January 5, 1988

Docket Nos. 50-266
and 50-301

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DWagner ACRS(10)
DWigginton GPA/PA
OGC-Bethesda ARM/LFMB

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. 110 and 113 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 April 10, 1986, as revised by your July 17, 1987 letter.

These amendments modify Technical Specification 15.6.10, "Plant Operating Records," and modify other Technical Specifications to correct minor administrative errors. Some of the administrative corrections requested by the April 10, 1986 submittal have been made by intervening amendments issued since the initial date of your application. Therefore, those changes are not included with these amendments. Additionally, as discussed with Mr. Ron Seizert of your staff on December 1, 1987, the proposed change to Item 4, "Reactor Coolant System Subcooling", of Table 15.3.5-5 has been withdrawn.

A copy of the Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's next bi-weekly Federal Register notice.

Sincerely,

Original Signed By:

David H. Wagner, Project Manager
Project Directorate III-3
Division of Reactor Projects

Enclosures:
1. Amendment No.110 to DPR-24
2. Amendment No.113 to DPR-27
3. Safety Evaluation

cc w/enclosures:
See next page

*SEE PREVIOUS CONCURRENCE

Office:	LA/PDIII-3	PM/PDIII-3	OGC	PD/PDIII-3
Surname:	*PKreutzer	*DWagner/rl	JOHNSON	~~DWigginton~~
Date:	08/18/87	08/20/87	12/8/87	~~12/ /87~~ 1/5/88

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These amendments modify Technical Specification 15.6.10, "Plant Operating Records," and modify other Technical Specifications to correct minor administrative errors. Some of the administrative corrections requested by the April 10, 1986 submittal have been made by intervening amendments issued since the initial date of your application. Therefore, those changes are not included with these amendments.

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Sincerely,

David H. Wagner, Project Manager
Project Directorate III-3
Division of Reactor Projects

Enclosures:
1. Amendment No. to DPR-24
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3. Safety Evaluation

cc w/enclosures:
See next page

DW

Office:	LA/PDIII-3	PM/PDIII-3	OGC	PD/PDIII-3
Surname:	PKreutzer	DWagner/rl		DWigginton
Date:	08/18/87	08/20/87	08/ /87	08/ /87

Mr. C. W. Fay
Wisconsin Electric Power Company

Point Beach Nuclear Plant
Units 1 and 2

cc:
Mr. Bruce Churchill, Esq.
Shaw, Pittman, Potts and Trowbridge
2300 N Street, N.W.
Washington, DC 20037

Mr. James J. Zach, Manager
Point Beach Nuclear Plant
Wisconsin Electric Power Company
6610 Nuclear Road
Two Rivers, Wisconsin 54241

Town Chairman
Town of Two Creeks
Route 3
Two Rivers, Wisconsin 54241

Chairman
Public Service Commission
of Wisconsin
Hills Farms State Office Building
Madison, Wisconsin 53702

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
Office of Executive Director
for Operations
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Resident Inspector's Office
U.S. Nuclear Regulatory Commission
6612 Nuclear Road
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. 110
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 April 10, 1986 as revised by letter dated July 17, 1987 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. 110 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective immediately upon issuance. The Technical Specifications are to be implemented within 20 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Kenneth E. Perkins

Kenneth E. Perkins, Director
Project Directorate III-3
Division of Reactor Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 5, 1988

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

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-301

POINT BEACH NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 113
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 April 10, 1986 as revised by letter dated July 17, 1987 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. 113 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective immediately upon issuance. The Technical Specifications are to be implemented within 20 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Kenneth E. Perkins

Kenneth E. Perkins, Director
Project Directorate III-3
Division of Reactor Projects

