April 8, 1985

Docket Nos. 50-266
and 50-301

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Mr. C. W. Fay, Vice President
Nuclear Power Department
Wisconsin Electric Power Company
231 West Michigan Street, Room 308
Milwaukee, Wisconsin 53201

Dear Mr. Fay:

The Commission has issued the enclosed Amendment Nos. 91 and 95 to Facility Operating License Nos. DPR-24 and DPR-27 for the Point Beach Nuclear Plant, Unit Nos. 1 and 2. The amendments consist of changes to the Technical Specifications in response to your application dated March 16, 1984 as modified September 25, 1984.

These amendments provide additional restrictions on the movement of heavy loads over the spent fuel pool, changes to position titles within the administrative section of the Technical Specifications (TS) and miscellaneous corrections and editorial changes.

They also modify the TS relating to safety related snubbers. Table 15.3.13-1 and references thereto have been deleted in accordance with NRC Generic Letter 84-13, "Technical Specifications for Snubbers", dated May 3, 1984. Additionally, Specification 15.4.13.2 has been changed to require the functional testing of a representative sample of approximately 10 percent of the safety-related snubbers. This change more accurately reflects the intent of the requirement of functional testing and makes the specification independent of the number of safety-related snubbers.

With regard to one of your proposed administrative TS changes, specifically the proposed change to section 15.6.3.2 regarding selected facility staff qualifications, we do not find your proposed change acceptable. That proposed change is, therefore, denied.

The details of the staff's review are contained in the enclosed Safety Evaluation.

The Notice of Issuance will be contained in the Commission's next monthly Federal Register notice. A copy of the Notice of Denial is enclosed.

Sincerely,

Original signed by:
James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 91 to DPR-24
2. Amendment No. 95 to DPR-27
3. Safety Evaluation
4. Notice of Denial

cc w/enclosures:
See next page

AD/DSI
D. MULLER
3/27/85

ORB#3:DL	ORB#3:DL	ORB#3:DL	OELD	AD:OR:DL
PMKreutzer	TColburn:dd	JRMiller		GCLainas
3/21/85	3/26/85	3/ /85	3/28/85	4/3/85

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Two Rivers, Wisconsin 54241

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

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-266

POINT BEACH NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 91
License No. DPR-24

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

   A. The application for amendment by Wisconsin Electric Power Company (the licensee) dated March 16, 1984 as modified September 25, 1984, 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.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-24 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 91 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective within 20 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 8, 1985

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-301

POINT BEACH NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 95
License No. DPR-27

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

   A. The application for amendment by Wisconsin Electric Power Company (the licensee) dated March 16, 1984 as modified September 25, 1984 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.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-27 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 95 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective within 20 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James R. Miller

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 8, 1985

ATTACHMENT TO LICENSE AMENDMENTS NOS. 91 AND 95
TO FACILITY OPERATING LICENSE NOS. DPR-24 AND DPR-27
DOCKET NOS. 50-266 AND 50-301

Revise Appendix A as follows:

Remove Pages	Insert Pages
15.3.1-2	15.3.1-2
15.3.1-3b	15.3.1-3b
15.3.1-14	15.3.1-14
15.3.3-6	15.3.3-6
15.3.4-2	15.3.4-2
15.3.4-2a	15.3.4-2a
15.3.5-3	15.3.5-3
15.3.8-2	15.3.8-2
15.3.8-3	15.3.8-3
15.3.8-4	15.3.8-4
15.3.8-5	15.3.8-5
15.3.8-6	-
15.3.13-1	15.3.13-1
Table 15.3.13-1	-
15.3.16-1	15.3.16-1
15.4.6-1	15.4.6-1
15.4.13-1	15.4.13-1
15.4.13-2	15.4.13-2
15.6.1/2-1	15.6.1/2-1
15.6.2-2	15.6.2-2
Figure 15.6.2-1	Figure 15.6.2-1
Figure 15.6.2-2	Figure 15.6.2-2
Figure 15.6.2-3	Figure 15.6.2-3
Figure 15.6.2-4	Figure 15.6.2-4
15.6.4/5-1	15.6.4/5-1
15.6.5-2	15.6.5-2
15.6.5-3	15.6.5-3
15.6.5-4	15.6.5-4
15.6.5-5	15.6.5-5
15.6.5-6	15.6.5-6
15.6.5-7	15.6.5-7
15.6.5-8	15.6.5-8
15.6.5-9	-
15.6.7-1	15.6.7-1
15.6.8-1	15.6.8-1
15.6.8-2	15.6.8-2
15.6.8-3	-

(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.

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*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.

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.

References

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

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 for indications of leakage within the containment will be conducted to enhance early detection of problems and to assure best on-line reliability.

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 requirements 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.2.a and b are not satisfied 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 by pass 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 initiated 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 a 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 high flow in coincidence with low T_{avg} and SIS or high steam flow in coincidence with 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 Pool*

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.

3. Loads greater than 1750 pounds but 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 no spent fuel assemblies are stored in that half of the spent fuel pool.

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

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

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 subcritical approximately by 10% $\Delta k/k$ in the cold condition with all rods inserted, and will also maintain the core subcritical even if no control rods were inserted into the reactor.[2] Periodic checks of refueling water boron concentration insure that proper shutdown margin is maintained. Part A6 allows the control room operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

During the refueling operation a substantial number of station personnel and perhaps some regulatory people will be in the containment. The requirements are to prevent an unsafe amount of radioactivity from escaping to the environment in the case of a refueling accident, and also to allow safe avenues of escape for the personnel inside the containment as required by the Wisconsin Department of Industry, Labor and Human Relations. To provide for these requirements, the personnel locks (both doors) are open for the normal refueling operations with a third temporary door which opens outward installed across the outside end of the personnel lock.[3] This hollow metal third door is equipped with weather stripping and an automatic door closer to minimize the exchange of inside air with the outside atmosphere under the very small differential pressures expected while in the refueling condition. Upon sounding of the containment evacuation alarm, all personnel will exit through the temporary door(s) and then all personnel lock doors shall be closed. As soon as possible, the fuel transfer gate value shall be closed to back up the 30 foot water seal to prevent escape of fission products.

