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Dockets Nos. 50-266/301

Wisconsin Electric Power Company
Wisconsin Michigan Power Company
ATTN: Mr. Sol Burstein
Executive Vice President
231 West Michigan Street
Milwaukee, Wisconsin 53201

Gentlemen:

The Commission has issued the enclosed Amendments Nos. 8 and 10 to Facility Operating Licenses Nos. DPR-24 and DPR-27 for the Point Beach Nuclear Plant Units 1/2. The amendments also incorporate Changes Nos. 13 and 16 in the Technical Specifications in accordance with your applications dated May 5, 1973 and April 11, 1975.

The amendments ~~permit modification to~~ modify the Technical Specifications for clarification of protective instrumentation settings for reactor trip interlocks and correction of typographical errors and omissions.

Copies of the Safety Evaluation and Federal Register Notice are also enclosed.

Sincerely,

/S/

George Lear, Chief
Operating Reactors Branch #3
Division of Reactor Licensing

Enclosures:
1. Amendments Nos. 8 and 10
2. Safety Evaluation
3. Federal Register Notice
Ans 50-266 encl

cc: See next page

OFFICE ➤	ORB#3	ORB#3	OELD	ORB#3		
SURNAME ➤	CParrish/dg	JWetmore/dg		GLear		
DATE ➤	7/ /75	7/ /75	7/ /75	7/ /75		

Form AEC-318 (Rev. 9-53) AECM 0240

U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

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

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Dockets Nos. 50-266/301

AUG 7 1975

Am-8 } DPR-24
ch-13 }

Wisconsin Electric Power Company
Wisconsin Michigan Power Company
ATTN: Mr. Sol Burstein
Executive Vice President
231 West Michigan Street
Milwaukee, Wisconsin 53201

DO NOT REMOVE

Gentlemen:

The Commission has issued the enclosed Amendments Nos. 8 and 10 to Facility Operating Licenses Nos. DPR-24 and DPR-27 for the Point Beach Nuclear Plant Units 1/2. The amendments also incorporate Changes Nos. 13 and 16 in the Technical Specifications in accordance with your applications dated May 5, 1973 and April 11, 1975.

The amendments modify the Technical Specifications for clarification of protective instrumentation settings for reactor trip interlocks and correction of typographical errors and omissions.

Copies of the Safety Evaluation and Federal Register Notice are also enclosed.

Sincerely,

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Reactor Licensing

Enclosures:
1. Amendments Nos. 8 and 10
2. Safety Evaluation
3. Federal Register Notice

cc: See next page

Wisconsin Michigan and Wisconsin Electric Power Company

cc: w/enclosure

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

WISCONSIN ELECTRIC POWER COMPANY
WISCONSIN MICHIGAN POWER COMPANY

DOCKET NO. 50-266

POINT BEACH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 8
License No. DPR-24

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

   A. The applications for amendment by Wisconsin Electric Power Company and Wisconsin Michigan Power Company (the licensees) dated May 5, 1973 and April 11, 1975, comply 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 1;

   B. The facility will operate in conformity with the applications, 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; and

   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.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 3.B. of Facility 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, are hereby incorporated in the license. The licensees shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 13."

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

FOR THE NUCLEAR REGULATION COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Reactor Licensing

Attachment:
Change No. 13
Technical Specifications

Date of Issuance: AUG 7 1975

ATTACHMENT TO LICENSE AMENDMENT NO. 8

CHANGE NO. 13 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-24

DOCKET NO. 50-266

Replace page 15.2.3-4, page 15.2.3-7, page 15.3.5-3, Table 15.3.5-1 page 15.3.7-3 and page 15.3.11-1 with the attached revised pages.

No change has been made on pages 15.2.3-3a, 15.3.5-4, 15.3.7-4 and 15.3.11-2.