Attachment:
Changes to the Technical Specifications

Date of Issuance: January 5, 1988

ATTACHMENT TO LICENSE AMENDMENT NOS. 110 AND 113

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

DOCKET NOS. 50-266 AND 50-301

Revise Appendix A and B Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

APPENDIX A

REMOVE	INSERT
15.1-2	15.1-2
15.3.1-14a	15.3.1-14a
15.3.1-15	15.3.1-15
Table 15.3.5-5 (1 pg.)	Table 15.3.5-5 (1 pg.)
15.3.10-6	15.3.10-6
15.3.12-1	15.3.12-1
15.3.13-2	15.3.13-2
15.4.4-7	15.4.4-7
15.4.4-11	15.4.4-11
15.4.5-4	15.4.5-4
15.4.6-2	15.4.6-2
15.4.15-3	15.4.15-3
Figure 15.6.2-2	Figure 15.6.2-2
15.6.3-1	15.6.3-1
15.6.3-2	15.6.3-2
15.6.4/5.1	15.6.4/5.1
15.6.9-1	15.6.9-1
15.6.9-2	15.6.9-2
15.6.10-1	15.6.10-1
15.6.10-2	15.6.10-2
15.6.12-1	15.6.12-1
15.7.8-1	15.7.8-1

d. Containment Integrity*

Containment integrity is defined to exist when:

1) All non-automatic containment isolation valves and blind flanges are closed as required.

2) The equipment hatch is properly closed.

3) At least one door in each personnel air lock is properly closed.

4) All automatic containment isolation valves are operable or are secured closed.

5) The uncontrolled containment leakage satisfies Specification 15.4.4.

e. Protective Instrumentation Logic

1) Analog Channel

An analog channel is an arrangement of components and modules as required to generate a single protective action signal when required by a plant condition. An analog channel loses its identity where single action signals are combined.

*Containment isolation valves are discussed in FSAR Section 5.2.

If leakage is to another system, it will be detected by the plant radiation monitors and/or water inventory control.

Continuous monitoring of steam generator tube leakage is accomplished by either the individual unit Air Ejector Radiation Monitor, the combined Air Ejector Radiation Monitor, or the Steam Generator Blowdown Radiation Monitor in combination with periodic surveillance of the primary coolant activity. Backup monitoring can be accomplished by sampling secondary coolant gross activity.

References

FSAR Section 6.5, 11.2.3

E. Maximum Reactor Coolant Oxygen and Chloride and Fluoride Concentration For Power Operation

Specification:

1. The concentration of oxygen in the reactor coolant shall not exceed 0.1 ppm.
2. The concentration of chloride and of fluoride in the reactor coolant shall not exceed 0.15 ppm each.
3. If the oxygen concentration and the chloride or fluoride concentration of the reactor coolant simultaneously exceed the limits given in 1) and 2) respectively, corrective action is to be taken immediately to return the system to within normal operation specifications. If the normal operational limits are not achieved within 24 hours, the reactor is to be brought to a hot shutdown condition. If the system is not brought to within specifications within an additional 24-hour period, the system is to be brought to a cold shutdown condition, and the cause of the out-of-specification operation ascertained and corrected.

Basis:

By maintaining the oxygen, chloride and fluoride concentration in the reactor coolant within the limits as specified by E 1), 2) and 3), the functional integrity of the materials of the Reactor Coolant System is assured under all operating conditions.[1]

If these limits are exceeded, measures can be taken to correct the condition, e.g., replacement of ion exchange resin or adjustment of the hydrogen concentration in the volume control tank[2], and further because of the time-dependent nature of any adverse effects arising from oxygen, chloride and fluoride concentration in excess of the limits, it is unnecessary to shut down immediately since the condition can be corrected. Thus the period of 24 hours for corrective action to restore the concentrations within the limits has been established. If the corrective action has not been effective at the end of the 24-hour period, then the reactor will be brought to the hot shutdown

TABLE 15.3.5-5 (Continued)

NO.	FUNCTIONAL UNIT	NO. OF CHANNELS	MINIMUM OPERABLE CHANNELS	OPERATOR ACTION IF CONDITIONS OF COLUMN 2 CANNOT BE MET
7.	Containment High Range Radiation Monitor	3	2	If operability cannot be restored within seven days after failure, prepare a special report to be submitted within thirty days in accordance with 15.6.9.2.D.
8.	Containment High Range Pressure Monitor	2	1	If operability cannot be restored within 48 hours, be in hot shutdown within twelve hours.
9.	a. Containment Water Level Keyway	2	1	Operation may continue up to thirty days. If operability cannot be restored, be in hot shutdown within the next twelve hours.
	b. Containment Water Level Sump B Continuous Indication	2	1	If the operability cannot be restored within 48 hours, be in hot shutdown within twelve hours.
10.	Containment Hydrogen Monitors	4	1	If operability cannot be restored within 72 hours, be in hot shutdown within the next six hours.
11.	Reactor Vessel Fluid Level System	4	1	If operability cannot be restored within 48 hours, be in hot shutdown within the next twelve hours.
12.	In-Core Thermocouples	4/core quadrant	2/core quadrant	If operability of at least two thermocouples per core quadrant cannot be restored within 48 hours, be in hot shutdown within the next twelve hours.
13.	Main Steam Line Radiation Monitors (SA-11)	1/steam line	1/steam line	If operability cannot be restored within seven days, prepare a special report to be submitted within thirty days in accordance with 15.6.9.2.E.

