October 9, 1996

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Mr. Robert E. Link, Vice President
Nuclear Power Department
Wisconsin Electric Power Company
231 West Michigan Street, Room P379
Milwaukee, WI 53201

SUBJECT: AMENDMENT NOS. 169 AND 173 TO FACILITY OPERATING LICENSE NOS. DPR-24 AND DPR-27 - POINT BEACH NUCLEAR PLANT, UNIT NOS. 1 AND 2 (TAC NOS. M95668 AND M95669)

Dear Mr. Link:

The Commission has issued the enclosed Amendment Nos. 169 and 173 to Facility Operating License Nos. DPR-24 and DPR-27 for the Point Beach Nuclear Plant, Unit Nos. 1 and 2, respectively. The amendments revise the Technical Specifications (TS) in response to your application dated May 29, 1996, as supplemented by letter dated August 20, 1996.

These amendments revise TS Section 15.4.4, "Containment Tests," to incorporate the provisions of 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," Option B. Revisions have also been made to TS Sections 15.1, "Definitions," 15.3.6, "Containment System," and 15.6, "Administrative Controls," to support the proposed changes to Section 15.4.4.

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

Sincerely,

Original signed by:

Allen G. Hansen, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-266
and 50-301

Enclosures:
1. Amendment No. 169to DPR-24
2. Amendment No. 173to DPR-27
3. Safety Evaluation

cc w/encls: See next page

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* See previous concurrence

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 9, 1996

Mr. Robert E. Link, Vice President
Nuclear Power Department
Wisconsin Electric Power Company
231 West Michigan Street, Room P379
Milwaukee, WI 53201

SUBJECT: AMENDMENT NOS. 169 AND 173 TO FACILITY OPERATING LICENSE NOS. DPR-24 AND DPR-27 - POINT BEACH NUCLEAR PLANT, UNIT NOS. 1 AND 2 (TAC NOS. M95668 AND M95669)

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Docket Nos. 50-266
and 50-301

Enclosures:
1. Amendment No. 169 to DPR-24
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3. Safety Evaluation

cc w/encls: See next page

Mr. Robert E. Link, Vice President
Wisconsin Electric Power Company

Point Beach Nuclear Plant
Unit Nos. 1 and 2

cc:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-266

POINT BEACH NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 169
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 May 29, 1996, as supplemented August 20, 1996, 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. <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 169 , 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 45 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Allen G. Hansen, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of issuance: October 9, 1996

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-301

POINT BEACH NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 173
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 May 29, 1996, as supplemented August 20, 1996, 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. 173 , 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 45 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Allen G. Hansen, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of issuance: October 9, 1996

ATTACHMENT TO LICENSE AMENDMENT NOS. 169 AND 173

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

DOCKET NOS. 50-266 AND 50-301

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

REMOVE	INSERT
15.1-2	15.1-2
15.3.6-4	15.3.6-4
15.3.6-8	15.3.6-8
15.3.6-9	15.3.6-9
15.4.4-1	15.4.4-1
15.4.4-2	15.4.4-2
15.4.4-2a	15.4.4-3
15.4.4-2b	15.4.4-4
15.4.4-3	15.4.4-5
15.4.4-4	15.4.4-6
15.4.4-5	15.4.4-7
15.4.4-6	15.4.4-8
15.4.4-6a	
15.4.4-6b	
15.4.4-7	
15.4.4-8	
15.4.4-9	
15.4.4-9a	
15.4.4-9b	
15.4.4-10	
15.4.4-11	
15.4.4-12	
15.4.4-13	
15.4.4-14	
15.4.4-15	
15.4.4-16	
15.6.9-4	15.6.9-4
--------	15.6.12-1

D. Containment Integrity*

Containment integrity is defined to exist when:

1) Penetrations required to be isolated during accident conditions are either:

   a. Capable of being closed by an operable automatic containment isolation valve,

      OR

   b. Closed by an operable containment isolation valve,

      OR

   c. Closed in accordance with Specifications 15.3.6.A.1.b and 15.3.6.A.1.c.

