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Docket Nos. 50-266
and 50-301

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

Gentlemen:

The Commission has issued the enclosed Amendments Nos. 10 and 12 to Facility Operating Licenses Nos. DPR-24 and DPR-27 for the Point Beach Nuclear Plant, Units Nos. 1 and 2. The amendments consist of changes to the Technical Specifications and are in accordance with your applications dated August 14, 1974 and August 30, 1975.

The amendments will revise the provisions in the Technical Specifications for primary to secondary leak rate limits and would add steam generator tube surveillance requirements to the Technical Specifications.

Copies of the related Safety Evaluation and the Federal Register Notice also are enclosed.

Sincerely,

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

Enclosures:
1. Amendment No. 10
2. Amendment No. 12
3. Safety Evaluation
4. Federal Register Notice

cc w/encls:
See next page

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

WISCONSIN ELECTRIC POWER COMPANY
WISCONSIN MICHIGAN POWER COMPANY

DOCKET NO. 50-266

POINT BEACH NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 10
License No. DPR-24

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

   A. The applications for amendment by Wisconsin Electric Power Company and Wisconsin Michigan Power Company (the licensees) dated August 14, 1974 and August 30, 1975, comply 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; and

   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.

   E. An environmental statement or negative declaration need not be prepared in connection with the issuance of this amendment.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 12, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 10

FACILITY OPERATING LICENSE NO. DPR-24

DOCKET NO. 50-266

Replace pages 15.3.1-11 through 15.3.1-14, 15.4.2-1 and Table 15.4.2-1 with the attached revised pages. No change has been made on pages 15.4.2-1 and 15.4.2-2.

Add pages 15.3.1-14a and 15.4.2-1a through 15.4.2-1d.

D. LEAKAGE OF REACTOR COOLANT

Specification:

1. If leakage of reactor coolant from the reactor coolant system is indicated to exceed 1 gpm by the means available such as water inventory balances, monitoring equipment or direct observation, a follow-up evaluation of the safety implications shall be initiated as soon as practicable but no later than within 4 hours. Any indicated leak shall be considered to be a real leak until it is determined that either (1) a safety problem does not exist or (2) that the indicated leak cannot be substantiated by direct observation or other indication.

2. If the indicated reactor coolant leakage is substantiated and is not evaluated as safe or is determined to exceed 10 gpm, reactor shutdown shall be initiated as soon as practicable, but no later than within 24 hours after the leak was first detected.

3. The nature of the leak as well as the magnitude of the leak shall be considered in the safety evaluation. If plant shutdown is necessary per specification 2 above, the rate of shutdown and the conditions of shutdown shall be determined by the safety evaluation for each case and justified in writing as soon thereafter as practicable. The safety evaluation shall assure that the exposure of offsite personnel to radiation from the primary system coolant activity is within the guidelines of 10 CFR 20.

4. If the leakage is determined to be primary to secondary steam generator leakage in excess of 500 GPD in either steam generator, the reactor shall be shutdown and the plant placed in the cold shutdown condition within 36 hours after detection.

5. If any reactor coolant leakage exists through a non-isolable fault in a reactor coolant system component (exterior wall of the reactor vessel, piping, valve body, pressurizer or steam generator head), the reactor shall be shut down, and cooldown to the cold shutdown condition shall be initiated within 24 hours of detection.

6. The reactor shall not be restarted until the leak is repaired or until the problem is otherwise corrected.

7. When the reactor is in power operation, two reactor coolant leak detection systems of different operating principles shall be in operation, with one of the two systems sensitive to radioactivity. The systems sensitive to radioactivity may be out-of-service for 48 hours provided two other means are available to detect leakage.

8. Secondary coolant gross radioactivity shall be monitored continuously by an air ejector gas monitor.
Secondary coolant gross radioactivity shall be measured weekly.
If the air ejector monitor is not operating, the secondary coolant gross radioactivity shall be measured daily to evaluate steam generator leak tightness.

Basis:

Water inventory balances, monitoring equipment, radioactive tracing, boric acid crystalline deposits, and physical inspections can disclose reactor coolant leaks. Any leak of radioactive fluid, whether from the reactor coolant system primary boundary or not, can be a serious problem with respect to in-plant radioactivity contamination and cleanup or it could develop into a still more serious problem; and therefore, first indications of such leakage will be followed up as soon as practicable.

