LESSONS LEARNED TECHNICAL SPECIFICATIONS

CROSS-REFERENCE

NUREG-0578 SECTION NO.	ABBREVIATED TITLE	PROPOSED TECHNICAL SPECIFICATION PAGE(S)
2.1.1	Emergency power supply requirement	
	Pressurizer level	Already in CR-3
	PORV and block valve indicators	3/4 3-38
		3/4 3-39 3/4 3-39a
	PORV and block valve	3/4 4-4a
2.1.3.a	Direct indication of valve position	3/4 3-38
		3/4 3-39 3/4 3-39a
2.1.3.b	Inadequate core cooling instrumentation	3/4 3-38
		3/4 3-39
2.1.4	Diverse containment isolation	3/4 3-12
		3/4 3-16 3/4 3-17 3/4 3-20
2.2.1.b	Shift technical advisor	6-4
		6-5

Technical Specification Change Request No. 64, Revision 1, Appendix A

Replace pages:		With:	
	IV		IV
	3/4 3-10		3/4 3-10
	3/4 3-12		3/4 3-12
	3/4 3-15		3/4 3-15
	3/4 3-16		3/4 3-16
	3/4 3-16a		3/4 3-16a
	3/4 3-17		3/4 3-17
	3/4 3-17a		3/4 3-17a
	3/4 3-18		3/4 3-18
	3/4 3-20		3/4 3-20
	3/4 3-38		3/4 3-38
	3/4 3-39		3/4 3-39
	3/4 4-3		3/4 4-3
	3/4 4-4		3/4 4-4
			3/4 4-4a
	3/4 7-5		3/4 7-5
	B 3/4 4-1		B 3/4 4-1
	B 3/4 4-3		B 3/4 4-3
	6-4		6-4
	6-5		6-5

Proposed Change

The proposed change incorporates the recommendations of the following NUREG-0578 items into the Crystal River Unit 3 Technical Specifications.

2.1.1	Emergency Power Supply Requirement (partially)
2.1.3.a	Direct Indication of Valve Position
2.1.3.b	Instrumentation for Inadequate Core Cooling
2.2.1.b	Shift Technical Advisor

The proposed change also deletes the ±1% acceptance value associated with the relief valve setpoint (specifications 3.4.2 and 3.4.3.1), due to inconsistency with ASME Code, and impropriety in having accuracy range in a Limiting Condition for Operation.

Reasons for Proposed Change

These changes are in response to Mr. Darrell Eisenhut's (Director, USNRC, Division of Licensing) letter of July 2, 1980. Items included in Mr. Eisenhut's letter which were not included in this Technical Specification Change Request are discussed in the letter to which this is attached.

Safety Analysis Justifying the Proposed Change

The changes proposed include installed modifications into the Technical Specifications. These modifications were required as part of the implementation of NUREG-0578, Category A items. The Staff reviewed these modifications as part of their review of all Lessons Learned changes and have issued a Safety Evaluation (dated May 5, 1980) on the system, as modified.

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION					
a. High Pressure Injection					
1. Manual Initiation	2	1	2	1, 2, 3, 4	13
2. Reactor Bldg. Pressure High	3	2	2	1, 2, 3	9#
3. RCS Pressure Low	3	2	2	1, 2, 3*	9#
4. RCS Pressure Low-Low	3	2	2	1, 2, 3**	9#
5. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	10
b. Low Pressure Injection					
1. Manual Initiation	2	1	2	1, 2, 3, 4	13
2. Reactor Bldg. Pressure High	3	2	2	1, 2, 3	9#
3. RCS Pressure Low-Low	3	2	2	1, 2, 3**	9#
4. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	10
2. REACTOR BLDG. COOLING					
a. Manual Initiation	2	1	2	1, 2, 3, 4	13
b. Reactor Bldg. Pressure High	3	2	2	1, 2, 3	9#
c. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	10

TABLE 3.3-3 (Cont'd)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
5. REACTOR BLDG. ISOLATION					
a. Manual Initiation	2	1	2	1, 2, 3, 4	13
b. Reactor Bldg. Pressure High	3	2	2	1, 2, 3	9#
c. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	10
d. Manual Initiation (HPI Isolation)	2	1	2	1, 2, 3, 4	13
e. RCS Pressure Low (HPI Isolation)	3	2	2	1, 2, 3*	13
f. Automatic Actuation Logic (HPI Isolation)	2	1	2	1, 2, 3, 4	10

