Docket File
DCS- MS-016

SEP 3 0 1981

Docket Nos. 50-266
and 50-301

RECEIVED
OCT 8 1981
U.S. NUCLEAR REGULATORY COMMISSION

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

Dear Mr. Burstein:

The Commission has issued the enclosed Amendment No. 55 to Facility Operating License No. DPR-24 and Amendment No. 60 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 completing the response to your application transmitted by letter dated February 4, 1981 which has been answered in part by Amendment Nos. 52 and 58 issued August 20, 1981.

These amendments address TMI-2 short term Lessons Learned Category A Technical Specifications changes requested by the NRC staff in our letter dated July 2, 1980. Specifically they modify the following sections of the Point Beach Technical Specifications:

1. Specifications have been added to Section 15.3.1, Reactor Coolant System, to describe limiting conditions for operation (LCO) concerning the pressurizer, the pressurizer power operated relief valves (PORVs) and the PORV block valves.

2. Specification 15.3.5.D and Table 15.3.5-5 have been added to Section 15.3.5, "Instrumentation Systems". These additions describe the LCOs for the accident monitoring instrumentation channels listed in the new table. Two new ESF initiation instrument settings have been added to Table 15.3.5-1 to list the setpoints for auxiliary feedwater initiation. Item 16 from Table 15.3.5-2 has been moved to the new Table 15.3.5-5. Table 15.3.5-3, Emergency Cooling, has also been expanded to include minimum operable channel requirements for the auxiliary feedwater initiation circuitry.

3. Table 15.4.1-1 has been changed to include minimum check, calibration, and testing requirements, for the instrumentation added to the LCOs discussed above.

4. Item 23 has been added to Table 15.4.1-2 to prescribe necessary surveillance of the PORV block valves. A surveillance requirement has also been added as Item 24 to evaluate the integrity of those systems outside containment used for post design basis accident recovery.

OFFICE							
SURNAME	8110200241 810930 PDR ADOCK 05000266 P PDR						

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

By our letter of September 13, 1979, we issued new requirements to all operating nuclear power plants established as a result of our review of the TMI-2 accident. Certain of these requirements, designated Lessons Learned Category "A" requirements, were to have been completed prior to any operation subsequent to January 1, 1980. Our evaluation and acceptance of your actions to comply with these Category "A" items was contained in our letter to you of April 9, 1980.

In order to provide reasonable assurance that operating reactor facilities are maintained within the limits determined acceptable following the implementation of these TMI-2 Lessons Learned Category "A" items, we requested that licensees include certain of these items in the operating license as license conditions and additional Technical Specifications. These requirements were contained in our letter to you of July 2, 1980, which contained model Technical Specifications that we had determined to be acceptable.

Your request for amendment dated February 4, 1981 is responsive to our request. Certain changes have been made to conform to our requirements. These have been discussed with and accepted by members of your staff. Our acceptance of these new requirements is documented in our evaluation letter to you of April 9, 1980 and our letter to you of July 2, 1980, which, together with this letter constitute our Safety Evaluation of this matter.

Certain portions of your amendment request regarding administrative requirements of the Duty Technical Advisor have been previously issued with amendments 52 and 58 for Point Beach Units 1 and 2 respectively on August 20, 1981. Certain other technical specification change requests, identified by you in your February 4 letter as addressing portions of our model technical specifications, you assert have been previously submitted for NRC staff review. These include LCOs for containment isolation valves and auxiliary feedwater pump operability. The staff acknowledges your submittal on these items. As they are still under review, they have not been included in these amendments.

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.

OFFICE ▶							
SURNAME ▶							
DATE ▶							

We have concluded 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.

A copy of the Notice of Issuance is also enclosed.

