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. 114
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 January 19, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 114 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective immediately upon issuance. The Technical Specifications are to be implemented within 20 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Kenneth E. Perkins

Kenneth E. Perkins, Director
Project Directorate III-3
Division of Reactor Projects - III,
IV, V and Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 18, 1988

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. 117
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 January 19, 1988 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-27 is hereby amended to read as follows:

   B. <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 117 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective immediately upon issuance. The Technical Specifications are to be implemented within 20 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Kenneth E. Perkins

Kenneth E. Perkins, Director
Project Directorate III-3
Division of Reactor Projects - III,
IV, V and Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 18, 1988

ATTACHMENT TO LICENSE AMENDMENT NOS. 114 AND 117

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

DOCKET NOS. 50-266 AND 50-301

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

REMOVE	INSERT
15.3.5-5	15.3.5-5
15.3.5-6	15.3.5-6
Table 15.3.5-1 (page 1)	Table 15.3.5-1 (page 1)
15.3.10-15	15.3.10-15
15.3.10-16	15.3.10-16
15.4.4-12	15.4.4-12
15.4.4-13	15.4.4-13
15.4.4-14	15.4.4-14
15.6.4/5-1	15.6.4/5-1
15.6.5-4	15.6.5-4
15.6.5-5	15.6.5-5
15.6.5-6	15.6.5-6
15.6.5-7	15.6.5-7
15.6.5-8	15.6.5-8
15.7.5-8	15.7.5-8
15.7.7-2	15.7.7-2

which automatically initiates appropriate action to prevent exceeding established limits. Safety is not compromised, however, by continuing operation with certain instrumentation channels out of service since provisions were made for this in the plant design. This specification outlines limiting conditions for operation necessary to preserve the effectiveness of the Reactor Control and Protection System when any one or more of the channels is out of service.

Almost all reactor protection channels are supplied with sufficient redundancy to provide the capability for channel calibration and test at power. Exceptions are backup channels such as reactor coolant pump breakers. The removal of one trip channel on process control equipment is accomplished by placing that channel bistable in a tripped mode; e.g., a two-out-of-three circuit becomes a one-out-of-two circuit. The source and intermediate range nuclear instrumentation system channels are not intentionally placed in a tripped mode since these are one-out-of-two trips, therefore the trips are bypassed during testing. Testing of the NIS power range channel requires bypassing the Dropped Rod protection from NIS, for the channel being tested. However, the Rod Position System still provides the dropped-rod protection. Testing does not trip the system unless a trip condition exists in a concurrent channel.

The operability of the accident monitoring instrumentation ensures that sufficient information is available in selected plant parameters to monitor and assess these variables during and following an accident. The PORV block valves have local, external indication of whether the block valve is open or shut. If necessary, this local indication can be visually verified during a containment entry inspection to verify the block valve is shut.

The subcooling displays are comprised of two separate channels which receive temperature and pressure information directly from installed instrumentation. These channels display the temperature differential between the sensed conditions in the reactor coolant system and the calculated saturation temperature. As a backup, the Plant Process Computer System (PPCS) displays subcooling margin by both addressable point and on the Safety Assessment System (SAS). A second backup display of subcooling information is available on seismically qualified plasma displays which receive input signals from seismically qualified multiplexing equipment. Control board indications and a saturation curve can be used if failure of all direct subcooling indications occurs.

