3. Pressurizer Safety Valves

   a. At least one pressurizer safety valve shall be operable whenever the reactor head is on the vessel.

   b. Both pressurizer safety valves shall be operable whenever the reactor is critical.

4. Pressurizer Power Operated Relief Valves (PORV) and PORV Block Valves.

   a. Two PORVs and their associated block valves shall be operable.

      1. If a PORV is inoperable, the PORV shall be restored to an operable condition within one hour or the associated block valve shall be closed.

      2. If a PORV block valve is inoperable, the block valve shall be restored to an operable condition within one hour or the block valve shall be closed with power removed from the block valve otherwise the unit shall be shutdown and in hot standby within the next six hours.

5. The pressurizer shall be operable with at least 100 KW of pressurizer heaters available and a water level greater than 10% during steady state power operation. At least one bank of pressurizer heaters shall be supplied by an emergency bus power supply.

Basis:

When the boron concentration of the reactor coolant system is to be reduced the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the primary system volume in approximately one half hour. The pressurizer is of little concern

because of the low pressurizer volume and because pressurizer boron concentration normally will be higher than that of the rest of the reactor coolant.

Part 1 of the specification requires that a sufficient number of reactor coolant pumps be operating to provide core cooling in the event that a loss of flow occurs. The flow provided in each case will keep DNBR well above 1.30 as discussed in FFDSAR Section 14.1.9. Therefore, cladding damage and release of fission products to the reactor coolant will not occur. Heat transfer analyses (1) show that reactor heat equivalent to 10% of rated power can be removed with natural circulation only; hence, the specified upper limit of 1% rated power without operating pumps provides a substantial safety factor.

Each of the pressurizer safety valves is designed to relieve 288,000 lbs. per hr. of saturated steam at setpoint. Below 350°F and 350 psig in the reactor coolant system, the residual heat removal system can remove decay heat and thereby control system temperature and pressure. If no residual heat is removed by any of the means available, the amount of steam which could be generated at safety valve relief pressure would be less than half the valves' capacity. One valve therefore provides adequate defense against over-pressurization. Part 1 c(2) permits an orderly reduction in power if a reactor coolant pump is lost during operation between 10% and 50% of rated power. Above 50% power, an automatic reactor trip will occur if either pump is lost. The power-to-flow ratio will be maintained equal to or less than 1.0 which ensures that the minimum DNB ratio increases at lower flow since the maximum enthalpy rise does not increase above its normal full-flow maximum value.(2)

A PORV is defined as OPERABLE if leakage past the valve is less than that allowed in Specification 15.3.1.D and the PORV has met its most recent channel test as specified in Table 15.4.1-1. The PORVs operate to relieve, in a controlled manner, reactor coolant system pressure increases below

the setting of the pressurizer safety valves. These PORVs have remotely operated block valves to provide a positive shutoff capability should a PORV become inoperable.

The requirement that 100 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 pressure control and natural circulation at hot standby.

Reference

(1) FSAR Section 14.1.6
(2) FSAR Section 7.2.3

15.3.5 INSTRUMENTATION SYSTEM

Operational Safety Instrumentation

Applicability:

Applies to plant instrumentation systems.

Objectives:

To provide for automatic initiation of the Engineered Safety Features in the event that principal process variable limits are exceeded, and to delineate the conditions of the plant instrumentation and safety circuits necessary to ensure reactor safety.

Specification:

A. The Engineered Safety Features initiation instrumentation setting limits shall be as stated in Table 15.3.5-1.

B. For on-line testing or in the event of a sub-system instrumentation channel failure, plant operation at rated power shall be permitted to continue in accordance with Tables 15.3.5-2 through 15.3.5-4.

C. In the event the number of channels of a particular sub-system in service falls below the limits given in the column entitled Minimum Operable Channels, or Minimum Degree of Redundancy cannot be achieved, operation shall be limited according to the requirement shown in Tables 15.3.5-2 through 15.3.5-4, Operator Action when minimum operable channels unavailable.

D. The accident monitoring instrumentation channels in Table 15.3.5-5 shall be operable. In the event the number of channels in a particular sub-system falls below the minimum number of operable channels given in Column 2, operation and subsequent operator action shall be in accordance with Column 3.