Sincerely,

Original signed by:

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

Enclosures:
1. Amendment No. 55 to DPR-24
2. Amendment No. 60 to DPR-27
3. Notice of Issuance

cc: w/enclosures
See next page

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9-29-81 no legal objection to amend or notice

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T. Colburn, ORB#3

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TO:

Sol. Burstein

ORIG.: X | CC: X | OTHER:

ACTION NECESSARY	☐	CONCURRENCE	☒ XX	DATE ANSWERED:
NO ACTION NECESSARY	☐	COMMENT	☐	BY:

CLASSIF.: U

POST OFFICE

REG. NO.:

FILE CODE: Point Beach

DESCRIPTION: (Must Be Unclassified)

Amdt to ol re Changes to T.S. Address TMI-2 Short term lesson learned Category A T.S. ~~###~~ changes

ENCLOSURES:

U R G E N T

for legal review of amdt and federal register ~~####~~ notice only

REMARKS: ELD Due Date 10-1-81

REFERRED TO	DATE	RECEIVED BY	DATE
Scinto	9/29		
Barth	9/29	has seen	9/29
~~Treby~~	9/29	has seen	9/29
Scinto			

U. S. NUCLEAR REGULATORY COMMISSION

MAIL CONTROL FORM

FORM NRC-326S (5-76)

Wisconsin Electric Power Company

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cc w/enclosure(s) and incoming
dtd: 2/4/81

Chairman
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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. 55
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 February 4, 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.

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. 55 , 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

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 30, 1981

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. 60
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 February 4, 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.

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. 60 , 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

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 30, 1981

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 55 TO FACILITY OPERATING LICENSE NO. DPR-24

AMENDMENT NO. 60 TO FACILITY OPERATING LICENSE NO. DPR-27

DOCKET NOS. 50-266 AND 50-301

Revise Appendix A as follows:

Remove Pages	Insert Pages
15.3.1.2	15.3.1.2
15.3.1.3	15.3.1.3
-	15.3.1.3A
15.3.5-1	15.3.5-1
15.3.5-5	15.3.5-5
-	15.3.5-6
Table 15.3.5-1	Table 15.3.5-1
Table 15.3.5-2 (Continued)	Table 15.3.5-2 (Continued)
Table 15.3.5-3	Table 15.3.5-3
Table 15.3.5-3 (Continued)	Table 15.3.5-3 (Continued)
-	Table 15.3.5-5
Table 15.4.1-1 (Continued)	Table 15.4.1-1 (Continued)
Table 15.4.1-1 (Continued)	Table 15.4.1-1 (Continued)
Table 15.4.1-2 (Continued)	Table 15.4.1-2 (Continued)

3. Pressurizer Safety Valves

    a. At least one pressurizer safety valve shall be operable whenever the reactor head is on the vessel.

    b. Both pressurizer safety valves shall be operable whenever the reactor is critical.

4. Pressurizer Power Operated Relief Valves (PORV) and PORV Block Valves.

    a. Two PORVs and their associated block valves shall be operable.

        1. If a PORV is inoperable, the PORV shall be restored to an operable condition within one hour or the associated block valve shall be closed.

        2. If a PORV block valve is inoperable, the block valve shall be restored to an operable condition within one hour or the block valve shall be closed with power removed from the block valve; otherwise, the unit shall be in the hot shutdown condition within the next six hours.

5. The pressurizer shall be operable with at least 100 KW of pressurizer heaters available and a water level greater than 10% and less than 95% during steady state power operation. At least one bank of pressurizer heaters shall be supplied by an emergency bus power supply.

Basis:

When the boron concentration of the reactor coolant system is to be reduced the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the primary system volume in approximately one half hour. The pressurizer is of little concern because of the low pressurizer volume and because pressurizer boron concentration normally will be higher than that of the rest of the reactor coolant.

Part 1 of the specification requires that a sufficient number of reactor coolant pumps be operating to provide core cooling in the event that a loss of flow occurs. The flow provided in each case will keep DNBR well above 1.30 as discussed in FFDSAR Section 14.1.9. Therefore, cladding damage and release of fission products to the reactor coolant will not occur. Heat transfer analyses (1) show that reactor heat equivalent to 10% of rated power can be removed with natural circulation only; hence, the specified upper limit of 1% rated power without operating pumps provides a substantial safety factor.

