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

PORTLAND GENERAL ELECTRIC COMPANY

THE CITY OF EUGENE, OREGON

PACIFIC POWER AND LIGHT COMPANY

DOCKET NO. 50-344

TROJAN NUCLEAR PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 56
License No. NPF-1

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

   A. The application for amendment by Portland General Electric Company, the City of Eugene, Oregon, and Pacific Power and Light Company (the licensee) dated December 28, 1979 and October 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 the addition of new paragraphs 2.C.(14) and 2.C.(15) as indicated below, and 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. NPF-1 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 56 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, except as noted in paragraphs 2.C.(10) through 2.C.(12) below.

   (14) Systems Integrity

   The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

   1. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

   2. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

   (15) Iodine Monitoring

   The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

   1. Training of personnel,

   2. Procedures for monitoring, and

   3. Provisions for maintenance of sampling and analysis equipment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 21, 1981

ATTACHMENT TO LICENSE AMENDMENT

ADMENDMENT NO.56 TO FACILITY OPERATING LICENSE NO. NPF-1

DOCKET NO. 50-344

Revise Appendix A as follows:

Remove Pages	Insert Pages
IV	IV
V	V
3/4 3-18	3/4 3-18
3/4 3-19	3/4 3-19
-	3/4 3-19a
3/4 3-20	3/4 3-20
3/4 3-24	3/4 3-24
3/4 3-27	3/4 3-27
3/4 3-28	3/4 3-28
3/4 3-31	3/4 3-31
-	3/4 3-51
-	3/4 3-52
-	3/4 3-53
3/4 4-4	3/4 4-4
-	3/4 4-4a
3/4 4-5	3/4 4-5
3/4 7-6	3/4 7-6
3/4 8-3	3/4 8-3
B 3/4 3-3	B 3/4 3-3
B 3/4 4-1	B 3/4 4-1
B 3/4 4-2	B 3/4 4-2
6-3	6-3
6-4	6-4
6-5	6-5
6-18	6-18

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
2) Containment Radioactivity-High					
Particulate	1	1	1	1, 2, 3, 4	17
Iodine	1	1	1		
High Level Noble Gas	1	1	1		
Low Level Noble Gas	1	1	1		
4. STEAM LINE ISOLATION					
a. Manual	1/steam line	1/steam line	1/operating steam line	1, 2, 3, 4	18
b. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	13
c Containment Pressure--High-High	4	2	3	1, 2, 3	16
d. Steam Flow in Two Steam Lines- High				1, 2, 3##	
Four Loops Operating	2/steam line	1/steam line	1/steam line		14
Three Loops Operating	2/operating steam line	1###/any operating steam line	1/operating		15

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
COINCIDENT WITH EITHER T_{avg}--Low-Low				1, 2, 3##	
Four Loops Operating	1 T_{avg}/loop	2 T_{avg} any loops	1 T_{avg} in any 3 loops		14
Three Loops Operating	1 T_{avg}/oper-acting loop	1### T_{avg} in any operating loop	1 T_{avg} in any two operating loops		15
OR, COINCIDENT WITH					
Steam Line Pressure-Low				1, 2, 3##	
Four Loops Operating	1 pressure/ loop	2 pressures any loops	1 pressure any 3 loops		14
Three Loops Operating	1 pressure/ operating loop	1### pressure in any oper-ating loop	1 pressure in any 2 oper-ating loops		15
5. TURBINE TRIP & FEEDWATER ISOLATION					
a. Steam Generator Water Level--High-High	3/loop	2/loop in any oper-ating loop	2/loop in each oper-ating loop	1, 2, 3	14

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. AUXILIARY FEEDWATER PUMPS START					
a. Steam Generator Water Level--Low-Low	3/Steam Generator	2/Steam Generator any Steam Generator	2/Steam Generator	1,2,3	18
b. Safety Injection	2/pump	1/pump	2/pump	1, 2, 3	18
c. Loss of Normal and Preferred Power	2/Bus	1/Bus	1/Bus	1,2,3	18
d. Manual (control room and Panel C-160)	2/pump	1/pump	2/pump	1, 2, 3	18

TABLE 3.3-3 (Continued)

TABLE NOTATION

#Trip function may be bypassed in this MODE below P-11.

##Trip function may be bypassed in this MODE below P-12.