Basis:

Instrumentation has been provided to sense accident conditions and to initiate operation of the Engineered Safety Features(1).

which automatically initiates appropriate action to prevent exceeding established limits. Safety is not compromised, however, by continuing operation with certain instrumentation channels out of service since provisions were made for this in the plant design. This specification outlines limiting conditions for operation necesssary to preserve the effectiveness of the Reactor Control and Protection System when any one or more of the channels is out of service.

Almost all reactor protection channels are supplied with sufficient redundancy to provide the capability for channel calibration and test at power. Exceptions are backup channels such as reactor coolant pump breakers. The removal of one trip channel on process control equipment is accomplished by placing that channel bistable in a tripped mode; e.g., a two-out-of-three circuit becomes a one-out-of-two circuit. The source and intermediate range nuclear instrumentation system channels are not intentionally placed in a tripped mode since these are one-out-of-two trips, therefore the trips are bypassed during testing. Testing of the NIS power range channel requires bypassing the Dropped Rod protection from NIS, for the channel being tested. However, the Rod Position System still provides the dropped-rod protection. Testing does not trip the system unless a trip condition exists in a concurrent channel.

The operability of the accident monitoring instrumentation ensures that sufficient information is available in selected plant parameters to monitor and assess these variables during and following an accident. The PORV block valves have local, external indication of whether the block valve is open or shut. If necessary, this local indication can be visually verified during a containment entry inspection to verify the block valve is shut.

If the process computer, which provides the reactor coolant system subcooling margin monitor, becomes inoperable, subcooling will be monitored by means of a backup plotter method or manually using control board instrumentation and a saturation curve.

Reference

(1) FSAR - Section 7.5
(2) FSAR - Section 14.3
(3) FSAR - Section 14.2.5

TABLE 15.3.5-1

ENGINEERED SAFETY FEATURES INITIATION INSTRUMENT SETTING LIMITS

NO.	FUNCTIONAL UNIT	CHANNEL	SETTING LIMIT
1	High Containment Pressure (Hi)	Safety Injection*	$\leq$ 6 psig
2	High Containmert Pressure (Hi-Hi)	a. Containment Spray b. Steam Line Isolation of Both Lines	$\leq$ 30 psig $\leq$ 20 psig
3	Pressurizer Low Pressure	Safety Injection*	$\geq$ 1715 psig
4	Low Steam Line Pressure	Safety Injection* Lead Time Constant Lag Time Constant	$\geq$ 500 psig $\geq$ 12 seconds $\leq$ 2 seconds
5	High Steam Flow in a Steam Line Coincident with Safety Injection and Low T_{avg}	Steam Line Isolation of Affected Line	d/p corresponding to $\leq 0.66 \times 10^6$ lb/hr at 1005 psig $\geq$ 540°F
6	High-high Steam Flow in a Steam Line Coincident with Safety Injection	Steam Line Isolation of Affected Line	$\leq$ d/p corresponding to 4×10^6 lb/hr at 806 psig
7	Low-low Steam Generator Water Level	Auxiliary Feedwater Initiation	$\geq$ 5% of narrow range instrument
8	Undervoltage on 4 KV Busses	Auxiliary Feedwater Initiation	$\geq$ 75% of normal voltage

*Initiates also containment isolation, feedwater line isolation and starting of all containment fans.

d/p means differential pressure

TABLE 15.3.5-2 (Cont'd)

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MIN. OPERABLE CHANNELS	4 MINIMUM DEGREE OF REDUNDANCY	5 PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 4 CANNOT BE MET
11.	Turbine Trip	3	2	2	1		Maintain <50% of rated power
12.	Steam Flow - Feed Water Flow mismatch	2/loop	1/loop	1/loop	1/loop		Maintain hot shutdown
13.	Lo Lo Steam Generator Water Level	3/loop	2/loop	2/loop	1/loop		Maintain hot shutdown
14.	Undervoltage 4 KV Bus	2/bus	1/bus (both buses)	1/bus	--		Maintain hot shutdown
15.	Underfrequency 4 KV Bus	2/bus	1/bus (both buses)	1/bus	--		Maintain hot shutdown

NOTE 1: When block condition exists, maintain normal operation.

