentially challenge the safety valves have been analyzed to determine the consequences of delayed or premature valve actuation at the extremes of the new setpoint tolerances. These analyses show that all applicable acceptance criteria are met using the wider tolerances.

The proposed amendment will not result in the creation of any new accident not previously evaluated. As noted above, the setpoint tolerance only affects the time at which the safety valve opens following or during a transient, and is not a contributor to the probability of an accident.

The proposed amendment will not result in a significant decrease in a margin of safety. The limiting transient in each accident category has been analyzed to determine the effect of the change in lift setpoint tolerance on the transient. In each case, the results of the analyses met all acceptance criteria.

Based on the above, it is concluded that no significant hazards exist.

LOW STEAM LINE SETPOINT PRESSURE CHANGE

Changing the Low Steam Line Pressure setpoint and removal of dynamic compensation will not increase the probability or consequences of any previously-reviewed accident. The higher steam line pressure setpoint is consistent with all licensing basis safety analyses. This change, in conjunction with the removal of the dynamic compensation of the steam pressure signal, is intended to reduce or eliminate spurious Engineered Safeguards Features (ESF) actuations which are caused by minor (but rapid) pressure decreases in the secondary system.

The proposed amendment will not result in a new accident not previously reviewed. A change in steam line pressure is a response to an existing transient condition, rather than a precursor or initiating event. A change in the steam line pressure setpoint is also not a precursor or initiating event.

The proposed amendment will not result in a significant decrease in a margin of safety. The reanalysis of the steam line break accident which was performed shows that all imposed Condition II acceptance criteria are met.

Based on the above, it is concluded that no significant hazards exist.

FEEDWATER AND MAIN STEAM LINE ISOLATION VALVE STROKE TIME

The proposed changes to the valve stroke times in Tables 3.3-5 and Table 3.6-2a will not significantly increase the probability or consequences of any previously evaluated accident. The effects of the delays in isolation times on the various transients affected have been analyzed and found to be acceptable. Since these valves do not recieve a containment isolation signal, and no credit is taken for operation of these valves in the dose analysis for a containment isolation function, a maximum stroke time does not apply for containment isolation.

The proposed changes will not significantly increase the possibility of a new accident not previously evaluated. Feedwater and main steam isolation are responses to ongoing transients, rather than initiators or precursors of transients. No equipment or component reconfiguration will occur as a result of this change.

The proposed changes will not significantly decrease any margin of safety. As noted above, the effects of the longer isolation times have been evaluated and found to be acceptable.

Based on the above, it is concluded that no significant hazards exist.

REVISE LIST OF ACCIDENTS REQUIRING REEVALUATION IN THE EVENT OF AN INOPERABLE RCCA

The proposed change to Table 3.3-1 will not change the probability or consequences of any accident or reduce any safety margin, because the table simply lists accident analyses which must be reevaluated in the event of an inoperable rod cluster control assembly (RCCA). The activities involved are analytical only, and do not introduce any operational considerations. Revision of the table to more accurately define the affected analyses is an administrative effort related to activities (analyses) which are conducted offsite after the fact of a postulated inoperable RCCA.

Based on the above, it is concluded that no significant hazards exist.

ENVIRONMENTAL IMPACT STATEMENT

The proposed Technical Specification change has been reviewed against the criteria of 10 CFR 51.22 for environmental considerations. As shown above, the proposed change does not involve any significant hazards consideration, nor increase the types and amounts of effluents that may be released offsite, nor increase the individual or cumulative occupational radiation exposures. Based on this, the proposed Technical Specification change meets the criteria given in 10 CFR 51.22 (c) (9) for categorical exclusion from the requirement for an Environmental Impact Statement.