Reference

(1) FSAR - Section 7.5

(2) FSAR - Section 14.3

(3) FSAR - Section 14.2.5

TABLE 15.3.5-1
(PAGE 1 OF 2)
ENGINEERED SAFETY FEATURES INITIATION INSTRUMENT SETTING LIMITS

NO.	FUNCTIONAL UNIT	CHANNEL	SETTING LIMIT
1	High Containment Pressure (Hi)	Safety Injection*	≤ 6 psig
2	High Containment Pressure (Hi-Hi)	a. Containment Spray b. Steam Line Isolation of Both Lines	≤ 30 psig ≤ 20 psig
3	Pressurizer Low Pressure	Safety Injection*	≥ 1715 psig
4	Low Steam Line Pressure	Safety Injection* Lead Time Constant Lag Time Constant	≥ 500 psig ≥ 12 seconds ≤ 2 seconds
5	High Steam Flow in a Steam Line Coincident with Safety Injection and Low T_{avg}	Steam Line Isolation of Affected Line	≤ d/p corresponding to 0.66 x 10^6 lb/hr at 1005 psig ≥ 540°F
6	High-high Steam Flow in a Steam Line Coincident with Safety Injection	Steam Line Isolation of Affected Line	≤ d/p corresponding to 4 x 10^6 lb/hr at 806 psig
7	Low-low Steam Generator Water Level	Auxiliary Feedwater Initiation	≥ 5% of narrow range instrument
8	Undervoltage on 4 KV Busses	Auxiliary Feedwater Initiation	≥ 75% of normal voltage

*Initiates also containment isolation, feedwater line isolation and starting of all containment fans.

d/p means differential pressure

in a deliberate manner without undue pressure on the operating personnel because of the unusual techniques to be used to accommodate the reactivity changes associated with the shutdown.

Misaligned RCCAS

The various control rod banks (shutdown banks and control banks, A, B, C, and D) are each to be moved as a bank; that is, with all rods in the bank within one step (5/8 inch) of the bank position. Direct information on rod position indication is provided by two methods: A digital count of actuating pulses which shows the demand position of the banks and a linear position indicator (LVDT) which indicates the actual rod position. The rod position indicator channel has a demonstrated accuracy of 5% of span (± 7.2 inches). Therefore, an analysis has been performed to show that a misalignment of 15 inches cannot cause design hot channel factors to be exceeded. A single fully misaligned RCCA, that is, an RCCA 12 feet out of alignment with its bank, does not result in exceeding core limits in steady-state operation at power levels less than or equal to rated power. In other words, a single dropped RCCA is allowable from a core power distribution viewpoint. If the misalignment condition cannot be readily corrected, the specified reduction in power to 75% will insure that design margins to core limits will be maintained under both steady-state and anticipated transient conditions. The eight (8) hour permissible limit on rod misalignment at rated power is short with respect to the probability of an independent accident.

Because the rod position indicator system may have a 7.5 inch error when a misalignment of 15 inches is occurring, the Specification allows only a 7.5 inch indicated misalignment. However, when the bank demand position is greater than or equal to 215 steps, or, less than or equal to 30 steps, the consequences of a misalignment are much less severe. The differential worth of an individual RCCA is less, and the resultant perturbation on power distributions is less than when the bank is in its high differential worth region. At the top and bottom of the core, an indicated 15 inch misalignment may be representing an actual misalignment of 22.5 inches.

The failure of an LVDT in itself does not reduce the shutdown capability of the rods, but it does reduce the operator's capability for determining the position of that rod by direct means. The operator has available to him the excore detector recordings, incore thermocouple readings and periodic incore flux traces for indirectly determining rod position and flux tilts should the rod with the inoperable LVDT become malpositioned. The excore and incore instrumentation will not necessarily recognize a misalignment of 15 inches because the concomitant increase in power density will normally be less than 1% for a 15 inch misalignment. The excore and incore instrumentation will, however, detect any rod misalignment which is sufficient to cause a significant increase in hot channel factors and/or any significant loss in shutdown capability. The increased surveillance of the core if one or more rod position indicator channels is out-of-service serves to guard against any significant loss in shutdown margin or margin to core thermal limits.

The history of malpositioned RCCA's indicates that in nearly all such cases, the malpositioning occurred during bank movement. Checking rod position after bank motion exceeds 24 steps will verify that the RCCA with the inoperable LVDT is moving properly with its bank and the bank step counter. Malpositioning of an RCCA in a stationary bank is very rare, and if it does occur, it is usually gross slippage which will be seen by external detectors. Should it go undetected, the time between the rod position checks performed every shift is short with respect to the probability of occurrence of another independent undetected situation which would further reduce the shutdown capability of the rods.

Any combination of misaligned rods below 10% rated power will not exceed the design limits. For this reason, it is not necessary to check the position of rods with inoperable LVDT's below 10% power; plus, the incore instrumentation is not effective for determining rod position until the power level is above approximately 5%.

