JUN 25 1982

DCS MS-016

Docket Nos. 50-266
and 50-301

Mr. C. W. Fay
Assistant Vice President
Wisconsin Electric Power Company
231 West Michigan Street
Milwaukee, Wisconsin 53201

Dear Mr. Fay:

The Commission has issued the enclosed Amendment No. 61 to Facility Operating License No. DPR-24 and Amendment No. 66 to Facility Operating License No. DPR-27 for the Point Beach Nuclear Plant, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated December 12, 1975 as modified by letters dated July 18, 1977, February 6, 1978 and February 25, 1981.

These amendments provide primary containment integrated leak rate test requirements and schedules consistent in part to the requirements of Appendix J to 10 CFR Part 50.

Your proposed Technical Specification for performance of Type A tests of less than 24 hours duration was not acceptable. Appendix J allows in part that Type A containment integrated leak rate tests of less than 24 hours duration are allowed if conducted in accordance with approved procedures. The staff has approved conduct of Type A tests of less than 24 hour duration only if done in accordance with the staff-approved Bechtel Topical Report BN-TOP1.

Your proposed Technical Specification 15.4.4.II.C.1 concerning Type B testing of the containment airlock has been reviewed and found to be unacceptable. Your present Technical Specifications should be modified to conform with the requirements of Section III.D.2 of Appendix J. Our basis for rejecting the above proposed Technical Specifications has been discussed with members of your staff by telephone. In that conversation we expressed our acceptance of the use of Appendix A of the Franklin Research Center Technical Evaluation Report (enclosed with the staff's Safety Evaluation) for use in extrapolating the results of reduced pressure tests to design test pressue.

Amendment

OFFICE							
SURNAME							
DATE							

You are requested to submit proposed Technical Specification changes, as necessary, to meet the requirements of Appendix J within 45 days of your receipt of this letter.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original signed by

Timothy G. Colburn, Project Manager
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 61 to DPR-24
2. Amendment No. 66 to DPR-27
3. Safety Evaluation
4. Notice of Issuance

cc: w/enclosures
See next page

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Amendment + Notice FR

OFFICE	ORB#3:DL	ORB#3:DL	ORB#3:DL	AD-OR:DL	OELD		
SURNAME	PMKreutzer	TColburn/pn	RAClark	TMNovak	R. Bachmann		
DATE	6/8/82	6/8/82	6/22/82	6/24/82	6/24/82		

NRC FORM 318 (10/80) NRCM 0240 **OFFICIAL RECORD COPY** ☆ USGPO: 1980—329-824

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

Docket No. 50-266/50-301

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: WISCONSIN ELECTRIC POWER COMPANY, Point Beach Nuclear Plant

Two signed originals of the Federal Register Notice identified below are enclosed for your transmittal to the Office of the Federal Register for publication. Additional conformed copies (12) of the Notice are enclosed for your use.

☐ Notice of Receipt of Application for Construction Permit(s) and Operating License(s).

☐ Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s): Time for Submission of Views on Antitrust Matters.

☐ Notice of Availability of Applicant's Environmental Report.

☐ Notice of Proposed Issuance of Amendment to Facility Operating License.

☐ Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.

☐ Notice of Availability of NRC Draft/Final Environmental Statement.

☐ Notice of Limited Work Authorization.

☐ Notice of Availability of Safety Evaluation Report.

☐ Notice of Issuance of Construction Permit(s).

☐ Notice of Issuance of Facility Operating License(s) or Amendment(s).

☒ Other: Amendment Nos. 61 and 66 and Exemption - Appendix J.

Reference documents have been provided.
Please publish these documents simultaneously.

Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

OFFICE→	ORB#3:DL					
SURNAME→	PMKreutzer/pr					
DATE→	6/28/82					

NRC FORM 102 7—79

Wisconsin Electric Power Company

cc:
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Shaw, Pittman, Potts and Trowbridge
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Mr. Gordon Blaha
Town Chairman
Town of Two Creeks
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Ms. Kathleen M. Falk
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U. S. Environmental Protection Agency
Federal Activities Branch
Region V Office
ATTN: Regional Radiation
Representative
230 S. Dearborn Street
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cc w/enclosure(s) and incoming
dtd: 12/12/75, 7/18/77, 2/6/78, 2/25/81

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

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

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. 61
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 December 12, 1975 as modified by letters dated July 18, 1977, February 6, 1978 and February 25, 1981, 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.

DESIGNATED ORIGINAL

Certified By Patricia J. Noonan

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

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 25, 1982

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. 66
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 December 12, 1975 as modified by letters dated July 18, 1977, February 6, 1978 and February 25, 1981, 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.

DESIGNATED ORIGINAL

Certified By Patricia J. Noonan

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

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: June 25, 1982

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 61 TO FACILITY OPERATING LICENSE NO. DPR-24

AMENDMENT NO. 66 TO FACILITY OPERATING LICENSE NO. DPR-27

DOCKET NOS. 50-266 AND 50-301

Revise Appendix A as follows:

Remove pages	Insert Pages
15.4.4-1	15.4.4-1
15.4.4-2	15.4.4-2
15.4.4-3	15.4.4-3
15.4.4-4	15.4.4-4
15.4.4-5	15.4.4-5
15.4.4-6	15.4.4-6
	15.4.4-6a
	15.4.4-6b

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. The test duration shall not be less than 24 hours unless test experiences of at least 2 prior tests provide evidence of the adequacy of shorter test duration. 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.

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 P_a 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.

4. The supplementary test described in I.A.2 is acceptable, provided the difference between the supplemental test data and the Type "A" test data is within 0.25 L_t. If results are not within 0.25 L_t, the reason shall be determined, corrective action taken, and a successful supplemental test performed.

5. If repairs are necessary to meet the acceptance criteria, the integrated leakage rate test need not be repeated provided local measured reductions in leakages achieved by repairs reduce the overall measured integrated leakage rate to a value not in excess of 0.75 L_t. Local leakage measurements taken to effect repairs to meet the acceptance criteria shall be taken at a test pressure P_a for full pressure tests and P_t for reduced pressure tests.

C. Frequency

1. Integrated leakage rate tests shall be performed as follows: After the initial preoperational leakage rate test, two integrated leakage rate tests shall be performed at approximately equal intervals between each major shutdown for in-service inspection to be performed at 10 year intervals. In addition, an integrated test shall be performed at each 10 year interval, coinciding with the in-service inspection shutdown. The test shall coincide with a shutdown for major fuel reloading.

2. If two consecutive Type "A" tests fail to meet the applicable

acceptance criteria in Section 1.B above, a Type "A" test shall be performed at each plant shutdown for refueling, or approximately every 18 months, whichever occurs first. The accelerated test schedule shall continue until two consecutive Type "A" tests pass, after which time the retest schedule in I.C.1 may be resumed.

D. Report of Test Results

1. Each Type "A" leakage rate test will be the subject of a summary technical report, which will include summaries of Type "B" and "C" tests (Items II and III below) that were performed since the last Type "A" test.

II. Type "B" Tests

A Type "B" test measures leakage across individual and/or portions of pressure containing or leakage-limiting boundaries of primary reactor containment penetrations as defined in II.A.3.

A. Test

1. Type "B" tests shall be performed at intervals specified in II.D below and at a pressure of not less than P_a.
2. Acceptable methods of testing are halogen leak detection, pressure decay and fluid flow using air or nitrogen. Another method may be used if it can be shown to have equivalent sensitivity.

3. The local leakage shall be measured for each of the following components:

   a. Containment penetrations that employ resilient seals, gaskets or sealant compounds, piping penetrations fitted with expansion bellows and electrical penetrations fitted with flexible metal seal assemblies.

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

B. Acceptance Criterion

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

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 II.B is required.

2. 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.

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.

   b. The personnel air lock door seals shall be tested at 6 month intervals, except when the air locks are not opened during the interval. In that case the test is to be performed after each opening, except that no test interval is to exceed 12 months.

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.2 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.3 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 effectedand 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.