###The channel(s) associated with the protective functions derived from the out of service Reactor Coolant Loop shall be placed in the tripped mode.

ACTION STATEMENTS

ACTION 13 - With the number of OPERABLE Channels one less than the Total Number of Channels, be in 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 14 - With the number of OPERABLE Channels one less than the Total Number of Channels:

a. Below P-11 or P-12, place the inoperable channel in the tripped condition within 1 hour and restore the inoperable channel to OPERABLE status within 24 hours after exceeding P-11 or P-12; otherwise be in at least HOT STANDBY within the following 6 hours.

b. Above P-11 and P-12, place the inoperable channel in the tripped condition within 1 hour; operation may continue until performance of the next required CHANNEL FUNCTIONAL TEST.

ACTION 15 - With a channel associated with an operating loop inoperable, restore the inoperable channel to OPERABLE status within 2 hours or be in HOT SHUTDOWN within the following 12 hours; however, one channel associated with an operating loop may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

ACTION 16 - With the number of OPERABLE Channels one less than the Total Number of Channels:

a. Below P-11 or P-12, place the inoperable channel in the bypass condition and restore the inoperable channel to OPERABLE status within 24 hours after exceeding P-11 or P-12; otherwise, be in at least HOT SHUTDOWN within the following 12 hours.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
4. STEAM LINE ISOLATION		
a. Manual	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High-High	$\leq$ 30 psig	$\leq$ 32 psig
d. Steam Flow in Two Steam Lines--High Coincident with T_{avg} --Low or Steam Line Pressure--Low	$\leq$ A function defined as follows: 40% of full steam flow between 0% and 20% load and then increasing linearly to 110% of full steam flow at full load. $T_{avg} \geq 553°F$ $\geq$ 600 psig steam line pressure	$\leq$ A function defined as follows: 44% of full steam flow between 0% and 20% load and then increasing linearly to 111.5% of full steam flow at full load $T_{avg} \geq 551°F$ $\geq$ 580 psig steam line pressure
5. TURBINE TRIP AND FEEDWATER ISOLATION		
a. Steam Generator Water Level--High-High	$\leq$ 75% of narrow range instrument span each steam generator	$\leq$ 76% of narrow range instrument span each steam generator
6. AUXILIARY FEEDWATER PUMPS START		
a. Steam Generator Water Level --Low-Low	$\geq$ 5% of narrow range instrument span each steam generator	$\geq$ 3% of narrow range instrument span each steam generator
b. Safety Injection	Not Applicable	Not Applicable
c. Loss of Normal and Preferred Power	$\geq$ 2520 volts	$\geq$ 2478 volts
d. Manual	Not Applicable	Not Applicable

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. Steam Flow in Two Steam Lines-High Coincident with Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq$ 13.0#/23.0##
b. Reactor Trip (from SI)	$\leq$ 3.0
c. Feedwater Isolation	$\leq$ 8.0
d. Containment Isolation	$\leq$ 18.0#/28.0##
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Service Water System	$\leq$ 14.0#/48.0##
h. Steam Line Isolation	$\leq$ 8.0
i. Emergency Fan Coolers	$\leq$ 10.0#/49.0##
7. Containment Pressure--High-High	
a. Containment Spray	$\leq$ 30.0
b. Steam Line Isolation	$\leq$ 7.0
8. Steam Generator Water Level--High-High	
a. Turbine Trip-Reactor Trip	$\leq$ 2.5
b. Feedwater Isolation	$\leq$ 11.0
9. Steam Generator Water Level --Low-Low	
a. Auxiliary Feedwater Pumps	$\leq$ 60.0

TABLE NOTATION

* Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps, SI and RHR pumps.