The spent fuel storage pool at the Point Beach Nuclear Plant consists of a single pool with a four foot thick reinforced concrete divider wall which separates the pool into a north half and south half. The divider wall is notched to a point sixteen feet above the pool floor to allow transfer of assemblies from one half of the pool to the other.

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 would 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 the 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)

Unit 1 - Amendment No. ~~35~~, 91
Unit 2 - Amenement No. ~~42~~, 95

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

15.3.13 SHOCK SUPPRESSORS (SNUBBERS)

Applicability

Applies to the operability of safety related snubbers.

Objective

To define those limiting conditions for operation for snubbers required to protect the primary coolant system and safety related systems.

Specification

1. During all modes of operation except Cold Shutdown and Refueling Shutdown, all safety related snubbers shall be operable except as noted in items 15.3.13.2 through 15.3.13.4 below.

2. If a snubber is determined to be inoperable, reactor operation may be continued no longer than 72 hours after the time the snubber was determined to be inoperable.

3. If the requirements of 15.3.13.1 and 15.3.13.2 cannot be met, an orderly shutdown shall be initiated and the reactor put in the Cold Shutdown condition within 36 hours.

4. If a snubber is determined to be inoperable while the reactor is in the Cold Shutdown or Refueling Shutdown condition, the snubber shall be made operable prior to reactor startup.

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 $200^{\circ}F$ and shutdown margin is less than 1% $\Delta K/K$.

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 deenergized.

Unit 1 - ~~Order dated 4/20/81~~, Amendment No. 91
Unit 2 - ~~Order dated 4/20/81~~, Amendment No. 95

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 2850 KW. 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 A-C 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.

15.4.13 Shock Suppressors (Snubbers)

Applicability

Applies to the periodic inspection and testing requirements of safety related snubbers.

Objective

To verify the operability of the snubbers.

Specifications

The following surveillance requirements apply to safety related snubbers:

1. All snubbers shall be visually inspected to verify operability in accordance with the following schedule:

Number of Snubbers Found Inoperable During Inspection or During Inspection Interval	Next Required Inspection Interval
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3, 4	124 days ± 25%
5, 6, 7	62 days ± 25%
≥8	31 days ± 25%

The required inspection interval shall not be lengthened more than one step at a time.

2. During each refueling shutdown, a representative sample of approximately 10% of the snubbers shall be functionally tested for operability. The hydraulic snubber functional test shall verify

   a. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

   b. Snubber bleed, or release rate, where required, is within the specified range in compression or tension. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

15.4.13 Shock Suppressors (Snubbers)(continued)

c. For each snubber found to be inoperable, an additional 10% of that type of snubber shall be tested until no more failures are found or all units have been tested.

3. A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained. Concurrent with the next inservice visual inspection and at least once per 18 months thereafter, the installation and maintenance records for all safety related snubbers shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be re-evaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This re-evaluation replacement or reconditioning shall be indicated in the records.

Bases

All safety related snubbers are visually inspected for overall integrity and operability. The inspection will include verification of proper orientation, adequate hydraulic fluid level and proper attachment of snubber to piping and structures. To further increase the assurance of snubber reliability, functional tests are performed once each refueling cycle on a representative

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 start-up, 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

15.6 ADMINISTRATIVE CONTROLS (Continued)

15.6.2 Organization (Continued)

Facility Staff (Continued)

Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

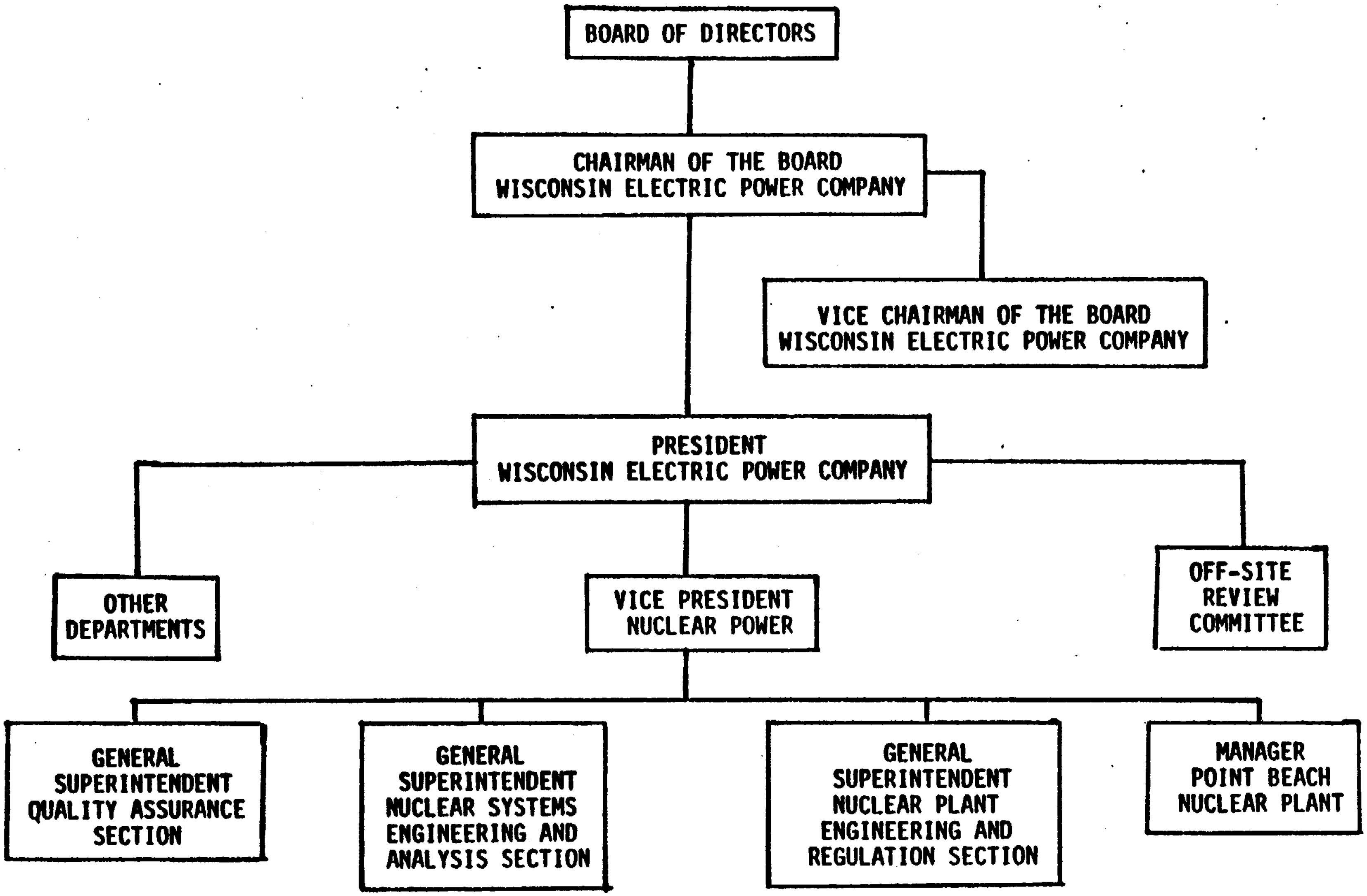
f. A Fire Brigade of at least 5 members shall be maintained onsite at all times*. This excludes 3 members of the minimum shift crew necessary for safe shutdown of the plant and any personnel required for other essential functions during a fire emergency.