a. The RCCA does not drop upon removal of stationary gripper coil voltage.

b. The RCCA does not step in properly when the proper voltage sequences are applied to the control rod drive mechanism coils. It shall then be assumed inoperable until it has been tested to verify that it does drop.

c. If the bank demand position is greater than or equal to 215 steps, or, less than or equal to 30 steps, and the rod position indicator channel shows a misalignment from the bank demand position of 15 inches, the RCCA shall be assumed inoperable until it has been tested to verify that it does step properly.

d. If the bank demand position is between 215 steps and 30 steps, and the rod position indicator channel shows a misalignment from the bank demand position of 7.5 inches, the RCCA shall be assumed inoperable until it has been tested to verify that it does step properly.

2. Specification 15.3.10.C.1.b can be modified by the following:

   a. If an RCCA does not step in upon demand, up to six hours is allowed to determine whether the problem with stepping is an electrical problem. If the problem cannot be resolved within six hours, the RCCA shall be assumed inoperable until it has been verified that it will step in or would drop upon demand.

   b. If more than one RCCA does not step in, apparently due to electrical problems, the situation shall be rectified or clearly defined that it is an electrical problem and the RCCAs are capable of dropping upon demand or an orderly shutdown shall commence within six hours.

3. No more than one inoperable RCCA shall be permitted during sustained power operation.

4. When it has been determined that an RCCA does not drop on removal of stationary gripper coil voltage, the shutdown margin shall be maintained by boration as necessary to compensate for the withdrawn worth of the inoperable RCCA. If sustained power operation is anticipated, the insertion limit shall be adjusted to reflect the worth of the inoperable RCCA.

15.3.12 CONTROL ROOM EMERGENCY FILTRATION

Applicability:

Applies to the operability of the control room emergency filtration.

Objective:

To specify functional requirements of the control room emergency filtration during power operation and refueling operation.

Specification:

1. Except as specified in 15.3.12.3 below, the control room emergency filtration system shall be operable at all times during power operation and refueling operation of either unit.

   a. The results of in-place cold DOP and halogenated hydrocarbon tests, conducted in accordance with Specification 15.4.11, on HEPA filter and charcoal adsorber banks shall show a minimum of 99% DOP removal and 99% halogenated hydrocarbon removal.

   b. The results of laboratory charcoal adsorbent tests, conducted in accordance with Specification 15.4.11, shall show a minimum of 90% removal of methyl iodide. If laboratory analysis results for in-place charcoal indicate less than 90% methyl iodide removal, this specification may be met by replacement with charcoal adsorbent which has been verified to achieve 90% minimum removal and which has been stored in sealed containers, and retesting the charcoal adsorber bank for halogenated hydrocarbon removal.

   c. The results of fan testing, conducted in accordance with specification 15.4.11, shall show operation within ± 10% of design flow.

Basis

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion during plant startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping as a result of seismic or other events initiating dynamic loads. It is therefore required that all snubbers required to protect the primary coolant system, and other safety related systems or components, be operable during reactor operation.

Because the snubber protection is required only during relatively low probability events, a period of 72 hours is allowed for repairs or replacement. In case a shutdown is required, the allowance of 36 hours to reach a Cold Shutdown condition will permit an orderly shutdown consistent with standard operating procedures. Since plant power operation should not commence with known defective safety related equipment, Specification 15.3.13.4 prohibits reactor startup with inoperable snubbers.

2. Visual inspection shall be made for excessive leakage from components of the system. Any significant leakage shall be measured by collection and weighing or by another equivalent method.

B. Acceptance Criterion

The maximum allowable leakage from the Residual Heat Removal System components (which includes valve stems, flanges and pump seals) shall not exceed two gallons per hour.