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

Basis

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

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

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

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

The performance of 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 preoperational 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 the as measured leak rate to be less than 75% of the allowable leakage rate. The basis for these deterioration allowances are arbitrary judgments, which are believed to be conservative and which will be confirmed or denied by periodic testing. If indicated to be necessary, the deterioration allowances will be altered based on experience.

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

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

15.6.4 TRAINING

15.6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Superintendent - Training and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and 10 CFR Part 55.

15.6.4.2 A training program for the Fire Brigade shall meet or exceed the requirements of Section 27 of the NFPA Code-1976, except that the meeting frequency may be quarterly.

15.6.5 REVIEW AND AUDIT

15.6.5.1 Manager's Supervisory Staff

15.6.5.1.1 The Manager's Supervisory Staff (MSS) shall function to advise the Manager on all matters related to nuclear safety.

15.6.5.1.2 The Manager's Supervisory Staff shall be selected from the following:

Chairman: Manager - Point Beach Nuclear Plant
Member: General Superintendent
Member: Superintendent - Operations
Member: Superintendent - Maintenance & Construction
Member: Superintendent - Engineering, Quality & Regulatory Services
Member: Superintendent - Training
Member: Superintendent - Technical Services
Member: Superintendent - Reactor Engineering
Member: Radiochemist
Member: Health Physicist
Member: Superintendent - Instrumentation & Control

15.6.5.1.3 Alternate members may be appointed by the MSS Chairman to serve on a temporary basis; however, no more than two alternates shall vote in MSS at any one time. Such appointment shall be in writing.

15.6.5.2 <u>OFF-SITE REVIEW COMMITTEE</u> (OSRC)

<u>FUNCTION</u>

15.6.5.2.1 The Off-Site Review Committee shall function to provide independent review and audit of designated activities in the areas of:

a) nuclear power plant operations

b) nuclear engineering

c) chemistry and radiochemistry

d) metallurgy

e) instrumentation and control

f) radiological safety

g) mechanical and electrical engineering

h) quality assurance practices

i) environmental monitoring

<u>COMPOSITION</u>

15.6.5.2.2 The Off-Site Review Committee is made up of a minimum of five regular members appointed by the President and one or more ex-officio members. Of the five or more regular members, at least two will be persons not directly employed by the Licensee. All members will be experienced in one or more aspects of the nuclear industry.

<u>ALTERNATES</u>

15.6.5.2.3 Alternate members may be appointed in writing by the OSRC Chairman to serve on a temporary basis; however, no more than two alternates shall participate in OSRC activities at any one time.

<u>CONSULTANTS</u>

15.6.5.2.4 Consultants shall be utilized as determined by the OSRC Chairman to provide expert advice to the OSRC.

MEETING FREQUENCY

15.6.5.2.5 The OSRC shall meet at least once per calendar quarter during the initial year of facility operation following fuel loading and at least twice per year at approximately six month intervals thereafter.

QUORUM

15.6.5.2.6 A quorum of OSRC shall consist of the Chairman or his designated alternate and three members. No more than a minority of the quorum shall have line responsibility for operation of the facility.

REVIEW

15.6.5.2.7 The OSRC shall review:

a) The safety evaluations for 1) changes to procedures, equipment or systems, and 2) tests or experiments completed under the provision of 10 CFR, Section 50.59, to verify that such actions did not constitute an unreviewed safety question.
b) Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in 10 CFR, Section 50.59.
c) Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR Section 50.59.
d) Proposed changes in Technical Specifications or Licenses.
e) Violations of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.
f) Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.
g) All reportable events.

15.6.5.2.7 (Continued)

h) Any indication of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

i) Reports and meeting minutes of the Manager's Supervisory Staff.

AUDITS

15.6.5.2.8 Audits of facility activities shall be performed under the cognizance of the OSRC. These audits shall encompass:

a) The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions at least once per year.

b) The performance, training and qualifications of the licensed operating staff at least once per year.

c) The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety at least twice per year at approximately six month intervals.

d) The results of quarterly audits by the Quality Assurance Division on the performance of activities required by the Quality Assurance Program to meet the criteria of Appendix B, 10 CFR 50, at least once per two years.

e) Any other area of facility operation considered appropriate by the President.

