(c) Residual Heat Removal Loop (A)*

(d) Residual Heat Removal Loop (B)*

(2) If the conditions of specification (1) above cannot be met, corrective action to return a second decay heat removal method to operable status as soon as possible shall be initiated immediately.

(3) At least one of the above decay heat removal methods shall be in operation except when required to be secured for testing.

(4) If no decay heat removal method is in operation, all operations causing an increase in the reactor decay heat load or a reduction in reactor coolant system boron concentration shall be suspended. Corrective actions to return a decay heat removal method to operation shall be initiated immediately.

b. Reactor Coolant Temperature Less Than 140°F

(1) Both residual heat removal loops shall be operable except as permitted in items (3) or (4) below.

(2) If no residual heat removal loop is in operation, all operations causing an increase in the reactor decay heat load or a reduction in reactor coolant system boron concentration shall be suspended. Corrective actions to return a decay heat removal method to operation shall be initiated immediately.

(3) One residual heat removal loop may be out of service when the reactor vessel head is removed and the refueling cavity flooded.

(4) One of the two residual heat removal loops may be temporarily out of service to meet surveillance requirements.

4 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.

*Mechanical design provisions of the residual heat removal system afford the necessary flexibility to allow an operable residual heat removal loop to consist of the RHR pump from one loop coupled with the RHR heat exchanger from the other loop. Electrical design provisions of the residual heat removal system afford the necessary flexibility to allow the normal or emergency power source to be inoperable or tied together when the reactor coolant temperature is less than 200°F.

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 shutdown.

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(1) FSAR Section 14.1.6

(2) FSAR Section 7.2.3

allowable leakage rate of 10 gpm has been established. The explained leakage rate of 10 gpm is also well within the capacity of one charging pump, and makeup would be available even under the loss of offsite power condition.

If leakage is to the containment, it may be identified by one or more of the following methods:

a. The containment air particulate monitor is sensitive to low leak rates. the rate of leakage to which the instrument is sensitive is 0.013 gpm within 20 minutes, assuming the presence of corrosion product activity.
b. The containment radiogas monitor is less sensitive but can be used as a backup to the air particulate monitor. The sensitivity range of the instrument is approximately 2 gpm to greater than 10 gpm.
c. The humidity detector provides a backup to a. and b. The sensitivity range of the instrumentation is from approximately 2 gpm to 10 gpm.
d. A leakage detection system which determines leakage losses from water and steam systems within the containment collects and measures moisture condensed from the containment atmosphere by cooling coils of the main recirculation units. This system provides a dependable and accurate means of measuring total leakage, including leaks from the cooling coils themselves which are part of the containment boundary. Condensate flows from approximately 1/2 gpm to 10 gpm can be measured by this system.
e. Indication of leakage from the above sources shall be cause to require a containment entry and limited inspection at power of the reactor coolant system. Visual inspection means, i.e., looking for steam, floor wetness, or boric acid crystalline formations, will be used. Periodic inspections

a. Four service water pumps are operable.

b. All necessary valves, interlocks and piping required for the functioning of the Service Water System during accident conditions are also operable.

2. During power operation, the requirements of 15.3.3.D-1 may be modified to allow one of the following components to be inoperable at any one time. If the system is not restored to meet the conditions of 15.3.3.D-1 within the time period specified, both reactors will be placed in the hot shutdown condition. If the requirements of 15.3.3.D-1 are not satisfied within an additional 48 hours, both reactors shall be placed in the cold shutdown condition.

   a. One of the four required service water pumps may be out of service provided a pump is restored to operable status within 24 hours.

   b. One of the two loop headers may be out of service for a period of 24 hours.

   c. A valve or other passive component may be out of service provided repairs can be completed within 48 hours.

Basis

The normal procedure for starting the reactor is, first, to heat the reactor coolant to near operating temperature, by running the reactor coolant pumps. The reactor is then made critical by withdrawing control rods and/or diluting boron in the coolant.(1) With this mode of start-up, the energy stored in the reactor coolant during the approach to criticality is substantially equal to that during power operation and therefore to be conservative most engineered safety system components and auxiliary cooling systems, shall be fully operable. During low temperature physics tests there is a negligible amount of stored energy in the reactor coolant, therefore an accident comparable in severity to the Design Basis Accident is not possible, and the engineered safety systems are not required.

3. A minimum of 10,000 gallons of water per operating unit in the condensate storage tanks and an unlimited water supply from the lake via either leg of the plant Service Water System.

4. System piping and valves required to function during accident conditions directly associated with the above components operable.

B. The iodine-131 activity on the secondary side of the steam generator shall not exceed 1.2 μCi/cc.

C. During power operation the requirments of 15.3.4.A.2.a and b may be modified to allow the following components to be inoperable for a specified time. If the system is not restored to meet the requirements of 15.3.4.A.2.a and b within the time period specified, the specified action must be taken. If the requirements of 15.3.4.A.2a and b are not satisified within an additional 48 hours, the appropriate reactor (s) shall be cooled down to less than 350°F.

1. Two Unit Operation - One of the four operable auxiliary feedwater pumps may be out-of-service for the below specified times. A turbine driven auxiliary feedwater pump may be out of service for up to 72 hours. If the turbine driven auxiliary feedwater pump cannot be restored to service within the 72 hour time period the associated reactor shall be in hot shutdown within the next 12 hours. A motor driven auxiliary feedwater pump may be out of service for up to 7 days. If the inoperable motor driven auxiliary feedwater pump cannot be restored to service within the 7 day time period both of the reactors shall be in hot shutdown within the next 12 hours.

