DCS/MS-016

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Docket File
NRC PDR
Local PDR
ORB# 3 Rdg
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NSIC
HRDenton

Docket Nos. 50-266
and 50-301

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

Dear Mr. Fay:

The Commission has issued the enclosed Amendment No. 76 to Facility Operating License No. DPR-24 and Amendment No. 80 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 May 4, 1983.

These amendments make various administrative changes to the Technical Specifications in order to clarify terminology used in a limiting condition for operation, clarify language relating to a periodic calibration interval requirement, and correct specific portions of the specifications and bases.

A copy of the Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's next monthly Federal Register notice.

Sincerely,

Original signed by

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

[illegible]

Enclosures:
1. Amendment No. 76 to DPR-24
2. Amendment No. 80 to DPR-27
3. Safety Evaluation

cc: w/enclosures
See next page

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DATE	9/23/83	9/23/83	9/27/83	9/ /83	9/28/83		

NRC FORM 318 (10-80) NRCM 0240 OFFICIAL RECORD COPY USGPO: 1981—335-960

Wisconsin Electric Power Company

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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. 76
License No. DPR-24

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Wisconsin Electric Power Company (the licensee) dated May 4, 1983, 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. 76, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

James R. Miller

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 6, 1983

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-301

POINT BEACH NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 80
License No. DPR-27

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Wisconsin Electric Power Company (the licensee) dated May 4, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   B. Technical specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 80 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

James R. Miller

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 6, 1983

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 76 TO FACILITY OPERATING LICENSE NO. DPR-24

AMENDMENT NO. 80 TO FACILITY OPERATING LICENSE NO. DPR-27

DOCKET NOS. 50-266 AND 50-301

Revise Appendix A as follows:

Remove Pages	Insert Pages
15.3.1.3a	15.3.1-3a
15.3.3-9	15.3.3-9
15.3.10.-6	15.3.10-6
15.3.10-7	15.3.10-7
Table 15.4.1-1 Page 4 of 4	Table 15.4.1-1 Page 4 of 4

Above 50% power, an automatic reactor trip will occur if either pump is lost. The power-to-flow ratio will be maintained equal to or less than 1.0, which ensures that the minimum DNB ratio increases at lower flow since the maximum enthalpy rise does not increase above its normal full-flow maximum value.[2]

Specification 15.3.1.A.3 provides limiting conditions for operation to ensure that redundancy in decay heat removal methods is provided. A single reactor coolant loop with its associated steam generator and a reactor coolant pump or a single residual heat removal loop provides sufficient heat removal capacity for removing the reactor core decay heat; however, single failure considerations require that at least two decay heat removal methods be available. Operability of a steam generator for decay heat removal includes two sources of water, water level indication in the steam generator, a vent path to atmosphere, and the Reactor Coolant System filled and vented so thermal convection cooling of the core is possible. If the steam generators are not available for decay heat removal, this Specification requires both residual heat removal loops to be operable unless the reactor system is in the refueling shutdown condition with the refueling cavity flooded and no operations in progress which could cause an increase in reactor decay heat load or a decrease in boron concentration. In this condition, the reactor vessel is essentially a fuel storage pool and removing a RHR loop from service provides conservative conditions should operability problems develop in the other RHR loop. Also, one residual heat removal loop may be temporarily out of service due to surveillance testing, calibration, or inspection requirements. The surveillance procedures follow administrative controls which allow for timely restoration of the residual heat removal loop to service if required.

Each of the pressurizer safety valves is designed to relieve 288,000 lbs. per hour of saturated steam at setpoint. If no residual heat is removed by any of the means available, the amount of steam which could be generated at safety valve relief pressure would be less than half the valves' capacity. One valve, therefore, provides adequate defense against overpressurization. Below 350°F and 400 psig in the Reactor Coolant System, the residual heat removal system can remove decay heat and thereby control system temperature and pressure.

A PORV is defined as OPERABLE if leakage past the valve is less than that allowed in Specification 15.3.1.D and the PORV has met its most recent channel test as specified in Table 15.4.1-1. The PORVs operate to relieve, in a controlled following combinations will provide sufficient cooling to reduce containment pressure: (1) four fan coolers, (2) two containment spray pumps, (3) two fan coolers plus one containment spray pump.[4] Sodium hydroxide addition via one spray pump reduces airborne iodine activity sufficiently to limit off-site doses to acceptable values. One of the four fan coolers is permitted to be inoperable for up to 48 hours during power operation.

The component cooling system is different from the other systems discussed above in that the components are so located in the Auxiliary Building as to be accessible for repair after a loss-of-coolant accident. One component cooling water pump together with one component cooling heat exchanger can accommodate the heat removal load on one unit either following a loss-of-coolant accident, or during normal plant shutdown. If during the post-accident phase the component cooling water supply is lost, core and containment cooling could be maintained until repairs were effected.[5]

A total of six service water pumps are installed, only three of which are required to operate during the injection and recirculation phases of a postulated loss-of-coolant accident,[6] in one unit together with a hot shutdown condition in the other unit.

References

(1) FSAR Section 3.2.1

(2) FSAR Section 6.2

(3) FSAR Section 6.3.2

(4) FSAR Section 6.3

(5) FSAR Section 9.3.2

(6) FSAR Section 9.6.2

a. The RCCA does not drop upon removal of stationary gripper coil voltage.

b. The RCCA does not step in properly when the proper voltage sequences are applied to the control rod drive mechanism coils. It shall then be assumed inoperable until it has been tested to verify that it does drop.

c. If the bank demand position is greater than or equal to 215 steps, or, less than or equal to 30 steps, and the rod position indicator channel shows a misalignment from the bank demand position of 15 inches. The RCCA shall be assumed inoperable until it has been tested to verify that it does step properly.

d. If the bank demand position is between 215 steps and 30 steps, and the rod position indicator channel shows a misalignment from the bank demand position of 7.5 inches. The RCCA shall be assumed inoperable until it has been tested to verify that it does step properly.

