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Wisconsin Electric Power Company
Wisconsin Michigan Power Company
ATTN: Mr. Sol Burstein
Executive Vice President
231 West Michigan Street
Milwaukee, Wisconsin 53201

Gentlemen:

By our letter dated July 12, 1976, we transmitted to you Amendments Nos. 10 and 12 to Facility Licenses Nos. DPR-24 and DPR-27 for the Point Beach Nuclear Plant, Units Nos. 1 and 2, which included Revised Technical Specifications for your facilities.

Due to an administrative error pages 15.4.2-1a and 15.4.2-1c included some incorrect data. As a result of a collating error, page 15.4.2-4 was omitted. Enclosed are corrected pages 15.4.2-1a, 15.4.2-1c, and 15.4.2-4 for inclusion in your copy.

Sincerely,

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
Revised Technical
Specification Pages

cc: See next page

OFFICE	ORB#3	ORB#3	ORB#3			
SURNAME	CParrish:kmb	JWetmore	GLear			
DATE	8/10/76	8/10/76	8/10/76			

Wisconsin Electric Power Company
Wisconsin Michigan Power Company

cc:

Mr. Bruce Churchill, Esquire
Shaw, Pittman, Potts and Trowbridge
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Mr. Norman Clap, Chairman
Public Service Commission
of Wisconsin
Hill Farms State Office Building
Madison, Wisconsin 53702

Mr. Arthur M. Fish
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University of Wisconsin -
Stevens Point Library
Stevens Point, Wisconsin 54481

Wisconsin Electric Power Company
ATTN: Mr. Glen Reed
Plant Superintendent
Point Beach Plant
231 West Michigan Street
Milwaukee, Wisconsin 53201

Category 1: less than 5% of the tota' umber of tubes examined are degraded but none are defective.

Category C-2: Between 5% and 10% of the total number of tubes examined are degraded, but none are defective or one tube to not more than 1% of the sample is defective.

Category C-3: More than 10% of the total number of tubes examined are degraded, but none are defective or more than 1% of the sample is defective.

In the first sample of a given steam generator during any inservice inspection, degraded tubes not beyond the plugging limit detected by the prior examinations in that steam generator shall be included in the above percentage calculations, only if these tubes are demonstrated to have a further wall penetration of greater than 10% of the nominal tube wall thickness.

(c) Tubes shall be selected for examination primarily from those areas of the tube bundle where service experience has shown the most severe tube degradation.

(d) In addition to the sample size specified in Table 15.4.2-1, the tubes examined in a given steam generator during the first examination of any inservice inspection shall include all non-plugged tubes in that steam generator that from prior examination were degraded.

(e) During the second and third sample examinations of any inservice inspection, the tube inspection may be limited to those sections of the tube lengths where imperfections were detected during the prior examination.

3. Examination Method and Requirements

(a) Steam generator tubes shall be examined in accordance with the method prescribed in Article 8 - "Eddy Current Examination of Tubular Products," as contained in ASME Boiler and Pressure Vessel Code - Section V - "Nondestructive Examination".

(b) The examination method of 15.4.2.A3(a) shall apply until Appendix IV, "Eddy Current Examination Method of Non-Ferromagnetic Steam Generator Heat Exchanger tubing" is incorporated and become effective rules of the ASME Boiler and Pressure Vessel Code, Section XI - Inservice Inspection of Nuclear Power Plant Components. At that time, the rules of ASME Code, Section XI shall be used in lieu of 15.4.2.A3(a).

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.

B. 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.

C. Reports

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

2. 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.

3. Reports shall include:

(a) Number and extent of tubes inspected

(b) Location and percent of all thickness penetration for each indication

(c) Identification of tubes plugged

4. Reports required by Table 15.4.2-1 - Steam Generator Tube Inspection shall provide the information required by Specification 15.4.2.C.2 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 Reactor Coolant System Components Other Than Steam Generator Tubes

The in-service inspection program is generally based on the recommendations of ASME Boiler and Pressure Vessel Code, Section XI, Summer 1971 Addenda as practical for a plant whose design and construction preceded issuance of the recommendations. The commitments herein are made assuming that the necessary inspection

NOTE (2): Threads in vessel not included: these are relatively low stress areas and ability to do a meaningful examination is doubtful. Stud stretching will be done after each refueling and is considered a better test.

NOTE (3): No particular patches prepared; general inspection will be made.

NOTE (4): Visual and surface of RV nozzle to pipe welds will be top surface only.

NOTE (5): Subsequent tests will be scheduled based on results of examinations made at first 40 month interval.

NOTE (6): Internals will be inspected as accessible during normal refueling. Removal of core barrel to allow additional inspection of reactor vessel internal areas shall be done once during inspection interval. If core barrel is removed prior to end of period (120 months), inspection for that period will be made when barrel is removed; otherwise, barrel will be removed at end of period specifically to allow inspection.

Bases

The proposed inspection program is, where practical, in compliance with the recommendations of ASME Boiler and Pressure Vessel Code, Section XI, Summer 1971 Addenda. It must be recognized, 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.