2. Single Unit Operation - The turbine driven auxiliary feedwater pump may be out-of-service for up to 72 hours. If the turbine driven auxiliary feedwater pump cannot be restored to service within that 72 hour time period, the reactor shall be in hot shutdown within the next 12 hours. Either one of the two motor driven auxiliary feedwater pumps may be out-of-service for up to 7 days. If the motor driven auxiliary feedwater pump cannot be restored to service within that 7 day period the operating unit shall be in hot shutdown within the next 12 hours.

Basis

A reactor shutdown from power requires removal of core decay heat. Immediate decay heat removal requirements are normally satisfied by the steam bypass to the condenser. Therefore, core decay heat can be continuously dissipated via the steam bypass to the condenser as feedwater in the steam generator is converted to steam by heat absorption. Normally, the capability to return feedwater flow to the steam generators is provided by operation of the turbine cycle feedwater system.

The eight main steam safety valves have a total combined rated capability of 6,664,000 lbs/hr. The total full power steam flow is 6,620,000 lbs/hr, therefore eight (8) main steam safety valves will be able to relieve the total full-power steam flow if necessary.

In the unlikely event of complete loss of electrical power to the station, decay heat removal would continue to be assured for each unit by the availability of either the steam-driven auxiliary feedwater pump or one of the two motor-driven auxiliary steam generator feedwater pumps, and steam discharge to the atmosphere via the main steam safety valves or atmospheric relief valves. One motor-driven auxiliary feedwater pump can supply sufficient feedwater for removal of decay heat from a unit. The minimum amount of water in the condensate storage tanks is the amount needed for 25 minutes of operation/unit, which allows sufficient time for operator action.

An unlimited supply is available from the lake via either leg of the plant service water system for an indefinite time period.

Containment Spray

The Engineered Safety Features also initiate containment spray upon sensing a high containment pressure signal (Hi-Hi). The containment spray acts to reduce containment pressure in the event of a loss of coolant or steam line break accident inside the containment. The containment spray cools the containment directly and limits the release of fission products by absorbing iodine should it be released to the containment.

Containment spray is designed to be actuated at a higher containment pressure (approximately 50% of design containment pressure) than the SIS (10% of design). Since spurious actuation of containment spray is to be avoided, it is initiated only on coincidence of high containment pressure (Hi-Hi) sensed by both sets of two-out-of-three containment pressure signals provided for its actuation.

Containment Isolation

A containment isolation signal is initiated by any signal causing automatic initiation of safety injection or may be intiated manually. The containment isolation system provides the means of isolating the various pipes passing through the containment walls as required to prevent the release of radioactivity to the outside environment in the event of loss-of-coolant accident.

Steam Line Isolation

Steam line isolation signals are initiated by the Engineered Safety Features closing the steam line stop valve of the affected line. In the event of a steam line break, this action prevents continuous, uncontrolled steam release from more than one steam generator by isolating the steam lines on high containment pressure (Hi-Hi) or high steam line flow in coincidence with low T_{avg} and SIS. Protection is afforded for breaks inside or outside the containment even when it is assumed that a steam line check valve does not function properly.

6. Direct communication between the control room and the operating floor of the containment shall be available whenever changes in core geometry are taking place.
7. The containment vent and purge system, including the radiation monitors which initiate isolation shall be tested and verified to be operable immediately prior to refueling operations.
8. If any of the specified limiting conditions for refueling are not met, refueling of the reactor shall cease. Work shall be initiated to correct the violated conditions so that the specified limits are met, and no operations which may increase the reactivity of the core shall be made.

B. Limitations on Load Movements Over a Spent Fuel *

1. A load of 1750 pounds shall be the maximum load allowed over either the north half or south half of the spent fuel storage pool when one or more spent fuel assemblies are stored in that half of the spent fuel pool.
2. Auxiliary building crane bridge and trolley positive acting limit switches shall be installed to prevent motion of the main crane hook over that half of the spent fuel pool which contains stored spent fuel which has been subcritical for less than one year.

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* These are interim requirements pending completion and implementation of NRC Generic Task A-36 "Control of Heavy Loads Near Spent Fuel."

3. Loads not exceeding 52,500 pounds may be carried over either pool half (or placed in the north half of the spent fuel pool) provided that that half of the pool contains no spent fuel assemblies.

Basis

The equipment and general procedures to be utilized during refueling are discussed in the Final Safety Analysis Report. Detailed instructions, the above specified precautions, and the design of the fuel handling equipment incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety.[(1)].

Whenever changes are not being made in core geometry one flux monitor is sufficient. This permits maintenance of the instrumentation. Continuous monitoring of radiation levels (A2 above) and neutron flux provides immediate indication of an unsafe condition. The residual heat pump is used to maintain a uniform boron concentration.

The shutdown margin indicated in Part A5 will keep the core subcritical, even if all control rods were withdrawn from the core. During refueling, the reactor refueling cavity is filled with approximately 275,000 gallons of borated water. The boron concentration of this water is sufficient to maintain the reactor

In order to preclude the possibility of dropping a heavy load onto spent fuel assemblies stored in the spent fuel pool and causing a release of radioactivity which could affect the public health and safety, a number of precautionary measures have been incorporated into these limiting conditions for operation. No loads are permitted to be carried over freshly discharged spent fuel assemblies other than single spent fuel assemblies, handling tools and items weighing less than 1750 pounds. Limit switches are installed to prevent motion of the auxiliary building crane main hook over the half of the spent fuel pool which contains freshly discharged fuel.