2) The equipment hatch is properly closed.

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

4) The overall uncontrolled containment leakage is less than La.**

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.

**Prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test, the applicable leakage limits specified in TS 15.6.12.D.2 must be met.

c. <u>Containment Purge Supply and Exhaust Valves</u>

The containment purge supply and exhaust valves shall be locked closed and may not be opened unless the reactor is in the cold shutdown or refueling shutdown condition.

(1) One of the redundant valves in the purge supply and exhaust lines may be opened to perform the repairs required to conform with the Containment Leakage Rate Testing Program.

(2) If containment purge supply and exhaust penetration leakage results in exceeding the overall containment leakage rate acceptance criteria (L_a), enter 15.3.6.A.1.a.

E. CONTAINMENT STRUCTURAL INTEGRITY

The structural integrity of the reactor containment shall be maintained in accordance with the surveillance criteria specified in the Containment Leakage Rate Testing Program and 15.4.4.II.

1. If more than one tendon is observed with a prestressing force between the predicted lower limit (PLL) and 90% of the PLL or if one tendon is observed with prestressing force less than 90% of the PLL, the tendon(s) shall be restored to the required level of integrity within 15 days or the reactor shall be in hot standby within the next six hours and in cold shutdown within the following 30 hours. An engineering evaluation of the situation shall be conducted and a special report submitted in accordance with Specification 15.4.4.II.D within 30 days.
2. With an abnormal degradation of the containment structural integrity in excess of that specified in 15.3.6.E.1, and at a level below the acceptance criteria of Specification 15.4.4.II, restore the containment structural integrity to the required level within 72 hours or be in hot shutdown within the next six hours and in cold shutdown within the following 30 hours. Perform an engineering evaluation of the containment structural integrity and provide a special report in accordance with Specification 15.4.4.II.D within 30 days.

Basis

Specification 15.3.6.A.1

The Reactor Coolant System conditions of cold shutdown assure that no steam will be formed and hence there would be no pressure buildup in the containment if the Reactor Coolant System ruptures.

Specification 15.3.6.A.1.a.

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the design basis LOCA without exceeding the design leakage rate. The design allowable leakage rate (L_a) is 0.4% of containment air weight per day at 60 psig (P_a).[1]

Containment operability is maintained by limiting the overall containment leakage rate to within the design allowable leakage rate (L_a). Prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test, however, the applicable leakage limits specified in TS 15.6.12.D.2 must be met. Compliance with Specification 15.3.6.A.1.a. will ensure a containment configuration that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

If penetration or air lock leakage results in exceeding L_a, Specification 15.3.6.A.1.a. shall be entered simultaneously with the LCO applicable to the penetration or air lock with the excessive leakage. Once the overall containment leakage rate is restored to less than L_a, Specification 15.3.6.A.1.a. may be exited and operation continued in accordance with the applicable LCO.

Specification 15.3.6.A.1.a.(1)

In the event the containment is inoperable, containment must be restored to operable status within one hour. The one hour completion time provides a period of time to correct the problem commensurate with the importance of maintaining containment integrity during plant operation. This time period also ensures that the probability of an accident (requiring containment integrity) occurring during periods when containment is inoperable is minimal.

15.4.4 CONTAINMENT TESTS

Applicability

Applies to containment leakage and structural integrity.

Objective

To verify that potential leakage from the containment and the pre-stressing tendon loads are maintained within acceptable values.

Specification

I. Perform required visual examinations and leakage rate testing in accordance with the Containment Leakage Rate Testing Program.

II. TENDON SURVEILLANCE

A. Object

In order to insure containment structural integrity, selected tendons shall be periodically inspected for symptoms of material deterioration or lift-off force reduction. The tendons for inspection shall be randomly but representatively selected from each group for each inspection; however, to develop a history and to correlate the observed data, one tendon from each group shall be kept unchanged after initial selection. Tendons selected for inspection will consist of five hoop tendons, three vertical tendons located approximately 120° apart, and three dome tendons, one from each of the three dome tendon groups.