Every reasonable effort will be made to reduce reactor coolant leakage to the lowest possible rate. Although some leak rates may be tolerable from a dose point of view, especially if they are to closed systems, it must be recognized that leaks in the order of drops per minute through any of the walls of the primary system could be indicative of materials failure such as stress corrosion cracking. If depressurization, isolation and/or other safety measures are not taken promptly, these small leaks could develop into much larger leaks. Therefore, the nature of the leak, as well as the magnitude of the leakage, must be considered in the safety evaluation. The provision pertaining to a non-isolable fault in a reactor coolant system component is not intended to cover steam generator tube leakages, valve or packings, instrument fittings or similar primary system boundaries not indicative of major component exterior wall leakage.

The specific leak rate limit identified for primary-to-secondary leakage of 500 GPD per steam generator provides an additional margin of safety with regard to the potential for large steam generator tube failure in that action to shutdown the plant will be explicitly required at a low leakage rate threshold.

When the source and location of leakage has been identified, the situation can be evaluated to determine if operation can safely continue. This evaluation will be performed by the Manager's Supervisory Staff according to routine established in Section 15.6. Under these conditions, an allowable leakage rate of 10 gpm has been established. The explained leakage rate of 10 gpm is also well within the capacity of one charging pump, and makeup would be available even under the loss of offsite power condition.

If leakage is to the cont   ment, it may be identified b 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.

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

FFDSAR Section 6.5, 11.2.3

15.4.2 IN-SERVICE INSPECTION OF PRIMARY SYSTEM COMPONENTS

Applicability

Applies to in-service inspection of Reactor Coolant System Components.

Objectives

To provide assurance of the continuing integrity of the Reactor Coolant System.

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:

TABLE 15.4.2-1

STEAM GENERATOR TUBE INSPECTION PER UNIT
POINT BEACH UNITS 1 & 2

1ST SAMPLE EXAMINATION			2ND SAMPLE EXAMINATION		3RD SAMPLE EXAMINATION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S tubes per Steam Generator (S.G.) S=3(N/n)% where: N is the number of steam generators in the plant = 2 n is the number of steam generators inspected during an examination	C-1	Acceptable for Continued Service	N/A	N/A	N/A	N/A
	C-2	Plug tubes exceeding the plugging limit and proceed with 2nd sample examination of 2S tubes in same steam generator	C-1	Acceptable for continued Service	N/A	N/A
			C-2	Plug tubes exceeding the plugging limit and proceed with 3rd sample examination of 4S tubes in same steam generator	C-1	Acceptable for Continued Service
					C-2	Plug tubes exc. plug limit. Acceptable f continued service
					C-3	Perform action requi under C-3 of 1st sample examination
			C-3	Perform action required under C-3 of 1st sample examination	N/A	N/A
	C-3	Inspect essentially all tubes in this S.G., plug tubes exceeding the plugging limit and proceed with 2nd sample examination of 2S tubes in the other steam generator. Report results to NRC within 24 hours in accordance with Technical Specification 15.6.5.2.A.3.	C-1 in other S.G.	Acceptable for Continued Service	N/A	N/A
			C-2 in other S.G.	Perform action required under C-2 of 2nd sample examination above	N/A	N/A
			C-3 in other S.G.	Inspect essentially all tubes in S.G. and plug tubes exceeding the plugging limit. Report to NRC within 24 hours in accordance with Technical	N/A	N/A

Category C-1: less than 5% of the total number 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 XI - "Inservice Inspection of Nuclear Power Plant Components."

(b) The examination method of 15.4.2.A3(a) shall be supplemented on an interim basis by the requirements specified in Appendix A of this Specification, 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 Appendix A.

4. Inspection Intervals

(a) Inservice inspections shall not be more than 24 calendar months apart.

(b) The inservice inspections may be scheduled to be coincident with refueling outages or any plant shutdown, provided the inspection intervals of 15.4.2.A.4(a) are not exceeded.

(c) If two consecutive inservice inspections covering a time span of at least 12 months yield results that fall in C-1 category, the inspection frequency may be extended to 40 month intervals.

(d) If the results of the inservice inspection of steam generator tubing conducted in accordance with Table 15.4.2-1 requires that a third sample examination must be performed, and the results of this fall in category C-3, the inspection frequency shall be reduced to not more than 20 months intervals. The reduction shall apply until a subsequent inspection demonstrates that a third sample examination is not required.

(e) Unscheduled inspections shall be conducted in accordance with Specifications 15.4.2.A.2 on any steam generator with primary-to-secondary tube leakage exceeding Specification 15.3.1.D.4. All steam generators shall be inspected in the event of a seismic occurrence greater than an operating basis earthquake, a LOCA requiring actuation of engineered safeguards, or a main steam line or feedwater line break.

5. Acceptance Limits

(a) Definitions:

Imperfection is an exception to the dimension, finish, or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

Degradation means a service induced cracking, wastage, wear, or general corrosion occurring on either inside or outside of a tube.

