September 29, 1994

Docket Nos. 50-266
and 50-301

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Mr. Robert E. Link, Vice President
Nuclear Power Department
Wisconsin Electric Power Company
231 West Michigan Street, Room P379
Milwaukee, Wisconsin 53201

Dear Mr. Link:

SUBJECT: AMENDMENT NOS. 155 AND 159 TO FACILITY OPERATING LICENSE NOS. DPR-24 AND DPR-27 (TACS M77369, M77370, M77442 AND M77443)

The Commission has issued the enclosed Amendment Nos. 155 and 159 to Facility Operating License Nos. DPR-24 and DPR-27 for the Point Beach Nuclear Plant, Unit Nos. 1 and 2. The amendments revise the Technical Specifications in response to your application dated May 30, 1991, and supplemented by letters dated May 7, 1993, and April 28, 1994.

These amendments revise Technical Specifications 15.3.1.A.5 and 15.3.15, and Tables 15.4.1-1 and 15.4.1-2. The changes specify more stringent limiting conditions for operation and surveillance requirements for pressurizer power-operated relief valves and block valves. These changes were proposed to conform to the NRC's plan for resolution of Generic Issue 70, "Power-Operated Relief Valve and Block Valve Reliability," and Generic Issue 94, "Additional Low-Temperature Overpressure Protection for Light Water Reactors," as conveyed in Generic Letter 90-06. Other related changes have also been made.

A copy of the Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,
Original signed by Allen G. Hansen
Allen G. Hansen, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 155 to DPR-24
2. Amendment No. 159 to DPR-27
3. Safety Evaluation

cc w/enclosures:
See next page

PD3-3:LA	PD3-3:PM	SRXB:BC	EMEB:BC	PD3-3:PD	OGC
MRushbrook	AHansen	RJones	RWessman	JHannon	C. Marco
/94	6/13/94	8/ /94	6/22/94	8/23/94	9/21/94

Document Name: G:PTBEACH\PTB77369.AMD

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Mr. Robert E. Link
Wisconsin Electric Power Company

Point Beach Nuclear Plant
Unit Nos. 1 and 2

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-266

POINT BEACH NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 155
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 30, 1991, as supplemented by letters dated May 7, 1993, and April 28, 1994, 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 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;

   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. 155 , 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 45 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Allen G. Hansen, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of issuance: September 30, 1994

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-301

POINT BEACH NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 159
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 30, 1991, as supplemented by letters dated May 7, 1993, and April 28, 1994, 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 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;

   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. 159 , 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 45 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Allen G. Hansen, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of issuance: September 30, 1994

ATTACHMENT TO LICENSE AMENDMENT NOS. 155 AND 159

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
15.3.1-3	15.3.1-3
15.3.1-3a	15.3.1-3a
15.3.1-3b	15.3.1-3b
15.3.1-3c	15.3.1-3c
15.3.1-3d	15.3.1-3d
- - - - -	15.3.1-3e
15.3.15-1	15.3.15-1
15.3.15-2	15.3.15-2
15.3.15-3	15.3.15-3
Table 15.4.1-1 (page 3 of 5)	Table 15.4.1-1 (page 3 of 5)
Table 15.4.1-1 (page 4 of 5)	Table 15.4.1-1 (page 4 of 5)
Table 15.4.1-2 (page 2 of 4)	Table 15.4.1-2 (page 2 of 4)
Table 15.4.1-2 (page 3 of 4)	Table 15.4.1-2 (page 3 of 4)
Table 15.4.1-2 (page 4 of 4)	Table 15.4.1-2 (page 4 of 4)

5. Pressurizer Power-Operated Relief Valves (PORV) and PORV Block Valves

If a unit is placed in the HOT SHUTDOWN condition in accordance with the requirements of Specifications a(1) through a(5) below, then the reactor coolant system temperature should be maintained greater than the minimum pressurization temperature for the inservice pressure test as defined in Figure 15.3.1-1. If cooldown to less than this temperature is required in order to take action to restore the inoperable component(s) to service, then the requirements of Specification 15.3.15 apply.