This requires the addition of strongbacks on the inside of the inner personnel airlock door, frustrating the possibility of meeting the requirements of the "after each opening rule" stated in Section III.D.2, Appendix J, since the strongback installer must again return through the airlock after the test. For this reason, the Licensee requested an exemption from Section III.D.2, and prefers testing only once every 6 months.

In References 5 and 6, further amplification of the Licensee's position was provided, as follows:

> Reference 5. We disagree strongly with the Staff's alternative of pressurizing between the "O" rings within 72 hours of a containment entry, and request that the Staff reconsider its position. Point Beach personnel have tested personnel hatches since 1970 for two units, with two hatches per unit. During that time period there have been several leaks from the hatches. However these leaks were less than allowable rates and none required a unit shutdown. Of the leaks, none have been past the two "O" rings on either door as long as the strongbacks were made up properly on the inside door. The only leaks of any significance have been from such areas as electrical penetrations and leakage through the vent valves. Also, there has been no instance of a leak through any hatch component during a Type A integrated leak rate test. These have been performed a total of four times on the two units and will be repeated every 3-1/3 years through the life of each unit. The Type A test most nearly duplicates the pressure conditions seen during a LOCA. This is a significant test of the ultimate performance of the Point Beach hatches.
>
> We wish to emphasize that the Point Beach door design will not allow a test between the gaskets near design pressure. It is anticipated that no more than three to five psig could be put between the "O" rings and repeatability is unknown at this time.
>
> It appears that the NRC considers entry into the containment a nonfrequent event. This is not the case at Point Beach. In order to insure that the reactor coolant leakage is quickly identified and incontainment equipment is operating satisfactorily, containment inspections are made at least every two weeks. In addition, radiation control personnel make frequent surveys to keep a close watch on the radiation levels and gaseous activity levels. The imposition of this Staff position on Point Beach would be an unnecessary burden costing personnel time, equipment expenditure and unnecessary radiation exposure for the testing personnel.
>
> Reference 6. As we have previously stated, we strongly disagree with the Appendix J, Section III.D.2, requirement to test the operated containment airlock (whether it be a complete airlock test or a reduced pressure test

between the "O" ring door seals) within 72 hours of every containment entry.

Apparently, the reasoning behind this additional test requirement is to provide for a means of assuring that the door seals have not been damaged or seated improperly during airlock use (as conveyed to us in Attachment "A" of Reference 3).

We are again requesting that an exemption be granted from the 72-hour containment airlock testing requirement of Appendix J, Section III.D.2, to allow for our present Technical Specification testing requirements, plus an additional procedural change requiring a visual examination of the "O" ring seals following periodic containment inspection entries.

The following considerations justify our above request:

1. In the numerous Type "A" containment tests and Type "B" airlock tests, there has never been a test failure caused by leakage through the door "O" ring seals (see Reference 2). This large amount of inservice test data surely verifies the ability of the airlock door seals to seat properly.

2. A visual inspection of the "O" ring seals will be required following periodic containment inspections upon exit to assure that the "O" rings were not damaged due to or during containment airlock use.

3. The design of the airlocks is such that a full design pressure test between the "O" ring seals is not possible. In addition, there appears to be no viable method of extrapolating the results of a reduced pressure test between the "O" ring seals to an equivalent design pressure leak rate.

4. The implementation of the 72-hour airlock testing requirement of Appendix J, Section III.D.2, will have a potential negative impact on containment and reactor systems surveillance at the Point Beach Nuclear Plant. This negative impact relates to the likelihood that fewer containment entries will be made for purposes of surveillance testing, inspections, and radiation monitoring (see Reference 2 for containment entry practices).

5. Existing airlock testing procedures require a containment entry to install a strongback on the containment side of the inner door. This containment entry, in turn, would require an additional 72-hour airlock test. The requirements of Appendix J, Section III.D.2, would dictate that this "round robin" testing scenario be implemented on the present design of the Point Beach Nuclear Plant. It becomes obvious that this leads to the absurd result of a continuous airlock testing activity.

FRC EVALUATION:

Section III.D.2 to Appendix J was modified by the NRC, effective October 22, 1980. The revised section, as published in the Federal Register on September 22, 1980, is provided below:

III. Leakage Testing Requirements

D. Periodic retest schedule-

2. Type B Tests.

(a) Type B tests, except tests for airlocks, shall be performed during reactor shutdown for refueling, or other convenient intervals, but in no case at intervals greater than 2 years. If opened following a Type A or B test, containment penetrations subject to Type B testing shall be Type B tested prior to returning the reactor to an operating mode requiring containment integrity. For primary reactor containment penetrations employing a continuous leakage monitoring system, Type B tests, except for tests of airlocks, may, notwithstanding the test schedule specified under III.D.1., be performed every other reactor shutdown for refueling but in no case at intervals greater than 3 years.

(b)(i) Airlocks shall be tested prior to initial fuel loading and at 6-month intervals thereafter at an internal pressure not less than P_a.

(ii) Airlocks opened during periods when containment integrity is not required by the plant's Technical Specifications shall be tested at the end of such periods at not less than P_a.

(iii) Airlocks opened during periods when containment integrity is required by the plant's Technical Specifications shall be tested within 3 days after being opened. For airlock doors opened more frequently than once every 3 days, the airlock shall be tested at least once every 3 days during the period of frequent openings. For airlock doors having testable seals, testing the seals fulfills the 3-day test requirements. In the event that the testing for this 3-day interval cannot be at P_a, the test pressure shall be as stated in the Technical Specifications. Airlock door seal testing shall not be substituted for the 6-month test of the entire airlock at not less than P_a.

(iv) The acceptance criteria for airlock testing shall be stated in the Technical Specifications.

The NRC initiated this rule change on the basis of a compilation of data taken from Licensee Event Reports (LERs) submitted since 1969. These data showed that there were approximately 70 reported instances when airlock testing results have exceeded allowable leakage limits. Of these events, 25% were caused by leakage other than that from improper seating of airlock door seals. These failures were generally caused by leakage past door operating mechanism hand-wheel packing, door operating cylinder shaft seals, equalizer valves, or test lines. These penetrations are similar to other Type B or Type C containment penetrations except that they may be operated more frequently. Since airlocks are tested at a pressure of Pa every 6 months, these penetrations are tested, at a minimum, four times more frequently than typical Type B or C penetrations. The 6-month test was therefore considered to be both necessary and adequate in the prompt identification of this leakage.

Improper seating of the airlock door seals, however, was not only the most frequent cause of airlock failures (the remaining 75%), but also represented the largest potential leakage path. While testing at a pressure of Pa after each opening would identify this seal leakage, seal leakage could also be identified by alternative methods such as pressurizing between double-gasketed door seals (for airlocks designed with this type of seal) or pressurizing the airlock to pressures less than Pa. Furthermore, experience gained in testing airlocks since the issuance of Appendix J indicated that the use of one of these alternative methods might be preferable to the full pressure test of the entire airlock.

Reactor plants designed prior to the issuance of Appendix J often do not have the capability to test airlocks at Pa without the installation of holding devices (strongbacks) or mechanical adjustments of the operating mechanisms of the inner doors. This is because the inner doors are designed to seat with accident pressure (i.e., accident pressure on the containment side of the door) and therefore the operating mechanisms were not designed to withstand accident test pressure in the opposite direction. When the airlock is pressurized for a local airlock test (i.e., pressurized between the doors), pressure is exerted on the airlock side of the inner door, causing the door to

unseat and preventing the conduct of a meaningful test. The strongback or mechanical adjustments prevent the unseating of the inner door, allowing the test to proceed. The installation of strongbacks or performance of mechanical adjustments is time consuming (often taking several hours), may result in additional radiation exposure of operating personnel, and may also cause degradation to the operating mechanism of the inner door with consequent loss of reliability of the airlock. In addition, when conditions require frequent openings over a short period of time, testing at Pa after each opening becomes both impractical (tests often take from 8 hours to several days) and accelerates the rate of exposure of personnel and degradation of mechanical equipment.

Consequently, the NRC determined that testing of airlock door seals within 72 hours of opening (or every 72 hours during periods of frequent openings), rather than "after each opening," satisfied the intent of Appendix J. The NRC further determined that this testing could be performed by pressurizing the entire airlock to a pressure less than Pa or by pressurizing between double gasketed seals (for airlocks so equipped), rather than entire airlock testing at Pa.

