UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

FLORIDA POWER CORPORATION
CITY OF ALACHUA
CITY OF BUSHNELL
CITY OF GAINESVILLE
CITY OF KISSIMMEE
CITY OF LEESBURG
CITY OF NEW SMYRNA BEACH AND UTILITIES COMMISSION, CITY OF NEW SMYRNA BEACH
CITY OF OCALA
ORLANDO UTILITIES COMMISSION AND CITY OF ORLANDO
SEBRING UTILITIES COMMISSION
SEMINOLE ELECTRIC COOPERATIVE, INC.
CITY OF TALLAHASSEE

DOCKET NO. 50-302

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 38
License No. DPR-72

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Florida Power Corporation, et al. (the licensees) dated September 15, 1980, as revised December 31, 1980, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-72 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No.38 , are hereby incorporated in the license. Florida Power Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 17, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 38

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages	
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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.1 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels 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 channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-3 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint Value.

b. With an ESFAS instrumentation channel inoperable, take the action shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1.1 Each ESFAS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the MODES and at the frequencies shown in Table 4.3-2.

4.3.2.1.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3.

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
3. REACTOR BLDG. SPRAY					
a. Reactor Bldg. Pressure High-High coincident with HPI Signal	3	2	2	1, 2, 3	12
b. Automatic Actuation Logic	2	1	2	1, 2, 3	10
4. OTHER SAFETY SYSTEMS					
a. Reactor Bldg. Purge Exhaust Duct Isolation on High Radioactivity					
Gaseous	1	1	1	1, 2, 3, 4	11#

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
b. Steam Line Rupture Matrix					
1. Low SG Pressure	2 per steam generator	1 per steam generator	2 per steam generator	1, 2, 3***	10
2. Automatic Actuation Logic	1 per steam generator	1 per steam generator	1 per steam generator	1, 2, 3	10
c. Emergency Feedwater					
1. Main Feedwater Pump Turbines A and B Control Oil Low	2	2	2	1, 2, 3##	10
2. OTSG A and B Level Low-Low	2	2	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 3.3-3 (Continued)

TABLE NOTATION

*Trip function may be bypassed in this MODE with RCS pressure below 1700 psig. Bypass shall be automatically removed when RCS pressure exceeds 1700 psig.

**Trip function may be bypassed in this MODE with RCS pressure below 900 psig. Bypass shall be automatically removed when RCS pressure exceeds 900 psig.

***Trip function may be bypassed in this MODE with steam generator pressure below 725 psig. Bypass shall be automatically removed when steam generator pressure exceeds 765 psig.

#The provisions of Specification 3.0.4 are not applicable.

##Trip function may be bypassed in this MODE prior to stopping the operating main feedwater pump. Bypass shall be manually removed after starting the first main feedwater pump.

ACTION STATEMENTS

ACTION 9 - With the number of OPERABLE Channels one less than the Total Number of Channels operation may proceed until performance of the next required CHANNEL FUNCTIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 10 - With the number of OPERABLE channels one less than the Total Number of Channels, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing per Specification 4.3.2.1.1.

ACTION 11 - With less than the Minimum Channels OPERABLE, operation may continue provided the containment purge and exhaust valves are maintained closed.

ACTION 12 - With the number of OPERABLE Channels one less than the Total Number of Channels operation may proceed provided the inoperable channel is placed in the bypassed condition and the minimum channels OPERABLE required is demonstrated within 1 hour; one additional channel may be bypassed for up to 2 hours for Surveillance testing per Specification 4.3.2.1.