All heavy load transfers exceeding 1750 pounds will be routed across the spent fuel pool half which contains no stored fuel. When this is no longer possible, heavy loads will not be permitted to be carried over the spent fuel pool until the auxiliary building crane has been certified as a single-failure-proof crane.

Pending additional analysis which demonstrates that dropping a spent fuel shipping cask into the cask loading area of the north spent fuel pool will not cause an uncontrollable loss of spent fuel pool coolant or installation of the redundant crane hoisting mechanism described in Licensee's submittal of March 21, 1978, as amended; specification B1 precludes placing a spent fuel shipping cask into the cask loading area of the north pool when spent fuel is stored in the north half of the spent fuel pool. These specifications also limit the size of the allowable load that can be placed in or carried across either the north or south half of the spent fuel pool when spent fuel assemblies are present in the respective half of the pool. The 52,500 pound limit is consistent with the analysis done for the potential effects upon spent fuel stored in the south spent fuel pool in the event of a postulated cask drop in the north spent fuel pool.[4]

References

(1) FSAR - Section 9.5.2

(2) FSAR - Table 3.2.1-1

(3) FSAR - Volume 5, Question 9.3

(4) FSAR - Appendix F

TABLE 15.3.13-1

SAFETY RELATED SHOCK SUPPRESSORS (SNUBBERS)

Location ID Number	Location	Elevation (ft.)	Snubber in High Radiation Area During Shutdown*	Snubber Especially Difficult to Remove	Snubber Inaccessible During Normal Operation	Snubber Accessible During Normal Operation
	Unit 1					
1-HS-1	"A" Main Steam Line - West	100				X
1-hS-2	"A" Main Steam Line - East	100				X
1-HS-5	"A" SG Side - North	66				X (2)
1-HS-6	"A" SG Side - Middle	66				X (2)
1-HS-7	"A" SG Side - South	66				X (2)
1-HS-3	"B" Main Steam Line - West	100				X
1-HS-4	"B" Main Steam Line - East	100				X
1-HS-8	"B" SG Side - North	66				X (2)
1-HS-9	"B" SG Side - Middle	66				X (2)
1-HS-10	"B" SG Side - South	66				X (2)
1-HS-11	"A" Main Feed Line at 66'	61				X
1-HS-15	Containment Spray Header Above 66'	120				X
1-HS-16	Containment Spray Header Above 66'	120				X
1-HS-12	SIS Line - Regen. HX Cubicle	34	X			X (2)
1-HS-13	SIS Line at 21' Elevation	40				X
1-HS-19	Reactor Vessel Keyway	3	X (1)		X	
1-HS-20	Reactor Vessel Keyway	3	X (1)		X	
HS-601R-37A	Pressurizer PORV Header	76				X
HS-601R-73	Pressurizer SRV Relief Line	80				X
HS-601R-80	Pressurizer SRV Relief Header	80				X
HS-2501R-15	Pressurizer PORV Header	78				X
HS-2501R-22A	Pressurizer PORV Header	78				X
HS-2501R-43	Pressurizer PORV Header	77				X
HS-2501R-51	Pressurizer PORV Header	77				X
EB-2-H7	Steam Dump to Condensers	20				X
EB-2-H17	Steam Dump to Condensers	17				X
R-EB-2-1	Steam Dump to Condensers	20				X
R-EB-2-3	Steam Dump to Condeners	20				X
R-EB-2-4	Steam Dump to Condensers	17				X
R-EB-2-6	Steam Dump to Condensers	20				X
R-EB-2-7	Steam Dump to Condensers	24				X
AC-601R-3-R-350	Pipeway No. 2 Valve Gallery (PAB)	8				X
AC-601R-3-R-356	Pipeway No. 2 Valve Gallery (PAB)	8				X

TABLE 15.3.13-1 (Continued)

Location ID Number	Location	Elevation (ft.)	Snubber in High Radiation Area During Shutdown*	Snubber Especially Difficult to Remove	Snubber Inacessible During Normal Operation	Snubber Accessible During Normal Operation
	Unit 2					
2-HS-36	"A" SG Side - North	66				X (2)
2-HS-37	"A" SG Side - Middle	66				X (2)
2-HS-38	"A" SG Side - South	66				X (2)
2-HS-32	"A" Main Steam Line - East	100				X
2-HS-33	"A" Main Steam Line - West	100				X
2-HS-39	"B" SG Side - North	66				X (2)
2-HS-40	"B" SG Side - Middle	66				X (2)
2-HS-41	"B" SG Side - South	66				X (2)
2-HS-34	"B" Main Steam Line - East	100				X
2-HS-35	"B" Main Steam Line - West	100				X
2-HS-21	Aux. Feed Line to "A" SG	72				X
2-HS-23	SIS Line at 46'	50				X
2-HS-26	SIS Line at 46'	34				X
2-HS-24	At Overhead to Keyway at 26'	36				X
2-HS-29	Downstream of Valve PCV 434	80				X
2-HS-30	Downstream of Valve PCV 435	80				X
2-HS-22	Beneath Valve 541 in "A" Loop Cubicle	41	X		X	
2-HS-31	Reactor Vessel Keyway	3	X (1)		X	
2-HS-27	Reactor Vessel Keyway	3	X (1)		X	
2-HS-25	Regen. HX Cubicle at 26'	34	X			X (2)
HS-601R-37	Pressurizer PORV Header	78				X
HS-2501R-15	Pressurizer PORV Header	78				X
HS-2501R-21A	Pressurizer PORV Header	78				X
HS-2501R-43	Pressurizer PORV Header	77				X
HS-2501R-49	Pressurizer PORV Header	77				X
2R-EB-2-1	Steam Dump to Condenser	19				X
2R-EB-2-2	Steam Dump to Condenser	18				X
2R-EB-2-3	Steam Dump to Condenser	18				X
2R-EB-2-4	Steam Dump to Condenser	17				X
2R-EB-2-5	Steam Dump to Condenser	19				X
2R-EB-2-6	Steam Dump to Condenser	19				X
2R-EB-2-7	Steam Dump to Condenser	24				X
EB-8-H206	Below "A" Loop Atmosphere Relief Valve Header	67				X