\# Diesel generator starting and sequence loading delays not included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

\## Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
4. STEAM LINE ISOLATION				
a. Manual	N.A.	N.A.	M(1)	1, 2, 3, 4
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
c. Containment Pressure--High-High	S	R	M(3)	1, 2, 3
d. Steam Flow in Two Steam Lines--High Coincident with T_{avg} -- Low or Steam Line Pressure--Low	S	R	M	1, 2, 3
5. TURBINE TRIP AND FEEDWATER ISOLATION				
a. Steam Generator Water Level--High-High	S	R	M	1, 2, 3
6. AUXILIARY FEEDWATER PUMPS START				
a. Steam Generator Water Level--Low-Low	S	R	M	1, 2, 3
b. Safety Injection	S	R	M	1, 2, 3
c. Loss of Normal and Preferred Power	N.A.	R	N.A.	1, 2, 3
d. Manual	N.A.	N.A.	R	1, 2, 3

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.9 The accident monitoring instrumentation channels shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of channels of operable accident monitoring instrumentation channels less than the MINIMUM NUMBER OF CHANNELS shown in Table 3.3-11, either restore the inoperable channel(s) to operable status within 72 hours, or:

   1. Establish an alternate method of monitoring the appropriate parameters, and

   2. Submit a Special Report in accordance with Specification 6.9.2

      a) by telephone within 24 hours

      b) confirmed by telegraph, mailgram or facsimile transmission no later than the first working day following the event, and

      c) in writing within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-11

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM NUMBER OF CHANNELS
1. Reactor Coolant System Subcooling Margin Monitor	2	1
*2. PORV Position Indicator	1/valve	1/valve
**3. PORV Block Valve Position Indicator	1/valve	1/valve
4. Safety Valve Position Indicator (Acoustical Flow)	1/valve	1/valve
5. Auxiliary Feedwater Flowrate	4 per pump	2 per pump
6. Pressurizer Water Level	3	2

* Not applicable if the associated block valve is in the closed position.

** Not applicable if the block valve is verified in the closed position and power removed.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Reactor Coolant System Subcooling Margin Monitor	M	R	
2. PORV Position Indicator	M		R
3. PORV Block Valve Position Indicator	M		R
4. Safety Valve Position Indicator (Acoustical Flow)	M		R
5. Auxiliary Feedwater Flowrate	M*	R	
6. Pressurizer Water Level	M	R	

* Not applicable in Modes 1 and 2

3/4.4.3 SAFETY AND RELIEF VALVES - OPERATING

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.4.3.1 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2485 PSIG ± 1%.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With a 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

4.4.3.1 Each pressurizer code safety valve shall be demonstrated OPERABLE with a lift setting of 2485 PSIG ± 1%, in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition.

3/4.4.3 SAFETY AND RELIEF VALVES - OPERATING

RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.3.2 Two power operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

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

c. The provisions of 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 Each PORV shall be demonstrated OPERABLE:

a. At least once per 31 days, the control circuit to each valve shall be demonstrated to have circuit continuity.

b. At least once per 18 months by performance of a test to verify that each valve opens at the proper setpoint.

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

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with at least 150 kw of pressurizer heaters and a water level above that necessary for heater operation but less than or equal to 1795 cu ft (92 percent indicated).

APPLICABILITY: MODES 1, 2, and 3

ACTION:

a. With the pressurizer inoperable due to an inoperable emergency power supply to the pressurizer heaters either restore the inoperable emergency power supply within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.4.1 The pressurizer water volume shall be determined to be within its limit 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 transferring power from the normal to the emergency power supply and energizing the heaters.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that each pump operates for at least 15 minutes.

4. Cycling each testable power operated or automatic valve in the flow path through at least one complete cycle of full travel.

5. 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. Cycling each power operated (excluding automatic) valve in the flow path that is not testable during plant operation, through at least once complete cycle of full travel.

2. Verifying that each automatic valve in the flow path actuates to its correct position on a safety injection test signal.

3. Verifying that each pump starts automatically upon receipt of each auxiliary feedwater actuation test signal in Table 4.3-2.

4.7.1.2.2 The auxiliary feedwater pump diesel shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying the fuel level in its day tank,

2. Verifying that a sample of diesel fuel from its fuel tank is within the acceptable limits specified in Table 1 of ASTM D975-78 when checked for viscosity, water and sediment, and

3. Verifying the diesel starts from ambient condition and operates for $\geq$ 15 minutes.

b. At least once per 18 months during shutdown by:

1. Subjecting the diesel to a inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service, and

2. Verifying the diesel and pump operate for $\geq$ 60 minutes.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that samples of diesel fuel from the day tanks and the fuel storage tanks are within the acceptable limits specified in Table 1 of ASTM D975-78 when checked for viscosity, water and sediment,

4. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day tank,

5. Verifying the diesels start from ambient condition,

6. Verifying the generator is synchronized, loaded to $\geq$ 1682 kw, and operates for $\geq$ 60 minutes with both diesel engines operating, and