ACTION 13 - With the number of OPERABLE Channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEMS INSTRUMENTATION 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 SYSTEMS INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
3. REACTOR BLDG. SPRAY		
a. Reactor Bldg. Pressure High-High coincident with HPI Signal	$\leq$ 30 psig See 1.a.2, 3, 4	$\leq$ 30 psig 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	$\geq$ 600 psig	$\geq$ 600 psig
2. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Emergency Feedwater		
1. Main Feedwater Pump Turbines A and B Control Oil Low	$\geq$ 55 psig	$\geq$ 55 psig
2. OTSG A and B Level Low-Low	$\geq$ 18 inches	$\geq$ 18 inches

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

TABLE 3.3-4 (Cont'd)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
5. REACTOR BLDG. ISOLATION		
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
4. Manual Initiation (HPI Isolation)	Not Applicable	Not Applicable
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 *
8. Main Feedwater Pump Turbines A and B Control Oil Low	
a. Emergency Feedwater Actuation	Not Applicable
9. OTSG A and B Level Low-Low	
a. Emergency Feedwater Actuation	Not Applicable

*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
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

TABLE 4.3-2 (Cont'd)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEMS INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
3. REACTOR BLDG. SPRAY				
a. Reactor Bldg. Pressure High-High coincident with HPI Signal	S	R	M(4)	1, 2, 3
b. Automatic Actuation Logic	N/A	N/A	M(3)	1, 2, 3
4. OTHER SAFETY SYSTEMS				
a. Reactor Bldg. Purge Exhaust Duct Isolation on High Radioactivity				
1. Gaseous	S	Q	M	All Modes
b. Steam Line Rupture Matrix				
1. Low SG Pressure	N/A	R	N/A	1, 2, 3
2. Automatic Actuation Logic	N/A	N/A	M(3)	1, 2, 3
c. Emergency Feedwater				
1. Main Feedwater Pump Turbines A and B Control Oil Low	S	R	N/A	1, 2, 3
2. OTSG A and B Level Low-Low	S	R	N/A	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

POST-ACCIDENT INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The post-accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE with readouts and recorders in the control room.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With the number of OPERABLE post-accident monitoring channels less than required by Table 3.3-10, either restore the inoperable channel to OPERABLE status within 30 days, or be in HOT SHUTDOWN within the next 12 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each post-accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-7.

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	$0-1.5x10^{6}$ 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 Detector)	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.

INSTRUMENTATION

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.7 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: Whenever equipment in that fire detection zone is required to be OPERABLE.

ACTION:

With one or more of the fire detection instrument(s) shown in Table 3.3-11, inoperable:

a. Within 1 hour, establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, and

b. Restore the inoperable instrument(s) to OPERABLE status within 14 days or in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the malfunction and the plans and schedule for restoring the instrument(s) to OPERABLE status.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7.1 Each of the above fire detection instruments shall be demonstrated OPERABLE at least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST.

4.3.3.7.2 The circuitry associated with the detector alarms listed in Table 3.3-11 shall be demonstrated OPERABLE at least once per 6 months for all National Fire Protection Association (NFPA) Code 72D Class B supervised circuits.

4.3.3.7.3 The non-supervised circuits between the local panels and the control room for the detectors listed in Table 3.3-11 shall be demonstrated OPERABLE at least once per 31 days.

POOR ORIGINAL

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 $\pm$ 1%.

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.

REACTOR COOLANT SYSTEM

RELIEF VALVES - OPERATING

CODE SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.4.3. All pressurizer code safety valves shall be OPERABLE with a lift setting of 2500 psig $\pm$ 1%.

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.

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.

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE with a water level between 18 and 360 inches.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one or more steam generators inoperable due to steam generator tube imperfections, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

b. With one or more steam generators inoperable due to the water level being outside the limits, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1. Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

TABLE 3.6-1

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	ISOLATION TIME (seconds)
A. CONTAINMENT ISOLATION		
1. BSV-27 check #	closed dur. nor. operation and open dur. RB spray	NA
BSV-3 #	"	60
BSV-26 check #	"	NA
BSV-4 #	"	60
2. CAV-126(A)	iso. CA sys. fr. RC letdn.	60
CAV-1**(A)	iso. CA sys. fr. pzr.	60
CAV-3 (A)	"	60
CAV-2 (B)	iso. CA sys. fr. RB	60
CAV-4 # (A)	isolate liquid sampling system	60
CAV-6 # (B)	"	60
CAV-5 # (A)	"	60
CAV-7 # (B)	"	60
3. CFV-20 check	iso. N_2 supply fr. CFT-1A	NA
CFV-28 (A/B)	"	60
CFV-17 check	iso. N_2 supply fr. CFT-1B	NA
CFV-27 (A/B)	"	60
CFV-18 check	iso. MU system fr. CFT-1B	NA
CFV-26 (A/B)	"	60
CFV-19 check	iso. MU system fr. CFT-1A	NA
CFV-25 (A/B)	"	60
CFV-42 (B)	iso. liquid sampling fr. CF system	60
CFV-15 (A)	iso. WD sys. fr. CF tanks	60
CFV-16 (A)	"	60
CFV-29 (B)	"	60
CFV-11 (A)	iso. CF tanks fr. liquid sampling system	60
CFV-12 (A)	"	60