Each of the pressurizer safety valves is designed to relieve 288,000 lbs. per hr. of saturated steam at setpoint. Below 350°F and 350 psig in the reactor coolant system, the residual heat removal system can remove decay heat and thereby control system temperature and pressure. 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 over-pressurization. Part 1 c(2) permits an orderly reduction in power if a reactor coolant pump is lost during operation between 10% and 50% of rated power. Above 50% power, an automatic reactor trip will occur if either pump is lost. The power-to-flow ratib 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)

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 manner, reactor coolant system pressure increases below the setting of the pressurizer safety valves. These PORVs have remotely operated block valves to provide a positive shutoff capability should a PORV become inoperable.

The requirement that 100 KW of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain pressure control and natural circulation at hot standby.

Reference

(1) FSAR Section 14.1.6
(2) FSAR Section 7.2.3

15.3.5 INSTRUMENTATION SYSTEM

Operational Safety Instrumentation

Applicability:

Applies to plant instrumentation systems.

Objectives:

To provide for automatic initiation of the Engineered Safety Features in the event that principal process variable limits are exceeded, and to delineate the conditions of the plant instrumentation and safety circuits necessary to ensure reactor safety.

Specification:

A. The Engineered Safety Features initiation instrumentation setting limits shall be as stated in Table 15.3.5-1.

B. For on-line testing or in the event of a sub-system instrumentation channel failure, plant operation at rated power shall be permitted to continue in accordance with Tables 15.3.5-2 through 15.3.5-4.

C. In the event the number of channels of a particular sub-system in service falls below the limits given in the column entitled Minimum Operable Channels, or Minimum Degree of Redundancy cannot be achieved, operation shall be limited according to the requirement shown in Tables 15.3.5-2 through 15.3.5-4, Operator Action when minimum operable channels unavailable.

D. The accident monitoring instrumentation channels in Table 15.3.5-5 shall be operable. In the event the number of channels in a particular sub-system falls below the minimum number of operable channels given in Column 2, operation and subsequent operator action shall be in accordance with Column 3.

Basis:

Instrumentation has been provided to sense accident conditions and to initiate operation of the Engineered Safety Features(1).

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 necesssary 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.

If the process computer, which provides the reactor coolant system subcooling margin monitor, becomes inoperable, subcooling will be monitored by means of a backup plotter method or manually using control board instrumentation and a saturation curve.

Reference

(1) FSAR - Section 7.5
(2) FSAR - Section 14.3
(3) FSAR - Section 14.2.5

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 b. Steam Line Isolation of Both Lines	$\leq$ 30 psig $\leq$ 20 psig
3	Pressurizer Low Pressure	Safety Injection*	$\geq$ 1715 psig
4	Low Steam Line Pressure	Safety Injection* Lead Time Constant Lag Time Constant	$\geq$ 500 psig $\geq$ 12 seconds $\leq$ 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
7	Low-low Steam Generator Water Level	Auxiliary Feedwater Initiation	$\geq$ 5% of narrow range instrument
8	Undervoltage on 4 KV Busses	Auxiliary Feedwater Initiation	$\geq$ 75% of normal voltage

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

d/p means differential pressure

Unit 1-Amendment No. 33, 47, 55
Unit 2-Amendment No. 43, 32, 60

NO.	FUNCTIONAL UNIT	CHANNEL	SETTING LIMIT
9.	Degraded Voltage (4.16 KV)	Disconnection of affected bus from offsite power	$\geq$ 3675 volts $\pm$ 2% Time delay: 13.6 sec $\pm$ 5% at 0-95% of voltage settin
10.	Loss of Voltage		
	a. 4.16KV	Disconnection of affected bus from offsite power Start Diesel	a. 2450 volts $\pm$ 3% Time delay: 0.3 sec. $\pm$ 5% at 0 volts 1.2 sec. $\pm$ 5% at 90% voltage setting
	b. 480 V	Load shedding	b. 256 volts $\pm$ 3% Time delay: 0.75 sec. $\pm$5% at 0 volts 3.5 sec. $\pm$ 15% at 90% of voltage setting