C. Other reactor trips

(1) High pressurizer water level - $\leq$ 95% of span

(2) Low-low steam generator water level - $\geq$5% of narrow range instrument span

(3) Steam-Feedwater Flow Mismatch Trip - $\leq 1.0 \times 10^6$ lb/hr

(4) Turbine Trip (Not a protection circuit)

(5) Safety Injection Signal

(6) Manual Trip

2. Protective instrumentation settings for reactor trip interlocks shall be as follows:

A. The "at power" reactor trips (low pressurizer pressure, high pressurizer level, and low reactor coolant flow for both loops) shall be unblocked when:

(1) Power range nuclear flux $\geq$ 9% ($\pm$1%) of rated power or, 13

(2) Turbine load $\geq$ 10% of full load turbine pressure.

B. The single loss of flow trip shall be unblocked when the power range nuclear flux $\geq$ 50% of rated power.

C. The power range high flux level low range trip, and intermediate range high flux level trip shall be unblocked 13 when power is $\leq$ 9% ($\pm$1%) of rated power.

D. The source range high flux reactor trip shall be unblocked when the intermediate range flux is $\leq 10^{-11}$ amperes.

the reactor coolant pump breaker opening as actuated by either high current, low supply voltage or low electrical frequency, or by a manual control switch. The significant feature of the breaker trip is the frequency set-point, 57.5 cps, which assures a trip signal before the pump inertia is reduced to an unacceptable value. The high pressurizer water level reactor trip protects the pressurizer safety valves against water relief. The specified set point allows adequate operating instrument error [2] and transient overshoot in level before the reactor trips.

The low-low steam generator water level reactor trip protects against loss of feedwater flow accidents. The specified set point assures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays for the auxiliary feedwater system.[9]

Numerous reactor trips are blocked at low power where they are not required for protection and would otherwise interfere with normal plant operations. The prescribed set point above which these trips are unblocked assures their availability in the power range where needed. Specifications 15.2.3.2.1(A) and 15.2.3.2.C have a ± 1% tolerance to allow for a 2% deadband of the P10 bistable which is used 13
to set the limit of both items.

Sustained operation with only one pump will not be permitted above 10% power. If a pump is lost while operating between 10% and 50% power, an orderly and immediate reduction in power level to below 10% is allowed. The power-to-flow ratio will be maintained equal to or less than unity, which ensures that the minimum DNB ratio increases at lower flow because the maximum enthalpy rise does not increase above the maximum enthalpy rise which occurs during full power and full flow operation.

References

(1)	FSAR 14.1.1	(4)	FSAR 14.3.1	(7)	FSAR 3.2.1
(2)	FSAR, page 14-3	(5)	FSAR 14.1.2	(8)	FSAR 14.1.9
(3)	FSAR 14.2.6	(6)	FSAR 7.2, 7.3	(9)	FSAR 14.1.11

Containment Spray

The Engineered Safety Features also initiate containment spray upon sensing a high containment pressure signal (Hi-Hi). The containment spray acts to reduce containment pressure in the event of a loss of coolant or steam line break accident inside the containment. The containment spray cools the containment directly and limits the release of fission products by absorbing iodine should it be released to the containment.

Containment spray is designed to be actuated at a higher containment pressure (approximately 50% of design containment pressure) than the SIS (10% of design). Since spurious actuation of containment spray is to be avoided, it is initiated only on coincidence of high containment pressure (Hi-Hi) sensed by both sets of two-out-of-three containment pressure signals provided for its actuation.

Containment Isolation

A containment isolation signal is initiated by any signal causing automatic initiation of safety injection or may be initiated manually. The containment isolation system provides the means of isolating the various pipes passing through the containment walls as required to prevent the release of radioactivity to the outside environment in the event of a loss-of-coolant accident.

Steam Line Isolation

Steam line isolation signals are initiated by the Engineered Safety Features closing the steam line stop valve of the affected line. In the event of a steam line break, this action prevents continuous, uncontrolled steam release from more than one steam generator by isolating the steam lines on high containment pressure (Hi-Hi) or high steam line high flow in coincidence with low T_{avg} and SIS or high steam flow in coincidence with SIS. Protection is afforded for breaks inside or outside the containment even when it is assumed that the steam line check valves do not function properly.