Nevertheless, the Licensee continues to object to the airlock testing requirements, including those of the new rule. The Licensee states that application of these requirements leads to the "absurd result of a continuous airlock testing activity." FRC does not agree.

Following completion of the semiannual airlock test at Pa, a test is required within 72 hours of the airlock opening to remove the strongbacks from the containment side of the inner door. At this point, however, the Licensee may conduct the test at a pressure less than Pa (this does not require the strongback) or by pressurizing between the double-gasketed seal. Following completion of this test, no further testing is required for 6 months or until 72 hours after the next containment entry.

With regard to other aspects of the Licensee's basis for an exemption, FRC has the following observations:

- The revised rule is based upon in-service test data from all operating plants, not a small sampling of plants.
- A visual inspection of the "O" ring seals for damage is not the equivalent of a positive pressure test that the seal is both undamaged and seated properly.
- Recommendations for extrapolating reduced pressure leakage to equivalent full pressure leakage are provided in Appendix A to this report.

FRC CONCLUSION:

An exemption from the revised containment airlock testing requirements of Section III.D.2 is not justified. Airlock testing at Point Beach should be conducted in accordance with Appendix J.

3.1.2.2 Type B Testing Methods

In Reference 3, WEPCO requested authority to alter its Type B test method from that specified in Appendix J, Section III.B.1(b), to that specified in proposed ANSI Standard N271, Appendix E. WEPCO proposes to follow the procedure E-3.3, Method 3, entitled "Procedure for Fluid Flow (Infinite Holding Volume)" described in the proposed ANSI Standard N271.

In Reference 5, WEPCO provided the following additional information:

> As stated in our submittal of December 12, 1975, per Appendix J Section III.B.1.(b), an acceptable test method for local leakage rate tests is measurement of the rate of pressure loss. It is our experience that with containment hatches the effect of temperature is quite difficult to overcome because small changes in indicated temperature cause large changes in indicated leakage rate. Alternatively, we have had excellent results with pressurizing the hatch and measuring the gas flow required to maintain the pressure in the hatch. The specific procedures for Point Beach are contained in Attachment "A" to this letter. The basic document containing the procedural instructions is Technical Specification Test No. 13 (TS-13). This document references Operating Instruction No. 58 for the detailed instructions on operation of the testing equipment.

FRC EVALUATION:

Section III.B.1.(b) of Appendix J specifies Type B testing by "measurement of the rate of pressure loss of the test chamber of the containment

penetration ... ." The acceptance criterion for this test, however, is in terms of a "leakage rate," which is defined in Section II.B as a percentage (by weight) of the original content of containment air that escapes in a 24-hour period. Consequently, after the pressure loss has been measured in accordance with III.B.1.(b), this loss must be analytically converted to a mass flow rate and ultimately to a percentage (by weight) of containment air per unit time.

WEPCO proposes to measure the volumetric flow rate (cc/min) of input gas necessary to maintain a constant test pressure. This measurement directly reflects the volumetric leakage rate out of the test chamber at the constant pressure. If anything, analytical conversion of this measured rate to an Appendix J "leakage rate" is more straightforward than that required when measuring pressure loss under Section III.B.1.(b).

Section III.B.1 provides other alternative methods for performing Type B tests (halide leak-detection and permanently installed monitoring). There is no indication that Section III.B.1 is intended to limit testing methods to the three methods described.

FRC CONCLUSION:

The proposed method of measuring the gas flow required to maintain constant test chamber pressure (Operating Instruction No. 58) is equivalent to the pressure loss method of Section III.B.1.(b) of Appendix J. The WEPCO test procedure is acceptable for use in performing Type B tests. No exemption from Appendix J is necessary since acceptable test methods are not limited to those described in Appendix J.

3.1.3 Exemptions from Type C Testing Requirements

3.1.3.1 Reverse Direction Testing

In Reference 3, WEPCO identified four valves (two diaphragm valves and two butterfly valves) which are tested in the reverse direction, i.e., test pressure is applied in a direction opposite to that of the post-accident

condition. WEPCO requested that the testing of diaphragm valves and butterfly valves in the reverse direction be considered equivalent to testing in the normal direction of flow. As a basis for this request, WEPCO noted that the reverse direction testing of diaphragm and butterfly valves was considered acceptable by proposed ANSI Standard N-271, November 1973, Section E-1.6.

In Reference 5, the Licensee further indicated that the acceptability of reverse-direction testing of these valves is also established by Section XI of the ASME Code. The Licensee stated:

> The testability of globe and butterfly valves in either direction is established in ASME Code, Section XI, Article IWV-3420. Thus, testing of the four identified valves in the reverse direction is considered to be equivalent to testing in the normal direction.

FRC EVALUATION:

Section III.C.1 of Appendix J permits reverse-direction testing of containment isolation valves where it can be determined that test results will be equivalent to or more conservative than the results of testing in the direction of post-accident pressure. Appendix J does not require that exemptions be granted before reverse direction testing is authorized nor is any other form of NRC staff approval specified. Having determined that reverse-direction testing satisfies the criteria of Section III.C.1, however, the Licensee should be prepared to justify the determination upon request.

The Licensee has indicated that proposed ANSI Standard N-271, as well as the ASME Code, serves as a basis for accepting reverse-direction testing for these valves. This, along with the fact that diaphragm valves and particularly butterfly valves are generally amenable to reverse-direction testing, indicates that reverse direction testing is undoubtedly acceptable.

FRC CONCLUSION:

Reverse-direction testing of these valves is acceptable because the Licensee has determined that the criteria of Section III.C.1 have been met. No exemption or NRC approval is required.

3.1.3.2 Containment Spray System Isolation Valves

In Reference 3, WEPCO requested an exemption from the Type C testing requirements for isolation valves in the containment spray system (penetration Nos. 54 and 55). The exemption would permit testing of these valves with water as a medium in lieu of air or nitrogen because the valves are check valves located in a vertical piping run. WEPCO noted that this request met the proposed test medium criteria of Section 6.3.6 of proposed ANSI Standard N-274, September 1974.

In Reference 5, WEPCO stated:

> There is no effect of leakage back through one of these valves with respect to radiation exposure, since the fluid on the outlet side is the same fluid as in the inlet side of the valves. There is no requirement at present to perform a hydrostatic test on the system and a fluid inventory is not required.
>
> The exemption was requested in our earlier submittal because (1) the line cannot be drained at present, and (2) it is felt that due to the valves' locations outside the containment boundary such an exemption was appropriate. Since there is no adverse effect due to leakage back through these valves, it is requested that an exemption be granted to delete the requirement II.H.3 of Appendix J with respect to the containment spray system check valves.

In Reference 6, WEPCO provided the following statement in reply to an NRC question regarding the available fluid inventory within the system:

> We cannot guarantee that there will be sufficient fluid inventory within the piping system, downstream of the containment spray system's containment isolation check valves, to assure that the valves' seating surfaces will be water-sealed for 30 days following a design basis accident.
>
> We are not requesting a total exemption from having to Type "C" leak test these valves. However, we are requesting an exemption from the requirements of Appendix J, Section III.C.2.a, in the method of testing these valves.
>
> Appendix J, Section III.C.2.a, requires that these valves be pressurized with air or nitrogen. Since the valves are installed in a vertical piping run, we cannot drain that piping section sufficiently to expose the valves' seating surfaces to the test medium required. We are requesting that an exemption be granted to allow us to test these valves with the undrainable water in the system present.

Considering the following, we feel that the request is justified:

1. A total system leakage rate test is performed per the requirements of NUREG-0578, Item 2.1.6.a.

2. In the event of backleakage through these valves, there are numerous other valve boundaries that would also have to be breached to allow for a release to the environment.

3. If unacceptable backleakage was identified, the line could be manually isolated via the closure of the manual isolation valve located at the containment penetration.