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 in writing 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.

*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 within the minimum requirements.

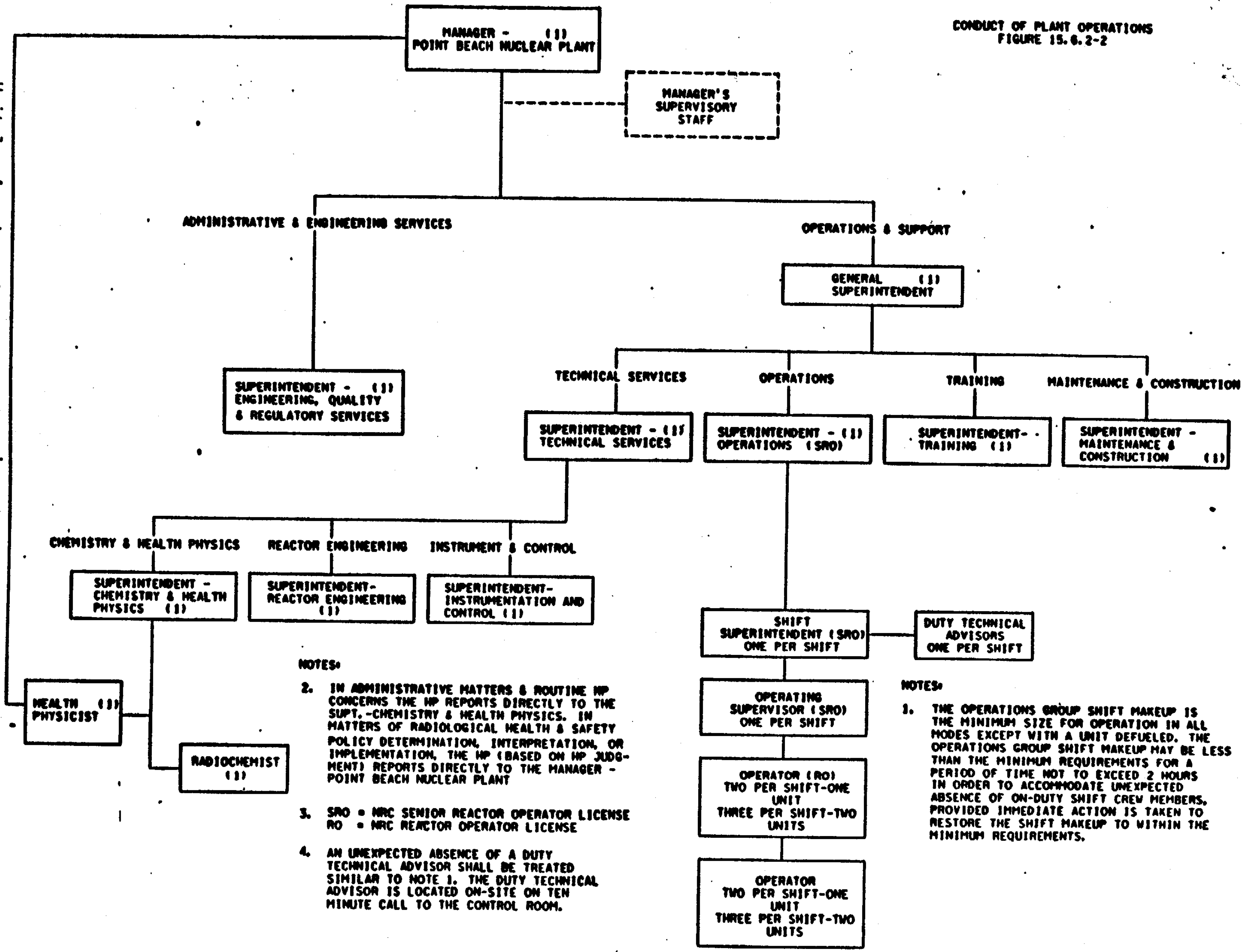