B. Frequency

Tendon surveillance shall be conducted at five-year intervals in accordance with the following schedule:*

Unit	Year	Surveillance Required
1	1984	Physical
2	1984	Visual
1	1989	Visual
2	1989	Physical

* Subsequent five-year interval inspections repeat this pattern.

C. <u>Inspections</u>

Tendon surveillance in accordance with 15.4.4.II.B shall consist of either a visual or physical inspection.

(1) Visual Inspection

a. Tendon anchorage assembly hardware of the randomly selected tendons shall be visually examined to the extent practicable without dismantling load bearing components of the anchorage. The immediate concrete area shall be checked visually for indications of abnormal material behavior.

(2) <u>Physical Inspection</u>

a. Tendons which are physically inspected shall first be visually inspected in accordance with C.(1).

b. All tendons which are physically inspected shall be subjected to a lift-off test to monitor their prestressing force.

(i) If the prestressing force of a selected tendon in a group lies above the predicted lower limit, the tendon is considered to be acceptable.

(ii) If the prestressing force of a selected tendon lies between the predicted lower limit and 90% of the predicted lower limit, two tendons, one on each side of the test tendon, shall be checked for their prestressing forces. If the prestressing forces for these tendons are above the predicted lower limit for the tendons, all three tendons shall be restored to the required level of integrity. A single deficiency shall be considered unique and acceptable. If the prestressing force of either of the adjacent tendons falls below the predicted lower limit of the tendon, additional lift-off testing should be done if necessary, so that the cause and extent of such occurrence can be determined and the condition shall be considered an abnormal degradation of the containment structure and the provisions of Specification 15.3.6.E are applicable.

(iii) If the prestressing force of the selected test tendon falls below 90% of the predicted lower limit, the tendon shall be completely detensioned and a determination shall be made as to the cause of the condition. Such a condition shall be considered an abnormal degradation of the containment structure and the provisions of Specification 15.3.6.E are applicable.

(iv) If the average of all measured tendon forces for each group (corrected for average condition) is found to be less than the minimum required prestress level at Anchorage location for that group, the condition should be considered as abnormal degradation of the containment structure and the provisions of 15.3.6.E are applicable. The average minimum design values adjusted for elastic losses are as follows:[6]

Hoop	134.5 ksi
Vertical	140.6 ksi
Dome	137.4 ksi

c. One randomly selected tendon from each group of tendons shall be subjected to complete detensioning in order to identify broken or damaged wires. During the retensioning of the detensioned tendon, simultaneous measurements of elongation and jacking force shall be made at a minimum of two levels of force between the required seating force and zero. During the detensioning and retensioning of the tendons tested, if the elongation corresponding to a specific load differs by more than 5% from that recorded during installation of the tendons, an investigation shall be made to ensure that such discrepancies are not related to wire failures or slippage of wires in anchorages.

d. A tendon wire shall be removed from the one tendon from each group which has been completely detensioned. The wire shall be inspected over its entire length to determine if evidence of corrosion or other deleterious effects are present. Tensile tests shall be made on three samples cut from each removed wire. The samples will be cut from the midsection and each end of the removed wire. Failure of the material to demonstrate the minimum required tensile strength of 240,000 psi shall be considered an abnormal condition of the containment structure and the engineering evaluation provisions of Specification 15.3.6.E.1 are applicable. If an acceptable justification for continued operation cannot be concluded from this evaluation, then the shutdown requirements of Specification 15.3.6.E.1 are applicable.

e. The sheathing filler grease will be sampled and inspected on each physically inspected tendon. The operability of the sheathing filler grease shall be verified by assuring:

1) There are no voids in the filler material in excess of 5% of net duct volume.

2) Complete grease coverage exists for the different parts of the Anchorage system, and

3) The chemical properties of the filler material are within the tolerance limits specified by the manufacturer.

D. <u>Reports</u>

A final report documenting the results of each tendon surveillance will be prepared and maintained as a permanent plant record.