7. Verifying the diesel generator set is aligned to provide standby power to the associated emergency busses.

b. At least once per 18 months during shutdown by:

1. Subjecting the diesels to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service,

2. Verifying the generator capability to reject a load of $\geq$ 828 kw without tripping,

3. Simulating a loss of offsite power with and without the presence of a safety injection signal, and:

a) Verifying de-energization of the emergency busses and load shedding from the emergency busses.

b) Verifying the diesels start from ambient condition on the auto-start signal, energize the emergency busses with permanently connected loads, energize the auto-connected emergency loads through the applicable load sequencer and operate for $\geq$ 5 minutes while the generator is loaded with the emergency loads.

c) Verifying that all diesel generator trips, except engine overspeed, generator phase overcurrent, generator neutral overcurrent or generator loss of field, are automatically bypassed upon loss of voltage on the emergency bus and/or safety injection actuation signal.

3/4.3.3.6 CHLORINE DETECTION SYSTEMS

The OPERABILITY of the chlorine detection system ensures that sufficient capability is available to promptly detect and initiate protective action in the event of an accidental chlorine release. This capability is required to protect control room personnel and is consistent with the recommendations of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," February 1975.

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.3.8 DECOUPLE SWITCHES

OPERABILITY of the decouple switches in the cable spreading room (CSR) ensures that the control cables passing through the CSR to certain equipment required for safe shutdown of the Plant will be isolated and local operation of the equipment can be achieved. In the event that a portion of the decouple switches becomes inoperable, a fire watch will be established in the CSR until the inoperable equipment is restored to OPERABILITY.

3/4.3.3.9 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with the recommendation of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations".

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with all reactor coolant loops in operation, and maintain DNBR above 1.73 during all normal operations and anticipated transients. With one reactor coolant loop not in operation, THERMAL POWER is restricted to < 38 percent of RATED THERMAL POWER until the Overtemperature ΔT trip is reset. Either action ensures that the DNBR will be maintained above 1.73. A loss of flow in two loops will cause a reactor trip if operating above P-7 (10 percent of RATED THERMAL POWER) while a loss of flow in one loop will cause a reactor trip if operating above P-8 (35 percent of RATED THERMAL POWER).

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 RHR 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 SAFETY VALVES

3/4.4.3 SAFETY AND RELIEF VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at 110% of 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 RHR 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 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip setpoint is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power-operated relief valves or steam dump valves.

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, 1974 Edition.

The power operated relief valves (PORVs) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable.

3/4.4.4 PRESSURIZER

The requirement that 150 kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at HOT STANDBY.

3/4.4.5 STEAM GENERATORS

One OPERABLE steam generator provides sufficient heat removal capability to remove decay heat after a reactor shutdown. The requirement for two OPERABLE steam generators, combined with other requirements of the Limiting Conditions for Operation ensures adequate decay heat removal capabilities for RCS temperatures greater than 350°F if one steam generator becomes inoperable due to single failure considerations. Below 350°F, decay heat is removed by the RHR system.

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 parameter limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these parameter 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