AUTHORITY

15.6.5.2.9 The OSRC shall report to and advise the President on those areas of responsibility specified in Section 15.6.5.2.7 and 15.6.5.2.8.

RECORDS

15.6.5.2.10 Records of OSRC activities shall be prepared, approved and distributed as indicated below:

a) Minutes of each OSRC meeting shall be prepared, approved and forwarded to the President within 14 days following each meeting.

b) Reports of reviews encompassed by Section 15.6.5.2.7.e, f and g above shall be prepared, approved and forwarded to the President within 14 days following completion of the review.

c) Audit reports encompassed by Section 15.6.5.2.8 above, shall be forwarded to the President and to the management positions responsible for the areas audited within 30 days after completion of the audit.

15.6.5.3 Fire Protection Audits

a) An independent fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified offsite license personnel or an outside fire protection firm.

b) An inspection and audit of the fire protection and loss prevention program shall be performed by an outside qualified fire consultant at intervals no greater than 3 years.

15.6.5.4 Emergency Plan Reviews

a) A review of the Emergency Preparedness Program shall be performed annually utilizing either offsite licensee personnel or an outside nuclear consulting firm. The review shall be conducted in accordance with 10 CFR 50.54(t) as effective on September 1, 1982.

releases upward to the point at which corresponding doses reach the applicable limit specified in Appendix I to 10 CFR Part 50.

The radioactive liquid and gaseous effluent instrumentation is provided to monitor and control the releases of radioactive materials in liquid and gaseous effluents during actual or potential releases. The trip setpoints for these instruments are calculated utilizing the methodology in the Offsite Dose Calculation Manual.

The requirement that the appropriate portions of the liquid and gaseous radwaste treatment systems be used when specified provides assurance that the releases of radioactive materials in liquid and gaseous effluents will be kept "as low as is reasonably achievable".

Compliance with the provisions of Appendix I to 10 CFR Part 50 constitutes adequate demonstration of conformance to the standards set forth in 40 CFR Part 190 regarding the dose commitment to individuals from the uranium fuel cycle. The Specifications require that if actual quantities of radioactive materials released exceed twice the quantities associated with the design dose objective of Appendix I to 10 CFR Part 50, actual doses will be calculated and a special report will be submitted.

<u>References</u>:

(1) FSAR, Section 10.2
(2) FSAR, Section 2
(3) FSAR, Sections 2.6 and 2.7

identified in the next Semiannual Monitoring Report. Figures and tables in the Environmental Manual are to be revised reflecting the new sample locations.

B. Detection Capabilities

1. Environmental samples shall be analyzed as specified in Table 15.7.7-2.
2. The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLDs).
3. If circumstances render the stated LLDs in Table 15.7.7-2 unachievable, the contributing factors shall be identified and described in next Semiannual Monitoring Report.

C. Notification Levels

1. If a measured level of radioactivity in any environmental medium exceeds the notification level listed in Table 15.7.7-3, resampling and/or reanalysis for confirmation shall be completed within 30 days of the determination of the anomalous result. If the confirmed measured level of radioactivity remains above the notification level, a written report shall be submitted to the NRC in accordance with Section 15.7.8.4.B within thirty days of the confirmation. This report is not required if the measured level of radioactivity was not the result of plant effluents.
2. If more than one of the radionuclides listed in Table 15.7.7-3 are detected in any environmental medium, a weighted sum calculation shall be performed if the measured concentration of a detected radionuclide is greater than 25% of the notification levels. For those radionuclides with LLDs in excess of 25% of the notification level, a weighted sum calculation need only be performed if the reported value exceeds the LLD. The weighted sum is calculated as follows:

$$\frac{\text{concentration (1)}}{\text{notification level (1)}} + \frac{\text{concentration (2)}}{\text{notification level (2)}} + \ldots = \text{weighted sum}$$

If the calculated weighted sum is equal to or greater than 1, resampling and/or reanalysis for confirmation shall be completed within 30 days of the determination of the anomalous result. If