Degraded Tube is a tube that contains imperfections caused by degradation greater than 20% of the nominal tube wall thickness.

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, as practical for a plant whose design and construction preceded issuance of the recommendations. The commitments herein are made assuming that the necessary inspection techniques will be commercially available and that necessary accessibility can be gained to components to allow inspection. At the end of the first five years of the inspection period, a review of the inservice inspection program will be conducted. This review will evaluate the results obtained to date in view of possible modifications to the inspection program. These modifications may increase or decrease surveillance requirements as experience dictates.

IN-SERVICE INSPECTION PROGRAM (NOTE 1)

By 1/3 of inspection period - 40 months

RV flange and head flange welds	Volumetric of 25% of each weld
RV nozzle to vessel welds and inside radii	Volumetric of 2 outlet nozzles
RV nuts and studs	Volumetric and visual on 25% (Note 2)
RV closure washers and bushings	Visual of 25%
Closure head cladding	Visual and surface of 2 patches
Pressurizer cladding	Visual (Note 3)
Reactor vessel nozzles to pipe; pressurizer surge nozzle to pipe; steam generator primary nozzles to pipe welds	Visual, surface, and volumetric of 25% of welds (Note 4)

Circumferential pipe welds	Visual and volumetric of [illegible]% of welds
Surveillance samples	Tensile, Charpy,. wedge-opening-load tests (Note 5)
Reactor coolant pump flywheels	Visual, as accessible without removing flywheel

<u>By 2/3 of inspection period - 80 months</u>

RV flange and head flange welds	Volumetric of additional (over previous inspection) 25% of each weld
RV nozzle to vessel welds and inside radii	Volumetric of 2 SIS nozzles
RV nuts and studs	Volumetric and visual on additional (over previous inspection) 25% (Note 2)
RV closure washers and bushings	Visual of additional (over previous inspection) 25%
Closure head cladding	Visual and surface of additional (over previous inspection) 2 patches
Pressurizer cladding	Visual (Note 3)
Reactor vessel nozzles to pipe; pressurizer surge nozzle to pipe; steam generator primary nozzles to pipe welds	Visual, surface and volumetric of additional (over previous inspection) 25% (Note 4)
Circumferential pipe welds	Visual and volumetric of additional (over previous inspection) 6% of welds
Reactor coolant pump flywheels	Volumetric, as accessible without removing flywheel

<u>End of inspection period - 120 months</u>

RV shell welds	Volumetric of 10% of longitudinal and 5% of circumferential welds
Reactor head welds	Volumetric of 10% of longitudinal and 5% of circumferential welds
RV flange and head flange welds	Volumetric of remainder (left from previous inspections) of each weld
RV nozzle to vessel welds and inside radii	Volumetric of 2 inlet nozzles
RV nuts and studs	Volumetric and visual of remainder (left from previous inspections) (Note 2)

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

WISCONSIN ELECTRIC POWER COMPANY
WISCONSIN MICHIGAN POWER COMPANY

DOCKET NO. 50-301

POINT BEACH NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 12
License No. DPR-27

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

   A. The applications for amendment by Wisconsin Electric Power Company and Wisconsin Michigan Power Company (the licensees) dated August 14, 1974 and August 30, 1975, comply 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; and

   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.

   E. An environmental statement or negative declaration need not be prepared in connection with the issuance of this amendment.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 12, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 12

FACILITY OPERATING LICENSE NO. DPR-27

DOCKET NO. 50-301

Replace pages 15.3.1-11 through 15.3.1-14, 15.4.2-1 and Table 15.4.2-1 with the attached revised pages. No change has been made on pages 15.4.2-1 and 15.4.2-2.

Add pages 15.3.1-14a and 15.4.2-1a through 15.4.2-1d.

D. LEAKAGE OF REACTOR COOLANT

Specification:

1. If leakage of reactor coolant from the reactor coolant system is indicated to exceed 1 gpm by the means available such as water inventory balances, monitoring equipment or direct observation, a follow-up evaluation of the safety implications shall be initiated as soon as practicable but no later than within 4 hours. Any indicated leak shall be considered to be a real leak until it is determined that either (1) a safety problem does not exist or (2) that the indicated leak cannot be substantiated by direct observation or other indication.

2. If the indicated reactor coolant leakage is substantiated and is not evaluated as safe or is determined to exceed 10 gpm, reactor shutdown shall be initiated as soon as practicable, but no later than within 24 hours after the leak was first detected.