Abnormal conditions observed during testing will be evaluated to determine the effect of such conditions on containment structural integrity. This evaluation should be completed within 30 days of the identification of the condition. Any condition which is determined in this evaluation to have a significant adverse effect on containment structural integrity will be considered an abnormal degradation of the containment structure.

Any abnormal degradation of the containment structure identified during the engineering evaluation of abnormal conditions shall be reported to the Nuclear Regulatory Commission pursuant to the requirements of 10 CFR 50.4 within thirty days of that determination. Other conditions that indicate possible effects on the integrity of two or more tendons shall be reportable in the same manner. Such reports shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedure and the corrective action taken.

III. <u>End Anchorage Concrete Surveillance</u>

A. Specific locations for surveillance will be determined by information obtained from design calculations, as-built end anchorage concrete and prestressing records, observations of the end anchorage concrete during and after prestressing, and results of deformation measurements made during prestressing and the initial structural test.

B. The inspection intervals will be approximately one-half year and one year after the initial structural test and shall be chosen such that the inspection occurs during the warmest and coldest part of the year following the initial structural test.

C. The inspections made shall include:

(1) Visual inspection of the end anchorage concrete exterior surfaces.

(2) A determination of the temperatures of the liner plate area or containment interior surface in locations near the end anchorage concrete under surveillance.

(3) Measurement of concrete temperatures at specific end anchorage concrete surfaces being inspected.

(4) The mapping of the predominant visible concrete crack patterns.

(5) The measurement of the crack widths, by use of optical comparators or wire feeler gauges.

(6) The measurement of movements, if any, by use of demountable mechanical extensometers.

D. The measurements and observations shall be compared with those to which prestressed structures have been subjected in normal and abnormal load conditions and with those of preceding measurements and observations at the same location on the reactor containment.

E. The acceptance criteria shall be as follows:

If the inspections determine that the conditions are favorable in comparison with experience and predictions, the close inspections will be terminated by the last of the inspections stated in the schedule and a report will be prepared which documents the findings and recommends the schedule for future inspections, if any. If the inspections detect symptoms of greater than normal cracking or movements, an immediate investigation will be made to determine the cause.

IV. <u>Liner Plate</u>

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 ± .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".

E. In addition to the preceding requirements, temperature readings will be obtained at the locations where inward deformations were measured. Temperature measurements will also be obtained on the outside of the containment building wall.

Basis

The containment is designed for an accident pressure of 60 psig.[1] While the reactor is operating, the internal environment of the containment will be air at approximately atmospheric pressure and a temperature of about 105°F. With these initial conditions, the temperature of the steam-air mixture at the peak accident pressure of 60 psig is 286°F.

Prior to initial operation, the containment was strength tested at 69 psig and then leak-tested. The design objective of this preoperational leakage rate test was established as 0.4% by weight per 24 hours at 60 psig. This leakage rate is consistent with the construction of the containment,[2] which is equipped with independent leak-testable penetrations and contains channels over all containment liner welds, which were independently leak-tested during construction.

Safety analyses have been performed on the basis of a leakage rate of 0.40% by weight per 24 hours at 60 psig. With this leakage rate and with minimum containment engineered safety systems for iodine removal in operation, i.e. one spray pump with sodium hydroxide addition, the public exposure would be well below 10 CFR 100 values in the event of the design basis accident.[3]

The safety analyses indicate that the containment leakage rates could be slightly in excess of 0.75% per day before a two-hour thyroid dose of 300R could be received at the site boundary.

The performance of periodic integrated leakage rate tests during plant life provide a current assessment of potential leakage from the containment in case of an accident that would pressurize the interior of the containment. These tests are performed in accordance with the Containment Leakage Rate Testing Program.

Periodic visual and physical inspection of the containment tendons is the method to be used to determine loss of load-carrying capability because of wire breakage or deterioration. The tendon surveillance program specified in 15.4.4.II is based on the recommendation of Regulatory Guide 1.35 Rev. 3. Containment tendon structural integrity was demonstrated for both units at the end of one, three and eight years following the initial containment structural integrity test.