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

   (1) If one or both PORVs are INOPERABLE due to seat leakage in excess of that allowed in Specifications 15.3.1.D, within one hour either restore the PORVs to an operable status or close the associated block valves(s). If these conditions cannot be met, place the unit in a HOT SHUTDOWN condition within the next six hours.

   (2) If one PORV is INOPERABLE due to causes other than excessive seat leakage, within one hour either restore the PORV to OPERABLE status or close its associated block valve and remove power from the block valve. If the PORV cannot be restored to operable status within 72 hours, place the unit in a HOT SHUTDOWN condition within the next six hours.

   (3) If both PORVs are INOPERABLE due to causes other than excessive seat leakage, within one hour restore at least one PORV to OPERABLE status. If this condition cannot be met, close the associated block valves, remove power from the block valves and place the unit in a HOT SHUTDOWN condition within the next six hours.

   (4) If one block valve is inoperable, within one hour either restore the block valve to OPERABLE status or place the associated PORV in manual control. Restore the block valve to OPERABLE status within 72 hours. If these conditions cannot be met, place the unit in a HOT SHUTDOWN condition within the next six hours.

(5) If both block valves are inoperable, restore the block valves to OPERABLE status within one hour or place the associated PORVs in manual control. Restore at least one block valve to OPERABLE status within the next hour. If these conditions cannot be met, then place the unit in a HOT SHUTDOWN condition within the next six hours.

6. 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.

7. Reactor Coolant Gas Vent System

These Specifications are not applicable during cold or refueling shutdown conditions:

a. At least one Reactor Coolant Gas Vent System vent path to the pressurizer relief tank (PRT) or containment atmosphere shall be operable from each of the following locations:

(1) Reactor vessel head

(2) Pressurizer

Each vent path from these locations to the common header includes two closed valves in parallel powered from emergency buses. The common header vents to the PRT and the containment atmosphere each contain a closed valve powered from an emergency bus which provides series isolation.

b. When unable to vent from the common header to the PRT or the containment atmosphere, reactor startup and/or power operations may continue provided that the series isolation valve in the inoperable vent path is maintained closed with power removed from the valve actuator.

c. If a vent path from the reactor vessel head or the pressurizer to the common header becomes inoperable, reactor startup and/or power operations may continue provided that the paralleled isolation valves in the inoperable vent path from that location to the common header are maintained closed with power removed from the valve actuator. This does not necessitate removing power from the PRT or containment atmosphere isolation valves. The inoperable vent path shall be restored to operable status within thirty days, or the reactor shall be placed in hot shutdown within six hours and in cold shutdown within the following thirty hours.

d. If the vent paths from both the reactor vessel head and the pressurizer to the common header are inoperable or the vent paths from the common header to both the PRT and the containment atmosphere are inoperable, then maintain all the inoperable vent path valves closed with power removed from the valve actuators of all the valves in the inoperable vent paths. Restore at least one of the vent paths from the reactor vessel head or pressurizer to the containment atmosphere or the PRT to operable status within 72 hours or be in hot shutdown within six hours and in cold shutdown within the following thirty hours.

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 lower pressurizer volume and because pressurizer boron concentration normally will be higher than that of the rest of the reactor coolant.

Specification 15.3.1.A.1 requires that at least one reactor coolant pump must be operating whenever the average reactor coolant temperature is above 350°F unless the listed restrictions are established. This is required so that the FSAR zero power transients (rod withdrawal from subcritical and rod ejection) are addressed from conservative conditions. With the reactor subcritical, with required shutdown margin, and with the trip breakers open, a single rod ejection will not result in criticality being reached. With the reactor subcritical and the average reactor coolant temperature above 350°F, a single reactor coolant pump provides sufficient decay heat removal capability. Heat transfer analyses[1] show that reactor heat equivalent to 3.5% of the rated power can be removed with natural circulation only.