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

	VALVE NUMBER	FUNCTION	ISOLATION TIME (seconds)
4.	CIV-41	iso. CI sys. fr. RB	60
	CIV-40	"	60
	CIV-34	"	60
	CIV-35	"	60
5.	DHV-93 check	iso. DH system fr. pzr.	NA
	CHV-91	"	60
	DHV-43 #	iso. DH sys. fr. RB sump	120
	DHV-42 #	"	120
	DHV-4# & 41#	iso. DH sys. fr. RC	120
	DHV-6 #	iso. DH system from Reactor Vessel	60
	DHV-5 #	"	60
6.	DWV-162 check	iso. DW system fr. RB	NA
	DWV-160 (A/B)	"	60
7.	FWV-44 check #	iso. feedwater from RCSG-1A	NA
	FWV-45 check #	"	NA
	FWV-43 check #	iso. feedwater from RCSG-1B	NA
	FWV-45 check #	"	NA
8.	MSV-130 #(A/B)	iso. MDT-1 from RCSG-1A	60
	MSV-148 #(A/B)	iso. MDT-1 from RCSG-1B	60
	MSV-411 #	iso. main steam lines from RCSG-1A	60
	MSV-412 #	iso. main steam lines from RCSG-1A	60
	MSV-413 #	iso. main steam lines from RCSG-1B	60
	MSV-414 #	iso. main steam lines from RCSG-1B	60

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER		FUNCTION	ISOLATION TIME (seconds)
9.	MUV-40 (A)	iso. MU system from RC	60
	MUV-41 (A)	"	60
	MUV-49 (B)	"	60
	MUV-253	"	60
	MUV-261	iso. MU system from control bleed-off	60
	MUV-260	"	60
	MUV-259	"	60
	MUV-258	"	60
	MUV-163 check #	open during HPI and closed dur. nor. operation	NA
	MUV-25 #	"	60
	MUV-164 check #	"	NA
	MUV-26 #	"	60
	MUV-160 check #	"	NA
	MUV-23 #	"	60
	MUV-161 check #	"	NA
	MUV-24 #	"	60
	MUV-27 #	open dur. nor. operation and closed during RB Isolation	60
10.	SWV-39 #	iso. NSCCC from AHF-1C	60
	SWV-45 #	"	60
	SWV-35 #	iso. NSCCC from AHF-1A	60
	SWV-41 #	"	60
	SWV-37 #	iso. NSCCC from AHF-1B	60
	SWV-43 #	"	60
	SWV-48 #	isolate NSCCC from MUHE-1A & 1B and WDT-5	60
	SWV-47 #	"	60
	SWV-49 #	"	60
	SWV-50 #	"	60
	SWV-80 #	iso. NSCCC from RCP-1A	60
	SWV-84 #	"	60
	SWV-82 #	iso. NSCCC from RCP-1C	60
	SWV-86 #	"	60