C. Corrective Action

Repairs shall be made as required to maintain leakage within the acceptance criterion of IV-B.

D. Test Frequency

Tests of the Residual Heat Removal System shall be conducted at shutdown for major refueling.

V. Annual Inspection

A detailed visual examination of the accessible interior and exterior surfaces of the containment structure and its components shall be performed annually and prior to any integrated leak test, to uncover any evidence of deterioration which may affect either the containment's structural integrity or leak-tightness. The discovery of any significant deterioration shall be accompanied by corrective actions in accordance with acceptable procedures, nondestructive tests and inspections, and local testing where practical, prior to the conduct of any integrated leak test. Such repairs shall be reported as part of the test results.

IX. Liner Plate

A. The liner plate will be examined before the initial pressure test to determine the following:

(1) Locate areas which have inward deformations. The magnitude of the inward deformations will be measured and recorded. The areas will be permanently marked for future reference. The inward deformations will be measured between the angle stiffeners which are on 15-inch centers. The measurements will be accurate to $\pm$ .01 inch.

(2) Try to locate areas having strain concentrations by visual examination paying particular attention to the condition of the liner surface. Record the location of any areas having strain concentrations.

B. Shortly after the initial pressure test and at about one year after initial start-up, reexamine the areas located in section (A). Measure and record inward deformations. Record observations pertaining to strain concentrations.

C. If the difference in the measured inward deformations exceeds 0.25 inch (for a particular location) and/or changes in strain concentration exist, then an investigation will be made. The investigation will determine the cause and any necessary corrective action.

D. The surveillance program will only be continued beyond the one year after initial start-up inspection if some corrective action was needed. If required, the frequency of inspection for a continued surveillance program will be determined shortly after the "one year after initial start-up inspection".

operability of these systems is therefore to combine systems tests to be performed during refueling shutdowns, with more frequent component tests, which can be performed during reactor operation.

The systems tests demonstrate proper automatic operation of the Safety Injection and Containment Spray Systems. With the pumps blocked from starting, a test signal is applied to initiate automatic action, and verification is made that the components receive the safety injection signal in the proper sequence. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry.(1)

During reactor operation, the instrumentation which is depended on to initiate safety injection and containment spray is generally checked weekly and the initiating circuits are tested monthly (in accordance with Specification 15.4.1). In addition, the active components (pumps and valves) are to be tested monthly to check the operation of the starting circuits and to verify that the pumps are in satisfactory running order. The test interval of one month is based on the judgement that more frequent testing would not significantly increase the reliability (i.e. the probability that the component would operate when required), yet more frequent testing would result in increased wear over a long period of time.

Other systems that are also important to the emergency cooling function are the accumulators, the Component Cooling System, the Service Water System and the containment fan coolers. The accumulators are a passive safeguard. In accordance with Specification 15.4.1 the water volume and pressure in the accumulators are checked periodically. The other systems mentioned operate when the reactor is in operation and by these means are continuously monitored for satisfactory performance.

References

(1) FSAR Section 6.2.

3. Each diesel generator shall be given an inspection, at least annually, following the manufacturer's recommendations for this class of stand-by service.

4. Each fuel oil transfer pump shall be run monthly.

The above tests will be considered satisfactory if all applicable equipment operates as designed.

B. Station Batteries

1. Every month the voltage of each cell (to the nearest 0.05 volt), the specific gravity and temperature of a pilot cell in each battery and each battery voltage shall be measured and recorded.

2. Every 3 months the specific gravity, the height of electrolyte, and the amount of water added, for each cell, and the temperature of every fifth cell, shall be measured and recorded.

3. At each time data is recorded, new data shall be compared with old to detect signs of abuse or deterioration.

4. Each battery shall be subjected to a load test at intervals recommended by the manufacturer but not exceeding five years. The battery voltage as a function of time shall be monitored to establish that the capacity is sufficient to carry the loads as delineated in FSAR Table 8.2-3 for the specified length of time. All electrical connections will be checked for tightness.