3. The nature of the leak as well as the magnitude of the leak shall be considered in the safety evaluation. If plant shutdown is necessary per specification 2 above, the rate of shutdown and the conditions of shutdown shall be determined by the safety evaluation for each case and justified in writing as soon thereafter as practicable. The safety evaluation shall assure that the exposure of offsite personnel to radiation from the primary system coolant activity is within the guidelines of 10 CFR 20.

4. If the leakage is determined to be primary to secondary steam generator leakage in excess of 500 GPD in either steam generator, the reactor shall be shutdown and the plant placed in the cold shutdown condition within 36 hours after detection.

5. If any reactor c lant leakage exists through on-isolable fault in a reactor coolant system component (exterior wall of the reactor vessel, piping, valve body, pressurizer or steam generator head), the reactor shall be shut down, and cooldown to the cold shutdown condition shall be initiated within 24 hours of detection.

6. The reactor shall not be restarted until the leak is repaired or until the problem is otherwise corrected.

7. When the reactor is in power operation, two reactor coolant leak detection systems of different operating principles shall be in operation, with one of the two systems sensitive to radioactivity. The systems sensitive to radioactivity may be out-of-service for 48 hours provided two other means are available to detect leakage.

8. Secondary coolant gross radioactivity shall be monitored continuously by an air ejector gas monitor.
Secondary coolant gross radioactivity shall be measured weekly.
If the air ejector monitor is not operating, the secondary coolant gross radioactivity shall be measured daily to evaluate steam generator leak tightness.

Basis:

Water inventory balances, monitoring equipment, radioactive tracing, boric acid crystalline deposits, and physical inspections can disclose reactor coolant leaks. Any leak of radioactive fluid, whether from the reactor coolant system primary boundary or not, can be a serious problem with respect to in-plant radioactivity contamination and cleanup or it could develop into a still more serious problem; and therefore, first indications of such leakage will be followed up as soon as practicable.

Every reasonable effort will be made to reduce reactor coolant leakage to the lowest possible ra' Although some leak rates r be tolerable from a dose point of view, especially if they are to closed systems, it must be recognized that leaks in the order of drops per minute through any of the walls of the primary system could be indicative of materials failure such as stress corrosion cracking. If depressurization, isolation and/or other safety measures are not taken promptly, these small leaks could develop into much larger leaks. Therefore, the nature of the leak, as well as the magnitude of the leakage, must be considered in the safety evaluation. The provision pertaining to a non-isolable fault in a reactor coolant system component is not intended to cover steam generator tube leakages, valve or packings, instrument fittings or similar primary system boundaries not indicative of major component exterior wall leakage.

The specific leak rate limit identified for primary-to-secondary leakage of 500 GPD per steam generator provides an additional margin of safety with regard to the potential for large steam generator tube failure in that action to shutdown the plant will be explicitly required at a low leakage rate threshold.

When the source and location of leakage has been identified, the situation can be evaluated to determine if operation can safely continue. This evaluation will be performed by the Manager's Supervisory Staff according to routine established in Section 15.6. Under these conditions, an allowable leakage rate of 10 gpm has been established. The explained leakage rate of 10 gpm is also well within the capacity of one charging pump, and makeup would be available even under the loss of offsite power condition.

If leakage is to the cont ıment, it may be identified l 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.

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

FFDSAR Section 6.5, 11.2.3

15.4.2 IN-SERVICE INSPECTION OF PRIMARY SYSTEM COMPONENTS

Applicability

Applies to in-service inspection of Reactor Coolant System Components.

Objectives

To provide assurance of the continuing integrity of the Reactor Coolant System.

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:

TABLE 15.4.2-1

STEAM GENERATOR TUBE INSPECTION PER UNIT
POINT BEACH UNITS 1 & 2

1ST SAMPLE EXAMINATION			2ND SAMPLE EXAMINATION		3RD SAMPLE EXAMINATION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S tubes per Steam Generator (S.G.) S=3(N/n)% where: N is the number of steam generators in the plant = 2 n is the number of steam generators inspected during an examination	C-1	Acceptable for Continued Service	N/A	N/A	N/A	N/A
	C-2	Plug tubes exceeding the plugging limit and proceed with 2nd sample examination of 2S tubes in same steam generator	C-1	Acceptable for continued Service	N/A	N/A
			C-2	Plug tubes exceeding the plugging limit and proceed with 3rd sample examination of 4S tubes in same steam generator	C-1	Acceptable for Continued Service
					C-2	Plug tubes exc. plug limit. Acceptable for continued service
					C-3	Perform action required under C-3 of 1st sample examination
			C-3	Perform action required under C-3 of 1st sample examination	N/A	N/A
	C-3	Inspect essentially all tubes in this S.G., plug tubes exceeding the plugging limit and proceed with 2nd sample examination of 2S tubes in the other steam generator. Report results to NRC within 24 hours in accordance with Technical Specification 15.6.5.2.A.3.	C-1 in other S.G.	Acceptable for Continued Service	N/A	N/A
			C-2 in other S.G.	Perform action required under C-2 of 2nd sample examination above	N/A	N/A
			C-3 in other S.G.	Inspect essentially all tubes in S.G. and plug tubes exceeding the plugging limit. Report to NRC within 24 hours in accordance with Technical Specification 15.6.5.2.A.3.	N/A	N/A

Category C-1: less than 5% of the total number 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 XI - "Inservice Inspection of Nuclear Power Plant Components."