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	ISOLATION TIME (seconds)
SWV-81 #	iso. NSCCC from RCP-1D	60
SWV-85 #	"	60
SWV-79 #	iso. NSCCC from RCP-1B	60
SWV-83 #	"	60
SWV-109#	iso. NSCCC from DRRD-1	60
SWV-110#	"	60
11. WDV-4 (B)	iso. WDT-4 from RB sump	60
WDV-3 (A)	"	60
WDV-60 (A)	iso. WDT-4 from WDT-5	60
WDV-61 (B)	"	60
WDV-94 (A)	iso. WDT-4 from WDP-8	60
WDV-62 (B)	"	60
WDV-406 (A)	iso. waste gas disposal from vents in RC system	60
WDV-405 (B)	"	60
12. WSV-3	iso. containment monitoring system from RB	60
WSV-4	"	60
WSV-5	"	60
WSV-6	"	60
B. CONTAINMENT PURGE AND EXHAUST		
1. AHV-1C (A)	iso. pur. sup. system fr. RB	60
AHV-1D (B)	"	60
AHV-1B (A)	iso. pur. exhaust system fr. RB	60
AHV-1A (B)	"	60
C. MANUAL		
1. IAV-28	iso. IA from RB	NA
IAV-29	"	NA
2. LRV-50	iso. leak rate test system from RB	NA
LRV-36	"	NA

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	ISOLATION TIME (seconds)
LRV-51	iso. atmos. vent and RB	NA
LRV-35 & 47	purge exhaust system from RB	NA
	"	
LRV-49	iso. atmos. vent from RB	NA
LRV-38 & 52	"	NA
LRV-45	iso. LR test panel from RB	NA
LRV-44	"	NA
LRV-46	"	NA
3. MSV-146#	iso. misc. waste storage tank from RCSG-1B	NA
4. NGV-62	iso. NG system from steam generators	NA
NGV-81 #	"	NA
NGV-82	iso. NG system from pzr.	NA
5. SAV-24	iso. SA from RB	NA
SAV-23 & 122	"	NA
6. SFV-18	iso. SF system	NA
SFV-19	"	NA
SFV-119#	iso. Fuel Transfer tubes from F.T. Canal	NA
SFV-120#	"	NA
7. WSV-1	iso. containment monitoring system from RB	NA
WSV-2	"	NA
D. PENETRATIONS REQUIRING TYPE B TESTS		
Blind Flange 119	iso. RB	NA
Blind Flange 120	"	NA
Blind Flange 202	"	NA

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	ISOLATION TIME (seconds)
Blind Flange 348	iso. fuel transfer tube from Transfer Canal	NA
Blind Flange 436	"	NA
Equipment Hatch	iso. RB	NA
Personnel Hatch	iso. RB	NA

\# Not subject to Type C Leakage Test

**The provisions of Specification 3.0.4 are not applicable until startup for Cycle 3 operation. Isolation valves closed to satisfy the requirements of Specification 3.6.3.1 ACTION b. and c. may be re-opened on an intermittent basis under administrative control for up to 4 hours in any 24 hour period as necessary for sampling.

(A) Isolates on Diverse Isolation Actuation Signal A
(B) Isolates on Diverse Isolation Actuation Signal B
(A/B) Isolates on Diverse Isolation Actuation Signal A or B

3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN ANALYZERS

LIMITING CONDITION FOR OPERATION

3.6.4.1 A containment hydrogen analyzer and a gas chromatograph shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one hydrogen analysis device inoperable, restore the inoperable device to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each hydrogen analysis device shall be demonstrated OPERABLE at least once per 92 days on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using sample gases containing:

a. One volume percent hydrogen, balance nitrogen, and

b. Four volume percent hydrogen, balance nitrogen.

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.

3. Verifying that the emergency feedwater ultrasonic flow rate detector is zero-checked.

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 the steam turbine driven pump and the motor driven pump start automatically:

a. Upon receipt of an emergency feedwater actuation OTSG A and B level low-low test signal, and

b Upon receipt of an emergency feedwater actuation main feedwater pump turbines A and B control oil low 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.

PLANT SYSTEMS

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with a minimum contained volume of 150,000 gallons of water.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the condensate storage tank inoperable, within 4 hours either:

a. Restore the CST to OPERABLE status or be in HOT SHUTDOWN within the next 12 hours, or

b. Demonstrate the OPERABILITY of the condenser hotwell as a backup supply to the emergency feedwater system and restore the condensate storage tank to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume to be within its limits when the tank is the supply source for the emergency feedwater pumps.