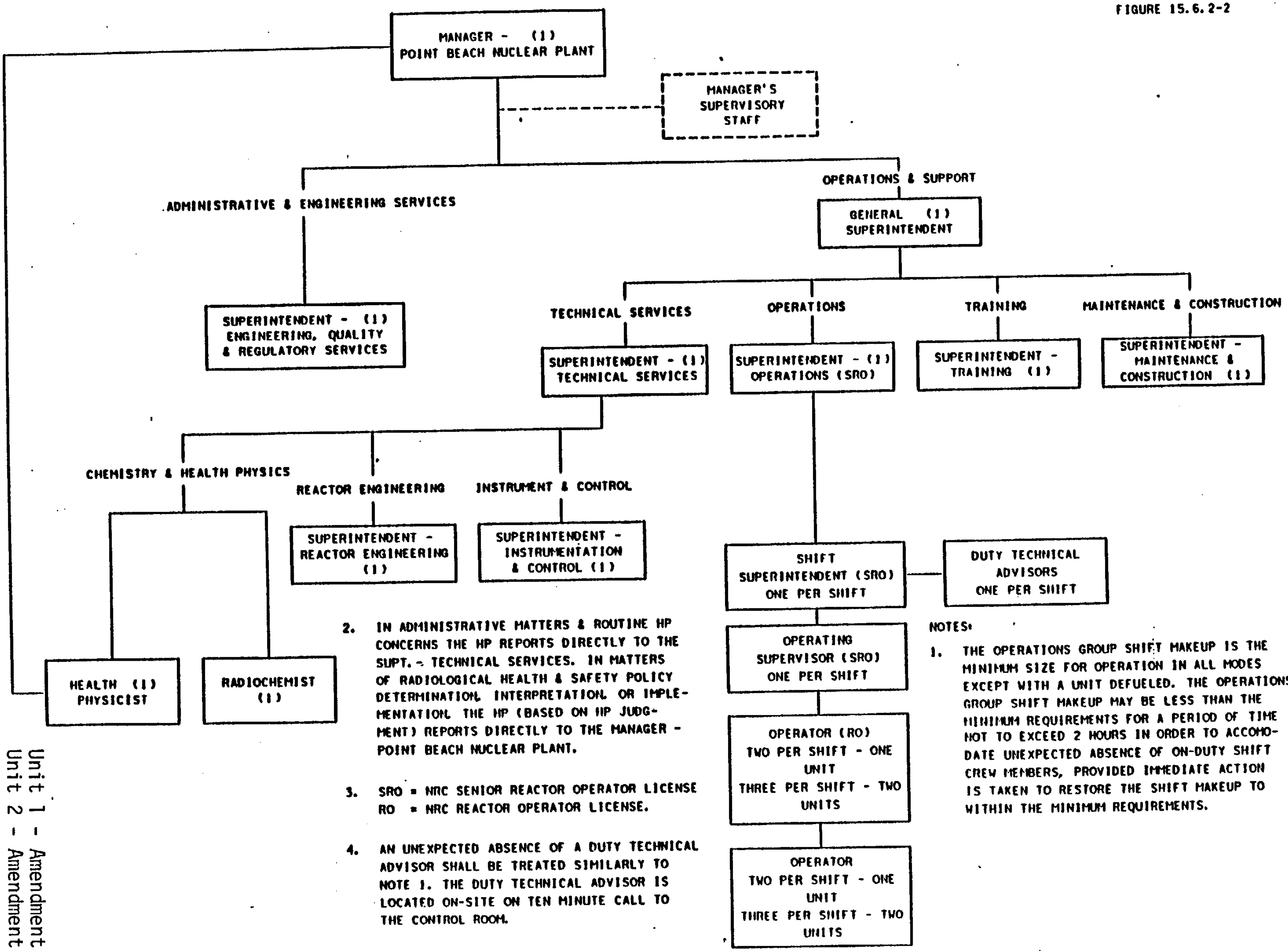
G. Fire Pump Diesel Battery and Charger

Test	Frequency
1. a. Verify electrolyte level above the plates	Weekly
b. Verify that the overall battery voltage is $\geq$ 24 volts	Weekly
2. Verify the specific gravity is appropriate for continued service of the battery	Quarterly
3. a. Verify that the battery, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration	18 months
b. Verify that the battery to battery and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material	18 months

Basis

Normally, the fire protection system is not in use. However, the system components are required to perform as designed in the event of a fire emergency. The National Fire Protection Association and the plant insurance carrier have specified periodic tests and inspections to demonstrate fire protection equipment operability. The listed tests and inspections include and exceed the requirements of these organizations. Testing more frequently than that listed is not considered necessary to insure operability and performance.