2. Specification 15.3.10.C.1.b can be modified by the following:

   a. If an RCCA does not step in upon demand, up to six hours is allowed to determine whether the problem with stepping is an electrical problem. If the problem cannot be resolved within six hours, the RCCA shall be assumed inoperable until it has been verified that it will step in or would drop upon demand.

   b. If more than one RCCA does not step in, apparently due to electrical problems, the situation shall be rectified or clearly defined that it is an electrical problem and the RCCAs are capable of dropping upon demand or an orderly shutdown shall commence within six hours.

3. No more than one inoperable RCCA shall be permitted during sustained power operation.

4. When it has been determined that an RCCA does not drop on removal of stationary gripper coil voltage, the shutdown margin shall be maintained by boration as necessary to compensate for the withdrawn worth of the inoperable RCCA. If sustained power operation is anticipated, the insertion limit shall be adjusted to reflect the worth of the inoperable RCCA.

D. <u>Misaligned or Dropped RCCA</u>

1. If the rod position indicator channel is functional and the associated RCCA is more than 7.5 inches indicated out of alignment with its bank demand position and cannot be aligned when the bank demand position is between 215 steps and 30 steps, then unless the hot channel factors are shown to be within design limits as specified in Section 15.3.10.B-1 within eight (8) hours, power shall be reduced to less than 75% of Rated Power. When the bank demand position is greater than or equal to 215 steps, or less than or equal to 30 steps, the allowable indicated misalignment is 15 inches between the rod position indicator and the bank demand position.
2. To increase power above 75% with an RCCA more than 7.5 inches indicated out of alignment with its bank demand position when the bank demand position is between 215 steps and 30 steps, an analysis shall first be made to determine the hot channel factors and the resulting allowable power level based on Section 15.3.10.B. When the bank demand position is greater than or equal to 215 steps, or less than or equal to 30 steps, the allowable indicated misalignment is 15 inches between the rod position indication and the bank demand position.
3. If it is determined that the apparent misalignment or dropped RCCA indication was caused by rod position indicator channel failure, sustained power operation may be continued if the following conditions are met:
    a. For operation between 10% power and Rated Power, the position of the RCCA(s) with the failed rod position indicator channel(s) will be checked indirectly by core instrumentation (excore detectors, and/or thermocouples, and/or moveable incore detectors) every shift and after associated bank motion exceeding 24 steps in one direction.
    b. For operation below 10% of Rated Power, no special monitoring is required.

E. <u>RCCA Drop Times</u>

1. At operating temperature and full flow, the drop time of each RCCA shall be no greater than 1.8 seconds from the loss of stationary gripper coil voltage to dashpot entry.

TABLE 15.4.1-1 (Continued)

S - Each shift	M - Monthly
D - Daily	P - Prior to each startup if not done previous week.
W - Weekly	R - Each Refueling Shutdown (But not to exceed 18 months).
B/W - Biweekly	N.A. - Not applicable.

** Not required during periods of refueling shutdown, but must be performed prior to starting up if it has not been performed during the previous surveillance period.

*** Not required during periods of refueling shutdown if steam generator vessel temperature is greater than 70°F.

**** When used for the overpressure mitigating system each PORV shall be demonstrated operable by:

a. Performance of a channel functional test on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required operable and at least once per 31 days thereafter when the PORV is required operable.

b. Testing valve operation in accordance with the inservice test requirements of the ASME Boiler and Pressure Vessel Code, Section IX.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 76 TO FACILITY OPERATING LICENSE NO. DPR-24

AND AMENDMENT NO. 80 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

Introduction

By letter dated May 4, 1983 Wisconsin Electric Power Company (licensee) requested several changes to the Technical Specifications (TS) and related bases. These changes were submitted to clarify and correct specific portions of the specifications and bases.

Discussion and Evaluation

A revision to the basis for TS 15.3.1.A at page 15.3.1-3a was proposed to be consistent with the intent of the limiting conditions for operation which require that, when no decay heat removal method is in operation, all operations causing an increase in the reactor decay heat load or a reduction in boron concentration shall be suspended.

A revision to the basis on page 15.3.3-9 has been made to correct an inconsistency with TS 15.3.3.B regarding operability of containment fan coolers. Proposed revisions to TS 15.3.10.C(1)(c), 15.3.10.C(1)(d), 15.3.10.D(1) and 15.3.10.D(2) have been submitted to clarify the conditions during which rod control cluster assemblies are considered misaligned or inoperable. The misalignment has been clarified to refer to a deviation between the rod position indication and the bank demand position. This is consistent with the wording in the Standard Technical Specifications for Westinghouse Pressurized Water Reactors.

The last requested TS change relates to the definition of the surveillance frequency code "R". The present TS defines "R" as each refueling shutdown (but not to exceed 20 months). The proposed change revises frequency code R to mean each refueling interval (not to exceed 18 months).. This allows surveillance testing not requiring shutdown conditions to be accomplished at power during the refueling interval and is consistent with the language of the Standard Technical Specifications for Westinghouse Pressurized Water Reactors.

The staff has reviewed the licensee's proposed changes and agrees that they are administrative in nature and serve to clarify or correct the intent of the technical specifications and their bases. The staff, therefore, finds them acceptable.

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 the amendments.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: October 6, 1983

Principal Contributors:

T. G. Colburn