Unit 1-Amendment No. 55
Unit 2-Amendment No. 60

TABLE 15.3.5-2 (Cont'd)

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MIN. OPERABLE CHANNELS	4 MINIMUM DEGREE OF REDUNDANCY	5 PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 4 CANNOT BE MET
11.	Turbine Trip	3	2	2	1		Maintain <50% of rated power
12.	Steam Flow - Feed Water Flow mismatch	2/loop	1/loop	1/loop	1/loop		Maintain hot shutdown
13.	Lo Lo Steam Generator Water Level	3/loop	2/loop	2/loop	1/loop		Maintain hot shutdown
14.	Undervoltage 4 KV Bus	2/bus	1/bus (both buses)	1/bus	--		Maintain hot shutdown
15.	Underfrequency 4 KV Bus	2/bus	1/bus (both buses)	1/bus	--		Maintain hot shutdown

NOTE 1: When block condition exists, maintain normal operation.

F.P. = Full Power

* Not Applicable

** One additional channel may be taken out of service for zero power physics testing.

TABLE 15.3.5-3

EMERGENCY COOLING

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MIN. OPERABLE CHANNELS	4 MIN. DEGREE OF REDUNDANCE	5 PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 4 CANNOT BE MET
1.	SAFETY INJECTION						
a.	Manual	2	1	1	1		Hot Shutdown***
b.	High Containment Pressure	3	2	2	1		Hot Shutdown***
c.	Steam Generator Low Steam Pressure/Loop	3	2	2	1	Primary Pressure is Less than 1800 psig	Hot Shutdown***
d.	Pressurizer Low Pressure	3	2	2	1	Primary Pressure is Less than 1800 psig	Hot Shutdown***
2.	CONTAINMENT SPRAY						
a.	Manual	2	2	2	--**		Hot Shutdown***
b.	Hi-Hi Containment Pressure (Containment Spray)	2 sets of 3	2 of 3 in each set	2 per set	1/set		Hot Shutdown***
3.	AUXILIARY FEEDWATER						
a.	Low-low Steam Generator Water						
	i. Start Motor Driven Pump	3/steam gen.	2/either gen.	2/steam gen.	1		Hot Shutdown***
	ii. Start Turbine Driven Pump	3/steam gen.	2/both gens.	2/steam gen.	1		Hot Shutdown***
b.	Trip of both Main Feedpumps starts motor driven pumps	2/pump	1/pump	1/pump	1		Hot Shutdown***
c.	Undervoltage on 4KV Busses starts Turbine driven pump	2/bus	1/bus	1/bus	--		Hot Shutdown***

Unit 1-Amendment No. ~~38~~, ~~47~~, 55
Unit 2-Amendment No. ~~30~~, ~~37~~. 60

TABLE 15.3.5-3

EMERGENCY COOLING

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MIN. OPERABLE CHANNELS	4 MIN. DEGREE OF REDUNDANCY	5 PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 4 CANNOT BE MET
3.	AUXILIARY FEEDWATER(Continued)						
d.	Safety Injection Signal Starts Motor Driven Pumps	S.I. INITIATING FUNCTIONS AND REQUIREMENT AS IN 1. ABOVE					
4.	SAFETY RELATED ELECTRICAL BUSES						
a.	Degraded Voltage	(4.16 KV)	3/bus	2/bus	2/bus	1/bus	****
b.	Loss of Voltage	(4.16 KV)	2/bus	1/bus	1/bus	1	****
c.	Loss of Voltage	(480 V)	3/bus	2/bus	2/bus	1	Hot Shutdown ***

** - Must activate 2 switches simultaneously.

*** - If minimum conditions are not met within 24 hours after reaching hot shutdown, the unit shall be in cold shutdown within 48 hours of the event causing the unit shutdown.