(b) The examination method of 15.4.2.A3(a) shall be supplemented on an interim basis by the requirements specified in Appendix A of this Specification, 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 Appendix A.

4. Inspection Intervals

(a) Inservice inspections shall not be more than 24 calendar months apart.

(b) The inservice inspections may be scheduled to be coincident with refueling outages or any plant shutdown, provided the inspection intervals of 15.4.2.A.4(a) are not exceeded.

(c) If two consecutive inservice inspections covering a time span of at least 12 months yield results that fall in C-1 category, the inspection frequency may be extended to 40 month intervals.

(d) If the results of the inservice inspection of steam generator tubing conducted in accordance with Table 15.4.2-1 requires that a third sample examination must be performed, and the results of this fall in category C-3, the inspection frequency shall be reduced to not more than 20 months intervals. The reduction shall apply until a subsequent inspection demonstrates that a third sample examination is not required.

(e) Unscheduled inspections shall be conducted in accordance with Specifications 15.4.2.A.2 on any steam generator with primary-to-secondary tube leakage exceeding Specification 15.3.1.D.4. All steam generators shall be inspected in the event of a seismic occurrence greater than an operating basis earthquake, a LOCA requiring actuation of engineered safeguards, or a main steam line or feedwater line break.

5. Acceptance Limits

(a) Definitions:

Imperfection is an exception to the dimension, finish, or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

Degradation means a service induced cracking, wastage, wear, or general corrosion occurring on either inside or outside of a tube.

Degraded Tube is a tube that contains imperfections caused by degradation greater than 20% of the nominal tube wall thickness.

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, as practical for a plant whose design and construction preceded issuance of the recommendations. The commitments herein are made assuming that the necessary inspection techniques will be commercially available and that necessary accessibility can be gained to components to allow inspection. At the end of the first five years of the inspection period, a review of the inservice inspection program will be conducted. This review will evaluate the results obtained to date in view of possible modifications to the inspection program. These modifications may increase or decrease surveillance requirements as experience dictates.

IN-SERVICE INSPECTION PROGRAM (NOTE 1)

By 1/3 of inspection period - 40 months

RV flange and head flange welds	Volumetric of 25% of each weld
RV nozzle to vessel welds and inside radii	Volumetric of 2 outlet nozzles
RV nuts and studs	Volumetric and visual on 25% (Note 2)
RV closure washers and bushings	Visual of 25%
Closure head cladding	Visual and surface of 2 patches
Pressurizer cladding	Visual (Note 3)
Reactor vessel nozzles to pipe; pressurizer surge nozzle to pipe; steam generator primary nozzles to pipe welds	Visual, surface, and volumetric of 25% of welds (Note 4)

Circumferential pipe welds	Visual and volumetric of 6% of welds
Surveillance samples	Tensile, Charpy, wedge-opening-load tests (Note 5)
Reactor coolant pump flywheels	Visual, as accessible without removing flywheel

By 2/3 of inspection period - 80 months

RV flange and head flange welds	Volumetric of additional (over previous inspection) 25% of each weld
RV nozzle to vessel welds and inside radii	Volumetric of 2 SIS nozzles
RV nuts and studs	Volumetric and visual on additional (over previous inspection) 25% (Note 2)
RV closure washers and bushings	Visual of additional (over previous inspection) 25%
Closure head cladding	Visual and surface of additional (over previous inspection) 2 patches
Pressurizer cladding	Visual (Note 3)
Reactor vessel nozzles to pipe; pressurizer surge nozzle to pipe; steam generator primary nozzles to pipe welds	Visual, surface and volumetric of additional (over previous inspection) 25% (Note 4)
Circumferential pipe welds	Visual and volumetric of additional (over previous inspection) 6% of welds
Reactor coolant pump flywheels	Volumetric, as accessible without removing flywheel