4.7.1.3.2 The condenser hotwell shall be demonstrated OPERABLE at least once per 12 hours by verifying a minimum contained volume of 150,000 gallons of water whenever the condenser hotwell is the supply source for the emergency feedwater system.

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.

Operational experience has shown that tube defects can be the result of unique operating conditions and/or physical arrangements in specific limited areas of the steam generators (for example, tubes adjacent to the open inspection lane or tubes whose 15th tube support plate hole is not broached but drilled). A full inspection of all of the tubes in such specific limited areas will provide complete assurance that degraded or defective tubes in these areas are detected. Because no credit is taken for these distinctive tubes in the constitution of the first sample or its results, the requirements for the first sample are unchanged. This requirement is essentially equivalent to and meets the intent of the requirements set forth in Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes", Rev. 1, July 1975, and does not reduce the margin of safety provided by those requirements.

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.

POOR ORIGINAL

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to detect and monitor leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems", May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

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 RCS, the UNIDENTIFIED LEAKAGE portion of this 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.

The total steam generator tube leakage limit of 1 GPM for all steam generators ensures that the dosage contribution from tube leakage will be limited to a small fraction of Part 100 limits 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 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 limit of 10 GPM restricts operation with a total RCS leakage from all RC pump seals in excess of 10 GPM.