F.P. = Full Power

* Not Applicable

** One additional channel may be taken out of service for zero power physics testing.

TABLE 15.3.5-3

EMERGENCY COOLING

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MIN. OPERABLE CHANNELS	4 MIN. DEGREE OF REDUNDANCE	5 PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 4 CANNOT BE MET
1.	SAFETY INJECTION						
a.	Manual	2	1	1	1		Hot Shutdown***
b.	High Containment Pressure	3	2	2	1		Hot Shutdown***
c.	Steam Generator Low Steam Pressure/Loop	3	2	2	1	Primary Pressure is Less than 1800 psig	Hot Shutdown***
d.	Pressurizer Low Pressure	3	2	2	1	Primary Pressure is Less than 1800 psig	Hot Shutdown***
2.	CONTAINMENT SPRAY						
a.	Manual	2	2	2	--**		Hot Shutdown***
b.	Hi-Hi Containment Pressure (Containment Spray)	2 sets of 3	2 of 3 in each set	2 per set	1/set		Hot Shutdown***
3.	AUXILIARY FEEDWATER						
a.	Low-low Steam Generator Water						
	i. Start Motor Driven Pump	2/steam gen.	2/either gen.	2	1		Hot Shutdown***
	ii. Start Turbine Driven Pump	3/steam gen.	2/both gens.	2	1		Hot Shutdown***
b.	Trip of both Main Feedpumps starts motor driven pumps	2/pump	1/pump	1/pump	1		Hot Shutdown***
c.	Undervoltage on 4KV Busses starts Turbine driven pump	2/bus	1/bus	1/bus	--		Hot Shutdown***

TABLE 15.3.5-3

EMERGENCY COOLING

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MIN. OPERABLE CHANNELS	4 MIN. DEGREE OF REDUNDANCY	5 PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 4 CANNOT BE MET
.	AUXILIARY FEEDWATER(Continued)						
.	Safety Injection Signal Starts Motor Driven Pumps	S.I. INITIATING FUNCTIONS AND REQUIREMENT AS IN 1. ABOVE					

TABLE 15.3.5-5

INSTRUMENT OPERATING CONDITIONS FOR INDICATION

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 MINIMUM OPERABLE CHANNEL	3 OPERATOR ACTION IF CONDITIONS OF COLUMN 2 CANNOT BE MET
1.	PORV Position Indicator	1/Valve	1/Valve	If the operability of the PORV position indicator cannot be restored within 48 hours, shut the associated PORV Block Valve.
2.	PORV Block Valve Position Indicator	1/Valve	1/Valve	If the operability of the PORV Block Valve Position Indicator cannot be restored within 48 hours, shut and verify the Block Valve shut by direct observation or declare the Block Valve inoperable.
3.	Safety Valve Position Indicator	1/Valve	1/Valve	If the operability of the Safety Valve Position Indicator cannot be restored within seven days, be in at least Hot Shutdown within the next 12 hours.
4.	Reactor Coolant System Subcooling	1	1	If the operability of a subcooling monitor cannot be restored or a backup monitor made functional within 48 hours, be in at least Hot Shutdown within the next 12 hours.
5.	Auxiliary Feedwater Flow Rate*	1	0	If auxiliary feedwater flow indication becomes inoperable while auxiliary feedwater is required for the Steam Generators, the Steam Generators should be monitored to establish adequacy of auxiliary feedwater flow.
6.	Control Rod Misalignment as Monitored by On-Line Computer	1	1	Log individual rod positions once/hr., after a load change >10% or after >30 inches of control rod motion.

*Applies to presently installed auxiliary feedwater pump discharge flow indicators.

TABLE 15.4.1-1 (CONTINUED)

	Channel Description	Check	Calibrate	Test	Remarks
24.	Containment Pressure	S	R	M**	Narrow range containment pressure (-3.0, +3 psig excluded)
25.	Steam Generator Pressure	S***	R	M***	
26.	Turbine First Stage Pressure	S**	R	M**	
27.	Emergency Plan Radiation Instruments	M	R	M	
28.	Environmental Monitors	M	N.A.	N.A.	
29.	Overpressure Mitigating	S	R	****	
30.	PORV Position Indicator	N.A.	R	N.A.	
31.	PORV Block Valve Position Indicator	Q	R	N.A.	
32.	Safety Valve Position Indicator	M	R	N.A.	
33.	PORV Operability	N.A.	R	M	Performance of a channel functional test but excluding valve operation.
34.	Subcooling Margin Monitor	M	R	N.A.	
35.	Undervoltage on 4KV Bus	N.A.	R	M**	For Auxiliary Feedwater Pump Initiation
36.	Auxiliary Feedwater Flow Rate	See Remarks	R	N.A.	Flow Rate indication will be checked at each unit startup and shutdown

S - Each Shift
D - Daily
W - Weekly
Q - Quarterly
B/W - Biweekly
M - Monthly
P - Prior to each startup if not done previous week.
R - Each Refueling Shutdown (But not to exceed 20 months).
N.A. - Not applicable.