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION		
a. High Pressure Injection ES Actuation "A" and "B"		
1. Manual Initiation	Not Applicable	Not Applicable
2. Reactor Bldg. Pressure High	$\leq$ 4 psig	$\leq$ 4 psig
3. RCS Pressure Low	$\geq$ 1500 psig	$\geq$ 1500 psig
4. RCS Pressure Low-Low	$\geq$ 500 psig	$\geq$ 500 psig
5. Automatic Actuation Logic	Not Applicable	Not Applicable
b. Low Pressure Injection ES Actuation "A" and "B"		
1. Manual Initiation	Not Applicable	Not Applicable
2. Reactor Bldg. Pressure High	$\leq$ 4 psig	$\leq$ 4 psig
3. RCS Pressure Low-Low	$\geq$ 500 psig	$\geq$ 500 psig
4. Automatic Actuation Logic	Not Applicable	Not Applicable
2. REACTOR BLDG. COOLING		
a. ES Actuation "A" and "B"		
1. Manual Initiation	Not Applicable	Not Applicable
2. Reactor Bldg. Pressure High	$\leq$ 4 psig	$\leq$ 4 psig
3. Automatic Actuation Logic	Not Applicable	Not Applicable
b. ES Actuation Indication "AB"		
1. Automatic Actuation Logic	Not Applicable	Not Applicable

TABLE 3.3-4 (Cont'd)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
3. REACTOR BLDG. SPRAY		
a. Reactor Bldg. Pressure High-high	≤ 30 psig	≤ 30 psig
Coincident with HPI Signal	See 1.a, 2, 3, 4	See 1.a, 2, 3, 4
b. Automatic Actuation Logic	Not Applicable	Not Applicable
4. OTHER SAFETY SYSTEMS		
a. Reactor Bldg. Purge Exhaust Duct Isolation on High Radioactivity		
Gaseous	*	Not Applicable
b. Steam Line Rupture Matrix		
1. Low SG Pressure	> 600 psig	> 600 psig
2. Automatic Actuation Logic	Not Applicable	Not Applicable
5. REACTOR BLDG. ISOLATION		
a. ES Actuation "A" and "B"		
1. Manual Initiation	Not Applicable	Not Applicable
2. Reactor Bldg. Pressure High	≤ 4 psig	≤ 4 psig
3. Automatic Actuation Logic	Not Applicable	Not Applicable
4. Manual Initiation (HPI Isolation)	Not Applicable	Not Applicable

*Determined by requirements of Appendix "B" Tech. Specs. Section 2.4.2 - Crystal River Unit 3 Operating License No. DPR-72.

TABLE 3.3-4 (Cont'd)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
5. RCS Pressure Low (HPI Isolation)	$<$ 1500 psig	$<$ 1500 psig
6. Automatic Actuation Logic (HPI Isolation)	Not Applicable	Not Applicable

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS*
1. Manual	
a. High Pressure Injection	Not Applicable
b. Low Pressure Injection	Not Applicable
c. Reactor Building Cooling	Not Applicable
d. Reactor Building Isolation	Not Applicable
e. Reactor Building Spray	Not Applicable
f. Reactor Building Purge Isolation	Not Applicable
g. Steam Line Rupture Matrix	
1. Emergency Feedwater Actuation	Not Applicable
2. Feedwater Isolation	Not Applicable
3. Steam Line Isolation	Not Applicable
h. HPI Isolation	Not Applicable
2. Reactor Building Pressure-High	
a. High Pressure Injection	25*
b. Low Pressure Injection	25*
c. Reactor Building Cooling	25*
d. Reactor Building Isolation	60*
3. Reactor Building Pressure High-High (with HPI signal)	
a. Reactor Building Spray	56*
4. RCS Pressure Low	
a. High Pressure Injection	25*
b. HPI Isolation	60*
5. RCS Pressure Low-Low	
a. High Pressure Injection	25*
b. Low Pressure Injection	25*
6. Low Steam Generator Pressure	
a. Feedwater Isolation	34
b. Steam Line Isolation	5

TABLE 3.3-5 (Cont'd)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS*
7. Containment Radioactivity-High	
a. Reactor Building Purge Isolation	15 *

*Diesel Generator starting and sequence loading delays included. Response time limit includes movement of valves and attainment of pump or blower discharge pressure.