(1) High radiation during shutdown with flux thimble withdrawn.
(2) Accessible during normal operation for visual inspection only.
* Modifications to this Table due to changes in high radiation areas shall be submitted to the NRC as part of the next license amendment.

15.3.16 REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

Applicability

Applies to the operational status of the reactor coolant system pressure isolation valves during power operation, startup and shutdown where reactor coolant temperature is greater than 200°F and shutdown margin is less than 1%ΔK/K.

Objective

To increase the reliability of reactor coolant system pressure isolation valves thereby reducing the potential for an intersystem loss of coolant accident.

Specification

A. Each pressure isolation valve listed in Table 15.3.16-1 shall be functional as a pressure isolation device, except as specified in B. Valve leakage shall not exceed the amounts indicated.

B. In the event that the integrity of any pressure isolation valve specified in Table 15.3.16-1 cannot be demonstrated, reactor operation may continue, provided that at least two valves in each high pressure line having a non-functional valve are in, and remain in, the mode corresponding to the isolated condition.[a]

C. If specifications A and B cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

Basis

The operational requirements for reactor coolant system pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA which bypasses containment.

(a) Manual valves shall be locked in the closed position; motor operated valves shall be placed in the closed position and power supplies de-energized.

15.4.6 EMERGENCY POWER SYSTEM PERIODIC TESTS

Applicability

Applies to periodic testing and surveillance requirements of the emergency power system.

Objective

To verify that the emergency power system will respond promptly and properly when required.

Specification

The following tests and surveillance shall be performed as stated:

A. Diesel Generators

1. Manually-initiated start of the diesel generator, followed by manual synchronization with other power sources and assumption of load by the diesel generator shall not exceed 2850KW. This test will be conducted monthly with a minimum running time of 30 minutes on each diesel generator. Normal plant operation will not be affected.
2. Automatic start of each diesel generator, load shedding, and restoration to operation of particular vital equipment, initiated by an actual interruption of normal AC station service power supplies to associated engineered safety systems busses together with a simulated safety injection signal. In addition, after the diesel generator has carried its load for a minimum of 5 minutes, automatic load shedding and restoration of vital loads are tested again by manually tripping the diesel generator output breaker. This test will be conducted during reactor shutdown for major fuel reloading of each reactor to assure that the diesel generator will start and assume required load in less than the time periods listed in the FSAR Section 8.2 after the initial starting signal. During this test a checkout of emergency lighting will be performed, including the changeover relay for DC lights.

15.6 ADMINISTRATIVE CONTROLS

15.6.1 Responsibility

15.6.1.1 The Manager - Point Beach Nuclear Plant (hereinafter referred to as the Manager) shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during absences from the Point Beach Nuclear Plant area of greater than 48 hours and where ready contact by telephone or other means is not assured.

15.6.2 Organization

Offsite

15.6.2.1 The offsite organization for facility management and technical support shall be as shown on Figure 15.6.2-1.

Facility Staff

15.6.2.2 The Facility organization shall be as shown on Figure 15.6.2-2 and:

a. Each on-duty shift shall normally be composed of at least the minimum shift crew composition as noted in Figure 15.6.2-2.

b. At least one licensed operator shall be in the control room when fuel is in either reactor.

c. At least two licensed operators shall be present in the control room during reactor startup, scheduled reactor shutdown and during recovery from reactor trips.

d. An individual qualified in radiation protection procedures shall be on site when fuel is in either reactor.

e. ALL CORE ALTERATIONS after the initial fuel loading shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limitied to Fuel Handling who has no other concurrent responsibilities during this operation.

f. A Fire Brigade of at least 5 members shall be maintained onsite at all times*. This excludes 3 members of the minimum shift crew necessary for safe shutdown of a unit and any personnel required for other essential functions during a fire emergency.

15.6.2.3 Duty & Call Superintendent

a. To assist and counsel the Shift Superintendent in case of significant operating events, a Duty & Call Superintendent group has been established. The Duty & Call Superintendent group shall consist of qualified persons designated by the Manager.

b. In the event of a reportable occurrence, the Shift Superintendent shall communicate with at least one Duty & Call Superintendent before taking other than the immediate on-the-spot action required. One Duty & Call Superintendent will be assigned to be "on call" at all times.