MANAGEMENT ORGANIZATION CHART

Figure 15.6.2-1

Unit 1 - Amendment No. 43, 52, 91
Unit 2 - Amendment No. 48, 58, 95

CONDUCT OF PLANT OPERATIONS
FIGURE 15.6.2-2



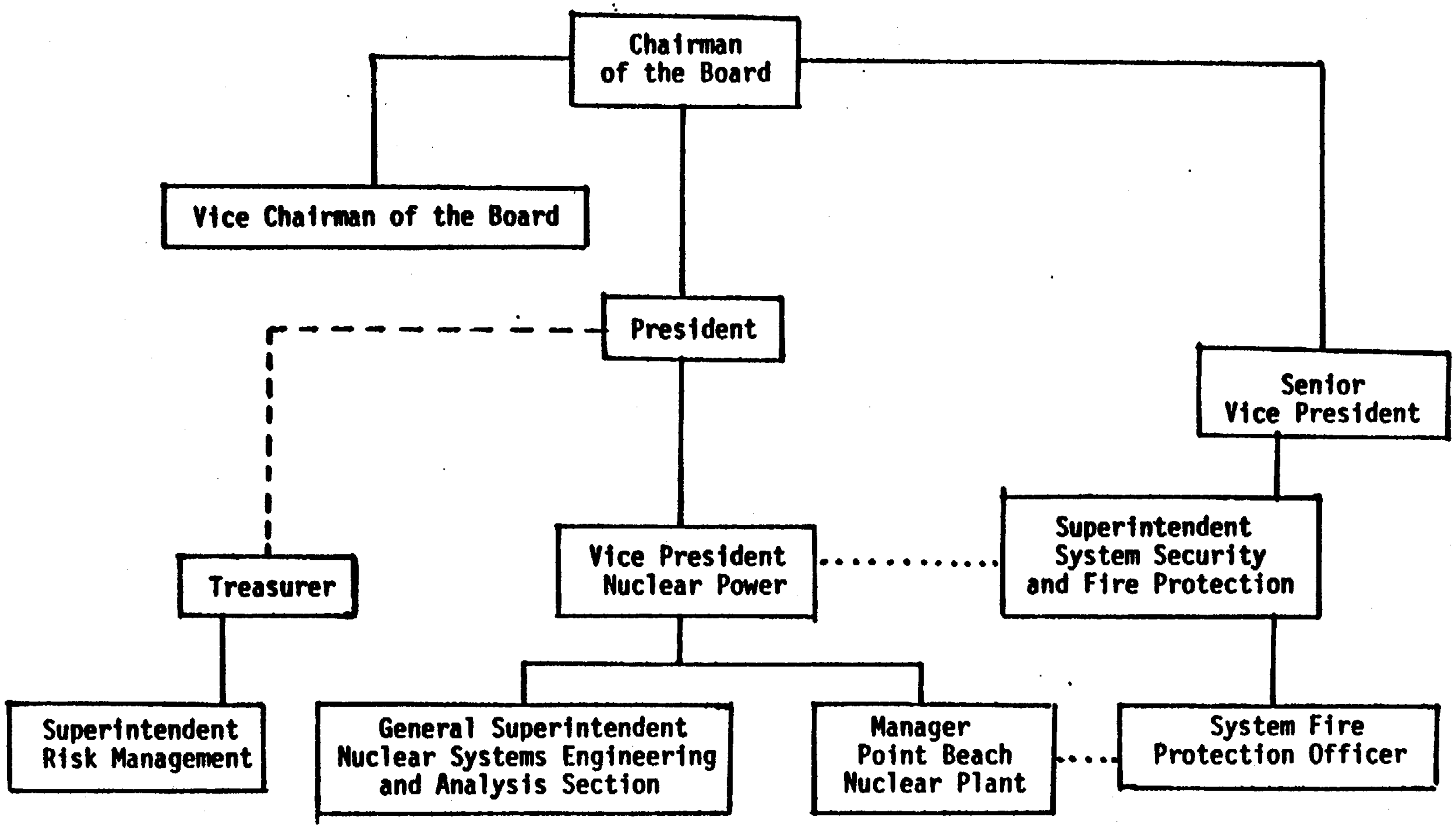
NOTES:

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

..... Policy, Procedure, Design Coordination

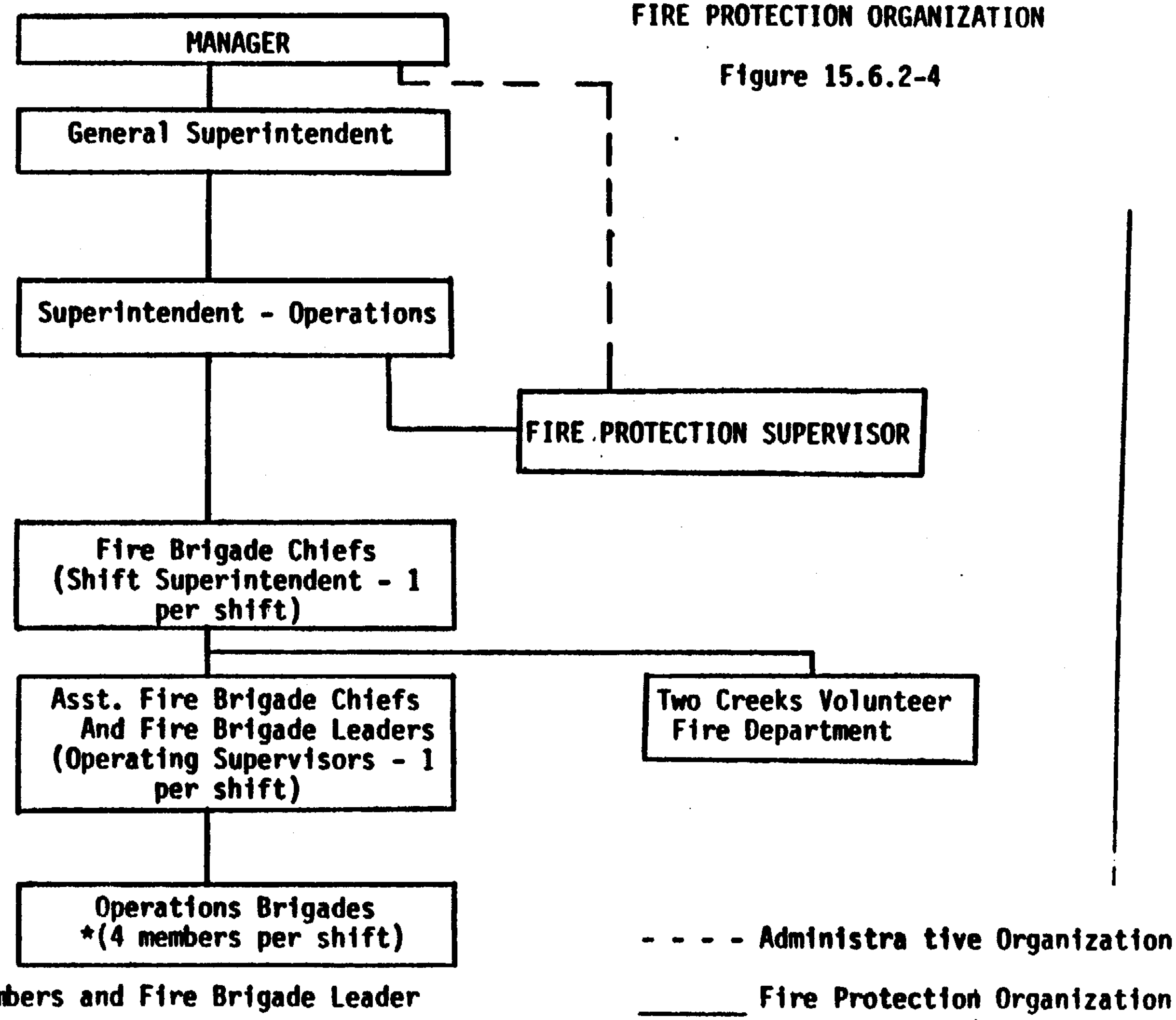
----- Administrative Organization

_____ Fire Protection Organization

Figure 15.6.2-3

POINT BEACH NUCLEAR PLANT
FIRE PROTECTION ORGANIZATION

Figure 15.6.2-4



*Five-man brigade includes members and Fire Brigade Leader

Unit 1 - Amendment No. ~~43~~, ~~52~~, 91
Unit 2 - Amendment No. ~~48~~, ~~58~~, 95

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 - Training
Member:	Superintendent - Technical Services
Member:	Superintendent - Chemistry & Health Physics
Member:	Superintendent - Reactor Engineering
Member:	Health Physicist
Member:	Superintendent - Instrumentation & Control

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 REVIEW AND AUDIT (Continued)

15.6.5.1 Manager's Supervisory Staff (Continued)

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.

15.6.5 REVIEW AND AUDIT (Continued)

15.6.5.1 Manager's Supervisory Staff (Continued)

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 Manager 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 reviewed by staff members and copies shall be provided to the Vice President - Nuclear Power and Chairman of the Off-Site Review Committee.

15.6.5.3 <u>OFF-SITE REVIEW COMMITTEE</u> (OSRC)

FUNCTION

15.6.5.3.1 The Off-Site Review Committee shall function to provide independent review and audit of designated activities in the areas of:

a) nuclear power plant operations
b) nuclear engineering
c) chemistry and radiochemistry
d) metallurgy
e) instrumentation and control
f) radiological safety
g) mechanical and electrical engineering
h) quality assurance practices
i) environmental monitoring

COMPOSITION

15.6.5.3.2 The Off-Site Review Committee is made up of a minimum of five regular members appointed by the President and one or more ex-officio members. Of the five or more regular members, at least two will be persons not directly employed by the Licensee. All members will be experienced in one or more aspects of the nuclear industry.

ALTERNATES

15.6.5.3.3 Alternate members may be appointed in writing by the OSRC Chairman to serve on a temporary basis; however, no more than two alternates shall participate in OSRC activities at any one time.

CONSULTANTS

15.6.5.3.4 Consultants shall be utilized as determined by the OSRC Chairman to provide expert advice to the OSRC.

MEETING FREQUENCY

15.6.5.3.5 The OSRC shall meet at least once per calendar quarter during the initial year of facility operation following fuel loading and at least twice per year at approximately six month intervals thereafter.

QUORUM

15.6.5.3.6 A quorum of OSRC shall consist of the Chairman or his designated alternate and three members. No more than a minority of the quorum shall have line responsibility for operation of the facility.

REVIEW

15.6.5.3.7 The OSRC shall review:

a) The safety evaluations for 1) changes to procedures, equipment or systems, and 2) tests or experiments completed under the provision of 10 CFR, Section 50.59, to verify that such actions did not constitute an unreviewed safety question.

b) Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in 10 CFR, Section 50.59.

c) Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR Section 50.59.

d) Proposed changes in Technical Specifications or Licenses.

e) Violations of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f) Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

g) All reportable events.

15.6.5.3.7 (Continued)

h) Any indication of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

i) Reports and meeting minutes of the Manager's Supervisory Staff.