**** - Normal operation provided both diesel generators are available, and the associated diesel generator is operating and providing power to the affected safeguards bus. If minimum conditions are not met within 7 days, the affected unit shall be placed in hot shutdown.

Unit 1-Amendment No. ~~28~~, ~~47~~, 55
Unit 2-Amendment No. ~~30~~, ~~52~~, 60

TABLE 15.3.5-5

INSTRUMENT OPERATING CONDITIONS FOR INDICATION

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 MINIMUM OPERABLE CHANNEL	3 OPERATOR ACTION IF CONDITIONS OF COLUMN 2 CANNOT BE MET
1.	PORV Position Indicator	1/Valve	1/Valve	If the operability of the PORV position indicator cannot be restored within 48 hours, shut the associated PORV Block Valve.
2.	PORV Block Valve Position Indicator	1/Valve	1/Valve	If the operability of the PORV Block Valve Position Indicator cannot be restored within 48 hours, shut and verify the Block Valve shut by direct observation or declare the Block Valve inoperable.
3.	Safety Valve Position Indicator	1/Valve	1/Valve	If the operability of the Safety Valve Position Indicator cannot be restored within seven days, be in at least Hot Shutdown within the next 12 hours.
4.	Reactor Coolant System Subcooling	1	1	If the operability of a subcooling monitor cannot be restored or a backup monitor made functional within 48 hours, be in at least Hot Shutdown within the next 12 hours.
5.	Auxiliary Feedwater Flow Rate*	1	1	If the operability of the auxiliary feedwater flow rate indicator cannot be restored within 48 hours, be in hot shutdown within 12 hours.
6.	Control Rod Misalignment as Monitored by On-Line Computer	1	1	Log individual rod positions once/hr., after a load change >10% or after >30 inches of control rod moti

*Applies to presently installed combination of auxiliary feedwater pump discharge flow indicators and auxiliary feedwater flow to steam generator indicators.

TABLE 15.4.1-1 (CONTINUED)

	Channel Description	Check	Calibrate	Test	Remarks
24.	Containment Pressure	S	R	M**	Narrow range containment pressure (-3.0, +3 psig excluded)
25.	Steam Generator Pressure	S***	R	M***	
26.	Turbine First Stage Pressure	S**	R	M**	
27.	Emergency Plan Radiation Instruments	M	R	M	
28.	Environmental Monitors	M	N.A.	N.A.	
29.	Overpressure Mitigating	S	R	****	
30.	PORV Position Indicator	S	R	R	
31.	PORV Block Valve Position Indicator	Q	R	N.A.	
32.	Safety Valve Position Indicator	M	R	N.A.	
33.	PORV Operability	N.A.	R	M	Performance of a channel functional test but excluding valve operation.
34.	Subcooling Margin Monitor	M	R	N.A.	
35.	Undervoltage on 4KV Bus	N.A.	R	M**	For Auxiliary Feedwater Pump Initiation
36.	Auxiliary Feedwater Flow Rate	See Remarks	R	N.A.	Flow Rate indication will be checked at each unit startup and shutdown
37.	Degraded 4.16 KV Voltage	S	R	M**	
38.	a. Loss of Voltage (4.16 KV)	S	R	M**	
	b. Loss of Voltage (4.16 KV)	S	R	M**	
39.	4160 V. Frequency	N.A.	R	N.A.	

TABLE 15.4.1-1 (CONTINUED)

S - Each Shift
D - Daily
W - Weekly
Q - Quarterly
B/W - Biweekly

M - Monthly
P - Prior to each startup if not done previous week.
R - Each Refueling Shutdown (But not to exceed 20 months).
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.

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

(1) A radiochemical analysis for this purpose shall consist of a quantative 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.

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. 55 to Facility Operating License No. DPR-24, and Amendment No. 60 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 address TMI-2 short term Lessons Learned Category A Technical Specification changes requested by the NRC staff. They incorporate new requirements in the form of limiting conditions for operation and surveillance requirements for instruments and equipment.

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.

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 February 4, 1981, (2) Amendment Nos. 55 and 60 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 30th day of September, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

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