End of inspection period - 120 months

RV shell welds	Volumetric of 10% of longitudinal and 5% of circumferential welds
Reactor head welds	Volumetric of 10% of longitudinal and 5% of circumferential welds
RV flange and head flange welds	Volumetric of remainder (left from previous inspections) of each weld
RV nozzle to vessel welds and inside radii	Volumetric of 2 inlet nozzles
RV nuts and studs	Volumetric and visual of remainder (left from previous inspections) (Note 2)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENTS NOS. 10 AND 12 TO LICENSES DPR-24 AND DPR-27

WISCONSIN ELECTRIC POWER COMPANY
WISCONSIN MICHIGAN POWER COMPANY

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

DOCKETS NOS. 50-266 AND 50-301

Introduction

By letters dated August 30, 1974 and August 14, 1975 Wisconsin Electric Power Company (WEPCO) requested changes to the Technical Specifications appended to Facility Licenses DPR-24 and DPR-27 for Point Beach Nuclear Plant, Units 1 and 2. The proposed changes would (1) establish surveillance requirements for steam generator tubes, and (2) revise the primary to secondary leak rate limits, and make editorial corrections.

Discussion

In July, 1974, we requested the licensees of pressurized water reactors (PWR's) to submit proposed changes to their Technical Specifications that would establish requirements for a program of steam generator tube inspection. To provide guidance in developing an inspection program, licensees were advised, at that time, to refer to Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes", dated June 1974. Per our request, WEPCO submitted a program for Point Beach, Units 1 and 2, by letter dated August 30, 1974. We delayed implementation of the program for Point Beach, as well as other operating reactors, because Regulatory Guide 1.83 was being revised based upon developments in the state of art of steam generator tube inspection techniques, and inspection experience that was being gained at operating plants. Furthermore, industry wide practice already included voluntary inspection of steam generator tubes that, in many respects, was comparable to Regulatory Guide 1.83. Then in July 1975, Revision 1 to Regulatory Guide 1.83 was issued after receiving comments from the industry. The NRC staff has subsequently reviewed Regulatory Guide 1.83 in light of steam generator operating experience and inspection experience and we are now taking steps to incorporate steam generator tube inservice inspection requirements into the Technical Specifications for all operating PWR's.

The inspection requirements are in general agreement with Regulatory Guide 1.83, Revision 1, dated July 1975, but may deviate in some areas where the NRC staff has determined that the overall program would be enhanced.

In the case of Point Beach, Units 1 and 2, we are not only implementing steam generator tube inservice inspection requirements, we are also instituting a revised primary-to-secondary leakage limit. The licensee proposed revisions to the primary-to-secondary leakage rate limit by letter dated August 14, 1975. The revised leakage limit is intended to provide an additional margin of safety with regard to steam generator tube integrity by requiring plant shutdown at a lower leakage rate threshhold. The revised leakage limit will also serve to bring the Technical Specifications for Point Beach, Units 1 and 2 into closer agreement with more recently licensed PWR's.

Evaluation

(1) Surveillance Requirements for Steam Generator Tubes:

Structures, systems, and components important to safety of a nuclear power plant are designed, fabricated, constructed, and tested so as to provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. To continuously maintain such assurance, General Design Criterion 32 requires that components which are part of the reactor coolant pressure boundary be designed to permit periodic inspection and testing of important areas and features to assess their structural and leaktight integrity. The steam generator tubing is part of the reactor coolant system pressure boundary and is an important part of a major barrier against fission product release to the environment. It also acts as a barrier against steam release to the containment in the event of a LOCA. To act as an effective barrier, this tubing must be free of cracks, perforations, and general deterioration. For this reason, a program of periodic inservice inspection is being established to assure the continued integrity of the steam generator tubes over the service life of the plant.

Generally, the major elements of the steam generator tube inservice inspection program for Point Beach Units 1 and 2, consist of specified: (a) sample selection, (b) examination methods, (c) inspection intervals, (d) acceptance criteria, and (e) reporting requirements. Each of these major elements of the program is separately evaluated below.

(a) Sample Selection

The proposed sampling scheme, as modified by the staff and concurred in by the licensee, is generally patterned after Regulatory Guide 1.83, Revision 1, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes". However, there are some deviations from Regulatory Guide 1.83 that the staff requires to improve the program and/or reduce the potential radiation exposure of personnel that must perform the inspections. The sampling procedure for Point Beach, Units 1 and 2 is contained in Table 15.4.2-1 of the Technical Specifications. The principal deviations from Regulatory Guide 1.83 supplementary sampling requirements are evaluated below:

(i) Regulatory Position C.5.a, "Supplementary Sampling Requirements" recommends that if the eddy current inspection results during an inservice inspection indicate any tubes with previously undetected imperfections of 20% or greater depth, additional steam generators, if any, should be inspected. In other words, because of a single tube in one steam generator with previously undetected imperfection of 20% or greater depth but still well below the plugging limit, all steam generators in the plant should be inspected. This requirement would be unreasonably too severe and would certainly increase the unnecessary radiation exposures to the inspection personnel. The supplementary sampling requirements, as modified, require inspection of the additional steam generators only if the inspection results of the particular steam generator fall in the rather severe category of C-3 as described in Table 15.4.2-1 and thus minimize the unnecessary inspection of other steam generators.