FRC EVALUATION:

Appendix J requires that Type C testing be performed with air or nitrogen as a test medium. This is because Appendix J is concerned with the escape of containment air to the outside atmosphere. Although Appendix J never addresses isolation valve testing with water in lieu of air or nitrogen as a test medium, the acceptability of this approach may be inferred where:

1. Measured leakage rates are converted to equivalent air leakage rates so as to be included as part of the total air leakage allowed by Section III.C.3; or

2. The hydraulic test is used to demonstrate the existence of a water seal at the isolation valves (and therefore no air leakage) throughout the post-accident period, i.e., the equivalent of the seal-water system of Section III.C.3.

In general, acceptable water-to-air correlations have not been demonstrated by licensees because of the low flow rates and unpredictable leakage path characteristics of the various isolation valves. Extremely conservative correlations may be developed but generally tend to be impractical. Consequently, as a practical matter, substitution of hydraulic testing for required pneumatic testing has been limited to demonstration of a water seal at the isolation valves throughout the post-accident period. Accordingly, Section 8.4 of the latest proposed version of ANSI-N274 (Revision 3 to Draft 2, November 1978) states:

> Systems that are designed to contain water subsequent to a design basis loss of coolant accident (DBA), such that the containment isolation valves seating surface remains water covered (considering the water

volume and water leakage of the isolation valve) for at least thirty (30) days, may be tested with water.

In the case of Point Beach, the Licensee states that sufficient inventory within the piping system to provide a 30-day liquid seal of the valve seating surfaces cannot be guaranteed. Nevertheless, the Licensee states that there are other reasons why this exemption should be granted including compliance with the testing recommended in NUREG-0578 (Recommendation 2.1.b.1) and the availability of other boundaries and isolation valves. In reviewing these justifications, however, FRC does not find that the intent of Appendix J is achieved. Namely, there is no assurance that either all gaseous leakage from the containment is precluded or, if containment air can escape, that it is measured and accounted for to ensure that acceptable levels are not exceeded.

FRC CONCLUSION:

Substitution of a hydraulic test for the required pneumatic test of the containment spray isolation check valves is not adequate because there is no guarantee that the valve seating surfaces will remain water-covered throughout the post-accident period. An exemption from the requirements of Appendix J is not acceptable.

3.2 PROPOSED TECHNICAL SPECIFICATION CHANGES

In Reference 3, WEPCO submitted proposed changes of the Technical Specifications for Point Beach, Section 15.4.4, Containment Tests. This section of the Technical Specifications provides for the frequency, acceptance criteria, and general requirements of the Type A, B, and C containment leakage tests. Paragraph A.2 of Subsection I of the proposed changes was subsequently modified by WEPCO on February 6, 1978 [10].

FRC has reviewed the proposed changes of Section 15.4.4 of Reference 3 (with the modification of Reference 10) and finds that the changes are acceptable because they comply with the requirements of 10CFR50, Appendix J, or are more conservative than the requirements of Appendix J, with the following exceptions:

Proposed Specification 15.4.4.I.A.2

Paragraph A.2 of subsection I (Periodic Integrated Leakage Rate Test) reads:

> The test shall not be less than 24 hours unless test experiences of at least 2 prior tests provide evidence of the adequacy of shorter test duration. 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.

FRC EVALUATION:

As discussed in Section 3.1.1.3 of this report, the question of conducting Type A tests in less than 24 hours is being resolved by the NRC staff on a generic basis.

FRC CONCLUSION:

This issue is not evaluated since it will be resolved by the NRC staff on a generic basis.

Proposed Specification 15.4.4.I.B.5

Paragraph B.5 of Subsection I (Periodic Integrated Leakage Rate Test) reads:

> If repairs are necessary to meet the acceptance criteria, the integrated leakage rate test need not be repeated, provided local measured reductions in leakages achieved by repairs, reduce the overall measured integrated leakage rate to a value not in excess of 0.75 L_t. Local leakage measurements taken to effect repairs to meet the acceptance criteria shall be taken at a test pressure P_a for full pressure tests and P_t for reduced pressure tests.

FRC EVALUATION:

Section III.A.1.(a) of Appendix J requires termination of the Type A test when excessive leakage paths are identified, the performance of local leakage tests of these paths, and a Type A test after repairs are made. The change in leakage rate due to these repairs is to be reported to the NRC.

Experience in conducting Type A tests since initiation of these requirements has shown that the objective of this portion of the regulation can be achieved without necessarily requiring a second complete Type A test. In order to achieve the objective of the regulation, however, it is necessary to establish the satisfactory completion of the test (i.e., achieve a leakage rate which meets the acceptance criteria) and to determine the pre-repair or "as is" condition of the containment.

To determine the specific procedure the Licensee intends to use when it is discovered that repairs are necessary in order to meet the acceptance criteria of the Type A test, it is necessary to consider proposed specifications 15.4.4.I.A.3 and 15.4.4.I.A.4. These sections are quoted below:

> 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 exercise 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 that test shall be included in the Test Report.
>
> 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.

Reading all three of these sections together (15.4.4.I.A.3, 15.4.4.I.A.4, and 15.4.4.I.B.5), it appears that the Licensee's procedure satisfies the objective of Appendix J. The Licensee either identifies valve leakage prior to the test and corrects this leakage (15.4.4.I.A.3) or makes repairs during the Type A test (15.4.4.I.A.4). When repairs are made during the test, the leaking penetration is apparently isolated from the test pressure until the criteria for completion of the test have been established. Following release of the test pressure, repairs are made and the post- and pre-repair local leakage rates are added to the integrated leakage rate to verify that the 0.75 L_t criterion has been met and to establish the "as is" condition of the containment, respectively.

To use this procedure, it is necessary to make the conservative assumption that all the measured local leakage is in a direction out of the containment when (1) adding the post-repair leakage rate to the Type A results

for the purpose of establishing satisfactory test results, or (2) when adding the pre-repair leakage rate to the Type A results for the purpose of establishing the "as is" condition, unless it can be positively determined that certain leakage was not containment out-leakage.

It should be noted that it is not acceptable to terminate the Type A test without achieving a leakage rate which meets the acceptance criteria and then to subtract differential leakage from repaired valves in order to meet the acceptance criteria. This procedure is unacceptable because subtraction of certain internal containment leakage may erroneously reduce the apparent overall containment leakage rate. From a complete reading of Specification 15.4.4, however, FRC does not believe this to be the intent of the Licensee.

FRC CONCLUSION:

The proposed specification is acceptable as an exemption to the requirements of Section III.A.1.(a) because the objective of Appendix J is achieved. In a future revision of this specification, FRC believes that this section should be reworded to more clearly define the procedure being used and to specify the requirements for imposing the conservative assumption of containment out-leakage.

Proposed Specification 15.4.4.II.C.1

Paragraph C.1 (Test frequency) of Subsection II (Type "B" Tests) reads:

> 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.
>
> b. The personnel airlock door seals shall be tested at 6 month intervals.

FRC EVALUATION:

In Section 3.1.2.1 of this report, FRC provided a detailed evaluation of the Licensee's request for exemption from the airlock testing requirements of

Appendix J. This evaluation concluded that an exemption from the revised containment airlock testing requirements of Section III.D.2 of Appendix J is not justified and that airlock testing at Point Beach should be conducted in accordance with Appendix J.

FRC CONCLUSION:

This proposed specification should be revised to require airlock testing in accordance with Section III.D.2 of Appendix J.

4. CONCLUSIONS

Technical evaluations of outstanding requests for exemption and proposed technical specification changes relative to the full implementation of 10CFR50, Appendix J, at Point Beach Nuclear Plant, Units 1 and 2, have been conducted. The following is a summary of conclusions:

Exemptions from the Requirements of Appendix J

- Type C testing of the service air supply line with measured leakage added to the Type A results is an adequate substitution for the Type A testing requirements of Appendix J because this is the procedure set forth in Section III.A.1.(d) for other systems which remain operational during the test. An exemption from the requirements of Appendix J is acceptable.

- Periodic hydrostatic testing of the RHR system is an adequate substitute for the pneumatic (Type C) testing required by Appendix J because the hydrostatic testing is utilized to ensure that the isolation valves are not relied upon to prevent the post-accident escape of containment air. Appendix J does not require further testing of these valves. An exemption from the requirements of Appendix J is acceptable.