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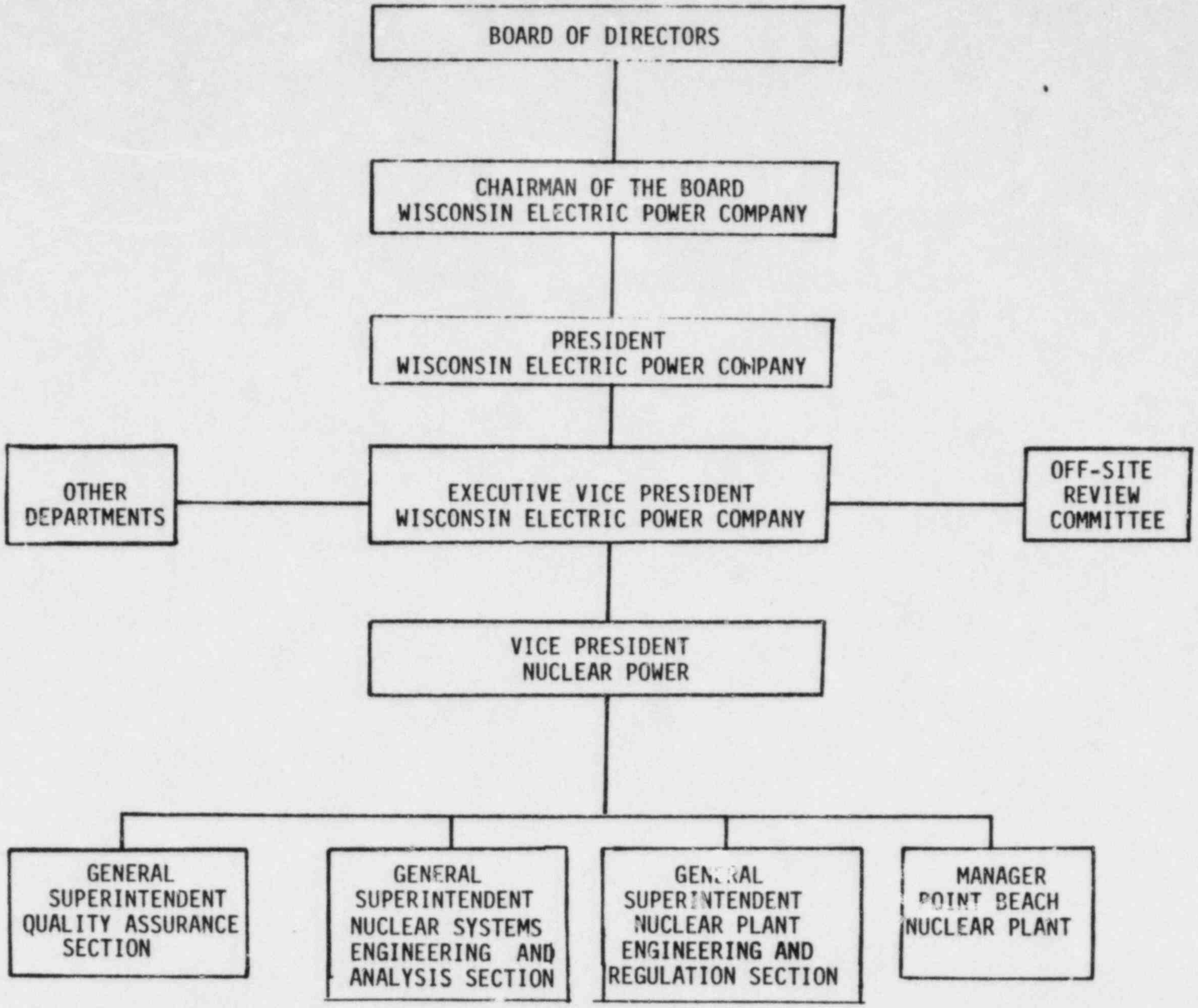
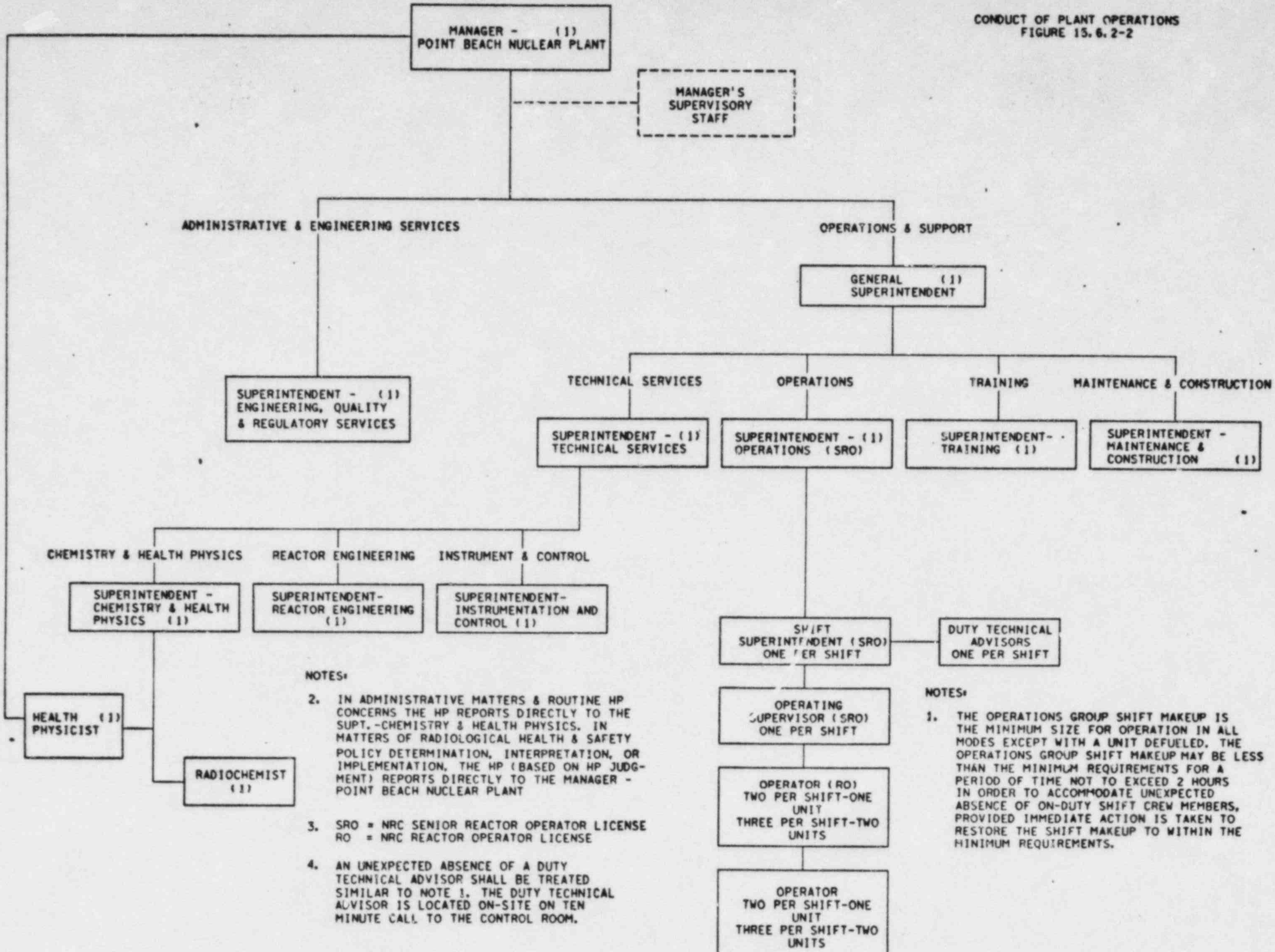
*Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of Fire Brigade members provided immediate action is taken to restore the Fire Brigade to at least the minimum requirements.