(ii) Regulatory Guide 1.83, Revision 1 requires two additional inspections if the initial inspection results indicate that more than 10% of the inspected tubes have detectable wall penetration of greater than 20% or that one or more tubes inspected have an indication in excess of the plugging limit. The additional inspections require a complete tube inspection of 3% and 6% of the tubes. On the other hand, the program for Point Beach requires that twice the number of tubes be inspected during the preceding sample inspection but require concentrating on tubes only in the areas of the tube sheet array and on the portion of the tube where tubes with imperfections were found during the first sample inspection. We understand that this sampling scheme is similar to that currently practiced by the industry. The primary purpose of the additional inspections is to reassure the initial inspection results and to ensure the steam generator integrity, thus we believe that the modified additional inspection scheme represents an improvement to Regulatory Guide 1.83.

Based on the considerations discussed above, we have concluded that the sample selection scheme, as modified by the staff and concurred in by the licensee, is acceptable.

(b) Examination Method

The proposed examination methods, as modified by the staff and concurred in by the licensee, include nondestructive examination by eddy current testing. The specified methods are capable of locating and identifying stress corrosion cracks and tube wall thinning from chemical wastage, mechanical damage or other causes. Based on our review of these methods, and experience gained using these methods by the industry, we have concluded that the examination methods are acceptable.

(c) Inspection Intervals

The proposed inspection intervals, as modified by the staff and concurred in by the licensee, are compatible with those recommended in Regulatory Guide 1.83; and thus, are acceptable.

(d) Acceptance Criteria

The principle parameter used to determine whether any one steam generator tube is acceptable for continued service is the measured imperfection depth. In order to specify what level of imperfection is acceptable, a tube "plugging limit" is established. The "plugging limit" is defined in the Technical Specifications as the imperfection depth beyond which the tube must be removed from service, because the tube may become defective prior to the next scheduled inspection. For Point Beach, Units 1 and 2 the "plugging limit", as modified by the staff and concurred in by the licensee, is 40% of the nominal tube wall thickness.

The "plugging limit" is based on (1) the minimum tube wall thickness needed to maintain steam generator tube integrity during the limiting stress loadings associated with a loss of coolant accident (LOCA) combined with a Safe Shutdown Earthquake (SSE), and (2) an operational allowance to account for the time interval between inspections. Based on other evaluations made by the NRC staff1/, and analyses performed by Westinghouse on steam generator tube designs similar to Point Beach, we have concluded that a minimum tube wall thickness of 50% is adequate to sustain all the forces associated with a LOCA combined with an SSE. To provide an additional margin of safety, however, an operational allowance of 10% is incorporated into the "plugging limit" to insure tube integrity will be maintained until the next inservice inspection. This allowance is adequate for the carefully controlled secondary water chemistry conditions that are normally maintained at Point Beach. Therefore, the acceptable tube wall thickness needed for continued service is 50% plus 10% or 60% or alternately, the "plugging limit" (imperfection depth) is established as 40%. This limit will provide adequate protection against wastage type corrosion or part thru wall cracks.

Based on our review, the acceptance criteria, as modified by the staff and concurred in by the licensee, are acceptable.

(e) Reporting Requirements

Regulatory Guide 1.83, Revision 1, requires the licensee to report to the Commission and to wait for resolution and approval of the proposed remedial action when the inspection results exceed the limits specified in the Guide. It also states that additional sampling and more frequent inspection may be required. In the proposed Technical Specifications, as modified by the staff and concurred by the licensee, it is clearly stated what additional inspection the licensee must do without reporting to the NRC and limits the reporting requirements only to the most severe cases described in Table 15.4.2-1 of the Technical Specifications.

It is our position that the reporting requirements, as modified, are reasonable and will facilitate reporting of pertinent information without unnecessarily increasing plant downtime; and thus, are acceptable.

1/Supplemental Testimony of James P. Knight before the Atomic Safety and Licensing Appeal Board in the matter of Northern States Power Company, Docket Nos. 50-282/306.

In summary, we have concluded that the proposed steam generator tube inservice inspection program will provide added assurance of the continued integrity of the steam generator tubes; and thus, is acceptable.