- An exemption to permit the Type A test to be terminated in less than 24 hours was not evaluated because this issue is being resolved by the NRC staff on a generic basis.

- An exemption from the revised containment airlock testing requirements of Section III.D.2 is not justified. Airlock testing at Point Beach should be conducted in accordance with Appendix J.

- The proposed method of measuring the gas flow required to maintain constant test chamber pressure (Operating Instruction No. 58) is equivalent to the pressure-loss method of Section III.B.1.(b) of Appendix J. The WEPCO test procedure is acceptable for use in performing Type B tests. No exemption from Appendix J is necessary since acceptable test methods are not limited to those described in Appendix J.

- Reverse-direction testing of certain containment isolation valves is authorized because the Licensee has determined that the criteria of Section III.C.1 have been met. No exemption is required.

- Substitution of a hydraulic test for the required pneumatic test of the containment spray isolation check valves is not acceptable

because there is no guarantee that the valve seating surfaces will remain water-covered throughout the post-accident period.

Proposed Technical Specification Changes

Proposed changes of Specification 15.4.4, submitted in Reference 3 and modified by WEPCO on February 6, 1978, were evaluated and found to be acceptable because they conformed to or were more conservative than Appendix J, subject to the following comments:

- Proposed specification 15.4.4.I.A.2 was not evaluated because the issue of performing a Type A test in less than a 24-hour period is being resolved by the NRC staff on a generic basis.
- Proposed Specifications 15.4.4.I.B.5 is acceptable as an exemption to the requirements of Section III.A.1.(a) of Appendix J. A future revision of this specification should reword this section to more specifically outline the procedure being used.
- Proposed specification 15.4.4.II.C.1 should be revised to require airlock testing in accordance with Section III.D.2 of Appendix J.

5. REFERENCES

1. NRC
   Generic letter regarding Containment Leakage Testing at Point Beach
   August 7, 1975

2. S. Burstein (WEPCO)
   Letter to K. Goller (NRC)
   September 5, 1975

3. S. Burstein (WEPCO)
   Letter to B. Rusche (NRC)
   December 12, 1975

4. G. Lear (NRC)
   Letter to S. Burstein (WEPCO)
   May 31, 1977

5. S. Burstein (WEPCO)
   Letter to G. Lear (NRC)
   July 18, 1977

6. C. W. Fay (WEPCO)
   Letter to R. A. Clark (NRC)
   February 25, 1981

7. R. A. Clark (NRC)
   Letter to S. Burstein (WEPCO)
   January 27, 1981

8. G. Lear (NRC)
   Letter to S. Burstein (WEPCO)
   October 4, 1977

9. S. Burstein (WEPCO)
   Letter to G. Lear (NRC)
   October 10, 1977

10. S. Burstein (WEPCO)
    Letter to E. Case (NRC)
    February 6, 1978

APPENDIX A

CONVERSION OF REDUCED PRESSURE AIR LEAKAGE MEASUREMENTS TO EQUIVALENT FULL PRESSURE AIR LEAKAGE

JULY 17, 1980

Dr. G. P. Wachtell

Franklin Research Center

APPENDIX A. AIR TO AIR LEAKAGE CONVERSION

In pneumatic leakage testing in which application of Pa psig is called for by Appendix J, it is sometimes necessary to request an exemption that permits pneumatic testing at a lower pressure, Pt psig. The leakage rate, Lt, measured under test conditions must then be converted mathematically to the leakage rate, La, that would occur if the pressure were equal to Pa. It is essential that the conversion be conservative. That is, the calculated value of La must not be lower than the actual leakage rate at Pa would be. On the other hand, the conversion should not be more conservative than necessary in the light of available data, because excessive conservatism could frequently result in the interpretation that a given leak exceeds its maximum allowable limit when in fact it would not exceed that limit if Pa were actually applied.

The meaning of the expression "if Pa were actually applied" should be carefully considered. The assumption is made that the geometry and dimensions of the leakage path would be the same with Pa applied as with Pt applied, or that any changes in geometry would not increase the leakage rate. In the case of airlock doors in which Pt is applied in the reverse direction, opposite to the direction in which Pa would be applied under function conditions, the use of the reverse direction of application of pressure is expected to tend to open the seal and increase the leakage rate. Under function conditions, in which pressure is applied in the forward direction, the seal should be improved if it changes at all. The expression "if Pa were actually applied" in this case means "if Pa were actually applied in the forward (normal for function) direction." In the case of valves and other penetrations, it is essential that increasing the applied pressure from Pt to Pa not change the geometry so as to increase the leakage rate. For example, increasing the pressure on a closed valve should tend to improve its sealing at the surfaces that provide the seal, and also in any other

potential leakage paths such as valve stem or packing that may have a connection to the applied pressure. Such other potential leakage paths are of course absent in valve designs in which the stem and packing have a connection only to the downstream side of the valve.

Reference 1, which is ASME Code, Section XI, paragraph IWV-3423 (e), states the following rule for tests at less than function differential pressure:

> "Leakage tests involving pressure differentials lower than function pressure differentials are permitted in those types of valves in which service pressure will tend to diminish the overall leakage channel opening, as by pressing the disk into or onto the seat with greater force. Gate valves, check valves, and globe-type valves having function pressure differential applied over the seat, are examples of valve applications satisfying this requirement. When leakage tests are made in such cases using pressures lower than function maximum pressure differential, the observed leakage shall be adjusted to function maximum pressure differential value. This adjustment shall be made by calculation appropriate to the test media and the ratio between test and function pressure differential, assuming leakage to be directly proportional to the pressure differential to the one-half power."

In the discussion below, it is shown that if (a) the test medium is air, (b) Pa is appreciable compared to one atmosphere, and (c) the leakage path is such as to produce laminar viscous flow (i.e., capillary-like rather than orifice-like), the calculation appropriate to this test medium yields a substantially higher calculated value of Pa than would be obtained by assuming leakage to be directly proportional to the pressure differential to the one-half power.

For air flow through an orifice, assuming uniform flow velocity over the orifice area, the mass flow rate per unit orifice area is ρv, where ρ is the density of air in the orifice and v is velocity in the orifice. Assuming that the discharge pressure is Pat = 1 atmosphere and the source pressure is Po, where Po and Pat are both absolute pressures, ρv is given by

$$(\rho v)^2 = \frac{2\gamma g}{\gamma - 1} \frac{Pat^2}{R_o T} \left[\frac{Po}{Pat} - 1 \right] G \qquad \text{(A-1)}$$

where $\gamma = 1.4$ is the specific heat ratio for air, $g = 32.2$ ft/sec^2 is the acceleration of gravity, T is source (upstream, at Po) temperature (°R), P is absolute pressure (psf), $R_o = 53.26$ ft-lb/lb°F is the gas constant for air and G is given by

$$G = \left(\frac{Pe}{Pat}\right)^2 \quad \frac{x^{\frac{\gamma-1}{\gamma}}\left(x^{\frac{\gamma-1}{\gamma}} - 1\right)}{\left(\frac{Po}{Pat} - 1\right)} \qquad \text{(A-2)}$$

$$x = \frac{Po}{Pe}$$

$Pe = Pat$ for subsonic flow

$Pe = 0.5283\ Po$ for choked flow

Choked flow occurs when

$$\frac{Pat}{Po} \leq \left(\frac{\gamma+1}{2}\right)^{-\frac{\gamma}{\gamma-1}} = 0.5283$$

$\sqrt{G}$ is proportional to $\rho v/\sqrt{Po-Pat}$. Values of $\sqrt{G}$ are listed in Table A-1. $\sqrt{G_o}$, the limiting value of $\sqrt{G}$ for small (Po-Pat), is $\sqrt{(\gamma-1)/\gamma} = 0.5345$.

In Table A-1, inspection of $\sqrt{G}/\sqrt{G_o}$ shows the accuracy of the assumption that for an orifice-like leakage flow resistance, leakage mass flow rate is proportional to pressure difference to the one-half power. For example, if Po = 60 psig (Po-Pat = 60 in Table A-1), $\sqrt{G}/\sqrt{G_o} = 1.210$. Extrapolation of mass flow rate measured with Pt = 15 psig to mass flow rate predicted for Pa = 60 psig will underestimate the mass flow rate by the factor 0.968/1.210 = 0.80, or 20%.

The foregoing argument tacitly assumes that the orifice coefficient is = 1.0. However, the same conclusion concerning extrapolation from low values of Pt to high values of Po can be drawn if the orifice coefficient is assumed to be constant, i.e., independent of Po. Consequently,

Table A-1. $\sqrt{G}$ for Various Values of Po - Pat for Orifice. (Pat taken = 15 psia.)

Po - Pat (psi)	$\sqrt{G}$	$\sqrt{G}/\sqrt{G_o}$
0.01	0.5345	1.000
1	0.5332	0.998
5	0.5282	0.988
13.3	0.5185	0.970
13.4*	0.5184	0.970
15 *	0.5176	0.968
20 *	0.5230	0.978
25 *	0.5346	1.000
30 *	0.5490	1.027
35 *	0.5648	1.057
40 *	0.5811	1.087
45 *	0.5977	1.118
50 *	0.6143	1.149
55 *	0.6307	1.180
60 *	0.6470	1.210

*Choked flow

for leakage paths that are known to be entirely orifice-like, the assumption that leakage mass flow rate is proportional to pressure difference to the one-half power gives a reasonably accurate correlation, underestimating the leakage mass flow rate by at most 20% for Pa $\leq$ 60 psig. To correct the underestimate, the factor $(\sqrt{G}/\sqrt{G_o})_a / (\sqrt{G}/\sqrt{G_o})_t$ has to be applied, where a and t mean Po = Pa and P_t respectively. References 2, 3, and 4 discuss the conversion formulas to be applied for various fluids (e.g., air and water) for various types of leakage path. For viscous flow of a gas, the mass flow rate from a source at absolute inlet pressure P_1 to absolute outlet pressure P_2 is proportional to $(P_1^2 - P_2^2)$. The proportionality factor is $C/\mu T$, where C is a function of geometry, T is absolute temperature, and μ is viscosity (which is a function only of temperature).

Assuming that test pressure Pt psig is applied at the same temperature as that at which function pressure Pa psig is applied, and assuming

further that the downstream pressure is one atmosphere, Pat psia, then the ratio of the mass flow rates is

$$\frac{\dot{m}a}{\dot{m}t} = \frac{(Pa + Pat)^2 - (Pat)^2}{(Pt + Pat)^2 - (Pat)^2} \qquad (A-3)$$

If the temperatures are not the same, the right side of Equation (A-3) has to be multiplied by

$$\frac{\mu(Tt) \cdot Tt}{\mu(Ta) \cdot Ta} \qquad (A-4)$$

Assuming that Tt = Ta, Table A-2 shows the ratio $\dot{m}a/\dot{m}t$ for various values of Pa and Pt, along with values of $(Pa\ psig/Pt\ psig)^{1/2}$. Pat is taken to be 15 psia in calculating $\dot{m}a/\dot{m}t$.

Table A-2. $\dot{m}a/\dot{m}t$ for Various Values of Pa and Pt.

Pt (psig)	$\dot{m}a/\dot{m}t$ Pa=50	55 (psig)	60	$(Pa/Pt)^{1/2}$ 50	55	60	$\frac{(\dot{m}a/\dot{m}t)}{(Pa/Pt)^{1/2}}$ 50	55	60
5	22.86	26.71	30.86	3.16	3.32	3.46	7.2	8.1	8.9
15	5.93	6.93	8.00	1.83	1.91	2.00	3.2	3.6	4.0
25	2.91	3.40	3.93	1.41	1.48	1.55	2.1	2.3	2.5
35	1.76	2.05	2.37	1.20	1.25	1.31	1.5	1.6	1.8
45	1.19	1.39	1.60	1.05	1.11	1.15	1.1	1.3	1.4

In all cases, the assumption that mass flow rate is proportional to pressure differential to the one-half power is unconservative for purely viscous flow. For Pa = 60 psig and Pt = 5 psig, it is unconservative by a factor of 8.9.

RECOMMENDED PROCEDURE

Any one of the following procedures, A, B, or C should be adopted.

A. Test Program

An extensive test program, covering several components of each type for which a correlation from Pt to Pa is sought, should be performed, in which sufficient experimental data showing the relation between Pt and leakage mass flow rate are obtained to permit a conservative empirical correlation to be established. Care must be taken to ensure that experimental orifice-like leaks are not used to represent actual, potentially capillary-like or viscous leaks.

B. Conservative Theoretical Correlation

Use Equation (A-3) as the correlation formula, including the factor (A-4) if necessary.

C. Measure Leakage Characteristic

For a given penetration, several values of Pt may be applied, so that an empirical correlation can be established. A statistical analysis of the data would be required to ensure at a 95% confidence level, that the predicted value of $\dot{m}a$ is not exceeded by the actual value of $\dot{m}a$.

REFERENCES

1. ASME Code, Section XI, paragraph IWV-3423(e).

2. Amesz, J., "Conversion of Leak Flow-Rates for Various Fluids and Different Pressure Conditions," 1966, EUR 2982.e, ORGEL Program, Ispra Establishment, Italy.

3. Maccary, R.R., DiNunno, J.J., Holt, A.E., and Arlotto, G.A., "Leakage Characteristics of Steel Containment Vessels and the Analysis of Leakage Rate Determinations," May, 1964, Division of Safety Standards, AEC, TID-20583.

4. Cottrell, Wm. B., and Savolainen, A.W., editors, "U.S. Reactor Containment Technology," ORNL-NSIC-5, Aug. 1965. Chapter 10, "Performance Tests," R.F. Griffin and G.H. Dyer. Sections 10.4.5 and 10.4.6 adapted from Reference 3.

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-266 AND 50-301

WISCONSIN ELECTRIC POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 61 to Facility Operating License No. DPR-24, and Amendment No. 66 to Facility Operating License No. DPR-27 issued to Wisconsin Electric Power Company (the licensee), which revised Technical Specifications for operation of Point Beach Nuclear Plant, Unit Nos. 1 and 2 (the facilities) located in the Town of Two Creeks, Manitowoc County, Wisconsin. The amendments are effective as of the date of issuance.

The amendments provide primary containment integrated leak rate test requirements and schedules consistent in part to the requirements of Appendix J to 10 CFR Part 50.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

DESIGNATED ORIGINAL

Certified By Patricia J. Moore

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

For further details with respect to this action, see (1) the application for amendments dated December 12, 1975 as modified by letters dated July 18, 1977, February 6, 1978 and February 25, 1981, (2) Amendment Nos. 61 and 66 to License Nos. DPR-24 and DPR-27, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. 20555, and at the Joseph Mann Library, 1516 16th Street, Two Rivers, Wisconsin 54241. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 25th day of June, 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

1. BACKGROUND

On August 5, 1975 [1],* the NRC requested the Wisconsin Electric Power Company (WEPCO) to review the containment leakage testing program for Point Beach Nuclear Plant, Units 1 and 2 (Point Beach 1 and 2) and to provide a plan for achieving full compliance with 10CFR50, Appendix J, including appropriate design modifications, changes in technical specifications, or requests for exemption from the requirements pursuant to 10CFR50.12, where necessary.

WEPCO responded to the NRC's request in a letter dated September 5, 1975 [2]. WEPCO followed with other correspondence, dated December 12, 1975 [3], which identified areas of compliance or non-compliance with Appendix J, requested exemption from various requirements, and proposed changes to the technical specifications for Point Beach. On May 31, 1977 [4], the NRC requested additional information to substantiate exemptions requested in Reference 3. This information was supplied to the NRC by WEPCO in correspondence dated July 18, 1977 [5]. On February 25, 1981 [6], WEPCO submitted additional information relative to these exemption requests in response to NRC questions provided on January 27, 1981 [7].

In addition, on October 4, 1977 [8], the NRC addressed certain concerns regarding WEPCO's proposed procedures for conducting the October 1977 Containment Integrated Leakage Rate Test (CILRT) of Point Beach Unit 1. WEPCO responded in a letter dated October 10, 1977 [9] and also provided additional information relative to this matter in Reference 6.

The purpose of this report is to provide technical evaluations of the outstanding submittals regarding the implementation of the requirements of 10CFR50, Appendix J, at Point Beach 1 and 2. Consequently, technical evaluations of the exemption requests discussed in References 3, 5, 6, and 9 are provided. Technical evaluations of proposed revisions of the technical specifications for Point Beach 1 and 2, initially submitted by WEPCO in Reference 3, are also included.

*Numbers in brackets refer to citations in the list of references, Section 5.

2. EVALUATION CRITERIA

Code of Federal Regulations, Title 10, Part 50 (10CFR50), Appendix J, Containment Leakage Testing, was specified by the NRC as containing the criteria for the technical evaluations. Where applied to the following evaluations, the criteria are either referenced or briefly stated, where necessary, in support of the conclusions. Furthermore, in recognition of plant-specific conditions not explicitly covered by the regulation, the NRC directed that the technical review constantly emphasize the intent of Appendix J: that potential containment atmospheric leakage paths be identified, monitored, and maintained below established limits.

3. TECHNICAL EVALUATION

3.1 EXEMPTIONS FROM THE REQUIREMENTS OF APPENDIX J

3.1.1 Exemptions from Type A Testing Requirements

3.1.1.1 Containment Service Air Supply Line

In Attachment 1, Section I.B of Reference 3, WEPCO requested an exemption from the Type A testing requirements for the air supply line used in performing the Type A containment leakage test. The Licensee stated that the "isolation requirements for the test and the temporary piping installed for the test prevent its being tested in accordance with Appendix J." The Licensee further stated that Type C testing would be performed with consequent leakage being added to the Type A test results.

In Reference 6, the Licensee provided the following additional information:

> The containment service air supply line, penetration 33C, is used during the Type "A" test to pressurize and depressurize the containment. The isolation requirements for the test and the temporary piping installed for the test prevent its being tested in accordance with Appendix J. We have previously requested an exemption from the requirements to vent this line to containment atmosphere during the Type "A" test. Instead, we perform a Type "C" test on the isolation valves in this line according to Appendix J, Section III.C, and add the leakage measured in this test to the overall leakage measured in the Type "A" test.

FRC EVALUATION:

Since this line is used during the Type A test, its testing requirements are considered comparable to those of other systems which penetrate the containment boundary and remain operational during the Type A test. Testing requirements for these systems are specified in Section III.A.1.(d) of Appendix J which states, in part:

> Systems that are required to maintain the plant in a safe condition during the test shall be operable in their normal mode, and need not be vented. . . . However, the containment isolation valves in the systems defined in III.A.1.(d) shall be tested in accordance with III.C. [Type C testing.]

Consequently, the Licensee has proposed to test the air supply line in the same manner used for other containment penetrations which are not Type A tested because they must remain operational during the Type A test.

FRC CONCLUSION:

Type C testing of the service air supply line, with measured leakage added to the Type A results, is an acceptable substitute for the Type A testing requirements of Appendix J because this is the procedure set forth in Section III.A.1.(d) for other systems which remain operational during the test. An exemption from the requirements of Appendix J is acceptable.

3.1.1.2 Residual Heat Removal System Isolation Valves

In Attachment 1, Section I.C. of Reference 3, WEPCO requested permission to substitute a hydrostatic test as described in their Technical Specifications, pages 15.4.4-6B, because the residual heat removal (RHR) system cannot be air tested in accordance with Appendix J.

In Reference 6, the Licensee provided the following additional information relative to this request:

> Appendix J, Section III.A.1(d), requires the performance of a Type "C" test on the containment isolation valves of the residual heat removal system. This is due to the fact that the residual heat removal system is required to maintain the plant in a safe condition during performance of the Type "A" test and, therefore, will not be drained and vented to containment atmosphere.
>
> We previously requested (Reference 1), and are again requesting, that an exemption from the Type "C" testing of the residual heat removal system containment isolation valves be granted to allow for the testing presently required by the approved Technical Specification Section 15.4.4.IV.
>
> The system is presently treated and tested as if it was an extension of the containment boundary as the system is "in use" following a design basis accident. The present testing requirements include a 350 psig hydrostatic test of the entire system, except for the containment sump piping. A 60 psig hydrostatic test is performed in that section. A total system leakage test is performed per the requirements of NUREG-0578, Item 2.1.6.a.

It can be seen that any leakage past the containment isolation valves of the residual heat removal system would only contribute to the total system fluid inventory, which in itself is leak-tested to assure that the overall system leakage is below the limits set to provide for the safe operation of the plant and the public health and safety during post-accident conditions.

You are fully aware that the residual heat removal system is always flooded and available for service. Removal of this system from active service availability for a Type "C" test of the system's containment isolation valves would present a significant potential risk to the reactor core in the event of a major accident and as such is inimical to the public health and safety. Removal of the entire core for the purpose of conducting such a test is totally unwarranted.

Considering the "in use" nature of the system, that the additional information gained by Type "C" testing of these containment isolation valves is of questionable value, the additional radiation exposure to plant personnel in performing the tests, unnecessary additional risk in having to perform a full core unload, and the totally adequate system testing programs presently provided, we believe that Type "C" testing of these valves is not justified.

FRC EVALUATION:

FRC concurs with the Licensee that Appendix J, Section III.A.1.(d), requires Type C testing of the containment isolation valves in the RHR system. At the same time, Appendix J (Sections II.B and II.D) defines containment isolation valves as those valves relied upon to prevent the escape of containment air to the outside atmosphere. Consequently, Appendix J requires Type C testing of RHR isolation valves only where they are relied upon to prevent the escape of containment air to outside atmosphere following an accident.

As the Licensee has pointed out, the RHR system is always flooded and available for service. The 350-psig hydrostatic test (60 psig in the case of the containment sump piping) is used to verify the integrity of the system. The integrity tests are part of a testing program designed to ensure the post-accident reliability of this system.

The design features of the RHR system along with the periodic testing program provide sufficient confidence that this system will perform its post-accident function and therefore will be liquid-filled and operational

throughout the post-accident period. In this condition, the system isolation valves will either be opened to permit cooling flow or will be closed and liquid-sealed by the operating pressure head of the system. In either case, these isolation valves are not being relied upon to prevent the escape of containment air to the outside atmosphere during the post-accident period.

FRC CONCLUSION:

Periodic hydrostatic testing of the RHR system is an adequate substitute for the pneumatic Type C testing required by Appendix J because the hydrostatic testing is used to ensure that the isolation valves are not relied upon to prevent the post-accident escape of containment air. Appendix J does not require further testing of these valves. An exemption from the requirements of Appendix J is acceptable.

3.1.1.3 Conduct of the Type A Test

As a result of an exchange of correspondence [8, 9] between the NRC and WEPCO regarding a Type A test at Point Beach Unit 1 in October 1977, certain issues regarding the test methods remained unresolved. These issues involve the isolation or venting of primary systems during the test and the duration of the test. In reply to certain questions from the NRC [7], WEPCO provided additional information on these issues in Reference 6.

With regard to isolating certain penetrations, the Licensee stated:

> There were no valves isolated from the Type "A" containment integrated leak rate test performed on Unit 1 in October 1977 because of excessive leakage. However, there were six penetrations isolated from the Type "A" test for purposes of maintaining the plant in a safe condition, for providing a flow path for establishing containment test pressure, and for providing an instrumentation channel for pressure test data. These penetrations were Type "C" tested following recovery from the Type "A" test. The resulting leakage data were added to the results of the Type "A" test.

Also, the Licensee discussed the duration of the test as follows:

> Please initially note that ANSI N45.4-1972 does not require a 24-hour test period as you have indicated in your correspondence. Section 7.6 of ANSI N45.4-1972 allows for a test period shorter than 24 hours as long as

"...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...."