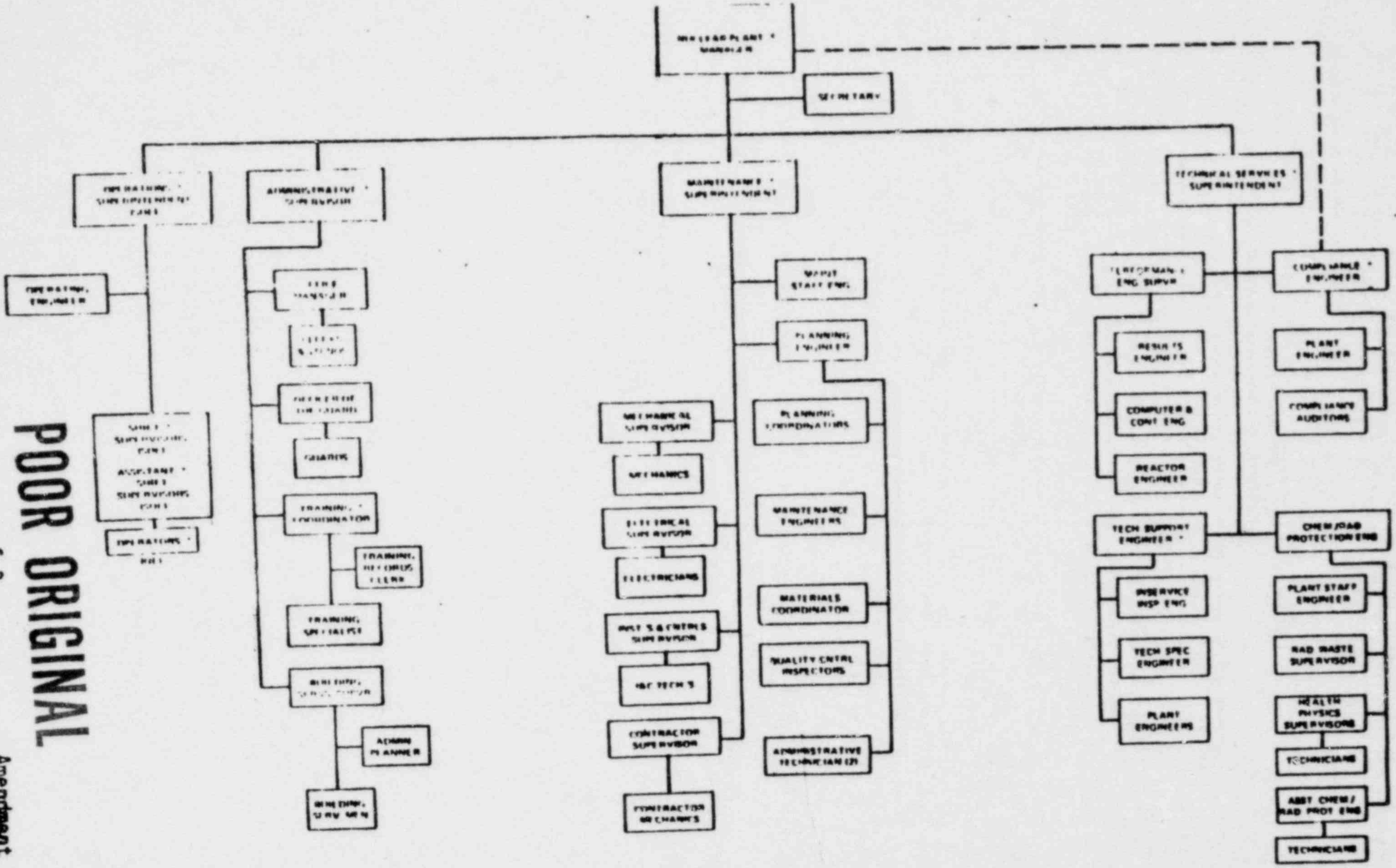

FIGURE 6.2-2 FACILITY ORGANIZATION

LEGEND

POOR ORIGINAL

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION#

LICENSE CATEGORY	APPLICABLE MODES	
	1, 2, 3, & 4	5 & 6
SOL	2	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 than the minumum requirement for a period of time not to exceed 2 hours in order to accommodate 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 10 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 Chairman or his designated alternate.

QUORUM

6.5.1.5 A quorum of the PRC shall consist of the Chairman or his designated alternate and four members including alternates.

RESPONSIBILITIES

6.5.1.6 The Plant Review Committee shall be responsbile for:

a. Review of 1) all procedures required by Specification 6.8 and changes thereto, 2) any other proposed procedures or changes thereto as determined by the Nuclear Plant Manager to affect nuclear safety.

b. Review of all proposed tests and experiments that affect nuclear Safety.

c. Review of all proposed changes to the Appendix "A" Technical Specifications.

d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Manager, Nuclear Operations and to the Chairman of the Nuclear General Review Committee.

f. Review of events requiring 24-hour written notification to the Commission.

g. Review of facility operations to detect potential nuclear safety hazards.

h. Performance of special reviews, investigations or analyses and reports thereon as requested by the Chairman of the Nuclear General Review Committee.

i. Review of the Plant Security Plan and implementing procedures and shall submit recommended changes to the Chairman of the Nuclear General Review Committee.

j. Review of the Emergency Plan and implementing procedures and shall submit recommended changes to the Chairman of the Nuclear General Review Committee.

RECORDS

6.5.2.11 Records of NGRC acti··· ·s shall be prepared, approved and distributed as indicated below.

a. Minutes of each NRGC meeting shall be prepared, approved and forwarded to the Senior Vice President Engineering and Construction within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.8 above, shall be prepared, approved and forwarded to the Senior Vice President-Engineering and Construction within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.2.9 above, shall be forwarded to the Senior Vice President Engineering and Construction and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.6 REPORTABLE OCCURRENCE ACTION

6.6.1 The following actions shall be taken for REPORTABLE OCCURRENCES:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 6.9.

b. Each REPORTABLE OCCURRENCE requiring 24 hour notification to the Commission shall be reviewed by the PRC and submitted to the NGRC and the Manager, Nuclear Operations.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The facility shall be placed in at least HOT STANDBY within one hour.

b. The Safety Limit violation shall be reported to the Commission, the Manager, Nuclear Operations and to the NGRC within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PRC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the NGRC and the Manager, Nuclear Operations within 14 days of the violation.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November, 1972.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

g. Systems Integrity Program implementation.

h. Iodine Monitoring Program implementation.

6.8.2 Each procedure and administrative policy of 6.8.1 above, and changes thereto, shall be reviewed by the PRC and approved by the Nuclear Plant Manager prior to implementation and reviewed periodically as set forth in administrative procedures.