POOR ORIGINAL

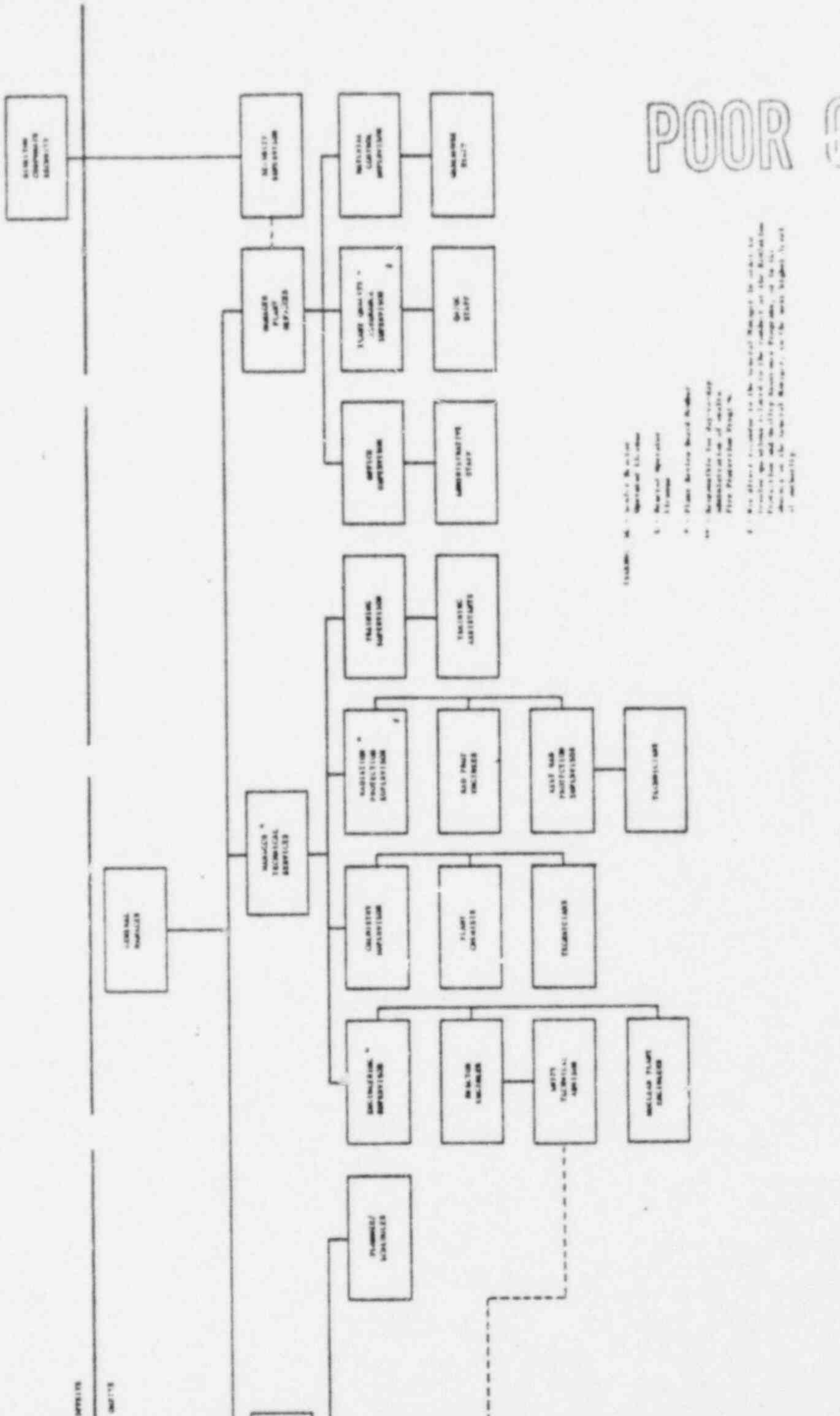

Figure 6.2-2 Facility Organization

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
Shift Technical Advisor	1	None Required

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

#Shift crew composition may be less than the minimum requirements 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. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming crewman being late or absent.

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 (1) the Radiation Protection Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975 and (2) the Shift Technical Advisor who shall have a bachelor's degree or 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 General 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 General Manager and shall meet or exceed the requirements of Section 27 of the NFPA Code-1975, except for Fire Brigade training sessions which shall be held at least quarterly.

6.5 REVIEW AND AUDIT

6.5.1 PLANT REVIEW BOARD (PRB)

FUNCTION

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

additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

d. Abnormal degradation of systems other than those specified in 6.9.1.8.c above designed to contain radioactive material resulting from the fission process.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Director of the Office of Inspection and Enforcement Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.3.3.

b. Inoperable Meteorological Monitoring Instrumentation, Specification 3.3.3.4.

c. Inservice Inspection Program Reviews, Specifications 4.4.10.1 and 4.4.10.2.

d. ECCS Actuation, Specifications 3.5.2 and 3.5.3.

e. Sealed Source Leakage in excess of limits, Specification 4.7.7.1.3.

f. Seismic Event Analysis, Specification 4.3.3.3.2.

g. Fire Detection Instrumentation, Specification 3.3.3.7.

h. Fire Suppression Systems, Specifications 3.7.8.1 and 3.7.8.2.

i. Accident Monitoring Instrumentation, Specification 3.3.3.9.

j. Control Building Modification Connection Bolts, Specifications 3.7.11 and 4.7.11.1.