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

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-266

POINT BEACH NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 89
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 2, 1984 as modified September 5, 1984 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-24 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 89 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Charles M. Tramell
for James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 7, 1985

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

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-301

POINT BEACH NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 94
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 2, 1984 as modified September 5, 1984 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. <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 94 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Charles M. Tramell

for James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 7, 1985

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

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 89 TO FACILITY OPERATING LICENSE NO. DPR-24

AMENDMENT NO. 94 TO FACILITY OPERATING LICENSE NO. DPR-27

DOCKET NOS. 50-266 AND 50-301

Revise Appendix A as follows:

Remove Page	Insert Page
15.3.6-2	15.3.6-2
15.3-6-3*	15.3.6-3*
15.4.4-8	15.4.4-8
15.4.4-9	15.4.4-9
-	15.4.4-9a
-	15.4.4-9b
15.4.4-15	15.4.4-15
15.4.4-16	15.4.4.16

*There are two pages 15.3.6-3 included for Unit 1. Please insert both pages into the Unit 1 TS, making careful note of the effectiveness instructions at the bottom of each page.

There is one page 15.3.6-3 included for Unit 2.

C. Containment Purge Supply and Exhaust Valves

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.

D. Containment Structural Integrity

The structural integrity of the reactor containment shall be maintained in accordance with the surveillance criteria specified in 15.4.4.V and 15.4.4.VII.

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.VII.D within 30 days.

2. With an abnormal degradation of the containment structural integrity in excess of that specified in 15.3.6.D.1, and at a level below the acceptance criteria of specification 15.4.4.VII, 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.VII.D within 30 days.

Basis

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.

The shutdown conditions of the reactor are selected based on the type of activities that are being carried out. When the reactor head is not to be removed, the specified cold shutdown margin of 1% $\Delta k/k$ precludes criticality under any occurrence. During refueling the reactor is subcritical by 5% $\Delta k/k$. Positive reactivity changes for the purpose of rod assembly testing will not result in criticality because no control bank worth exceeds 3%. Positive reactivity changes by boron dilution may be required or small concentration fluctuations may occur during preparation for, recovery from, or during refueling but maintaining the boron concentration greater than 1800 ppm precludes criticality under these circumstances. 1800 ppm is a nominal value that ensures 5% shutdown for typical reload cores. Should continuous dilution occur, the time intervals for this incident are discussed in Section 14.1.5 of the FSAR.

Regarding internal pressure limitations, the containment design pressure of 60 psig would not be exceeded if the internal pressure before a major loss-of-coolant accident were as much as 6 psig.[1] The containment is designed to withstand an internal vacuum of 2.0 psig.[2]

The containment purge supply and exhaust valves are required to be locked closed during plant operations since these valves have not been demonstrated capable of closing from the full open position during a design basis loss-of-coolant accident. Maintaining these valves locked closed during plant operation ensures that excessive quantities of radioactive materials will not be released via the containment purge system in the event of a design basis loss-of-coolant accident. The containment purge supply and exhaust valves will be locked closed by providing locking devices on the control board operators for these valves.

<u>References</u>

(1) FSAR - Section 14.3.4

(2) FSAR - Section 5.5.2

removed, the specified cold shutdown margin of 1% $\Delta K/K$ precludes criticality under any occurrence. During refueling the reactor is subcritical by 10% $\Delta K/K$. This precludes criticality under any circumstances even though fuel is being moved or control rods withdrawn. Positive reactivity addition by rod motion from an initial 10% $\Delta K/K$ subcritical reactor condition precludes criticality because the reactor would be substantially subcritical even if all control rods were completely withdrawn. Positive reactivity changes by boron dilution may be required or small fluctuations may occur during preparation for, recovery from, or during refueling but maintaining the boron concentration greater than 1800 ppm precludes criticality under any circumstances. Should continuous dilution occur, the time intervals for this incident are discussed in Section 14.1.5 of the FFDSAR.

Regarding internal pressure limitations, the containment design pressure of 60 psig would not be exceeded if the internal pressure before a major loss-of-coolant accident were as much as 6 psig.[1] The containment is designed to withstand an internal vacuum of 2.0 psig.[2]

The containment purge supply and exhaust valves are required to be locked closed during plant operations since these valves have not been demonstrated capable of closing from the full open position during a design basis loss-of-coolant accident. Maintaining these valves locked closed during plant operation ensures that excessive quantities of radioactive materials will not be released via the containment purge system in the event of a design basis loss-of-coolant accident. The containment purge supply and exhaust valves will be locked closed by providing locking devices on the control board operators for these valves.

References

(1) FSAR - Section 14.3.4

(2) FSAR - Section 5.5.2

This page is to be cancelled upon completion of the refueling outage which ends approximately May 30, 1985. Amendment No. 86 is effective at that time.

removed, the specified cold shutdown margin of 1% $\Delta k/k$ precludes criticality under any occurrence. During refueling the reactor is subcritical by 5% $\Delta k/k$. Positive reactivity changes for the purpose of rod assembly testing will not result in criticality because no control bank worth exceeds 3%. Positive reactivity changes by boron dilution may be required or small concentration fluctuations may occur during preparation for, recovery from, or during refueling but maintaining the boron concentration greater than 1800 ppm precludes criticality under these circumstances. 1800 ppm is a nominal value that ensures 5% shutdown for typical reload cores. Should continuous dilution occur, the time intervals for this incident are discussed in Section 14.1.5 of the FSAR.

Regarding internal pressure limitations, the containment design pressure of 60 psig would not be exceeded if the internal pressure before a major loss-of-coolant accident were as much as 6 psig.[1] The containment is designed to withstand an internal vacuum of 2.0 psig.[2]

The containment purge supply and exhaust valves are required to be locked closed during plant operations since these valves have not been demonstrated capable of closing from the full open position during a design basis loss-of-coolant accident. Maintaining these valves locked closed during plant operation ensures that excessive quantities of radioactive materials will not be released via the containment purge system in the event of a design basis loss-of-coolant accident. The containment purge supply and exhaust valves will be locked closed by providing locking devices on the control board operators for these valves.

References

(1) FSAR - Section 14.3.4
(2) FSAR - Section 5.5.2

UNIT 1 ONLY
This page is effective upon completion of the refueling outage which ends approximately May 30, 1985. Amendment 86 is incorporated herein.

VI. CONTAINMENT MODIFICATIONS

Any major modification or replacement of components of the containment performed after the initial preoperational leakage rate test shall be followed by either an integrated leakage rate test or a local leak detection test and shall meet the acceptance criteria of I.B and II.B, respectively. Modifications or replacements performed directly prior to the conduct of an integrated leakage rate test shall not require a separate test.

VII. 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^{0} 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

C. Inspections

Tendon surveillance in accordance with 15.4.4.VII.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

*Subsequent five-year interval inspections repeat this pattern.

without dismantling load bearing components of the anchorage. The immediate concrete area shall be checked visually for indications of abnormal material behavior.

(2) Physical Inspection

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 liftoff 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 life-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.D 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.D 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.D 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.D.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.D.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. Reports

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 Regional Administrator, Region III, 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.

VIII. End Anchorage Concrete Surveillance

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.

containment. A Residual Heat Removal System leakage of 2 gal/hr will limit off-site exposures due to leakage to insignificant levels relative to those calculated for leakage directly from the containment in the Design Basis Accident. The dose calculated as a result of this leakage is 7.7 mr for a 2 hour exposure at the site boundary.[5]

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.VII 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-carrying 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.VII.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. With the exception of the test frequency and acceptance criteria, leakage tests of the containment purge supply and exhaust valves shall be conducted in accordance with 15.4.4.III.

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