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION				
a. High Pressure Injection				
1. Manual Initiation	N/A	N/A	M(1)	1, 2, 3, 4
2. Reactor Bldg. Pressure High	S	R	M(2)	1, 2, 3
3. RCS Pressure Low	S	R	M	1, 2, 3
4. RCS Pressure Low-Low	S	R	M	1, 2, 3
5. Automatic Actuation Logic	N/A	N/A	M(3)	1, 2, 3, 4
b. Low Pressure Injection				
1. Manual Initiation	N/A	N/A	M(1)	1, 2, 3, 4
2. Reactor Bldg. Pressure High	S	R	M(2)	1, 2, 3
3. RCS Pressure Low-Low	S	R	M	1, 2, 3
4. Automatic Actuation Logic	N/A	N/A	M(3)	1, 2, 3, 4
2. REACTOR BLDG. COOLING				
a. Manual Initiation	N/A	N/A	M(1)	1, 2, 3, 4

TABLE 4.3-2 (Cont'd)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
5. REACTOR BLDG. ISOLATION				
a. Manual Initiation	N/A	N/A	M(1)	1, 2, 3, 4
b. Reactor Bldg. Pressure High	S	R	M(2)	1, 2, 3
c. Automatic Actuation Logic	N/A	N/A	M(3)	1, 2, 3, 4
d. Manual Initiation (HPI Isolation)	N/A	N/A	M(1)	1, 2, 3, 4
e. RCS Pressure Low (HPI Isolation)	S	R	M	1, 2, 3
f. Automatic Actuation Logic (HPI Isolation)	N/A	N/A	M(3)	1, 2, 3, 4

TABLE 3.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT		MEASUREMENT RANGE	MINIMUM CHANNELS OPERABLE
1.	Power Range Nuclear Flux	0-125%	2
2.	Reactor Building Pressure	0-70 psia	2
3.	Source Range Nuclear Flux	10^{-1} to 10^{6} cps	2
4.	Reactor Coolant Outlet Temperature	520 °F - 620 °F	2 per loop
5.	Reactor Coolant Total Flow	0-110% full flow	1
6.	RC Loop Pressure	0-2500 psig 0-600 psig 1700-2500 psig	2 1 2
7.	Pressurizer Level	0-320 inches	2
8.	Steam Generator Outlet Pressure	0-1200 psig	2/steam generator
9.	Steam Generator Operating Range Level	0-100%	2/steam generator
10.	Borated Water Storage Tank Level	0-50 feet	2
11.	Startup Feedwater Flow	$0-1.5 \times 10^{6}$ lb/hr.	2
12.	Reactor Coolant System Subcooling Margin Monitor	-658 F° to +668 F°	1
13.	PORV Position Indicator (Primary Detector)	N/A	1
14.	PORV Position Indicator (Backup Detector)	N/A	0
15.	PORV Block Valve Position Indicator	N/A	0
16.	Safety Valve Position Indicator (Primary Detector)	N/A	1/Valve
17.	Safety Valve Position Indicator (Backup Detector)	N/A	0

TABLE 4.3-7

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Power Range Nuclear Flux	M	Q*
2. Reactor Building Pressure	M	R
3. Source Range Nuclear Flux	M	R*
4. Reactor Coolant Outlet Temperature	M	R
5. Reactor Coolant Total Flow Rate	M	R
6. RC Loop Pressure	M	R
7. Pressurizer Level	M	R
8. Steam Generator Outlet Pressure	M	R
9. Steam Generator Level	M	R
10. Borated Water Storage Tank Level	M	R
11. Startup Feedwater Flow Rate	M	R
12. Reactor Coolant System Subcooling Margin Monitor	M	R
13. PORV Position Indicator (Primary Monitor)	M	R
14. PORV Position Indicator (Backup Detector)	M	R
15. PORV Block Valve Position Indicator	M	R
16. Safety Valve Position Indicator (Primary Detector)	M	R
17. Safety Valve Position Indicator (Backup Detector)	M	R

*Neutron detectors may be excluded from CHANNEL CALIBRATION.

RELIEF VALVES - SHUTDOWN

CODE SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.4.2 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2500 psig.

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 DHR loop into operation.

SURVEILLANCE REQUIREMENTS

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

RELIEF VALVES - OPERATING

CODE SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.4.3.1 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2500 psig.

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 HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

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

REACTOR COOLANT SYSTEM

RELIEF VALVES - OPERATING

POWER OPERATED RELIEF VALVE

LIMITING CONDITION FOR OPERATION

3.4.3.2 The Power Operated Relief Valve (PORV) and its associated block valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the PORV inoperable, within 1 hour, either restore the PORV to OPERABLE status or close and remove power from the associated block valve; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 30 hours.

b. With the associated block valve inoperable, within 1 hour, either restore the associated block valve to OPERABLE status, or close and remove power from the associated block valve; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The PORV shall be demonstrated OPERABLE at least once per 18 months by performance of a CHANNEL CALIBRATION.

