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. 63
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 February 17, 1977, 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.

DESIGNATED ORIGINAL

Certified By ______________________

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. 63 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: August 31, 1982

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 63 TO FACILITY OPERATING LICENSE NO. DPR-24

AMENDMENT NO. 68 TO FACILITY OPERATING LICENSE NO. DPR-27

DOCKET NOS. 50-266 AND 50-301

Revise Appendix A as follows:

Remove Pages	Insert Pages
15.i	15.i
15.4.2-1	15.4.2-1
15.4.2-1c	15.4.2-1c
15.4.2-1d	15.4.2-2
15.4.2-2	-
15.4.2-3	-
15.4.2-4	-
15.4.2-5	-
15.6.9-7	15.6.9-7

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15.4.2 IN-SERVICE INSPECTION OF SAFETY CLASS COMPONENTS

Applicability

Applies to in-service inspection of Safety Class Components.

Objectives

To provide assurance of the continuing integrity of the safety class systems.

Specifications

A. Steam Generator Tube Inspection Requirements

1. Tube Inspection

Entry from the hot-leg side with examination from the point of entry completely around the U-bend to the top support of the cold-leg is considered a tube inspection.

2. Sample Selection and Testing

Selection and testing of steam generator tubes shall be made on the following basis:

(a) One steam generator of each unit shall be inspected during inservice inspection in accordance with the following requirements:

1. The inservice inspection may be limited to one steam generator on an alternating sequence basis. This examination shall include at least 6% of the tubes if the results of the first or a prior inspection indicate that both generators are performing in a comparable manner.

2. When both steam generators are required to be examined by Table 15.4.2-1 and if the condition of the tubes in one generator is found to be more severe than in the other steam generator of a unit, the steam generator sampling sequence at the subsequent inservice inspection shall be modified to examine the steam generator with the more severe condition.

(b) The minimum sample size, inspection result classification and the associated required action shall be in conformance with the requirements specified in Table 15.4.2-1. The results of each sampling examination of a steam generator shall be classified into the following three categories:

Defect is an imperfection of such severity that it exceeds the minimum acceptable tube wall thickness of 50%. A tube containing a defect is defective.

Plugging limit is the imperfection depth beyond which the tube must be removed from service, because the tube may become defective prior to the next scheduled inspection. The plugging limit is 40% of the nominal tube wall thickness.

6. Corrective Measures

All tubes that leak or have degradation exceeding the plugging limit shall be plugged prior to return to power from a refueling or inservice inspection condition.*

7. Reports

(a) After each inservice examination, the number of tubes plugged in each steam generator shall be reported to the Commission as soon as practicable.

(b) The complete results of the steam generator tube inservice inspection shall be included in the Operating Report for the period in which the inspection was completed. In addition all results in Category C-3 of Table 15.4.2-1 shall be reported to the Commission prior to resumption of plant operation.

(c) Reports shall include:

1. Number and extent of tubes inspected

2. Location and percent of all thickness penetration for each indication

3. Identification of tubes plugged

(d) Reports required by Table 15.4.2-1 - Steam Generator Tube Inspection - shall provide the information required by Specification 15.4.2.A.7(b) and a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

B. In-Service Inspection of Safety Class Components Other than Steam Generator Tubes

1. Inservice inspection of ASME Code Class 1, Class 2 and Class 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g) modified by Section 50.55a(b), except where specific written relief is granted by the NRC, pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

*Point Beach Nuclear Plant Unit 1 may be operated at power with up to six tubes in one steam generator having degradation exceeding the plugging limit provided those tubes have been repaired by insertion of sleeves into the tubes to bridge the degraded or defective portion of the tube. The plugging limit is 35% of the nominal sleeve wall thickness for tubes that have been repaired by sleeving.

2. Inservice testing of ASME Code Class 1, Class 2 and Class 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g) modified by Section 50.55a(b), except where specified written relief is granted by the NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

3. Containment isolation valves will be tested in accordance with Technical Specification 15.4.4 instead of Section IWV-3420, Valve Leak Rate Test.

Bases

The proposed inspection program , where practical, in compliance with the recommendations of ASME Boiler a d Pressure Vessel Code, Section XI, Summer 1971 Addenda. It must be recogn zed, however, that equipment and techniques to perform the inspection are still in development. It is recognized, however, that examinations in certain areas are necessary and therefore a schedule is proposed that includes areas and frequencies that are believed practical at this time for this reactor. In most areas scheduled for test, a detailed pre-service mapping will be conducted using techniques which can be used for post-operation inspections. The areas indicated for inspection represent those of relatively high stress and therefore will serve to indicate potential problems before significant flaws develop there or at other areas. As more experience is gained in operation of pressurized-water reactors, the recommended time schedule and location of inspection might be altered, or should new techniques be developed, consideration will be given to incorporate these new techniques into this inspection program.

The use of conventional non-destructive, direct visual and remote visual test techniques can be applied to the inspection of all primary loop components except for the reactor vessel. The reactor vessel presents special problems because of the radiation levels and remote underwater accessibility to this component. Because of these limitations on access to the reactor vessel, several steps have been incorporated into the design and manufacturing procedures in preparation for non-destructive test techniques which may be available in the future.(1)

The techniques for in-service inspection include the use of visual inspections, volumetric (ultrasonic or radiographic) and surface (dye penetrant or magnetic particle) testing of selected parts during refueling periods.

The intent of the inspection is the detection of flaws large enough to initiate fast fracture and gross leakage prior to subsequent inspection. At this time it is judged that such a flaw is substantially larger than 1/2 inch by 1 inch which is the degree of detectability. The inspection method is designed to detect flaws of this magniture.

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(1) FSAR - Section 4.4

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

4. Abnormal degradation of systems other than those specified in 15.6.8.2.A.3 above designed to contain radioactive material resulting from the fission process.

15.6.9.3 UNIQUE REPORTING REQUIREMENTS

The following written reports shall be submitted to the Director, Office of Nuclear Reactor Regulation, USNRC:

A. Each integrated leak test shall be the subject of a summary technical report, including results of the local leak rate tests and isolation valve leak rate tests since the last report. The report shall include analysis and interpretations of the results which demonstrate compliance with specified leak rate limits.

B. Deleted

C. Submission of a report within 60 days after January 1 and after July 1 each year for the six-month period or fraction thereof, ending June 30 and December 31 containing:

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. 68
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 November 27, 1978, 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.

DESIGNATED ORIGINAL

Certified By ______________________

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. 68 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 31, 1982

ATTACHMENT TO LICENSE AMENDMENTS

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2. When both steam generators are required to be examined by Table 15.4.2-1 and if the condition of the tubes in one generator is found to be more severe than in the other steam generator of a unit, the steam generator sampling sequence at the subsequent inservice inspection shall be modified to examine the steam generator with the more severe condition.

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__Plugging Limit__ is the imperfection depth beyond which the tube must be removed from service, because the tube may become defective prior to the next scheduled inspection. The plugging limit is 40% of the nominal tube wall thickness.

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All tubes that leak or have degradation exceeding the plugging limit shall be plugged prior to return to power from a refueling or inservice inspection condition.

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