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 ACCOMODATE 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. - TECHNICAL SERVICES. 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 SIMILARLY TO NOTE 1. THE DUTY TECHNICAL ADVISOR IS LOCATED ON-SITE ON TEN MINUTE CALL TO THE CONTROL ROOM.

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, the Health Physicist shall meet the following requirements:

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

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

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

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

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

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

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

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

15.6.3.3 In the event the position of Health Physicist is vacated and the proposed replacement does not meet all the qualifications of 15.6.3.2, but is determined to be otherwise well qualified, then concurrence of NRC shall be sought in approving the qualification of that individual.

15.6.3.4 The Duty Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design and response and analysis of the plant for transients and accidents. 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 - Training
Member:	Superintendent - Technical Services
Member:	Superintendent - Reactor Engineering
Member:	Radiochemist
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.

15.6.9 PLANT REPORTING REQUIREMENTS

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following program for reporting of operating information shall be followed. Reports should be addressed to the Regional Administrator, Region III, unless otherwise noted.

15.6.9.1 Routine Reports

A. Startup Report

1. A summary report of plant startup and power escalation testing which addresses each of the tests identified in the FSAR and includes a general description of the measured values obtained during the test program and a comparison of these values with design predictions and specifications must be submitted under the following conditions:

   a. Receipt of an operating license.
   b. Amendment to the license involving a planned increase in power level.
   c. Installation of fuel that has a different design or has been manufactured by a different fuel supplier.
   d. Modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

   Any corrective actions that were required to obtain satisfactory operations shall also be described.

2. This report shall be submitted within the earliest time frame of the following:

a. 90 days following completion of the startup tests.
b. 90 days following resumption or commencement of commercial power operation.
c. 9 months following initial criticality.

B. Annual Results and Data Report

1. A results and data report covering the period of the previous calendar year shall be submitted prior to March 1 of each year.

2. This report shall include:

    a. Complete results of steam generator tube inservice inspection completed during the calendar year as required by specification 15.4.2.A.7
    b. A tabulation on an annual basis of the number of station, utility, and other personnel receiving exposures greater than 100 mrem/year and their associated man-rem exposure according to work and job functions. The dose assignments to various duty functions may be estimates based on pocket dosimeter, TLD or film badge measurements. Small exposures totalling less than 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.
    c. A description of facility changes, tests or experiments as required pursuant to 10 CFR 50.59(b).
    d. A tabulation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves:

15.6.10 PLANT OPERATING RECORDS

Specification

Records and logs relative to the following items shall be retained for 5 years unless a longer period is required by applicable regulations.

A. Records of normal plant operation, including power levels and periods of operation at each power level shall be retained for 5 years except those records of transient or operational cycles for reactor coolant system (RCS) components having a limited number of design transients, shall be retained for the duration of the operating license.

B. Records of principal maintenance activities, including inspection, repair, substitution, or replacement of items of equipment pertaining to nuclear safety shall be retained for a period of 5 years where these requirements do not conflict with requirements of 10 CFR 50.49(j), 10 CFR 50.59, and surveillance requirements of these Technical Specifications. The quality assurance, environmental qualification, installation, and service life records of components covered by these requirements shall be retained for the duration of the Operating License.

C. Records of Licensee Event Reports.

D. Records of installation, environmental qualification, periodic checks, inspections, and calibrations of equipment pertaining to nuclear safety to verify that surveillance requirements are being met will be retained for the duration of the Operating License. All other records of this type will be retained for 5 years.

E. Records of new and spent fuel inventory and assembly histories (5 years following transfer).

F.* Records of design modifications made to systems and equipment, including drawings, as described in the FSAR.

G.* Records of plant radiation and contamination surveys.

H.* Records of off-site environmental surveys.

I.* Records of radiation exposure of all individuals entering radiation controlled areas of the plant, including records for preparation of NRC-4 forms, bioassay and whole body counting results; and records of

* Items will be retained for the duration of the Operating License.

individual exposures exceeding 40-MPC hour limits, including evaluations and actions taken.

J.* Records of gaseous and liquid radioactive material released to the environment and dilution of these wastes.

K.* Records of any special reactor tests or experiments.

L. Records of changes made in the Operating Procedures.

M. Records of sealed source and fission detector leak tests and results performed pursuant to Specification 15.4.12, including annual physical inventory results verifying accountability of sources.