The pre-stress lift-off test provides a direct measure of the load-carring capability of the tendon. A deterioration of the corrosion preventive properties of the sheathing filler will be indicated by a change in the physical appearance of the filler. If the surveillance program indicates, by extensive wire breakage, tendon stress-strain relations, or other abnormal conditions, that the pre-stressing tendons are not behaving as expected, the abnormal conditions will be subjected to an engineering analysis and evaluation in accordance with Specification 15.4.4.II.D to determine whether the condition could result in a significant adverse impact on the containment structural integrity. The specified acceptance criteria are such as to alert attention to the situation well before the tendon load-carrying capability would deteriorate to a point that failure during a design basis accident might be possible. Thus, the cause of the incipient deterioration could be evaluated and corrective action studied without need to shut down the reactor. If the engineering evaluation determines that the abnormal condition could result in a significant adverse impact on the containment structural integrity, an abnormal degradation situation will be declared and a report submitted to the NRC in accordance with the specifications.

The purpose of the leakage tests of the isolation valves in the containment purge supply and exhaust lines is to identify excessive degradation of the resilient seals for these valves.

References

(1) FSAR Section 5.1.2.3

(2) FSAR Section 5.1.2

(3) FSAR Section 14.3.5

(4) FSAR Section 14.3.4

(5) FSAR Section 6.2.3

(6) FSAR pages 5.1-86 and 5.1-87

15.6.9.2 <u>Unique Reporting Requirements</u>

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

A. Deleted

B. <u>Poison Assembly Removal From Spent Fuel Storage Racks</u>

Plans for removal of any poison assemblies from the spent fuel storage racks shall be reported and described at least 14 days prior to the planned activity. Such report shall describe neutron attenuation testing for any replacement poison assemblies, if applicable, to confirm the presence of boron material.

C. <u>Overpressure Mitigating System Operation</u>

In the event the overpressure mitigating system (power operated relief valves in the low temperature overpressure protection mode) or residual heat removal system relief valves are operated to relieve a pressure transient which, by licensee's evaluation, could have resulted in an overpressurization incident had the system not been operable, a special report shall be prepared and submitted to the Commission within 30 days. The report shall describe the circumstances initiating the transient, the effect of the system on the transient and any corrective action necessary to prevent recurrence.

15.6.12 CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the following exceptions:

   1. The interval between the 1992 Unit 2 Type A test and the next Unit 2 Type A test shall be 60 months.

B. The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 53 psig.

C. The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.4% of containment air weight per day.

D. Leakage rate acceptance criteria are:

   1. The containment leakage rate acceptance criterion is $\leq 1.0\ L_a$.

   2. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.6\ L_a$ for the combined Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests.

E. The provisions of Specification 15.4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

F. The provisions of Specification 15.4.0.3 are applicable to the Containment Leakage Rate Testing Program.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 169 AND 173 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

On September 12, 1995, the U.S. Nuclear Regulatory Commission (NRC) approved issuance of a revision to 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors" which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The NRC added Option B, "Performance-Based Requirements," to allow licensees to voluntarily replace the prescriptive testing requirements of 10 CFR Part 50, Appendix J, with testing requirements based on both overall leakage rate performance and the performance of individual components.

By application dated May 29, 1996, and supplemented by letter dated August 20, 1996, Wisconsin Electric Power Company (WEPCo, the licensee) requested changes to the Technical Specifications (TS) for the Point Beach Nuclear Plant (PBNP), Units 1 and 2. The supplemental information did not change the staff's initial no significant hazards consideration determination. The proposed changes would permit implementation of 10 CFR Part 50, Appendix J - Option B. The licensee has established a "Containment Leakage Rate Testing Program" and proposed adding this program to the TS. The program references Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak Test Program," dated September 1995, which specifies a method acceptable to the NRC for complying with Option B.

2.0 BACKGROUND

Compliance with 10 CFR Part 50, Appendix J, provides assurance that the primary containment, including those systems and components which penetrate the primary containment, do not exceed the allowable leakage rate specified in the TS and Bases. The allowable leakage rate is determined so that the leakage assumed in the safety analyses is not exceeded.