AUDITS

15.6.5.3.8 Audits of facility activities shall be performed under the cognizance of the OSRC. These audits shall encompass:

a) The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions at least once per year.

b) The performance, training and qualifications of the licensed operating staff at least once per year.

c) The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety at least twice per year at approximately six month intervals.

d) The results of quarterly audits by the Quality Assurance Division on the performance of activities required by the Quality Assurance Program to meet the criteria of Appendix B, 10 CFR 50, at least once per two years.

e) Any other area of facility operation considered appropriate by the President.

AUTHORITY

15.6.5.3.9 The OSRC shall report to and advise the President on those areas of responsibility specified in Sections 15.6.5.3.7 and 15.6.5.3.8.

RECORDS

15.6.5.3.10 Records of OSRC activities shall be prepared, approved and distributed as indicated below:

a) Minutes of each OSRC meeting shall be prepared, approved and forwarded to the President within 14 days following each meeting.

b) Reports of reviews encompassed by Section 15.6.5.3.7.e, f and g above shall be prepared, approved and forwarded to the President within 14 days following completion of the review.

c) Audit reports encompassed by Section 15.6.5.3.8 above shall be forwarded to the President and to the management positions responsible for the areas audited within 30 days after completion of the audit.

15.6.5.4 Fire Protection Audits

a) An independent fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified offsite license personnel or an outside fire protection firm.

b) An inspection and audit of the fire protection and loss prevention program shall be performed by an outside qualified fire consultant at intervals no greater than 3 years.

15.6.5.5 Emergency Plan Audits

a) An audit of the Emergency Plan and Implementing Procedures (EPIP) shall be performed annually utilizing either offsite licensee personnel or an outside nuclear consulting firm. The audit shall be conducted in accordance with 10 CFR 50.54(t) as effective on September 1, 1982.

15.6.7 ACTION TO BE TAKEN IF A SAFETY LIMIT IS EXCEEDED

Specification

A. If a safety limit is exceeded, the affected reactor shall be shut down and reactor operation shall not be resumed until approval is received from the NRC.

B. An immediate report shall be made to the Vice President - Nuclear Power and the Chairman of the Off-Site Review Committee.

C. The Vice President - Nuclear Power shall report the circumstances to the NRC.

D. A Safety Limit Violation Report including a complete analysis of the circumstances leading to and resulting from the occurrence, effects upon facility components, systems or structures, together with recommendations to prevent a recurrence, shall be prepared. This report shall be submitted to the Vice President - Nuclear Power and the Chairman of the Off-Site Review Committee. A Safety Limit Violation Report shall be submitted to the NRC by the Vice President - Nuclear Power within 10 days of the occurrence.

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. <u>Minor</u> procedures (checkoff lists, operating 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 PLANT OPERATING PROCEDURES (Continued)

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 minor 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 91 AND 95 TO

FACILITY OPERATING LICENSE NOS. DPR-24 AND DPR-27

WISCONSIN ELECTRIC POWER COMPANY

POINT BEACH NUCLEAR PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-266 AND 50-301

Introduction

By letter dated March 16, 1984 as modified September 25, 1984, Wisconsin Electric Power Company (licensee) requested changes to the Technical Specifications (TS) for the Point Beach Nuclear Plant Units 1 and 2. These changes would revise sections of the TS relating to movement of heavy loads over spent fuel. They would also delete the list of safety related snubbers and make several changes to the administrative controls section of the TS. These latter changes include updating of organizational charts to reflect position title changes of plant and corporate personnel, clarification of current requirements and deletion of some redundant or unnecessary administrative requirements. Upgraded shift staffing levels are also reflected in the plant organization charts. Various other typographical and editorial change corrections were also proposed by the licensee.

Discussion

Licensees of operating PWRs that do not have a single-failure-proof overhead crane in the fuel storage pool area are required to include a specification comparable to PWR Standard Technical Specification (STS) 3.9.7, "Crane Travel-Spent Fuel Storage Pool Building" per NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," Section 5.3, Interim Protection Measure 1 (IPM-1). This specification prohibits handling of heavy loads over the stored spent fuel in the Spent Fuel Pool (SFP) until the measures which satisfy the guidelines of Section 5.1 of NUREG-0612 are implemented. The auxiliary building crane (the crane in the fuel storage pool area) for Point Beach Units 1 and 2 is not currently certified as a single-failure-proof crane. Further, the existing Point Beach 1 and 2 TS 15.3.8.B, "Limitations on Load Movements over a Spent Fuel Pool," only partially satisfies the intent of NUREG-0612, Section 5.3, IPM-1. It was pointed out in the staff's Heavy Loads Phase I SE that the existing TS 15.3.8.B should be modified appropriately so as to comply with the guidelines of NUREG-0612, Section 5.3, IPM-1.

The proposed TS 15.3.8.B.1 restricts the maximum load that can be moved over the stored spent fuel in either the north or south half of the Point Beach SFP to 1750 lbs. For Point Beach, a heavy load is defined as any load weighing greater than 1750 lbs. This change replaces the existing TS 15.3.8.B.1 which restricts movement of loads over the SFP to a maximum of one ton. Also, the proposed TS changes delete the existing TS 15.3.8.B.3 and 15.3.8.B.4 since the licensee concluded that these specifications are not consistent with either the intent of NUREG-0612 provisions or with the proposed change to Point Beach 1 and 2 TS 15.3.8.B.1. The licensee has further proposed changes in the corresponding bases for the proposed TS changes.

The licensee also proposed several changes to the "Administrative Controls" section of the Point Beach TS to reflect title changes and personnel changes within the plant and corporate staffs. These proposed changes were compared to the appropriate corresponding sections of the STS for evaluation.

Last, the licensee proposed to delete the list of Safety Related Snubbers (shock suppressors) contained on Table 15.3.13-1 in accordance with staff guidance contained in Generic Letter 84-13, "Technical Specifications for Snubbers," dated May 3, 1984 and further to require functional testing of approximately 10 percent of the safety-related snubbers per section 15.4.13.2 rather than a specific number as is currently required.