** Not required during periods of refueling shutdown, but must be performed prior to starting up if it has not been performed during the previous surveillance period.

*** Not required during periods of refueling shutdown if steam generator vessel temperature is greater than 70°F.

**** When used for the overpressure mitigating system each PORV shall be demonstrated operable by:

a. Performance of a channel functional test on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required operable and at least once per 31 days thereafter when the PORV is required operable.

b. Testing valve operation in accordance with the inservice test requirements of the ASME Boiler and Pressure Vessel Code, Section IX.

TABLE 15.4.1-2 (CONTINUED)

		Test	Frequency
14.	Refueling System Interlocks	Functioning	Each refueling shutdown
15.	Service Water System	Functioning	Each refueling shutdown
16.	Primary System Leakage	Evaluate	Monthly (6)
17.	Diesel Fuel Supply	Fuel inventory	Daily
18.	Turbine Stop and Governor Valves	Functioning	Monthly (6)
19.	Low Pressure Turbine Rotor Inspection (5)	Visual and magnetic particle or liquid penetrant	Every five years
20.	Boric Acid System	Storage Tank Temperature	Daily
21.	Boric Acid System	Visual observation of piping temperatures (all $\geq$145°F)	Daily
22.	Boric Acid Piping Heat Tracing	Electrical circuit operability	Monthly
23.	PORV Block Valves	Complete Valve Cycle	Quarterly (6)
24.	Integrity of Post Accident Recovery Systems Outside Containment	Evaluate	Yearly

(1) A radiochemical analysis for this purpose shall consist of a quantative measurement of each radionuclide with half life of >30 minutes such that at least 95% of total activity of primary coolant is accounted for.

(2) $\bar{E}$ determination will be started when the gross activity analysis of a filtered sample indicates $\geq$10 µc/cc and will be redetermined if the primary coolant gross radioactivity of a filtered sample increases by more than 10 µc/cc.

(3) Drop tests shall be conducted at rated reactor coolant flow. Rods shall be dropped under both cold and hot conditions, but cold drop tests need not be timed.

(4) Drop tests will be conducted in the hot condition for rods on which maintenance was performed.

(5) As accessible without disassembly of rotor.

(6) Not required during periods of refueling shutdown.

(7) At least once per week during periods of refueling shutdown.

(8) At least three times per week (with maximum time of 72 hours between samples) during periods of refueling shutdown.

15.6.3 Facility Staff Qualifications

15.6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions or as clarified in 15.6.3.2 through 15.6.3.4.

15.6.3.2 Except as provided in 15.6.3.3, either the Radiochemical Engineer or the Health Physicist shall meet the following requirements:

a. The individual shall have a bachelor's degree or the equivalent in a science or engineering subject, including some formal training in radiation protection. For purposes of this paragraph, "equivalent" is as follows:

(1) Four years of formal schooling in science or engineering; or

(2) Four years of applied radiation protection experience at a nuclear facility; or

(3) Four years of operational or technical experience or training in nuclear power; or

(4) Any combination of the above totalling four years.

b. Except as provided in d., below, the individual shall have at least five years of professional experience in applied radiation protection. A master's degree in a related field is equivalent to one year of experience and a doctor's degree in a related field is equivalent to two years of experience.