On February 4, 1992, the NRC published a notice in the Federal Register (57 FR 4166) discussing a planned initiative to begin eliminating requirements marginal to safety which impose a significant regulatory burden. Appendix J of 10 CFR Part 50 was considered for this initiative and the staff undertook a study of possible changes to this regulation. The study examined the previous performance history of domestic containments and examined the effect on risk of a revision to the requirements of Appendix J. The results of this study are reported in NUREG-1493, "Performance-Based Leak-Test Program."

Based on the results of this study, the staff developed a performance-based approach to containment leakage rate testing. On September 12, 1995, the NRC approved issuance of this revision to 10 CFR Part 50, Appendix J, which became effective on October 26, 1995. The revision added Option B, "Performance-Based Requirements," to Appendix J to allow licensees to voluntarily replace the prescriptive testing requirements of Appendix J with testing requirements based on both overall and individual component leakage rate performance.

Regulatory Guide 1.163, was developed as a method acceptable to the NRC staff for implementing Option B. This regulatory guide states that the Nuclear Energy Institute (NEI) guidance document NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" provides methods acceptable to the NRC staff for complying with Option B with four exceptions which are described therein.

Option B requires that the RG or other implementation document used by a licensee to develop a performance-based leakage rate testing program must be included, by general reference, in the plant TS. The licensee has referenced RG 1.163 in the Point Beach TS.

Regulatory Guide 1.163 specifies an extension in Type A test frequency to at least one test in 10 years based upon two consecutive successful tests. Type B tests may be extended up to a maximum interval of 10 years based upon completion of two consecutive successful tests and Type C tests may be extended up to 5 years based on two consecutive successful tests.

By letter dated October 20, 1995, NEI proposed TS to implement Option B. After some discussion, the staff and NEI agreed on final TS which were attached to a letter from C. Grimes (NRC) to D. Modeen (NEI) dated November 2, 1995. These TS are to serve as a model for licensees to develop plant specific TS in preparing amendment requests to implement Option B.

For a licensee to determine the performance of each component, factors that are indicative of or affect performance, such as an administrative leakage limit, must be established. The administrative limit is selected to be indicative of the potential onset of component degradation. Although these limits are subject to NRC inspection to assure that they are selected in a reasonable manner, they are not TS requirements. Failure to meet an administrative limit requires the licensee to return to the minimum value of the test interval.

Option B requires that the licensee maintain records to show that the criteria for Type A, B and C tests have been met. In addition, the licensee must maintain performance comparisons of the overall containment system and individual components to show that the test intervals are adequate. These records are subject to NRC inspection.

3.0 EVALUATION

In its May 29, 1996, letter, the licensee proposed establishing a "Containment Leakage Rate Testing Program" and proposed adding this program to the TS. The program references RG 1.163, which specifies a method acceptable to the NRC for complying with Option B. The proposal requires a change to existing TS Sections 15.4.4, "Containment Tests," 15.1, "Definitions," 15.3.6, "Containment System," and 15.6, "Administrative Controls."

Option B permits a licensee to choose Type A; or Type B and C; or Type A, B and C; testing to be done on a performance basis. The licensee has elected to perform Type A, B and C testing on a performance basis.

The TS changes proposed by the licensee are in compliance with the requirements of Option B and consistent with the guidance of RG 1.163, and the generic TS of the November 2, 1995, letter, with one exception. The licensee's proposed TS change includes a one-time exception to RG 1.163 in that the next Type A test for Unit 2 will be performed at an interval of 60, rather than 48, months since the last Type A test.

RG 1.163 endorses NEI 94-01 which states that periodic Type A tests shall be performed at intervals of 48 months until acceptable performance is established to extend the test intervals. Acceptable performance history is defined as completion of two consecutive periodic Type A tests where the calculated performance leakage rate was less than 1.0 L_a. At least one of these tests must be performed at peak accident pressure. Since the periodic Type A tests at PBNP are conducted at reduced pressure, the licensee must perform a full pressure test in order to adopt the extended Type A test interval provisions of Option B. Under the provision of Option A, the next Type A test for PBNP Unit 2 must be performed during the Fall 1996 outage. The licensee's proposal would delay the next Unit 2 Type A test until the Fall 1997 outage.