Evaluation

The staff has reviewed the existing Point Beach 1 and 2 TS 15.3.8.B and the associated bases as well as the proposed changes to TS 15.3.8.B and the associated bases. Furthermore, the staff compared the licensee's proposed changes to TS 15.3.8.B with the corresponding PWR Standard TS 3.9.7.

The staff review indicates that the licensee's proposed TS 15.3.8.B.1 satisfies the recommended value for the maximum load (a "heavy load") stated in the draft TER for compliance with NUREG-0612, Section 5.3, IPM-1 guidelines, as the 1750 maximum load for Point Beach corresponds to the weight of one fuel assembly and its associated handling devices. Furthermore, in the staff's previous SE for Point Beach 1 and 2 regarding Control of Heavy Loads - Phase I, dated March 27, 1984 the staff concluded that the licensee complies with NUREG-0612, Section 5.1.1, Guideline 1 which addresses safe load paths for handling heavy loads and, therefore, the reference in the proposed TS to movement over either the north or south half of the Point Beach SFP does not conflict with IPM-1 guidelines.

The staff reviewed the Point Beach 1 and 2 existing TS 15.3.8.B.2 for which no change has been proposed and has determined that this specification meets the intent of NUREG-0612, Section 5.3, IPM-1 as it is intended to prevent the motion of the main crane hook over that half of the SFP which contains spent fuel which has been subcritical for less than one year. Therefore, the staff concludes that the retention of this specification is acceptable.

With regard to the existing Point Beach 1 and 2 TS 15.3.8.B.3 and 15.3.8.B.4, the staff previously identified the need for modifying these specifications since they are not consistent with either the intent of NUREG-0612, Section 5.3, IPM-1 guidelines or with the proposed change to TS 15.3.8.B.1. Specifically, the existing TS 15.3.8.B.3 refers to the movement of loads exceeding one ton over the stored fuel in the SFP and the existing TS 15.3.8.B.4 contains vague wording with regard to the prevention of movement of loads over stored fuel in the SFP. Therefore, the staff concludes that the licensee's proposed deletions of these specifications are acceptable since the proposed change to TS 15.3.8.B.1 satisfies the concerns regarding movement of heavy loads over the SFP.

With regard to proposed Point Beach 1 and 2 TS 15.3.8.B.3, the staff concurs with the licensee's retention of this specification. This specification is identical to existing TS 15.3.8.B.5 and states that loads not exceeding 52,500 lbs can be carried over either pool half (or placed in the north half of the SFP) provided that half of the pool contains no spent fuel assemblies. The staff's concurrence is based on the staff's previous finding that the movement of heavy loads over the half of the SFP which does not contain any irradiated fuel complies with NUREG-0612, Section 5.1.1, Guideline 1 and Section 5.3, IMP-1.

The staff's review of the proposed changes in the associated bases for TS 15.3.8.B indicates that they are consistent with the corresponding technical specification changes and are, therefore, acceptable.

Based on the above, the staff concludes that the licensee's proposed changes to TS 15.3.8.B and the associated bases meet the guidelines of NUREG-0612, Section 5.3, IPM-1 and are, therefore, acceptable.

The staff has reviewed the licensee's proposed changes to the TS relating to safety-related snubbers. The deletion of the list of safety-related snubbers contained in Table 15.3.13-1 is in accordance with the guidance contained in Generic Letter 84-13 dated May 3, 1984. Further, the change to the surveillance requirement TS 15.4.13.2 (whereby the number of safety related snubbers functionally tested is changed to "approximately 10 per cent") more accurately reflects the intent of the surveillance requirement in that all snubbers would be functionally tested during a ten year interval. Based on the above, the staff finds the proposed changes to be acceptable.

With regard to the proposed changes to the "Administrative Controls" section of the Point Beach 1 and 2 TS, a brief description of the changes and their evaluation follows.

1. Section 15.6.1 - The title of the plant manager is changed from Manager Nuclear Operations to Manager - Point Beach Nuclear Plant. This is a title change only and involves no change of function. It is, therefore, acceptable.

2. Figure 15.6.2-1 - The title of Director - Nuclear Power Department is changed to Vice President - Nuclear Power. This is a title change only and involves no change in function. It is, therefore, acceptable.

3. Figure 15.6.2-1 - The previous Manager - Nuclear Engineering Section has been replaced by two General Superintendents, one for the Nuclear Systems Engineering and Analysis Section and one for the Nuclear Plant Engineering and Regulation Section. These two General Superintendents are responsible for the same functions previously performed by the Manager - Nuclear Engineering Section. This change allows each General Superintendent to concentrate his attention to particular aspects of the engineering support activity which should result in better engineering support of plant activities. The change does have a potential negative aspect since it results in an increased span of control for the Vice President - Nuclear Power. Under the revised organization, the Vice President will directly supervise the two General Superintendents mentioned above, plus the General Superintendent - Quality Assurance Section and the Manager - Point Beach Nuclear Plant. The staff considers that this span of control is not excessive, however, and thus concludes that the proposed change is acceptable.

4. Figure 15.6.2-2 - The old position of Shift Supervisor is now designated the Shift Superintendent. This is a title change only and is acceptable.

5. Figure 15.6.2-2 - The minimum shift staffing has been revised to conform to the new shift staffing requirements of 10 CFR 50.54(m). The revisions are, therefore, acceptable.

6. Figure 15.6.2-4 - The composition of the fire brigade has been revised to delete the use of one contract security guard as a member. The Point Beach fire protection organization no longer relies on the use of contract security guards. The fire brigade now is composed of either the Shift Superintendent (Fire Brigade Chief) or the Operations Supervisor (Assistant Fire Brigade Chief) and four shift members. Fire brigade membership excludes three members of the minimum shift crew necessary for safe shutdown of the plant and any personnel required for other essential functions during a fire emergency. With this change, the licensee still is in conformance with the requirements for fire brigade membership and for adequate coverage of other plant operations. The change is, therefore, acceptable.