Setting Limits

1. The high containment pressure limit is set at about 10% of design containment pressure. Initiation of Safety Injection protects against loss of coolant[2] or steam line break[3] accidents as discussed in the safety analysis.

2. The hi-hi containment pressure limit is set at about 50% of design containment pressure for initiation of containment spray and at about 30% for initiation of steam line isolation. Initiation of Containment Spray and Steam Line Isolation protects against large loss of coolant[2] or steam line break accidents[3] as discussed in the safety analysis.

3. The pressurizer low pressure limit is set substantially below system operating pressure limits. However, it is sufficiently high to protect against a loss of coolant accident as shown in the safety analysis.[2] The pressurizer low level limit is set sufficiently high to protect against a loss of coolant accident as shown in the accident analysis.

4. The steam line low pressure signal is lead/lag compensated and its setpoint is set well above the pressure expected in the event of a large steam line break accident as shown in the safety analysis.[3]

5. The high steam line flow limit is set at approximately 20% of nominal full load flow at the no-load pressure and the high-high steam line flow limit is set at approximately 120% of nominal full load flow at the full load pressure in order to protect against large steam break accidents. The coincident low Tavg setting limit for steam line isolation initiation is set below its hot shutdown value. The safety analysis shows that these settings provide protection in the event of a large steam break.[3]

Instrument Operating Conditions

During plant operations, the complete instrumentation systems will normally be in service. Reactor safety is provided by the Reactor Protection System,

TABLE 15.3.5-1

ENGINEERED SAFETY FEATURES INITIATION INSTRUMENT SETTING LIMITS

NO.	FUNCTIONAL UNIT	CHANNEL	SETTING LIMIT
1	High Containment Pressure (Hi)	Safety Injection*	$\leq$ 6 psig
2	High Containment Pressure (Hi-Hi)	a. Containment Spray	$\leq$ 30 psig
		b. Steam Line Isolation of Both Lines	$\leq$ 20 psig
3	Pressurizer Low Pressure and Low Level	Safety Injection*	$\geq$ 1715 psig
			$\geq$ 5 per cent (of distance) between the instrument taps.
4	Low Steam Line Pressure	Safety Injection*	$\geq$ 500 psig 13
		Lead Time Constant	$\geq$ 12 seconds
		Lag Time Constant	$\leq$ 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 $\leq 0.66 \times 10^6$ lb/hr at 1005 psig
			$\geq$ 540°F
6	High-high Steam Flow in a Steam Line Coincident with Safety Injection	Steam Line Isolation of Affected Line	$\leq$ d/p corresponding to 4×10^6 lb/hr at 806 psig

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

d/p means differential pressure

b. If both 345/13.8 KV auxiliary transformers are out of service and only the gas turbine is operating, only one reactor will remain operating and it will be limited to 50% power. The second reactor will be placed in the hot shutdown condition.

c. If the 13.8/4.16 KV auxiliary transformers are reduced to only one, the reactor associated with the out of service transformer must be placed in the hot shutdown condition.

d. Either bus A03 or A04 may be out of service for a period not exceeding 7 days provided both diesel generators are operable and the associated diesel generator is operating and providing power to the engineered safeguard bus normally supplied by the out of service bus.

e. One diesel generator may be inoperable for a period not exceeding 7 days provided the other diesel generator is tested daily to ensure operability and the engineered safety features associated with this diesel generator shall be operable.

f. One battery may be inoperable for a period not exceeding 24 hours provided the other battery and two battery chargers remain operable with one charger carrying the DC load of the inoperable battery's DC supply system.

Basis

This two unit plant has four 345 KV transmission line interconnections. A 20 MW gas turbine generator and two 2850 KW diesel generators are installed at the plant. All of these energy sources will be utilized to provide depth and reliability of service to the Engineered Safeguards equipment through redundant station auxiliary power supply systems.

The electrical system equipment is arranged so that no single contingency can inactivate enough safeguards equipment to jeopardize the plant safety. The 480-volt equipment is arranged on 4 buses per unit. The 4160-volt equipment is supplied from 6 buses per unit.