MANAGEMENT ORGANIZATION CHART

Figure 15.6.2-1

CONDUCT OF PLANT OPERATIONS
FIGURE 15.6.2-2



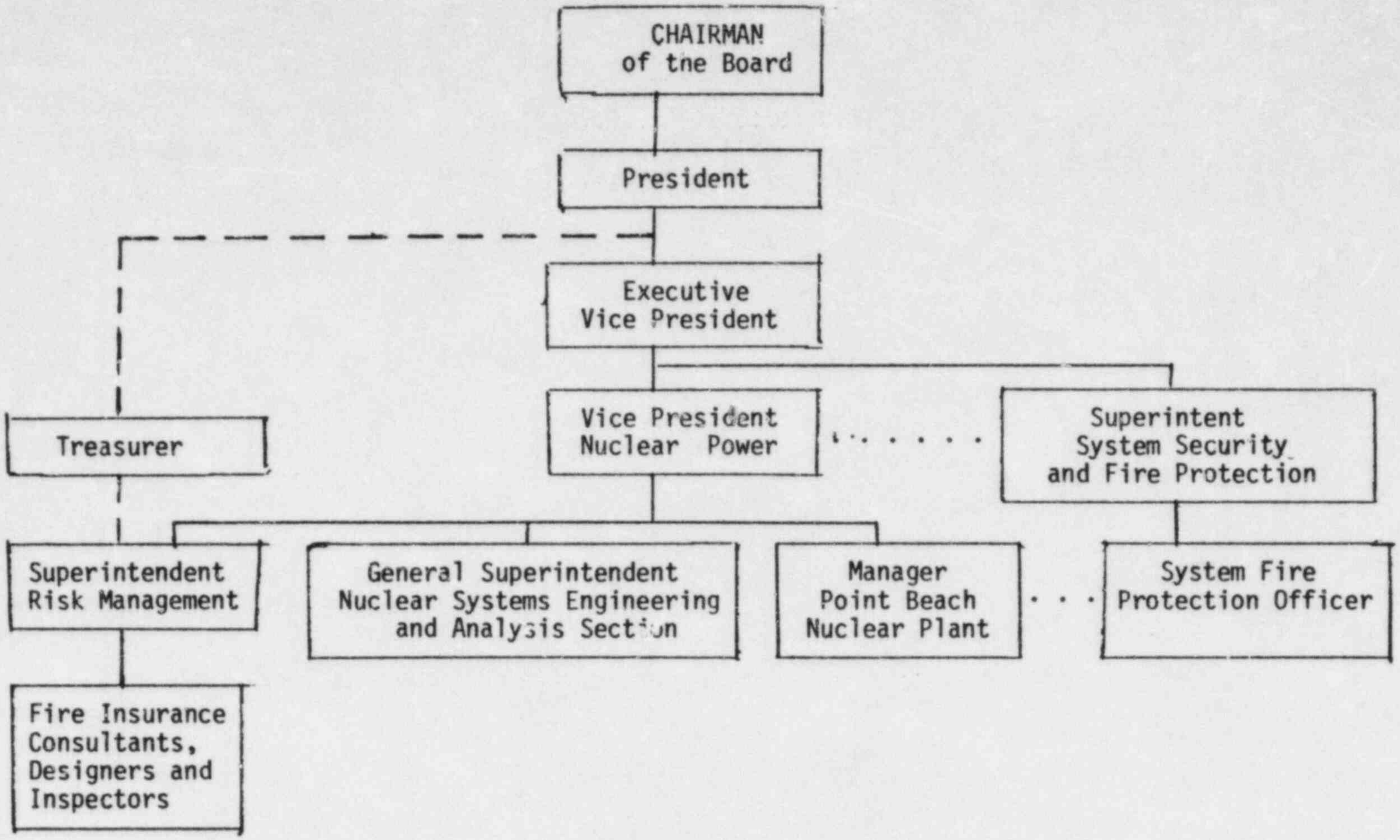
NOTES:

1. THE OPERATIONS GROUP SHIFT MAKEUP IS THE MINIMUM SIZE FOR OPERATION IN ALL MODES EXCEPT WITH A UNIT DEFUELED. 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 & ROUTINE HP CONCERNS THE HP REPORTS DIRECTLY TO THE SUPT.-CHEMISTRY & HEALTH PHYSICS. IN MATTERS OF RADIOLOGICAL HEALTH & SAFETY POLICY DETERMINATION, INTERPRETATION, OR IMPLEMENTATION, THE HP (BASED ON HP JUDGMENT) REPORTS DIRECTLY TO THE MANAGER - POINT BEACH NUCLEAR PLANT

3. SRO = NRC SENIOR REACTOR OPERATOR LICENSE
RO = NRC REACTOR OPERATOR LICENSE

4. AN UNEXPECTED ABSENCE OF A DUTY TECHNICAL ADVISOR SHALL BE TREATED SIMILAR TO NOTE 1. THE DUTY TECHNICAL ADVISOR IS LOCATED ON-SITE ON TEN MINUTE CALL TO THE CONTROL ROOM.



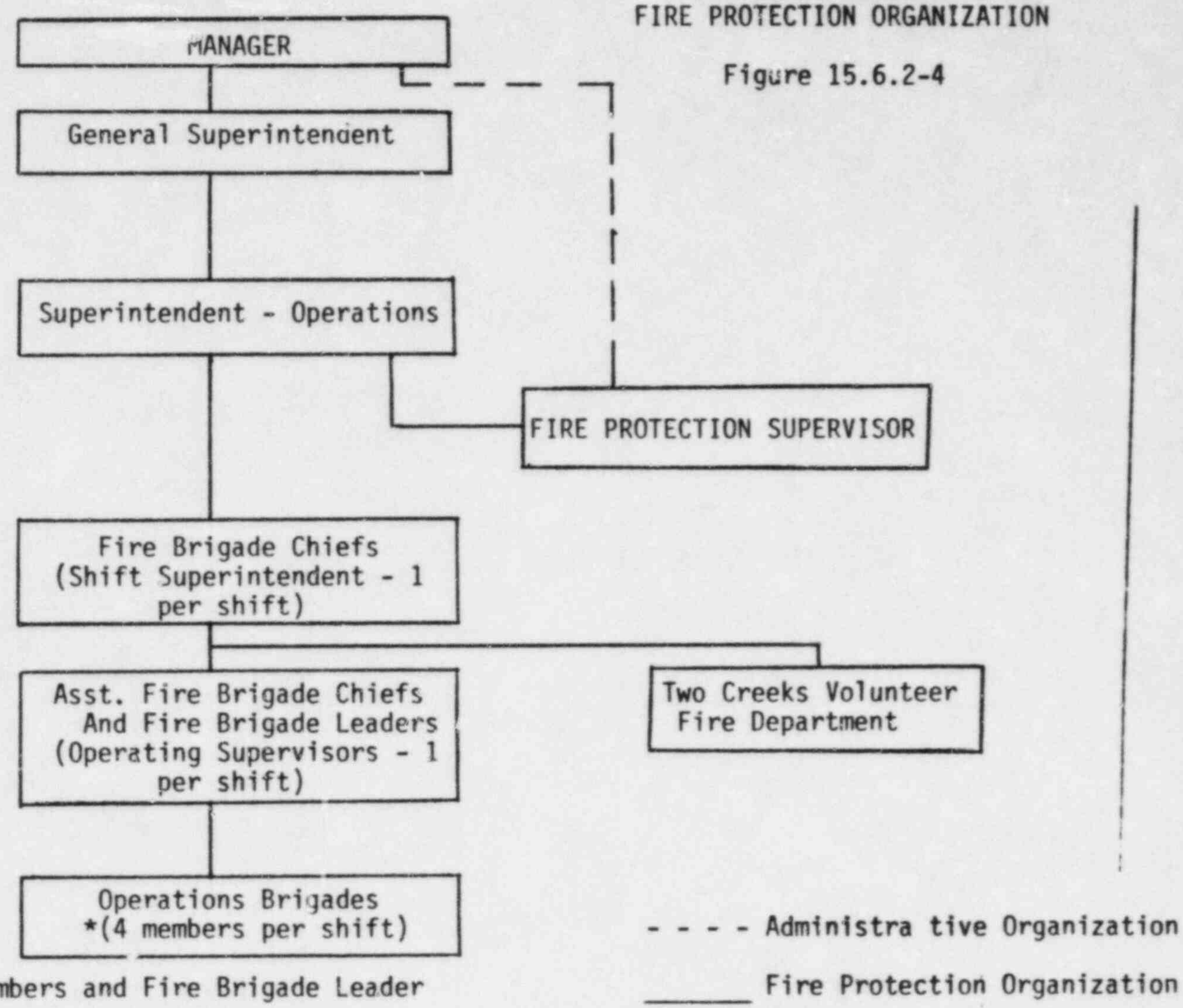
WISCONSIN ELECTRIC POWER COMPANY
OFF-SITE MANAGEMENT
FIRE PROTECTION ORGANIZATION

Figure 15.6.2-3

.....Policy, Procedure, Design Coordination
-----Administrative Organization
_____Fire Protection Organization

POINT BEACH NUCLEAR PLANT
FIRE PROTECTION ORGANIZATION

Figure 15.6.2-4



*Five-man brigade includes members and Fire Brigade Leader

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 Health Physicist or a designated alternate 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 Health Physist or designated alternate 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 concurrance 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. The Duty Technical Advisor shall also receive training in plant design and layout including the capabilities of instrumentation and controls in the control room.

