August 27, 1986

Docket Nos. 50-266
and 50-301

Mr. C. W. Fay, Vice President
Nuclear Power Department
Wisconsin Electric Power Company
231 West Michigan Street, Room 308
Milwaukee, Wisconsin 53201

Dear Mr. Fay:

The Commission has issued the enclosed Amendment Nos. 104 and 107 to Facility Operating License Nos. DPR-24 and DPR-27 for the Point Beach Nuclear Plant, Unit Nos. 1 and 2. The amendments consist of changes to the Technical Specifications in response to your application dated October 25, 1983 as revised February 7 and April 18, 1984.

These amendments revise the requirements for conducting containment integrated leak rate testing (CILRT) to allow for reduced duration testing (less than 24 hours); to allow inclusion of purge supply and exhaust and exhaust valves to be open for repairs; and to make minor editorial changes.

Your proposed change relating to reduced duration Type "A" CILRT; specifically, proposed change 15.4.4.I.A.5a was found to be unacceptable by the staff. Therefore, the staff denies this proposed change to your Technical Specifications. The specific reasons for the staff's denial are contained in the enclosed Safety Evaluation. Proposed change 15.4.4.I.A.5b also related to reduced duration Type "A" CILRT was found to be acceptable. We have renumbered this proposed change to maintain consistency within the Technical Specification. We also have modified your proposed basis for this Technical Specification accordingly.

Also, as discussed with members of your staff during a conference call on April 25, 1986, a typographical error was discovered in your April 18, 1984 submittal. Specifically, proposed Technical Specification (TS) 15.3.6.c incorrectly cross referenced TS 15.4.4.II.B.2 as the requirement to complete repairs or conduct a reactor shutdown within 48 hours. The correct reference is 15.4.4.II.B.1.b. Your proposed TS page has been modified to correct this typrographical error.

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A copy of the Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice. A copy of the Notice of Denial is enclosed.

Sincerely,

/s/

Timothy G. Colburn, Project Manager
PWR Project Directorate #1
Division of PWR Licensing-A

Enclosures:
1. Amendment No. 104 to DPR-24
2. Amendment No. 107 to DPR-27
3. Safety Evaluation
4. Notice of Denial

cc w/enclosures:
See next page

Office:	LA/PAD#1	PM/PAD#1	OGC	PD/PAD#1
Surname:	PShuttleworth	TColburn/tg	R Bachmann	GLear
Date:	08/14/86	08/19/86	08/21/86	08/26/86

8/26/86

cc:
Mr. Bruce Churchill, Esq.
Shaw, Pittman, Potts and Trowbridge
1800 M Street, N.W.
Washington, DC 20036

Mr. James J. Zach, Manager
Point Beach Nuclear Plant
Wisconsin Electric Power Company
6610 Nuclear Road
Two Rivers, Wisconsin 54241

Mr. Gordon Blaha
Town Chairman
Town of Two Creeks
Route 3
Two Rivers, Wisconsin 54241

Chairman
Public Service Commission
of Wisconsin
Hills Farms State Office Building
Madison, Wisconsin 53702

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
Office of Executive Director
for Operations
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Resident Inspector's Office
U.S. Nuclear Regulatory Commission
6612 Nuclear Road
Two Rivers, Wisconsin 54241

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. 104
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 October 25, 1983 as revised February 7 and April 18, 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.

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

3. This license amendment is effective 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Timothy G. Colburn

Timothy G. Colburn, Project Manager
PWR Project Directorate #1
Division of PWR Licensing-A

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 27, 1986.

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. 107
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 October 25, 1983 as revised February 7 and April 18, 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. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 107 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Timothy G. Colburn, Project Manager
PWR Project Directorate #1
Division of PWR Licensing-A

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 27, 1986

ATTACHMENT TO LICENSE AMENDMENT NOS. 104 AND 107

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 the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
15.3.6-2	15.3.6-2
15.3.6-3	15.3.6-3*
15.4.4-1	15.4.4-1
15.4.4-2	15.4.4-2
---	15.4.4-2a
---	15.4.4-2b
15.4.4-5	15.4.4-5
15.4.4-6	15.4.4-6
15.4.4-6a	15.4.4-6a
15.4.4-12	15.4.4-12
15.4.4-13	15.4.4-13
15.4.4-16	15.4.4-16

*Separate pages for Unit 1 and 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.

a. One of the redundant valves in the purge supply and exhaust lines may be opened to perform the repairs required to conform with TS 15.4.4.II.B. The time duration and shutdown requirements of TS 15.4.4.II.B.1.b shall be applied.

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.

References

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

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References

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

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. Type A Periodic Integrated Leakage Rate Test

A. Test

1. The Type A periodic in-service integrated leakage rate test shall be performed at intervals specified in I-C below at an initial pressure P_t at or above 30 psig (50% of design pressure (P_a)).
2. Test accuracy shall be verified by supplementary means such as measuring the quantity of air required to return to the starting pressure (P_t) or by imposing a known leak rate to demonstrate the validity of measurements.
3. Closure of the containment isolation valves for the purpose of the test shall be accomplished by the means provided for normal operation of the valves without preliminary exercises or adjustment. Repairs of maloperating or leaking valves shall

be made as necessary. Description of valve closure malfunction or valve leakage that requires corrective action before the test shall be included in the Test Report.

4. Leak repairs, if required during the integrated leakage test, shall be preceded and followed by local leakage rate measurements. A description of the repairs and the leakage rates measured prior to and after the repairs shall be included in the Test Report.

5. The test duration shall not be less than 24 hours unless the criteria listed in "a" below are met.

   a. For the Absolute Method, Total Time technique, the test duration may be shortened to less than 24 hours provided the following Bechtel Corporation Topical Report (BN-TOP-1) acceptance criteria for short duration testing are met:

      (1) For the containment atmosphere stabilization: Once the containment is at test pressure the containment atmosphere shall be allowed to be stabilized for about four hours. The atmosphere is considered stabilized when:

         i. The rate of change of average temperature is less than 1.0°F/hour/hour averaged over the last two hours.

         or

         ii. The rate of change of temperature changes less than 0.5°F/hour/hour averaged over the last two hours.

      (2) For the data recording and analysis, using the absolute method, Total Time technique:

         i. The Trend Report based on Total Time calculations shall indicate that the magnitude of the calculated leak rate is trending to stabilize at a value less than the maximum allowable leak rate (L_a).

(Note: The magnitude of the calculated leak rate may be increasing slightly as it tends to stabilize. In this case, the average rate shall be determined from the accumulated data over the last five hours or last twenty data points, whichever provides the most points. Using this average rate, the calculated leak rate can then be linearly extrapolated to the 24th hour data point. If this extrapolated value of the calculated leak rate exceeds 75% of the maximum allowable leak rate (L_a) then the leak rate test is continued).

and

ii. The end of test upper 95% confidence limit for the calculated leak rate based on Total Time calculations shall be less than the maximum allowable leak rate.

and

iii. The mean of the measured leak rates based on Total Time calculations over the last five hours of test or last twenty data points, whichever provides the most data, shall be less than the maximum allowable leak rate.

and

iv. Data shall be recorded at approximately equal intervals and in no case at intervals greater than one hour.

and

v. At least twenty data points shall be provided for proper statistical analysis.

and

vi. In no case shall the minimum test duration be less than six hours.

B. Acceptance Criteria

1. The governing criteria for acceptance of peak pressure tests is that the maximum allowable leakage (L_a) shall not exceed 0.40 weight percent per day of containment atmosphere at 60 psig (P_a) which is the design pressure.

2. The allowable in-service leakage rate (L_t) at the reduced test pressure (P_t) shall not exceed L_a (L_{tm}/L_{am}), except if L_{tm}/L_{am} is greater than 0.7, L_t shall be equal to $L_a(P_t/P_a)1/2$. Where: L_a is the maximum allowable leakage rate at pressure Pa for the preoperational tests; the subscript "m" refers to values of the leakage measured during initial preoperational tests; and the subscripts "a" and "t" refer to tests at accident pressure and reduced test pressure, respectively.

3. The measured leakage rate (L_{tm}) for in-service tests shall not exceed 0.75 L_t, as determined under B-1 above.

Unit 1-Amendment No. 104
Unit 2-Amendment No. 107

b. Airlock and equipment door seals, including operating mechanism and penetrations with resilient seals which are part of the containment boundary in the airlock structure.

c. Fuel transfer tube flange seal.

d. The containment purge supply and exhaust valves.

e. Other containment components which require leak repair in order to meet the acceptance criterion for any integrated leakage rate test.

B. Acceptance Criterion

1. The total leakage from items II.A.5 and III.A.3 shall not exceed 0.6 L_a.

   a. If at any time it is determined that 0.6 L_a is exceeded, repairs shall be initiated immediately. After repair, a retest to confirm conformance to the acceptance criterion of II.B. is required.

   b. If repairs are not completed and conformance to the acceptance criterion of II.B. is not demonstrated within 48 hours, the reactor shall be taken to cold shutdown conditions until repairs are effected and the local leakage meets this acceptance criterion.

2. The leakage from the airlock doors seal test, resulting from the 3 day testing requirement in II.C.1.d, shall be considered acceptable if the leakage sum from the worst door in each airlock, extrapolated to P_a, and added to the total of items II.A.5 and III.A.3, is less than 0.6 L_a.

   a. If the total identified in II.B.2, above, exceeds 0.6 L_a, then the airlock containing the worst door shall be full pressure tested to determine the actual leakage performance.

3. The leakage rate for the containment purge supply and exhaust valves shall be compared to the previously measured leakage rate to detect excessive valve degradation.

C. Test Frequency

1. Individual penetrations shall be tested during each shutdown for major fuel reloading except as specified in a and b below. In no case shall the interval be greater than two years.

   a. The containment equipment hatch flange seals and the fuel transfer tube flange seals shall be tested at each shutdown for major fuel reloading or after each time used, if that be sooner.

Unit 1 - Amendment No. 61, 70, 104
Unit 2 - Amendment No. 66, 75, 107

b. The air locks shall be tested at 6-month intervals at test pressure not less than P_a.

c. Personnel airlocks shall be tested at a pressure of no less than P_a following periods when containment integrity is defeated through the use of the airlock.

d. Personnel airlocks opened during periods when containment integrity is established shall be tested within 3 days after being opened. Personnel airlocks opened more frequently than once every 3 days shall be tested at least once every 3 days during the period of frequent openings.

e. The containment purge supply and exhaust valves shall be tested at 6-month intervals.

III. Type "C" Tests

A Type "C" test measures the leakage across an individual valve or across a group of valves used to isolate an individual penetration through the primary reactor containment as defined in III.A.3.

A. Test

1. Type "C" tests shall be performed at intervals specified in III.D below and at a pressure of not less than P_a.

2. Acceptable methods of testing are by local pressurization and the methods described in II.A.4 above. The pressure shall be applied in the same direction as that when the valve would be required to perform its safety function, unless it can be determined that the results from the tests for a pressure applied in a different direction will provide equivalent or more conservative results. Each valve to be tested shall be closed by normal operation and without any preliminary exercising or adjustments.

3. Local leakage shall be measured for containment isolation valves that:

   a. Provide a direct connection between the inside and outside atmospheres of the primary reactor containment under normal operation.

   b. Are required to close automatically upon receipt of a containment isolation signal.

   c. Are required to operate intermittently under post-accident conditions.

B. Acceptance Criterion

The total leakage from items II.A.5 and III.A.3 shall not exceed 0.6 L_a.

C. Corrective Action

1. If at any time it is determined that 0.6 L_a is exceeded, repairs shall be initiated immediately. After repair, a retest to confirm conformance to the acceptance criterion of III.B is required.

2. If repairs are not completed and conformance to the acceptance criterion of III.B is not demonstrated within 48 hours, the reactor shall be taken to cold shutdown conditions until repairs are effected and the local leakage meets this acceptance criterion.

D. Test Frequency

1. The above tests of the isolation valves shall be conducted during each shutdown for major fuel reloading but in no case at intervals greater than two years.

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 pre-operational 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 side boundary.

The performance of a periodic integrated leakage rate test during plant life provides a current assessment of potential leakage from the containment in case of an accident that would pressurize the interior of the containment. In order to provide a realistic appraisal of the integrity of the containment under accident conditions, this periodic test is to be performed without preliminary leak detection surveys or leak repairs, and containment isolation valves are to be closed in the normal manner. The test pressure of 30 psig or greater for the periodic integrated leakage rate test is sufficiently high to provide an accurate measurement of the leakage rate and it duplicates the pre-operational leakage rate test at 30 psig. The specification provides relationships for relating in a conservative manner, the measured leakage of air at 30 psig or greater to the potential leakage of steam-air mixture at 60 psig and 286°F. The specification also allows for possible deterioration of the leakage rate between tests, by requiring the as measured leak rate to be less than 75% of the allowable leakage rate. The basis for these deterioration allowances are arbitrary judgments, which are believed to be conservative and which will be confirmed or denied by periodic testing. If indicated to be necessary, the deterioration allowances will be altered based on experience.

The duration of the integrated leak rate test will be 24 hours unless the reduced time duration acceptance criteria are met. In 1972, the AEC approved a Bechtel Corporation Topical Report, BN-TOP-1, entitled "Testing Criteria for Integrated Leakage Rate Testing of Primary Containment Structures for Nuclear Power." This report provides criteria for short duration testing for the Absolute Method using the Total Time technique. The Bechtel short duration testing criteria contains requirements for stabilization, leakage rate trending, confidence level, sufficient data for statistical convergence, and allowed leakage rate.

The frequency of the periodic integrated leakage rate test is keyed to the refueling schedule for the reactor and shutdown for inservice inspection because these tests can only be performed during refueling shutdowns. The initial core loading was designed for approximately 24 months of power operation, thus the first refueling occurred approximately 30 months after initial

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 104 AND 107 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

Introduction

10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," specifies requirements for preoperational and periodic verification by test of the leak tight integrity of the primary reactor containment, and systems and components which penetrate containment, and establishes the acceptance criteria for such tests. Section III A.3 of Appendix J specifies the methods to be used when performing an integrated (Type A) leak test of a containment. This section requires, in part, that all Type A tests shall be conducted in accordance with the provisions of American National Standard N45.4-1972 (ANSI N45.4-1972), "Leakage Rate Testing of Containment Structures for Nuclear Reactors," dated March 16, 1972. Section 7.6 of ANSI N45.4-1972 specifies that the leakage rate test period, for any method, shall extend to 24 hours of retained internal pressure unless it can be demonstrated to the satisfaction of those responsible for the acceptance of the containment structure that the leakage rate can be accurately determined during a shorter test period, in which case the agreed upon shorter period may be used.

Sections III B and III C of Appendix J specify the test methods and acceptance criteria to be employed in determining the leak tightness of certain pressure-containing or leakage-limiting components penetrating the primary containment boundary including penetrations whose design incorporates resilient seals, gaskets, or sealant compounds and containment isolation valves.

By letters dated October 25, 1983, February 7, and April 18, 1984, Wisconsin Electric Power Company (the licensee) submitted proposed changes to the Point Beach Nuclear Plant (PBNP) Units 1 and 2 Technical Specifications (TS) for containment leakage rate testing requirements. The proposed changes modify operability and leakage rate testing requirements for the containment purge supply and exhaust valves, and incorporates specific methodologies and test acceptance criteria to be employed when performing a containment integrated leak rate test (CILRT) of less than 24 hour duration, as currently authorized by TS 15.4.4.I.A.2., and correct some editorial errors in the existing TS.

This Safety Evaluation addresses the licensee's proposed changes and their impact on the operation and administration of activities at PBNP.

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Summary of Evaluation

The changes proposed by the licensee for operability and leak rate testing of containment purge supply and exhaust valves and the editorial changes discussed in detail below, are consistent with the requirements of 10 CFR Part 50, Appendix J or conditions of operability as described in the existing PBNP TS. The staff agrees with these proposed changes.

With respect to the first of the two methodologies proposed by the licensee for performing CILRT's of less than 24 hours duration, the staff finds that insufficient justification of the statistical validity of performing a short duration mass point absolute method test was provided. The staff notes that the second methodology, that contained in Bechtel Topical Report BN-TOP-1, has been previously approved generically. Thus, the licensee's proposed changes for performing reduced duration testing in accordance with BN-TOP-1 are approved and those proposed changes associated with reduced duration testing in accordance with the mass-point absolute methodology are denied.

Evaluation

Proposed Change to TS 15.3.6.C, "Containment Purge Supply and Exhaust Valves" and Associated Bases

Description of Change

An Item a is added to the subject specification which would allow one of the redundant valves in the purge supply and exhaust lines to be opened to perform repairs during other than cold or refueling shutdown conditions when required due to leakage in excess of that allowed pursuant to TS 15.4.4.II.B. This condition would be allowed to exist for up to 48 hours at which time the reactor would have to be taken to cold shutdown.

Evaluation

Existing PBNP TS 15.3.6.C requires that the containment purge supply and exhaust valves be locked closed during other than cold or refueling shutdown conditions with no allowance for opening one of the redundant valves to facilitate repair of excessive leakage identified during the semiannually required leakage testing. The valves are required to be locked closed because they have not been demonstrated capable of closing from the full open position during a design basis loss-of-coolant accident.

The existing TS create the potential for requiring a reactor shutdown if one of the purge supply or exhaust valves is identified as having excessive leakage during the semiannual leakage tests. The proposed TS would allow one of the redundant valves to be opened to facilitate repairs, while the remaining valve would remain locked closed. With this configuration, at least one of the two valves in series in the purge supply line and the purge exhaust line would remain closed. Consequently, the containment boundary would be maintained at all times, and if the opened valve needed to be closed, it could be as it would have no differential pressure acting against it.

Existing TS 15.4.4.II.B.1.b allows 48 hours to repair and retest containment penetrations employing gaskets or resilient seals that have been identified as leaking in excess of allowable limits. As the purge supply and exhaust valves have been rendered passive components of the containment boundary by the requirement to keep them locked closed during other than cold or refueling shutdown, and as they employ resilient seals, the 48 hour proposed requirement for repair of these valves is consistent with existing requirements.

Based on the fact that one purge supply or exhaust valve may be opened without breaching the containment boundary and the fact that the licensee's proposal regarding a 48-hour time limit to effect repairs for identified leakage is consistent with their existing TS on repair of penetration leakage, the staff finds the proposed change acceptable.

During the review of this proposed change, it was identified that the licensee had incorrectly cross referenced TS 15.4.4.II.B.2 as the requirement to complete repairs or conduct a reactor shutdown in 48 hours. The correct reference is TS 15.4.4.II.B.1.b. This was confirmed to be a typographical error during discussions with the licensee onsite on April 25, 1986. The licensee's proposed Technical Specification has been changed to correct this typographical error.

Proposed Change to TS 15.4.4.II, "Type B Tests"

Description of Changes

The licensee has proposed the following changes:

1. A requirement to perform Type "B" leakage testing as defined by 10 CFR Part 50, Appendix J is added to Section A.5 of the subject TS for the containment purge supply and exhaust valves.

2. A requirement to compare the measured leakage for the containment purge supply and exhaust valves to previously measured leakage for these valves is added to Section B of the subject TS.

3. A requirement to perform Type "B" leakage testing as defined by 10 CFR Part 50, Appendix J at six month intervals is added to Section C of the subject TS for the containment purge supply and exhaust valves.

4. Corresponding changes to the bases are made.

Evaluation

Existing PBNP TS 15.3.6.C requires that the containment purge supply and exhaust valves be locked closed during other than cold shutdown or refueling conditions, because the valves have not been demonstrated capable of closing from the full open position during a design basis loss-of-coolant accident. These valves

employ resilient seals. Because the valves are required to be locked closed, they have no active safety function and are passive, resilient sealed containment penetrations. As such, they are subject to the Type "B" leak testing requirements of 10 CFR Part 50, Appendix J.

Item 1 above is consistent with the requirements of 10 CFR Part 50, Appendix J and is acceptable to the staff.

Item 2 above is consistent with the requirements of Section IX of the ASME Boiler and Pressure Vessel Code for inservice testing of valves, is reflective of a need to monitor for resilient seal degradation, and represents a new requirement in the PBNP TS. As such, this change is acceptable to the staff.

Item 3 above is also consistent with 10 CFR Part 50, Appendix J requirements. It further requires more frequent testing than that required by Appendix J. A portion of the licensee's change fails to address the 10 CFR Part 50, Appendix J, Section III.D.2.(a) requirement that any penetration subject to Type "B" testing be Type "B" tested if opened prior to returning the reactor to an operating mode requiring containment integrity; however, the licensee has not requested relief from this requirement and no such relief is implied or granted. Consequently, if the containment purge supply and/or exhaust valves are opened when containment integrity is not required, they must be Type "B" tested prior to returning to a condition requiring containment integrity. This change is acceptable to the staff.

The changes proposed to the bases, as identified in Item 4 above, are consistent with the proposed changes and are acceptable to the staff.

Proposed Change to TS 15.4.4.III

Description of Change

A reference to TS 15.4.4.II.A.3 is changed to TS 15.4.4.II.A.5.

Evaluation

The change corrects a previous typographical error and is acceptable to the staff.

Proposed Changes to TS 15.4.4.I, "Type A Periodic Integrated Leakage Rate Test"

Description of Changes

The licensee proposes the following changes:

1. TS 15.4.4.I.A.2 is revised to delete the existing requirement that, "The test duration shall not be less than 24 hours unless test experiences of at least two prior tests provide evidence of the adequacy of shorter test duration." The remaining requirement concerning verification of test accuracy remains unchanged.

2. A new TS 15.4.4.I.A.5 is added specifying two sets of criteria, one of which must be satisfied for a CILRT of less than 24 hours duration. The first set of criteria duplicates the criteria contained in the December 1983 Electric Power Research Institute (EPRI) Report EPRI NP-3400, "Criteria for Determining the Duration of Integrated Leakage Rate Tests of Reactor Containments." The second set of criteria duplicates the criteria contained in Bechtel Topical Report BN-TOP-1, previously approved by the NRC.

Evaluation

The change proposed to TS 15.4.4.I.A.2 is primarily administrative. No NRC regulations or policies currently exist which would explicitly require a licensee to perform two tests demonstrating the acceptability of a CILRT methodology requiring a test duration of less than 24 hours. The NRC has approved the test methodology of Bechtel Topical Report BN-TOP-1 without prior plant-specific demonstration of applicability. As such, this change is acceptable to the staff.

The proposed new TS 15.4.4.I.A.5 incorporates two CILRT methodologies for tests of less than 24 hours duration. Proposed TS 15.4.4.I.A.5.a reiterates the methodology and acceptance criteria contained in the Electric Power Research Institute Report (EPRI) EPRI NP-3400 dated December 1983. This change is discussed below. Proposed TS 15.4.4.I.A.5.b reiterates the methodology and acceptance criteria contained in Bechtel Topical Report BN-TOP-1. Use of BN-TOP-1 has been previously approved on a generic basis by the NRC; thus, the staff finds that this portion of the proposed change is acceptable.

Proposed TS 15.4.4.I.A.5.a would allow use of a short duration CILRT methodology not previously approved by the NRC. The staff reviewed the methodology as contained in the proposed TS against EPRI Report NP-3400, dated December 1983, 10 CFR Part 50, Appendix J requirements, and a previous PBNP CILRT performed on Unit 1 in 1984 with the following results.

1. Proposed TS 15.4.4.I.A.5.a.6 establishes an acceptance criteria as follows: The calculated LSF (Least Squares Fit) leakage rate as a function of time shall have stabilized with a negligible positive or negative slope, as demonstrated by the following:

$$\frac{(L_n - L_{n-i}) \times 100}{(t_n - t_{n-i})(L_c - L_n)} \leq 10$$

Where L_n = Final Test Point LSF leakage rate.

L_{n-i} = Leakage rate for data points taken within the previous hour.

t_n = Time in hours of the last data point.

t_{n-i} = Time in hours for the data point used for L_{n-i}.

L_c = Test leakage criteria (75% of allowable leakage at test pressure).

When compared to the corresponding EPRI NP-3400 criteria, the following discrepancy was noted:

> EPRI NP-3400 proposes the above criteria as an adjunct to the criteria contained in ANSI/ANS 56.8-1981, "Containment System Leakage Testing Requirements." The licensee's criteria do not include those from ANSI/ANS 56.8-1981, most notably criteria on minimum test duration and number of data points. Further, no justification as to the validity of this criteria with regard to the statistical acceptability of test data is provided.

2. Neither the licensee's proposed methodology nor that contained in EPRI NP-3400 specify containment test condition stabilization prior to the start of the test as required by 10 CFR Part 50, Appendix J, Section III.A.1.(c). Failure to stabilize conditions prior to the start of the test could produce false results of a satisfactory test. It is worthy of note, that the licensee did include stabilization criteria in that portion of their proposed TS change incorporating the Bechtel Topical Report BN-TOP-1 methodology.

3. Neither the licensee's proposal nor the methodology contained in EPRI NP-3400 specify a minimum data acquisition frequency. ANSI N45.4-1972 requires data to be taken at least hourly; however, this would appear to be inadequate for the purposes of the licensee's methodology. Proposed TS 15.4.4.I.A.5.a.(7) establishes an acceptance criteria for a successful CILRT that the difference between the 95 percent confidence level LSF leakage and the calculated LSF leakage must be constant or decreasing for all points taken during the last test hour. If data is taken only at hourly intervals, this is a two point comparison only which may be insensitive to excessive data scatter. This is particularly true as the test duration increases and the sensitivity of both the measured and 95 percent confidence level LSF leakages to data scatter decreases.

4. As noted in Section 5 of EPRI NP-3400, more validation of the criteria contained therein is necessary as the original effort was based on only 53

tests. This was discussed with the licensee and a representative of the Quadex Corporation which prepared EPRI NP-3400. During this discussion, it was determined that significantly more validation had taken place using data from successful CILRT's; however, only limited work had been done using data from failed CILRT's and none of those were marginal failures where a more significant potential for falsely predicting a successful result exists. It would thus appear, that additional validation is warranted.

5. Between the period March 30 through April 1, 1984, the licensee conducted a Unit 1 CILRT. During this CILRT, witnessed by NRC Region III inspection personnel, the licensee attempted to validate their proposed criteria. Application of the criteria would have allowed test termination after 10 1/2 hours; however, subsequent data indicated that the TS allowable leakage was exceeded between the 14th through 28th hour of the test. While the test was ultimately successful after approximately 30 hours, the final leakage rate measured was greater than that predicted after 10 1/2 hours. Thus, the result obtained using the proposed criteria was potentially nonconservative.

Based on the discrepancies in and uncertainties associated with the licensee's proposed short duration CILRT methodology, their proposal for using the mass-point methodology is denied.

Environmental Consideration

These amendments involve a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment relating to operability and testing of the containment purge supply and exhaust valves 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 published a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). These amendments also involve changes in recordkeeping, reporting or administrative procedures or requirements. Accordingly, with respect to these items, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above relative to operability and testing requirements for the containment purge supply and exhaust valves, 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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors:
T. Colburn
W. Guldemond

Date: August 27, 1986

U. S. NUCLEAR REGULATORY COMMISSION

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NOS. 50-266 AND 50-301

NOTICE OF DENIAL OF REQUEST FOR AMENDMENT

TO FACILITY OPERATING LICENSES AND OPPORTUNITY FOR A HEARING

The U. S. Nuclear Regulatory Commission (the Commission) has denied in part a request by the licensee for amendments to Facility Operating License Nos. DPR-24 and DPR-27, issued to the Wisconsin Electric Power Company (the licensee), for operation of the Point Beach Nuclear Plant, Unit Nos. 1 and 2 (the facilities), located in the Town of Two Creeks, Manitowoc County, Wisconsin.

The amendments as proposed by the licensee would modify the Point Beach Technical Specifications (TS), to allow reduced duration testing during Type "A" containment integrated leak rate tests (CILRT) using either Bechtel BN-TOP-1 criteria (absolute method, total-time technique) or the EPRI NP-3400 criteria (absolute method, mass-point technique). The amendments would also allow containment purge supply and exhaust valves to be treated as Type "B" containment penetration for purposes of testing and repair. The amendments would also make some minor editorial changes. The licensee's application for the amendments was dated October 25, 1983 as revised February 7 and April 18, 1984. Notice of consideration of issuance of the amendments was published in the FEDERAL REGISTER on January 26, 1984 (49 FR 3344 at 3358) and on June 20, 1984 (49 FR 25350 at 25382).

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Notice of issuance of Amendment Nos. 104 and 107 will be published in the Commission's next regular biweekly FEDERAL REGISTER notice.

The portion of the application which proposed a change regarding reduced duration CILRT using the EPRI criteria was denied. The proposed TS for reduced duration testing (using the EPRI criteria) contain several discrepancies with the requirements of 10 CFR Part 50, Appendix J, with the EPRI NP-3400 report and with a previous CILRT performed on Point Beach Unit 1. These discrepancies include lack of criteria on minimum test duration, data acquisition frequency and number of data points, failure to require containment test condition stabilization prior to test start, the existence of potential nonconservatisms observed when using the proposed criteria during a CILRT on Unit 1 and lack of adequate validation for the proposed criteria. Based on the discrepancies in and uncertainties associated with the licensee's proposed short duration testing using the EPRI criteria, the staff denied the licensee's proposed Technical Specification.

The licensee was notified of the Commission's denial of this request by letter dated August 27, 1986.

By September 29th , the licensee may demand a hearing with respect to the denial described above and any person whose interest may be affected by this proceeding may file a written petition for leave to intervene.

A request for a hearing or petition for leave to intervene must be filed with the Secretary of the Commission, U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch, or may be

delivered to the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., by the above date.

A copy of the petition should also be sent to the Executive Legal Director, U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, and to Gerald Charnoff, Esq., Shaw, Pittman, Potts & Trowbridge, 1800 M Street, N. W., Washington, D. C. 20036, attorney for the licensee.

For further details with respect to this action, see (1) the application for amendment dated October 25, 1983 as revised February 7 and April 18, 1984 and (2) the Commission's letter to Wisconsin Electric Power Company dated August 27, 1986, which are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., and at the Joseph P. Mann Public Library, Two Rivers, Wisconsin. A copy of item (2) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Division of PWR Licensing-A.

Dated at Bethesda, Maryland this 27th day of August 1986.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard F. Dudley

Richard F. Dudley, Acting Director
PWR Project Directorate #1
Division of PWR Licensing-A

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