Two separate outside sources can serve either unit's low voltage station auxiliary transformer. One is a direct feed from the unit's high voltage station auxiliary transformer and the second is from the other unit's high voltage station auxiliary transformer or the gas turbine via the 13,800 volt system tie bus H01.

Separation is maintained in the 4160-volt system to allow the plant auxiliary equipment to be arranged electrically so that redundant items receive their power from the two different buses. For example, the safety injection pumps are supplied from the 4160 volt buses 1-A05 and 1-A06 for Unit No. 1 and 2-A05 and 2-A06 for Unit No. 2; the six service water pumps are arranged on 480-volt buses as follows: two on bus 1-B03, one on bus 1-B04, one on bus 2-B03 and two on bus 2-B04; the four containment fans are divided between 480-volt buses 1-B03 and 1-B04 for Unit No. 1 and 2-B03 and 2-B04 for Unit No. 2 and so on. Redundant valves are supplied from motor control centers 1-B32 and 1-B42 for Unit No. 1 and 2-B32 and 2-B42 for Unit No. 2.

One battery charger shall be in service on each battery so that the batteries will always be at full charge in anticipation of a loss-of-ac power incident. This insures that adequate dc power will be available for starting the emergency generators and other emergency uses.

The emergency generator sets are General Motors Corporation, Electro-Motive Division, Model 999-20 Units rated at 2850 KW continuous, 0.8 power factor 900 RPM, 4160 volts 3 phase, 60 cycle and consume 205 gallons of fuel per hour. Thus the 11,000 gallon supply in the Emergency Fuel Tank provides sufficient fuel to operate one diesel at design load for more than 48 hours. In addition, it will be normal for Point Beach to keep one, or the equivalent of one, bulk storage tank full at all times (55,000 gal. which is equal to about 10 days' supply). They are each capable of providing 3050 kw for a 30 minute period. The gas turbine is capable of providing 20,000 kw.

15.3.11 MOVABLE IN-CORE INSTRUMENTATION

Applicability:

Applies to the operability of the movable detector instrumentation system.

Objective:

To specify functional requirements on the use of the in-core instrumentation systems for the recalibration of the excore axial off-set detection system.

Specification:

A. A minimum of 2 thimbles per quadrant and sufficient movable in-core detectors shall be operable during re-calibration of the excore axial off-set detection system.

B. Power shall be limited to 90% of rated power if the calibration requirements for excore axial off-set detection system, identified in Table 15.4.1-1, are not met.

Basis:

The Movable In-Core Instrumentation System (1) has four drives, four detectors, and 36 thimbles in the core. The A and B detectors can be routed to eighteen thimbles. The C and D detectors can be routed to twenty-seven 13
thimbles. Consequently, the full system has a great deal more capability than would be needed for the calibration of the ex-core detectors.

To calibrate the excore detectors channels, it is only necessary that the Movable In-Core System be used to determine the gross power distribution in the core as indicated by the power balance between the top and bottom halves of the core.

After the excore system is calibrated initially, recalibration is needed only infrequently to compensate for changes in the core, due for example to fuel depletion, and for changes in the detectors.

If the recalibration is not performed, the mandated power reduction assures safe operation of the reactor since it will compensate for an error of 10% in the excore protection system. Experience at Beznau (Switzerland) and Ginna has shown that drift due to changes in the core or instrument channels is very slight. Thus, the 10% reduction is considered to be very conservative.

Reference

(1) FSAR - Section 7.4

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENTS NOS 8 AND 10 TO LICENSES DPR-24/27

(CHANGES NOS. 13 AND 16 TO THE TECHNICAL SPECIFICATIONS)

WISCONSIN ELECTRIC POWER COMPANY
WISCONSIN MICHIGAN POWER COMPANY

POINT BEACH NUCLEAR PLANT, UNITS 1/2

DOCKETS NOS. 50-266/301

Introduction

By letters dated May 5, 1973 and April 11, 1975, Wisconsin Electric Power Company (WEPCO) requested changes to the Technical Specifications appended to Facility Licenses DPR-24 and 27. The proposed changes would (1) clarify protective instrumentation settings for reactor trip interlocks and (2) correct typographical errors and omissions.