15.6.4 TRAINING

15.6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Superintendent - Training 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.

15.6.4.2 A training program for the Fire Brigade shall meet or exceed the requirements of Section 27 of the NFPA Code-1976, except that the meeting frequency may be quarterly.

15.6.5 REVIEW AND AUDIT

15.6.5.1 Manager's Supervisory Staff

15.6.5.1.1 The Manager's Supervisory Staff (MSS) shall function to advise the Manager on all matters related to nuclear safety.

15.6.5.1.2 The Manager's Supervisory Staff shall be selected from the following:

Chairman:	Manager - Point Beach Nuclear Plant
Member:	General Superintendent
Member:	Superintendent - Operations
Member:	Superintendent - Maintenance & Construction
Member:	Superintendent - Engineering, Quality & Regulatory Services
Member:	Superintendent - Technical Services
Member:	Superintendent - Chemistry & Health Physics
Member:	Superintendent - Instrumentation & Control
Member:	Superintendent - Reactor Engineering
Member:	Superintendent - Training
Member:	Health Physicist

15.6.5.1.3 Alternate members may be appointed by the MSS Chairman to serve on a temporary basis; however, no more than two alternates shall vote in MSS at any one time. Such appointment shall be in writing.

15.6.5.1.4 The MSS shall meet at least once per calendar month and as convened by the MSS Chairman.

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

15.6.5.1.6 The MSS shall have the following duties:

a. Review procedures as required by these Technical Specifications. Review other procedures or changes thereto which affect nuclear safety as determined by the Manager.

b. Review all proposed tests and experiments related to nuclear safety and the results thereof when applicable.

c. Review all proposed changes to Technical Specifications.

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

e. Periodically review plant operations for nuclear safety hazards.

f. Investigate violations or suspected violations of Technical Specifications, such investigations to include reports, evaluations and recommendations.

g. Perform special reviews, investigations or prepare reports thereon as requested by the Chairman of the Off-Site Review Committee.

h. Review the Facility Fire Protection Program and implementing procedures at least once per 24 months.

i. Investigate, review, and report on all reportable events.

15.6.5.1.7 The Manager's Supervisory Staff shall have the following responsibility:

a. Serve as an advisory committee to the Manager.

b. Make recommendations to the Manger for proposals under items (a) through (d) above. In the event of disagreement between

a majority of the Supervisory Staff and decisions by the Manager, the course of action will be determined by the Manager and the disagreement recorded in the Staff minutes.

c. Make recommendations as to whether or not proposals considered by the Staff involve unreviewed safety questions.

d. Review and approve the contents of a report for each reportable event. Copies of all such reports shall be submitted to the Vice President - Nuclear Power and the Chairman of the Off-Site Review Committee.

e. Written minutes of each meeting shall be routed through staff members and copies shall be provided to the Vice President - Nuclear Power and Chairman of the Off-Site Review Committee.

15.6.8 <u>PLANT OPERATING PROCEDURES</u>

15.6.8.1 The plant shall be operated and maintained in accordance with approved procedures. <u>Major</u> procedures, supported by appropriate <u>minor</u> procedures (such as checkoff lists, operating instructions, data sheets, alarm responses, chemistry analytical procedures, etc.) shall be provided for the following operations where these operations involve nuclear safety of the plant:

1. Normal sequences of startup, operation and shutdown of components, systems and overall plant.
2. Refueling.
3. Specific and foreseen potential malfunctions of systems or components including abnormal reactivity changes.
4. Security Plan implementation.
5. Emergencies which could involve release of radioactivity.
6. Nuclear core testing.
7. Surveillance and testing of safety related equipment.
8. Fire protection implementation.

15.6.8.2 Approval of Procedures.

A. All <u>major</u> procedures of the categories listed in 15.6.8.1 (except 15.6.8.1.4) and 15.6.11, and modifications to the intent thereof, shall be reviewed by the Manager's Supervisory Staff and approved by the Manager prior to implementation.

B. Minor procedures (checkoff lists, operator instructions, data sheets, alarm responses, chemistry analytical procedures, technical instructions, special and routine maintenance procedures, laboratory manuals, etc.) shall, prior to initial use, be reviewed by the Manager's Supervisory Staff and approved by the Manager.

15.6.8.3 Changes to Procedures

A. Temporary changes to major procedures, of the categories listed in 15.6.8.2A, which do not change the intent of the approved procedure, may be made provided such changes are approved by the cognizant group head (Duty Shift Superintendent in Operations) and one of the Duty and Call Superintendents.

B. All temporary changes to major procedures (made by a Duty and Call Superintendent and either a cognizant group head or the Duty Shift Superintendent) shall subsequently be reviewed by the Manager's Supervisory Staff and approved by the Manager within 2 weeks: Temporary changes to major procedures made to a given unit during its refueling outage may be reviewed and approved at any time prior to initial criticality of the reload core; Temporary changes only become permanent changes after the Manager's Supervisory Staff review and Manager's approval steps.

C. All temporary or permanent changes to <u>minor</u> procedures shall be approved by a supervisor of the cognizant group (Duty Shift Superintendent in Operations) and shall be subsequently reviewed and approved by the group head of the cognizant group.