7. Section 15.6.5.1 - The description of the Duty and Call Superintendent (the title of the Shift Technical Advisor) has been moved from this section to Section 15.6.2.3. The new location is a more appropriate position for this description and the change, therefore, is acceptable.

8. Section 15.6.5.2 - (Erroneously referred to as Section 15.6.9.1 in the application for change) - The description of the duties and responsibilities of the Manager's Supervisory Staff (MSS) have been editorially revised to clarify and simplify the specifications. The requirement for the MSS to review plant operations for industrial safety has been deleted. The staff agrees with this deletion since industrial safety is not addressed in the Standard Technical Specifications. Also deleted as part of the revision is a requirement for the MSS to cause periodic drills on emergency procedures to be conducted. This is not a normal function for plant review groups and, thus, the deletion is acceptable. According to the licensee, requirements for

Emergency Plan drills are now incorporated in the Emergency Plan, audits of which are required by Section 15.6.5.5 of the Technical Specifications. Except for the deletions of the requirements regarding industrial safety and emergency drills, the coverage of the reviewed specification remains the same as before. The change, therefore, is acceptable.

9. Section 15.6.8.3 - The wording of this section has been changed to simplify and clarify the instructions. There has been no change to the requirements regarding changes to procedures. The proposed wording is an improvement over the existing wording and is, therefore, acceptable.

10. Section 15.6.3.2 - The licensee proposed changing this section to allow the use of either a "designated alternate" or the Health Physicist to meet requirements formerly required of either the Superintendent - Chemistry and Health Physics or the Health Physicist.

The staff finds the licensee's proposed use of "designated alternate" in Section 15.6.3.2 unacceptable for the following reasons: 1) The "designated alternate" is not limited to the Radiation Protection Staff (i.e., the individual could be part of the operations staff); 2) There is no assurance that the individual would have access to the plant manager; and 3) There is no assurance that the appointed individual would have line authority over Health Physics Foremen and Technicians.

It is the staff's position that existing TS 15.6.3.3 gives the licensee sufficient flexibility to operate the Point Beach facility. Therefore, the licensee's proposed TS change is denied.

Other administrative changes in terminology and corrections of typographical errors have been made and these are acceptable. Clarifications have been made to some sections to reduce the potential for ambiguity or misinterpretation. These have been reviewed and are also, acceptable.

Environmental Consideration

The amendments involve a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). These amendments also relate to changes in recordkeeping, reporting or administrative procedures or requirements and the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

CONCLUSION

We have concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: April 8, 1985

Principal Contributors:

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7590-01

U. S. NUCLEAR REGULATORY COMMISSION

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NOS. 50-266 AND 50-301

NOTICE OF DENIAL OF REQUEST FOR AMENDMENT

TO FACILITY OPERATING LICENSES AND OPPORTUNITY FOR A HEARING

The U. S. Nuclear Regulatory Commission (the Commission) has denied in part a request by the licensee for amendments to Facility Operating License Nos. DPR-24 and DPR-27, issued to the Wisconsin Electric Power Company (the licensee), for operation of the Point Beach Nuclear Plant, Unit Nos. 1 and 2 (the facilities), located in the Town of Two Creeks, Manitowoc County, Wisconsin.

The amendments as proposed by the licensee modified the Point Beach Technical Specifications (TS), to provide additional restrictions on the movement of heavy loads over the spent fuel pool, changed position titles within the administrative section of the TS, and made miscellaneous corrections and editorial changes. The licensee's application for the amendments was dated March 16, 1984 as modified September 25, 1984. Notice of consideration of issuance of the amendments was published in the FEDERAL REGISTER on June 20, 1984 (49 FR 25350 at 25381) and on January 23, 1985 (50 FR 3047 at 3057). The requested changes were granted except for a proposed change regarding selected facility staff qualifications, which was denied.

Notice of issuance of Amendment Nos. 91 and 95 will be published in the Commission's next regular monthly FEDERAL REGISTER notice.

The portion of the application which proposed a change regarding selected facility staff qualifications was denied. The request to allow the use of either a "designated alternate" or the Health Physicist to meet the requirements formerly required of either the Superintendent-Chemistry and Health Physics or the Health Physicist was found unacceptable for the following reasons: (1) The designated alternate" is not limited to the Radiation Protection Staff (i.e., the individual could be part of the operations staff); (2) There is no assurance that the individual would have access to the plant manager; and (3) There is no assurance that the appointed individual would have line authority over Health Physics foremen and technicians. The existing TS were found to give the licensee sufficient flexibility to operate the Point Beach facilities.

The licensee was notified of the Commission's denial of this request by letter dated April 8, 1985.

By May 13, 1985 the licensee may demand a hearing with respect to the denial described above and any person whose interest may be affected by this proceeding may file a written petition for leave to intervene.

A request for a hearing or petition for leave to intervene must be filed with the Secretary of the Commission, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Docketing and Service Branch, or may be delivered to the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C., by the above date.

A copy of the petition should also be sent to the Executive Legal Director, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, and to Gerald Charnoff, Esq., Shaw, Pittman, Potts & Trowbridge, 1800 M Street, N.W., Washington, D. C. 20036, attorney for the licensee.

For further details with respect to this action, see (1) the application for amendment dated March 16, 1984 as modified September 25, 1984 and (2) the Commission's letter to Wisconsin Electric Power Company dated April 8, 1985, which are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., and at the Joseph P. Mann Public Library, Two Rivers, Wisconsin. A copy of item (2) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Division of Licensing.

Dated at Bethesda, Maryland this 8th day of April, 1985.

FOR THE NUCLEAR REGULATORY COMMISSION

James R. Miller

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing