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. 64
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 August 28, 1981 as modified by letter dated January 28, 1982, 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. 64 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 4, 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. 69
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 August 28, 1981 as modified by letter dated January 28, 1982, 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. 69 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: October 4, 1982

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 64 TO FACILITY OPERATING LICENSE NO. DPR-24

AMENDMENT NO. 69 TO FACILITY OPERATING LICENSE NO. DPR-27

DOCKET NOS. 50-266 AND 50-301

Revise Appendix A as follows:

Remove Pages	Insert Pages
15.3.6-2	15.3.6-2
15.3.6-3	15.3.6-3
Table 15.4.1-2 page 2 of 2	Table 15.4.1-2 page 2 of 2
15.4.4-12	15.4.4-12
15.4.4-13	15.4.4-13
15.4.4-14	15.4.4-14
15.4.4-15	15.4.4-15

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.

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 This precludes criticality under any circumstances even though fuel is being moved or control rods withdrawn.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precludes criticality because the reactor would be substantially subcritical even if all control rods were completely withdrawn.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

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

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

References

(1) FSAR - Section 14.3.4

(2) FSAR - Section 5.5.2

TABLE 15.4.1-2 (Continued)

	Test	Frequency
14. Refueling System Interlocks	Functioning	Each refueling shutdown
15. Service Water System	Functioning	Each refueling shutdown
16. Primary System Leakage	Evaluate	Monthly (6)
17. Diesel Fuel Supply	Fuel inventory	Daily
18. Turbine Stop and Governor Valves	Functioning	Monthly (6)
19. Low Pressure Turbine Rotor Inspection (5)	Visual and magnetic particle or liquid penetrant	Every five years
20. Boric Acid System	Storage Tank Temperature	Daily
21. Boric Acid System	Visual observation of piping temperatures (all $\geq$ 145°F)	Daily
22. Boric Acid Piping Heat Tracing	Electrical circuit operability	Monthly
23. PORV Block Valves	Complete Valve Cycle	Quarterly (6)
24. Integrity of Post Accident Recovery Systems Outside Containment	Evaluate	Yearly
25. Containment Purge Supply and Exhaust Isolation Valves	Verify valves are locked closed	Monthly (9)

---

(1) A radiochemical analysis for this purpose shall consist of a quantitative measurement of each radionuclide with half life of >30 minutes such that at least 95% of total activity of primary coolant is accounted for.
(2) $\bar{E}$ determination will be started when the gross activity analysis of a filtered sample indicates $\geq$10 µc/cc and will be redetermined if the primary coolant gross radioactivity of a filtered sample increases by more than 10 µc/cc.
(3) Drop tests shall be conducted at rated reactor coolant flow. Rods shall be dropped under both cold and hot conditions, but cold drop tests need not be timed.
(4) Drop tests will be conducted in the hot condition for rods on which maintenance was performed.
(5) As accessible without disassembly of rotor.
(6) Not required during periods of refueling shutdown.
(7) At least once per week during periods of refueling shutdown.
(8) At least three times per week (with maximum time of 72 hours between samples) during periods of refueling shutdown.
(9) Not required during periods of cold or refueling shutdown.

Unit 1 - Amendment No. ~~32~~, ~~45~~, ~~55~~, 64
Unit 2 - Amendment No. ~~30~~, ~~33~~, ~~60~~, 69

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.

X. Leakage Test of Containment Purge Supply and Exhaust Valves

The containment purge supply and exhaust valves shall be demonstrated to be leak tight at intervals not to exceed 6 months. Valve operability shall be determined by verifying that when the measured leakage rate is added to the leakage rates last determined pursuant to Specifications 15.4.4.II and 15.4.4.III for other penetrations and isolation valves, the combined leakage rate is less than or equal to 0.6 L_a. 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.

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 will be strength tested at 69 psig and then will be leak-tested. The design objective of this pre-operational leakage rate test has been established as 0.4% by weight per 24 hours at 60 psig. This leakage rate is consistent with the construction of the containment,(2) which is equipped with independent leak-testable penetrations and contains channels over all containment liner welds, which were independently leak-tested during construction.

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

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

The performance of 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 a steam-air mixture at 60 psig and 286°F. The specification also allows for possible deterioration of the leakage rate between tests, by requiring that only 75% of the allowable leakage rates actually be measured. 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 24 hours for the integrated leakage rate test is established to provide a minimum level of accuracy and to allow for daily cyclic variation in temperature and thermal radiation.

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 is designed for approximately 24 months of power operation, thus the first refueling will occur approximately 30 months after initial criticality. Subsequent refueling shutdowns are scheduled at approximately 12-18 month intervals.

The specified frequency of periodic integrated leakage rate tests is based on three major considerations. First is the low probability of leaks in the liner, because of (a) the use of weld channels to test the leak tightness of the welds during erection, (b) conformance of the complete containment to a low leak rate at 60 psig during pre-operational testing which is consistent with 0.4% leakage at design basis accident conditions, and (c) absence of any significant stresses in the liner during reactor operation. Second is the more frequent testing, at the full accident pressure, of those portions of the containment envelope that are most likely to develop leaks during reactor operation (penetrations and isolation valves) and the low value (0.6 L_a) of the leakage that is specified as acceptable from penetrations and isolation valves. Third is the tendon stress surveillance program, which provides assurance that an important part of the structural integrity of the containment is maintained. A final point is that the 0.40%/day acceptance criterion for the integrated leakage test is indicated to be a factor of about 2 lower than necessary to meet 10 CFR 100 values.

The basis for specification of a leakage rate of 0.6 L_a from penetrations and isolation valves is that only six-tenths of the allowable integrated leakage rate should be from each of those sources, in order to provide assurance that the integrated leakage rate would remain within the specified limits during the intervals between integrated leakage rate tests. The allowable value of 0.6 L_a is found in 10 CFR Part 50, Appendix J.

The limiting leakage rates from the Residual Heat Removal System are judgment values based primarily on assuring that the components could operate without mechanical failure for a period on the order of 200 days after a Design Basis Accident. The test pressure (350 psig) achieved either by normal system operation or by hydrostatically testing gives an adequate margin over the highest pressure within the system after a design basis accident. Similarly, the pressure test for the return lines from the containment to the Residual Heat Removal System (60 psig) is equivalent to the design pressure of the containment. A Residual Heat Removal System leakage of 2 gal/hr will limit off-site exposures due to leakage to insignificant levels relative to those calculated for leakage directly from the containment in the Design Basis Accident. The dose calculated as a result of this leakage is 7.7 mr for a 2 hour exposure at the site boundary.[5]

Periodic visual inspection is the method to be used to determine loss of load-carrying capability because of wire breakage. The pre-stress lift-off test provides a direct measure of the load-carrying capability of the tendon. A deterioration of the corrosion preventive properties of the sheathing filler will be indicated by a change in the physical appearance of the filler. If the surveillance program indicated, by extensive wire breakage or tendon stress relation, that the pre-stressing tendons are not behaving as expected, the situation will be evaluated immediately. The specified acceptance criteria are such as to alert attention to the situation well before the tendon load-carrying capability would deteriorate to a point that failure during a design basis accident might be possible. Thus, the cause of the incipient deterioration could be evaluated and corrective action studied without need to shut down the reactor.

The purpose of the leakage tests of the isolation valves in the containment purge supply and exhaust lines is to identify excessive degradation of the resilient seals for these valves. With the exception of the test frequency and acceptance criteria, leakage tests of the containment purge supply and exhaust valves shall be conducted in accordance with 15.4.4.III.

References

(1) FSAR Section 5.1.2.3
(2) FSAR Section 5.1.2
(3) FSAR Section 14.3.5
(4) FSAR Section 14.3.4
(5) FSAR Section 6.2.3