(2) Primary to Secondary Leak Rate Limit

(a) The existing Technical Specification 15.3.1.D specifies a primary leak rate limit that is intended to envelope various leakage paths, including primary to secondary leakage. However, it does not contain an explicit primary to secondary leak rate limit. Consequently, by letter dated August 14, 1975, the licensee proposed revision of the primary to secondary leakage rate limit. The proposed change, as modified by the NRC staff and concurred in by the licensee, would specify a primary to secondary leak rate limit of 500 GPD (about 0.35 GPM) in either steam generator, and would require that the plant be placed in cold shutdown within 36 hours if the leak rate limit is exceeded.

The purpose of establishing a specific steam generator tube (primary to secondary) leakage rate limit is to assure that an acceptable level of tube integrity will be maintained during all normal or postulated accident conditions. Steam generator tube integrity needs to be maintained to ensure that (1) secondary coolant activity levels are maintained within acceptable limits during normal operation, (2) for postulated Loss of Coolant Accidents, excessive secondary to primary inleakage, that could aggravate the accident consequences, would not occur, and (3) for postulated Main Steam Line Break Accidents, excessive primary to secondary leakage, with resultant activity releases to the environment, would not occur.

Based on other evaluations made by the NRC staff[2/], and the results of tests performed on steam generator tubes like those at Point Beach, we have determined that it unlikely that a tube failure could occur in any tube having a through-wall crack limited to 0.5 inch in length under any normal or accident condition. This crack length, as demonstrated by test[2/], results in a primary to secondary leakage rate of 0.4 GPM under normal operating conditions. Consequently, it is our position that, with carefully controlled secondary water chemistry conditions like those normally maintained at Point Beach, steam generator tube integrity would be maintained under all normal and postulated accident

2/Testimony of Raymond R. Maccary before the Atomic Safety and Licensing Appeal Board, in the matter of Northern States Power Company, Docket Nos. 50-282/306.

conditions if primary to secondary leakage is kept below 0.4 GPM. On this basis, we believe that a leakage limit of 500 GPD (about 0.35 GPM) in either steam generator for Point Beach, Units 1 and 2, would provide a substantial margin of safety with regard to the potential for large tube failures. Therefore, we have concluded that the proposed change, as modified by the staff and concurred in by the licensee, is acceptable.

(b) The proposed editorial changes to Technical Specification 15.3.1.D would serve to correct and clarify the Technical Specifications. The specific change to Technical Specification 15.3.1.D.7 would delete the specific power level of 2% for specifying primary coolant system leak detection equipment operability requirements by simply stating that "power operation requires certain leak detection equipment operability. "Power operation" is defined in existing Technical Specification 15.1.h as reactor operation at power levels greater than 2%. Therefore, the proposed editorial change to Technical Specification 15.3.1.D.7 would not be material but would only be administrative; and thus, is acceptable.

The other editorial change to Technical Specification 15.3.1.D would delete the specific assumptions listed in the basis of the Technical Specification that were used to determine a steam generator leakage rate limit that was applicable to a previous Unit 1 core cycle only. These assumptions are not specifically applicable to the current core cycle for either unit or to the revised primary to secondary leak limits; and thus their deletion is acceptable.

We have concluded that the proposed editorial changes have no safety significance; and thus, are acceptable.

Environmental Finding

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental statement, negative declaration, or environmental impact appraisal need not 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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: July 12, 1976

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS NOS. 50-266 AND 50-301

WISCONSIN ELECTRIC POWER COMPANY
WISCONSIN MICHIGAN POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 10 and 12 to Facility Operating Licenses Nos. DPR-24 and DPR-27 issued to Wisconsin Electric Power Company and Wisconsin Michigan Power Company, which revised Technical Specifications for operation of the Point Beach Nuclear Plant Units Nos. 1 and 2, located in the town of Two Creeks, Manitowoc County, Wisconsin. The amendments are effective as of their date of issuance.

The amendments will revise the provisions in the Technical Specifications for primary to secondary leak rate limits and would add steam generator tube surveillance requirements to the Technical Specifications.

The applications for the amendments comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Notice of Proposed Issuance of Amendments to Facility Operating Licenses in connection with this action was published in the FEDERAL REGISTER on November 4, 1975 (40 F.R. 1247). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental statement, negative declaration or environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the applications for amendments dated August 14, 1974 and August 30, 1975, (2) Amendment No. 10 to License No. DPR-24, (3) Amendment No. 12 to License No. DPR-27, and (4) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Manitowoc Public Library, 808 Hamilton Street, Manitowoc, Wisconsin 54220.

A copy of items (2), (3) and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 12 day of July 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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