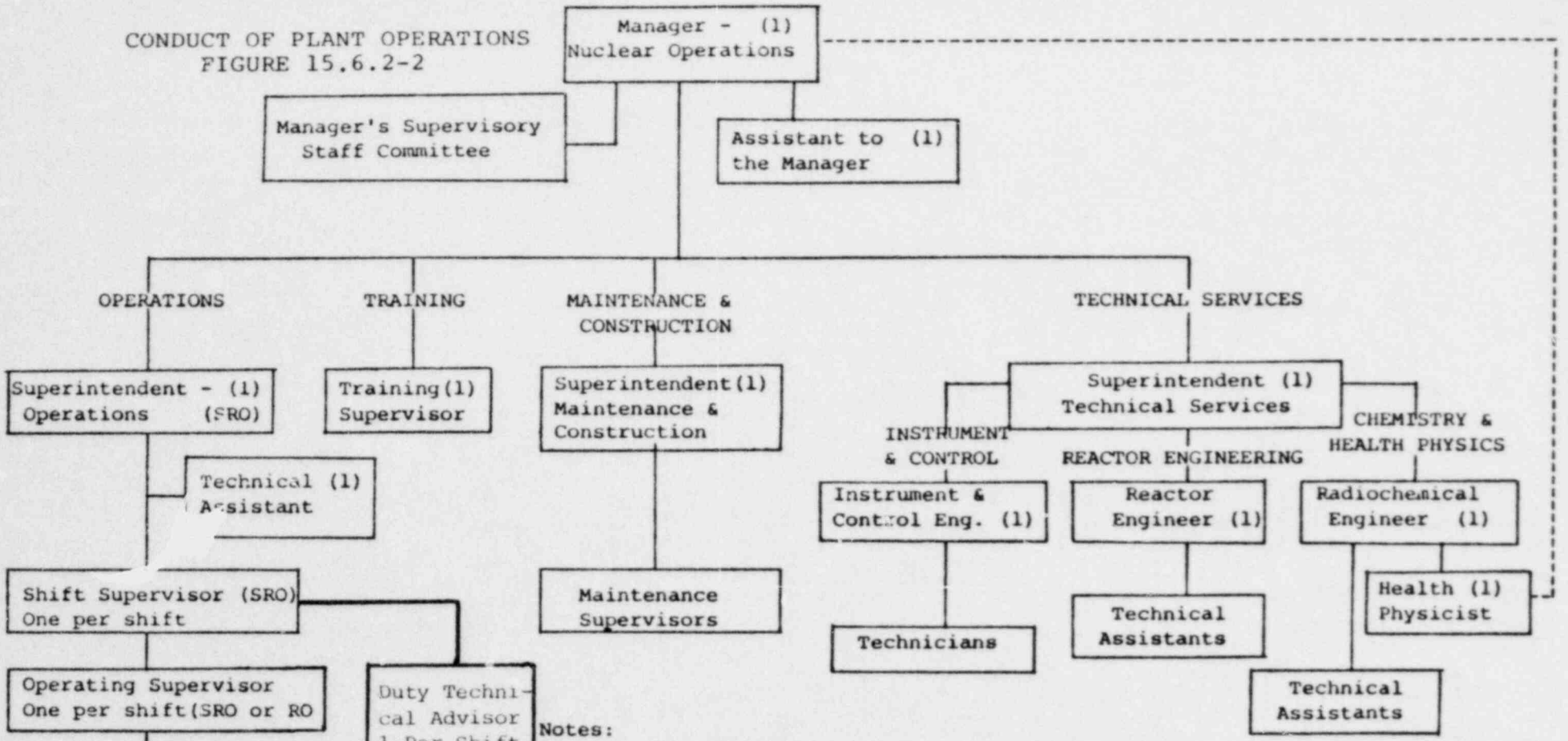
c. Except as provided in d., below, at least three of the five years of experience shall be in applied radiation protection work in a nuclear facility dealing with radiological problems similar to those encountered in nuclear power plants.

d. If the individual has a bachelor's degree specifically in health physics, radiological health, or radiation protection, at least three years of professional experience is required; if the individual has a master's or a doctor's degree specifically in health physics, radiological health, or radiation protection, at least two years of professional experience is required. This experience shall be in applied radiation protection in a nuclear facility dealing with radiological problems similar to those encountered in nuclear power plants.

15.6.3.3 In the event the position of Radiochemical Engineer or Health Physicist is vacated and neither the remaining individual nor the proposed replacement meets the qualifications of 15.6.3.2, but one of these individuals is determined to be otherwise well qualified, then concurrence of NRC shall be sought in approving the qualification of that individual.

15.6.3.4 The Duty Technical Advisor 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.

CONDUCT OF PLANT OPERATIONS
FIGURE 15.6.2-2



Notes:

1. The Operations Group shift makeup is the minimum size for operation in all modes except cold shutdown of Point Beach Unit Nos. 1 and 2. The Operations Group shift makeup 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 makeup to within the minimum requirements.

2. In administrative matters and routine HP concerns, the HP reports directly to the Radiochemical Engineer In matters of radiological health and safety policy determination, interpretation, or implementation, the HP (based on HP judgement) reports directly to the Manager-Nuclear Operations.

3. SRO - NRC Senior Reactor Operator License
   RO - NRC Reactor Operator License

4. The Duty Technical Advisor is located on site on 10 minute call to the control room. Unexpected absences will be treated as in Note 1.