The licensee is planning to replace the Unit 2 steam generators during the Fall 1996 outage. The licensee feels it is more prudent to focus its resources on the safe replacement of the steam generators rather than on obtaining the equipment and changing the implementing procedures necessary to account for performing a full pressure Type A test. Deferral of the Type A test will reduce the Fall 1996 outage scope and duration and will allow time to adequately prepare for a full pressure test.

In order to justify their proposal, the licensee reviewed the PBNP Type A test performance history. The Unit 2 containment has never failed a Type A test. The five Type A tests conducted since plant start-up have all been less than 63% of the allowable test leakage rate at the 95% confidence level.

The licensee also reviewed its activities and concluded that there have not been any alterations or challenges to the Unit 2 containment since the last Type A test. There are also no major modifications to the containment structure itself planned for the Fall 1996 outage. Transportation of the existing and replacement steam generators out of and into containment will be done via the existing equipment hatch. No cutting of the containment structure or liner plate is required. Welding of the main steam and feedwater lines after installation of the replacement steam generators will be followed by appropriate inspections and testing in accordance with approved codes and standards to ensure the integrity of these containment penetrations is maintained. No other work that could affect the containment structure is scheduled for the Fall 1996 outage.

Based on Unit 2 Type A test performance history, as discussed above, the staff finds the licensee's proposal to delay the next Unit 2 Type A test until the Fall 1997 outage acceptable. Since the proposed TS changes are otherwise in compliance with the requirements of Option B and consistent with the guidance of RG 1.163, and the generic TS of the November 2, 1995, letter, the staff finds the proposed TS changes acceptable.

Option B states that specific existing exemptions to Option A are still applicable to Option B, if necessary, unless specifically revoked by the NRC. The current PBNP TS contain three exemptions to Appendix J, Option A. These exemptions are: (1) an exemption from Section III.A.1.(d) related to the service air supply line used in conjunction with the Type A test; (2) an exemption from Section III.A.1.(d) related to leakage testing of the residual heat removal system; and (3) and exemption from Section III.A.1.(a) related to the termination of a Type A test if excessive leakage paths are identified. The licensee evaluated these existing exemptions from Option A against the new requirements of Option B and determined that the exemptions are no longer applicable.

The present Point Beach TSs require that the containment purge supply and exhaust valves be tested every six months. The licensee's May 29, 1996, letter also requested a revision to the TSs which would delete this requirement. These valves would then be tested in accordance with Regulatory Guide 1.163 which specifies a test interval of 30 months. The requirement to leak test these valves at a frequency of every 6 months is not an Appendix J requirement. The current 6 month test interval is based on the findings of Generic Issue B-20, "Containment Leakage Due to Seal Degradation," that valves with resilient seals should be tested more frequently than required by Appendix J. The background for this conclusion is discussed in IE Circular 77-11, "Leakage of Containment Isolation Valves With Resilient Seats," issued on September 6, 1977. However, by letter dated August 20, 1996, the licensee reported the results of a review of purge supply and exhaust valve leakage test results and maintenance history from 1992 to the present. The licensee stated that 36 leakage tests were performed, nine per penetration, and "there have been no failures when compared to Technical Specifications and Appendix J limits." In addition, the licensee stated that no valve has exceeded the licensee's administrative limit of 2000 sccm (standard cubic centimeter per minute). Based on these results, the staff finds it acceptable to perform leakage rate tests on these valves at the 30 month interval specified in Regulatory Guide 1.163, rather than the previous 6 month interval.

The staff has reviewed the licensee's proposed disposition of its existing (Option A) Appendix J exemptions as they relate to the Option B requirements. Pursuant to the provisions of 10 CFR Part 50, Appendix J - Option B, paragraph III.V.B.1, the staff finds it acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Wisconsin state Official was notified of the proposed issuance of the amendment. The state official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. This also changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (61 FR 34901). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date: October 9, 1996