Items 15.3.1.A.1.a.(2) permits an orderly reduction in power if a reactor coolant pump is lost during operation between 3.5% and 50% of rated power.

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.

Additionally, with reactor coolant temperature between 350°F and 140°F, all operating decay heat removal pumps (either reactor coolant pumps or residual heat removal pumps) are allowed to be deenergized for a short time (1 hour) with the stipulation that boron dilution activities are not allowed and that core outlet temperature remain 10°F below saturation.

The operation of one reactor coolant pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the reactor coolant system. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

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, provide 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 most recent associated channel test, as specified in Table 15.4.1-1. is acceptable. Additionally, the PORV must have the capability of operating manually to relieve reactor coolant system pressure increases.

A block valve is defined as OPERABLE if the valve can operate manually and if it can control identified PORV leakage.

When a PORV is INOPERABLE due to excessive seat leakage, the block valve is shut with power maintained to the block valve so that the block valve(s) is readily available and may be used to allow the PORV to control reactor pressure. Excessive primary system leakage is defined in specification 15.3.1.D. The block valve may remain shut to isolate the leaking PORV for a limited period of time not to exceed the next refueling shutdown. When a PORV is INOPERABLE for reasons other than excessive seat leakage, the block valve is shut with power removed; this precludes any inadvertent opening of the block valve.

When a block valve is INOPERABLE, the associated PORV is placed in manual control; this precludes the undesired automatic opening of the PORV.

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 shutdown.

The requirement to have a reactor coolant system gas vent operable from the reactor vessel or the pressurizer steam space assures that non-condensible gases can be released from the Reactor Coolant System if necessary. The Reactor Coolant Gas Vent System (RCGVS) provides an orificed vent path from the pressurizer steam space and an orificed vent path from the reactor vessel. Both vent paths include two parallel solenoid-operated isolation valves which are powered from emergency buses and vent to a common header. From the common header, gases may be vented via separate lines, each with a single solenoid operated isolation valve powered from the emergency bus to the pressurizer relief tank or containment atmosphere. The orifice in these vent lines restricts leakage so that, in the event of a pipe break or isolation valve failure, makeup water for the leakage can be provided by a single coolant charging pump. If a RCGVS vent path from either the pressurizer or reactor vessel head is inoperable, Specification 15.3.1.A.7.c requires the remotely operable valves in that inoperable path to be shut with power removed. If a vent path from the common header to the pressurizer relief tank or containment atmosphere is inoperable, the isolation valve in that path must be shut but reactor operations may continue. If both vent paths to or both vent paths from the common header are inoperable, the RCGVS is inoperable and the steps in specification 15.3.1.A.7.d must be taken.

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[1] FSAR Section 14.1.11.

[2] FSAR Section 7.2.3.

15.3.15 OVERPRESSURE MITIGATING SYSTEM OPERATIONS

Applicability

Applies to operability of the overpressure mitigating system when the reactor coolant system temperature is less than the minimum temperature for the inservice pressure test.

Objective

To specify functional requirements and limiting conditions for operation on the use of the pressurizer power operated relief valves when used as part of the overpressure mitigating system and to specify further limiting conditions for operation when the reactor coolant system is operated without a pressure absorbing volume in the pressurizer.