N. Records of training, qualification and requalification for NRC-licensed personnel shall be retained for the duration of the operator's license per 10 CFR 55 requirements. Records of fire brigade member training, including drill critiques shall be maintained for 3 years in accordance with 10 CFR 50, Appendix R, Section I.4 requirements.

O.* Records of in-service inspections performed pursuant to these technical specifications.

P.* Records of Quality Assurance activities required by the QA Manual shall be maintained for the duration of the Operating License except those QA records relating to radioactive materials shipping packages, which shall be maintained for the lifetime of the packaging per 10 CFR 71.91(c) requirements.

Q.* Records of reviews performed for changes made to procedures or equipment, or reviews of tests and experiments pursuant to 10 CFR 50.59 and as required per Specification 15.6.5.1.6.

R.* Records of meetings of the Manager's Supervisory Staff and the Off-Site Review Committee.

S.* Records of analyses for radiological environmental monitoring.

T. Records of radioactive material shipments having a specific activity of greater than 0.002 microcurie/gram shall be retained for a period of 2 years in accordance with 10 CFR 71.91(a).

U. Records concerning the Security Plan, procedures, testing, maintenance, and audit shall be maintained in accordance with the Commission-approved PBNP Modified Amended Security Plan.

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* Items will be retained for the duration of the Operating License.

15.6.12 ENVIRONMENTAL QUALIFICATION

A. By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of: Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors" (DOR Guidelines); or NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," December 1979. Copies of these documents are attached to Order for Modification of Licenses DPR-24 and DPR-27 dated October 24, 1980.

B. By no later than December 1, 1980, complete and auditable records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

15.7.8 ADMINISTRATIVE CONTROLS

15.7.8.1 Duties of the Manager's Supervisory Staff

The duties of the Manager's Supervisory Staff with respect to these radiological effluent technical specifications are listed in specification 15.6.5.1.6 at items j. and k.

15.7.8.2 Audits

A. An audit of the activities encompassed by the Offsite Dose Calculation Manual and the Process Control Program and its implementing procedures shall be performed at least once every 24 months utilizing either offsite licensee personnel or a consulting firm.

B. An audit of the radiological environmental monitoring program and the results thereof shall be performed at least once every 12 months utilizing either offsite licensee personnel or a qualified consulting firm.

C. The results of the audits in A and B above shall be transmitted to the Vice-President - Nuclear Power and the Chairman of the Offsite Review Committee.

15.7.8.3 Plant Operating Procedures

The ODCM and the PCP shall be established and maintained in accordance with the provisions of specification 15.6.8. Effluent and environmental monitoring shall be addressed in the Quality Assurance Program.

15.7.8.4 RETS Reporting Requirements

The following written reports shall be submitted to the Administrator, U.S. Nuclear Regulatory Commission Region III with a copy to the Director, Office of Inspection and Enforcement, USNRC, Washington, D.C. 20555 within the time periods specified.

A. Semiannual Monitoring Report

A report within 60 days after January 1 and July 1 each year for the six month period or fraction thereof, ending June 30 and December 31 containing:

1. Information relative to the quantities of liquid, gaseous and solid radioactive effluents released from the facility, and effluent volumes used in maintaining the releases

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NOS. 110 AND 113 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

1.0 INTRODUCTION

In a letter dated April 10, 1986, Wisconsin Electric Power Company (the licensee), submitted an application for amendment of the Point Beach, Units 1 and 2 Technical Specifications (TS). The proposed amendment would revise 15.6.10, "Plant Operation Records." Additionally, the proposed amendment would revise numerous other TS to correct minor administrative errors.

The staff reviewed the licensee's April 10, 1986 amendment application and determined that additional information was required. Subsequently, by letter dated May 5, 1987, the staff issued a Request for Additional Information (RAI) to the licensee. The licensee responded to the RAI in a letter dated July 17, 1987.

The staff based its review on NUREG-0452, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors, Rex. 4," Title 10 of the Code of Federal Regulations, the Point Beach Nuclear Power Plant Modified Amended Security Plan, and the Point Beach 1 and 2 Technical Specifications.