Justification for the use of a test duration less than 24 hours is provided in the procedural excerpts below. The Technical Specifications, Section 15.4.4.1.A.2, state 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 procedure used for the integrated leak rate test on the Unit 1 containment in October 1977 stated that "the test duration shall be 24 hours; however, based on the adequacy of two prior tests, the test duration may be shortened to no less than 8 hours provided there is evidence of a 95 percent certainty that the leak rate test is within allowable limits."

Please note that the containment integrated leakage rate test procedures used during the Unit 1 Type "A" test of October 1977 were prepared by Nuclear Services Corporation, approved for use by the Wisconsin Electric Power Company Point Beach Nuclear Plant Manager's Supervisory Staff, and reviewed by the NRC prior to implementation.

FRC EVALUATION:

From the Licensee's statement regarding the isolation of certain valves during the Type A test, it appears that these procedures comply with the requirements of Section III.A.1.(d); therefore, no further discussion of this issue is necessary.

With regard to terminating the Type A test in less than 24 hours, no evaluation is provided because the NRC staff is resolving this issue separately on a generic basis.

FRC CONCLUSION:

The remaining issue, the required duration of a Type A test, is being resolved by the NRC on a generic basis.

3.1.2 Exemptions from Type B Testing Requirements

3.1.2.1 Airlock Testing

In Reference 3, the Licensee stated that its present airlock testing procedure specifies the pressurizing of the entire personnel airlock chamber.

IV. Residual Heat Removal System

A. Test

1. (a) The portion of the Residual Heat Removal System, except as specified in (b), that is outside the containment shall be tested either by use in normal operation or hydrostatically tested at 350 psig at the interval specified in IV.D below.

(b) Piping from the containment sump to the residual heat removal pump suction isolation valve shall be pressure tested at no less than 60 psig at the interval specified in IV.D below.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 61 TO FACILITY OPERATING LICENSE NO. DPR-24

AND AMENDMENT NO. 66 TO FACILITY OPERATING LICENSE NO. 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 August 5, 1975(1), the NRC requested Wisconsin Electric Power Company (WEPCo) to review its containment leakage testing program for the Point Beach Nuclear Plant, Units 1 and 2 (Point Beach 1/2) and the associated Technical Specifications, for compliance with the requirements of Appendix J to 10 CFR Part 50.

Appendix J to 10 CFR Part 50 was published on February 14, 1973. Since by this date there were already many operating nuclear plants and a number of more in advanced stages of design or construction, the NRC decided to have these plants re-evaluated against the requirements of this new regulation. Therefore, beginning in August 1975, requests for review of the extent of compliance with the requirements of Appendix J were made of each licensee. Following the initial responses to these requests, NRC staff positions were developed which would assure that the objectives of the testing requirements of the above cited regulation were satisfied. These staff positions have since been applied in our review of the submittals filed by the licensee for Point Beach 1/2. The results of our evaluation are provided below.

2.0 Discussion

Our consultant, the Franklin Research Center (FRC), has reviewed the licensee's submittals (2, 3, 5, 6, 9 and 10) and prepared the attached evaluation of containment leak rate tests for Point Beach 1/2. We have reviewed this evaluation and concur in its bases and findings.

3.0 Evaluation

Based on our review of the enclosed technical evaluation report (TER) as prepared by the FRC, the following conclusions are made regarding the Appendix J review for Point Beach 1/2:

DESIGNATED ORIGINAL

Certified By Patricia J. Noonan

1. The service air supply system is used in conjunction with the Type A test and, consequently, its leakage integrity is not factored into the test results. However, Type C testing of the service air supply line, with measured leakage added to the Type A results, is an adequate substitution for the Type A testing requirements of Appendix J. This approach is described in Section III.A.1.(d) of Appendix J. An exemption from the requirements of Appendix J is necessary, however, because, unlike the systems addressed in Section III.A.1.(d), the service air supply system is not needed to maintain the plant in the safe shutdown mode. The staff and its consultant both conclude that an exemption is acceptable.

2. Periodic hydrostatic testing of the RHR system is an adequate substitute for the pneumatic (Type C) testing required by Appendix J because the hydrostatic testing is utilized to ensure that the isolation valves are not relied upon to prevent the post-accident escape of containment air. Appendix J does not require further air (Type C) testing of these valves; therefore, an exemption from the requirements of Appendix J is acceptable.

3. The request for exemption to permit the Type A test to be terminated in less than 24 hours was not evaluated since an exemption from the requirements of Appendix J is not needed if the licensee commits, in writing, to conduct Type A tests in accordance with the staff-approved Bechtel Topical Report BN-TOP1, and the plant Technical Specifications reflect this commitment.

4. An exemption from the revised containment airlock testing requirements of Section III.D.2 is not justified. Airlock testing at Point Beach should be conducted in accordance with Appendix J.

5. The proposed method of measuring the gas flow required to maintain constant test chamber pressure (Operating Instruction No. 58) is equivalent to the pressure-loss method of Section III.B.1.(b) of Appendix J. The Wisconsin Electric Power Company test procedure is acceptable for use in performing Type B tests. No exemption from Appendix J is necessary since acceptable test methods are not limited to those described in Appendix J.

6. Reverse-direction testing of certain containment isolation valves is authorized because the licensee has determined that the criteria of Section III.C.1 have been met. No exemption is required.

7. Substitution of a hydraulic test for the required pneumatic test of the containment spray isolation check valves is not acceptable.

8. Proposed specification 15.4.4.I.A.2 was not evaluated because the issue of performing a Type A test in less than a 24 hour period should be resolved in accordance with Item 3, above.

9. Proposed specification 15.4.4.I.B.5 is acceptable as an exemption to the requirements of Section III.A.1.(a) of Appendix J. Our acceptance is based upon our understanding that the licensee's sequence and methodology for conducting Type A containment tests is as described in the attached FRC TER. WEPCo's actual test procedures should be sufficiently clear and detailed to reflect this.

10. Proposed specification 15.4.4.II.C.1 should be revised to require airlock testing in accordance with Section III.D.2 of Appendix J.

Environmental Consideration

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date:

Attachment:
FRC TER

Principal Contributors:
P. Hearn
T. Colburn

4.0 References

1) NRC Generic letter regarding Containment Leakage Testing at Point Beach, dated August 7, 1975.

2) S. Burstein (WEPCO) letter to K. Goller (NRC), dated September 5, 1975.

3) S. Burstein (WEPCO) letter to B. Rusche (NRC), dated December 12, 1975.

4) G. Lear (NRC) letter to S. Burstein (WEPCO), dated May 31, 1977.

5) S. Burstein (WEPCO) letter to G. Lear (NRC), dated July 18, 1977.

6) C. W. Fay (WEPCO) letter to R. A. Clark (NRC), dated February 25, 1981.

7) R. A. Clark (NRC) letter to S. Burstein (WEPCO), dated January 27, 1981.

8) G. Lear (NRC) letter to S. Burstein (WEPCO), dated October 4, 1977.

9) S. Burstein (WEPCO) letter to G. Lear (NRC), dated October 10, 1977.

10) S. Burstein (WEPCO) letter to E. CAse (NRC), dated February 6, 1978.

DESIGNATED ORIGINAL

Certified By Patricia J. Moore

TECHNICAL EVALUATION REPORT

CONTAINMENT LEAKAGE RATE TESTING

WISCONSIN ELECTRIC POWER COMPANY
POINT BEACH NUCLEAR PLANT UNITS 1 AND 2

NRC DOCKET NO. 50-266, 50-301

NRC TAC NO. 07711, 08404

NRC CONTRACT NO. NRC-03-79-118

FRC PROJECT C5257

FRC ASSIGNMENT 1

FRC TASKS 41, 42

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March 30, 1982

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CONTENTS

FOREWORD

This Technical Evaluation Report was prepared by Franklin Research Center under a contract with the U.S. Nuclear Regulatory Commission (Office of Nuclear Reactor Regulation, Division of Operating Reactors) for technical assistance in support of NRC operating reactor licensing actions. The technical evaluation was conducted in accordance with criteria established by the NRC.

Mr. T. J. DelGaizo and Mr. J. S. Scherrer contributed to the technical preparation of this report through a subcontract with WESTEC Services, Inc.