Specification

A. System Operability

1. Except as specified in 15.3.15.A.2 below, the overpressurization mitigating system shall be operable whenever the reactor coolant system is not open to the atmosphere and the temperature is less than the minimum pressurization temperature for the inservice pressure test, as specified in Figure 15.3.1-1. Operability requirements are:
   a. Both pressurizer power operated relief valves operable at a setpoint of ≤425 psig.
   b. Both power operated relief valve block valves are open.
2. The requirements of 15.3.15.A.1 may be modified as specified below:
   a. With one PORV inoperable while reactor coolant system temperature is >200°F but less than the minimum pressurization temperature for the inservice pressure test, either restore the inoperable PORV to operable status within 7 days, or depressurize and vent reactor coolant system within the next 8 hours.
   b. With one PORV inoperable while reactor coolant system temperature is ≤200°F, either restore the inoperable PORV to operable status within 24 hours, or depressurize and vent the reactor coolant system within a total of 32 hours.

c. With both power operated relief valves inoperable while the reactor coolant system temperature is less than the minimum pressurization temperature for the inservice pressure test, the reactor coolant system must be depressurized and vented within 8 hours.

3. If the reactor coolant system is vented per Specification 15.3.15.A.2.a, b, or c, the pathway must be verified at least once every 31 days when it is provided by a non-isolable pathway or by a valve(s) that is locked, sealed, or otherwise secured in the open position; otherwise, verify the pathway every 12 hours.

B. Additional Limitations

1. When the reactor coolant system is not open to the atmosphere and the temperature of one or both reactor coolant system cold legs is $\leq 275^{\circ}F$, no more than one high pressure safety injection pump shall be operable. The second high pressure safety injection pump shall be demonstrated inoperable whenever the temperature of one or both reactor coolant system cold legs is $\leq 275^{\circ}F$ by verifying that the motor circuit breakers have been removed from their electrical power supply circuits or by verifying that the discharge valves from the high pressure safety injection pumps to the reactor coolant system are shut and that power is removed from their operators.

2. A reactor coolant pump shall not be started when the reactor coolant system temperature is less than the minimum temperature for the inservice pressure test unless:

   a. There is a pressure absorbing volume in the pressurizer or in the steam generator tubes or

   b. The secondary water temperature of each steam generator is less than $50^{\circ}F$ above the temperature of the reactor coolant system.

<u>Basis</u>

The Overpressurization Mitigating System consists of a diverse means of relieving pressure during periods of water solid operation and when the system temperature is below the value permitted to perform the primary system leak test. This method of water relief utilizes the pressurizer power operated relief valves (PORV's). The PORV's are made operational for low pressure relief by utilizing a dual setpoint where the low pressure circuit is energized and de-energized by the operator with a keylock switch depending on plant conditions. The logic required for the low pressure setpoint is in addition to the existing PORV actuation logic and will not interfere with existing automatic or manual actuation of the PORV's. The OPERABILITY of the PORVs is determined on the basis of their being capable of automatically mitigating an overpressure event during low temperature operation.

During plant cooldown prior to reducing reactor coolant system temperature below the minimum temperature allowable for the inservice pressure test, the operator under administrative procedures shall place the keylock switch in the "Low Pressure" position. This action enables the Overpressure Mitigating System. The redundant PORV channels shall remain enabled and operable while the Overpressure Mitigation system is required to be in operation.

The reactor coolant system is defined as vented if there is an opening in the reactor coolant system pressure boundary to atmosphere or the pressurizer relief tank that has an equivalent system pressure relieving capability as a PORV. Some examples of such openings include an open or removed PORV, open steam generator or pressurizer manways, a removed pressurizer safety valve, and the top of the reactor vessel when the reactor vessel head has been unbolted or removed.

The mass input transient used to determine the PORV setpoint assumed a worse case transient of a single high pressure safety injection pump discharging to the reactor coolant system while the system is solid. Therefore, when the reactor coolant system is less than 275°F, only one high pressure safety injection pump shall be operable at any time except when the reactor coolant system is open to the atmosphere.

The heat input transient used to determine the PORV setpoint assumes a temperature difference between the reactor coolant system and the steam generator of 50°F. Therefore, before starting a reactor coolant pump when the reactor coolant system is solid, the operator shall insure that the secondary temperature of each steam generator is less than 50°F above the temperature of the reactor coolant system unless a pressure absorbing volume has been verified to exist in the pressurizer or steam generator tubes.