2.0 EVALUATION

The licensee's proposed amendment to the Point Beach, Units 1 and 2 Technical Specifications consists of two parts: 1) revision to TS 15.6.10, "Plant Operation Records," and 2) revisions to numerous TS to correct administrative errors. Each part is discussed below.

The licensee's proposed revision to TS 15.6.10 changes the required retention period of plant operating records. Parts A, B, C, D, L, and M of the specification would be revised to require a retention period of 5 years. The current retention period is 6 years. Parts E, G, H, I, J, K, P, Q, R, V, and S of the specification would be revised to require retention for the "duration of operating license." The current retention period is "permanent." These changes to TS 15.6.10 are consistent with the Standard Technical Specifications for Westinghouse Pressurized Water Reactors, and therefore are acceptable.

Additional changes to TS 15.6.10 include:

- Parts N, T and U of the specifications will be deleted. Part N will be incorporated into Part M, and Parts T and U will be incorporated into Parts B and D. These changes are administrative in nature, have no effect on safety and are, therefore, acceptable.

- In accordance with 10 CFR 71.91(a) Parts T and U will be added to the specification. Part T requires retention of records regarding shipment of radioactive material having a specific activity of greater than 0.002 microcurie/gram for 2 years. Part U establishes requirements in the specification for retention of records concerning the Point Beach Modified Amended Security Plan. These changes are administrative in nature, have no effect on safety and are, therefore, acceptable.

- Part O will be revised to require that records of training, qualification and requalification for NRC-licensed personnel be retained until the operator's license is renewed. This change is in accordance with 10 CFR Part 55 (55.59(c)(5)(i)), and is, therefore, accept-able. Additionally Part O will be revised to require record retention for fire brigade member training for 3 years. This change is an accordance with 10 CFR Part 50 Appendix R, Section III. I.4 requirements and is, therefore, acceptable.

The licensee also proposed revisions to numerous TS to correct administrative errors. Each of these revisions is discussed below.

- TS 15.6.3.2, 15.6.3.3, 15.6.5.1.2 and figure 15.6.2-2 will be revised to reflect a reorganization which eliminated the position of Superintendent-Chemistry and Health Physics. The revision also designated the position of Radiochemist as being a regular member of the Point Beach Manager's Supervisory Staff.

- The wording of TS 15.7.8.1 will be changed from "responsibilities" to "duties". This will make this specification compatible with the referenced specification.

- TS 15.4.4.III.B, 15.6.9.1.B.2.a, and Table 15.3.5-5, Items 7 and 13 will be revised to correct erroneous references to other specifications.

- The basis of T.S.15.4.5 will be rewritten to remove ambiguities existing in the present wording.

- References to "FFDSAR" will be changed to "FSAR" on pages 15.1-2, 15.3.1-14A, 15.4.6-2, 15.6.9-1, and 15.6.10-1.

- Errors in spelling/punctuation will be corrected on pages 15.3.1-15, 15.3.10-6, 15.3.12-1, 15.3.13-2, 15.4.4-7, 15.4.4-11, 15.4.15-3 and 15.6.12-1.

In its April 10, 1986 letter, the licensee requested that TS 15.6.9.2.F be revised to remove ambiguity in the existing wording. Amendments 102 (for Unit 1) and 105 (for Unit 2) dated June 27, 1986 deleted TS 15.6.9.2.D; therefore, TS 15.6.9.2.F was relettered TS 15.6.9.2.E. TS 15.6.9.2.E currently in the Technical Specification is identical to the revision requested by the licensee. Accordingly, the revision originally requested by the licensee is not needed and has not been made.

Additionally, the licensee's letter dated July 17, 1987 retracted its request for amendments to Technical Specification 15.6.10.0, regarding the deletion of the term "key personnel." Accordingly, no such change to this specification will be made. Similarly, based on discussions with the licensee, the proposed change to Item 4, "Reactor Coolant System Subcooling," of Table 15.3.5-5 has been withdrawn. The licensee will evaluate the need for this change, and if necessary, request the change in a future Technical Specification amendment application. Accordingly, Table 15.3.5-5 is not revised by these amendments.

The staff has reviewed the above changes and concludes that these changes are administrative in nature and have no effect on safety, and are, therefore, acceptable.

ENVIRONMENTAL CONSIDERATION

These amendments relate to changes in recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the amendments meet 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 the amendments.

CONCLUSION

The staff has 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: January 5, 1988

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