4.4.3.2 The associated block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel.

4.4.3.3 The emergency power supply for the PORV and associated block valve shall be demonstrated OPERABLE at least once per 18 months by transferring motive and control power from the normal to the emergency power supply and operating the valves through a complete cycle of full travel.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that each valve (manual, power operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. 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 an emergency feedwater actuation test signal.

2. Verifying that each emergency feedwater pump starts automatically upon receipt of an emergency feedwater actuation test signal.

3. Verifying that the operating air accumulators for FWV-39 and FWV-40 maintain $\geq$27 psig for at least one hour when isolated from their air supply.

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with both reactor coolant loops in operation, and maintain DNBR above 1.30 during all normal operations and anticipated transients. With one reactor coolant pump not in operation in one loop, THERMAL POWER is restricted by the Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE, ensuring that the DNBR will be maintained above 1.30 at the maximum possible THERMAL POWER for the number of reactor coolant pumps in operation or the local quality at the point of minimum DNBR equal to 22%, whichever is more restrictive.

A single reactor coolant loop provides sufficient heat removal capability for removing core decay heat while in HOT STANDBY; however, single failure considerations require placing a DHR loop into operation in the shutdown cooling mode if component repairs and/or corrective actions cannot be made within the allowable out-of-service time.

3/4.4.2 RELIEF VALVES - SHUTDOWN

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psig. Each safety valve is designed to relieve 317,973 lbs per hour of saturated steam at the valve's setpoint.

The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating DHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from any transient.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.3 RELIEF VALVES - OPERATING

The power operated relief valve (PORV) operates to relieve RCS pressure below the setting of the pressurizer code safety valves. This relief valve has a remotely operated block valve to provide a positive shutoff capability should the PORV become inoperable. The electrical power for both the relief valve and the block valve is capable of being supplied from an emergency power source to ensure the ability to seal this possible RCS leakage path.

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and power operated relief valves against water relief.

The low level limit is based on providing enough water volume to prevent a pressurizer low level or a reactor coolant system low pressure condition that would actuate the Reactor Protection System or the Engineered Safety Feature Actuation System as a result of a reactor scram. The high level limit is based on maximum reactor coolant inventory assumed in the safety analysis.

The power operated relief valves and steam bubble function to relieve RCS pressure during all design transients. Operation of the power operated relief valves minimizes the undesirable opening of the spring-loaded pressurizer code safety valves.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these chemistry limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 1 GPM). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 1 GPM can be detected by monitoring the secondary coolant. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.1 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

The steam generator water level limits are consistent with the initial conditions assumptions in the FSAR.

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION#

LICENSE CATEGORY	APPLICABLE MODES	
	1, 2, 3, & 4	5 & 6
SOL	1	1*
OL	2	1
Non-Licensed	3	1
Operations Tech. Advisor	1	0

*Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling Individual supervising CORE ALTERATIONS after the initial fuel loading.

#Shift crew composition may be less that the minumum requirement for a period of time not to exceed 2 hours in order to accomodate unexpected absence of on duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the Chemistry and Radiation Protection Engineer who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and the Operations Technical Advisor, who shall have a Bachelor's degree, or the equivalent, in a scientific or engineering discipline with specific training in plant design and response and analysis of the plant for transients and accidents.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Nuclear Plant Manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 19 CFR Part 55.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Nuclear Plant Manager and shall meet or exceed the requirements of Section 27 of the NFPA Code-1976, except for Fire Brigade training sessions which shall be held at least quarterly.

6.5 REVIEW AND AUDIT

6.5.1 PLANT REVIEW COMMITTEE (PRC)

FUNCTION

6.5.1.1 The Plant Review Committee shall function to advise the Nuclear Plant Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The Plant Review Committee shall be composed of the:

Chairman:	Technical Services Superintendent
Member:	Operations Superintendent
Member:	Technical Support Engineer
Member:	Maintenance Superintendent
Member:	Chemistry and Radiation Protection Engineer
Member:	At large (Designated by Chairman)

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PRC Chairman to serve on a temporary basis; no more than two alternates shall participate as voting members in PRC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The PRC shall meet at least once per calendar month and as convened by the PRC Chariman or his designated alternate.

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with:

a. A steam bubble,

b. A water level between 40 and 290 inches, and

c. At least 126 kW of pressurizer heaters.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the pressurizer inoperable, be in at least HOT STANDBY with the control rod drive trip breakers open within 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.4.1 The pressurizer shall be demonstrated OPERABLE by verifying pressurizer level to be within limits at least once per 12 hours.

4.4.4.2 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by manually transferring power from the normal to the emergency power supply and energizing the heaters.