Evaluation

(1) Protective Instrumentation Interlocks: The existing Technical Specification 15.2.3.2.A(1) specifies that the "at power" reactor trips (low pressurizer pressure, high pressurizer level, and low reactor coolant flow for both loops) shall be unblocked when power range nuclear flux is equal to or greater than 10% of rated power. Technical Specification 15.2.3.2.C currently specifies that the power range-high flux level-low range trip, and the intermediate range-high flux level trip shall be unblocked when power is equal to or less than 10% of rated power. Literal simultaneous compliance with both of these Technical Specifications is impossible, due to the 2% deadband of the P-10 bistable which is used to set the limit for both Specifications. The proposed changes to Technical Specifications 15.2.3.2.A(1) and 15.2.3.2.C would add a tolerance band of 2% to the setting of the P-10 bistable. The tolerance band would eliminate the conflict between these two Technical Specifications.

We recognize that the reactor protection system has tolerance bands associated with each of its various settings. We also recognize that these tolerance bands are an inherent characteristic of electro-mechanical systems, and that the design of the plant takes this into account. However, the NRC staff position is that the P-10 bistable setting of 10% of rated power should not be exceeded because the safety analysis for the plant assumes that the "at power"

AMERICAN REVOLUTION BICENTENNIAL 1776-1976 trips are unblocked for all power levels greater than 10%. Since the proposed change could result in a condition wherein the "at power" trips are unblocked for power levels greater than 12% (i.e. 10% setting plus a tolerance of +2%), the range of power levels over which the "at power" trips are in effect would be reduced. Although this condition would have a very small effect, the NRC staff considers it to be prudent to require that the 10% of rated power setting of the P-10 bistable not be exceeded.

Therefore, to accommodate this requirement for a maximum value of the trip setting as well as the 2% deadband associated with the P-10 bistable, the NRC staff has determined that a bistable trip setting of 9% with a tolerance band of ± 1% would be appropriate. Discussion with the licensee has resulted in his concurrence with this modification of the licensee's request. The proposed changes, as modified by the NRC staff and concurred in by the licensee, would eliminate the conflict between Technical Specifications 15.2.3.2.A(1) and 15.2.3.2.C and would not alter the effect of the setpoints in the Technical Specifications; therefore, they are acceptable.

(2) Typographical Errors: The proposed changes to correct typographical errors and omissions on Technical Specification pages 15.3.5-3, 15.3.7-3, 15.3.11-1 and Table 15.3.5-1 are editorial only and have no safety significance; therefore, they are acceptable.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the changes 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 changes 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

DATED: AUG 7 1975

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS NOS. 50-266/301

WISCONSIN ELECTRIC POWER COMPANY
WISCONSIN MICHIGAN POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 8 and 10 to Facility Operating Licenses Nos. DPR-24 and DRP-27 issued to Wisconsin Electric Power Company and Wisconsin Michigan Power Company which revised Technical Specifications for operation of the Point Beach Nuclear Plant, Units 1/2, located in Manitowac County, Wisconsin. The amendments are effective as of their date of issuance.

The amendments modify (1) clarification of protective instrumentation settings for reactor trip interlocks and (2) correction of typographical errors and omissions.

The applications for the amendments comply 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 the amendments is not required since the amendments do not involve significant hazards considerations.

For further details with respect to this action, see (1) the applications for amendments dated May 5, 1973, and April 11, 1975, (2) Amendments Nos. 8 and 10 to Licenses Nos. DPR-24 and DPR-27, with Changes Nos. 13 and 16 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. and at the Document Department, University of Wisconsin - Stevens Point Library Stevens Point, Wisconsin 54481.

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 Reactor Licensing.

DatedDated at Bethesda, Maryland, this 7th day of August, 1975

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Reactor Licensing