No.	Channel Description	Check	Calibration	Test	Remarks
24.	Containment Pressure	S	R	Q**	Narrow range containment pressure (-3.0, +3 psig excluded)
25.	Steam Generator Pressure	S***	R	Q**	
26.	Emergency Plan Radiation Survey Instruments	Q	R	Q	
27.	Environmental Monitors	M	N.A.	N.A.	
28.	Overpressure Mitigating	S	R	****	
29.	PORV Position Indicator	S	R	R	Check required only when the overpressure mitigation system is in operation.
30.	PORV Block Valve Position Indicator	Q	R	N.A.	
31.	Safety Valve Position Indicator	M	R	N.A.	
32.	PORV Operability	N.A.	R	Q	Performance of a channel functional test but excluding valve operation
33.	Subcooling Margin Monitor	M	R	N.A.	
34.	Undervoltage on 4 KV Bus	N.A.	R	M**	For Auxiliary Feedwater Pump Initiation
35.	Auxiliary Feedwater Flow Rate	See Remarks	R	N.A.	Flow Rate indication will be checked at each unit startup and shutdown
36.	Degraded 4.16 KV Voltage	S	R	M**	
37.	a. Loss of Voltage (4.16 KV)	S	R	M**	
	b. Loss of Voltage (480 V)	S	R	M**	
38.	4160 V Frequency	N.A.	R	N.A.	

Unit 1 - Amendment No. ~~38~~,~~47~~,~~55~~,~~59~~,~~113~~,~~140~~,155
Unit 2 - Amendment No. ~~50~~,~~55~~,~~60~~,~~64~~,~~116~~,~~144~~,159

No.	Channel Description	Check	Calibrate	Test	Remarks
39.	Containment High Range Radiation	S**	R	M**	Calibration to be verification of response to a source.
40.	Containment Hydrogen Monitor	D	R/Q	N.A.	Gas Calibration - Q, Electronic Calibration - R Sample gas for calibration at 2% and 6% hydrogen.
41.	Reactor Vessel Fluid Level System	M	R	N.A.	
42.	In-Core Thermocouple	M	R	N.A.	Calibration to be verification of response to a source.

** 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.

*** During cold or refueling shutdown, a check of one pressure channel per steam generator is required when the steam generator could be pressurized.

**** 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.

TABLE 15.4.1-2 (Continued)

		Test	Frequency
8.	Secondary Coolant	Gross Beta-gamma Activity or gamma isotopic analysis	Weekly(6)
		Iodine concentration	Weekly when gross Beta-gamma activity equals or exceeds 1.2 μCi/cc (6)
9.	Control Rods	Rod drop times of all full length rods (3)	Each refueling or after maintenance that could affect proper functioning (4)
10.	Control Rod	Partial movement of all rods	Every 2 weeks (6)
11.	Pressurizer Safety Valves	Set point	Every five years (11)
12.	Main steam Safety Valves	Set Point	Every five years (11)
13.	Containment Isolation Trip	Functioning	Each refueling shutdown
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	Annually (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	a. Complete Valve Cycle b. Open position check	Quarterly (13) Every 72 hours (14)

TABLE 15.4.1-2 (Continued)

		Test	Frequency
24.	Integrity of Post Accident Recovery Systems Outside Containment	Evaluate	Each refueling cycle
25.	Containment Purge Supply and Exhaust Isolation Valves	Verify valves are locked closed	Monthly[9]
26.	Reactor Trip Breakers	a. Verify independent operability of automatic shunt and undervoltage trip functions.	Monthly[9]
		b. Verify independent operability of manual trip to shunt and undervoltage trip functions.	Each refueling shutdown
27.	Reactor Trip Bypass Breakers	a. Verify operability of the undervoltage trip function.	Prior to breaker use
		b. Verify operability of the shunt trip functions.	Each refueling shutdown
		c. Verify operability of the manual trip to undervoltage trip functions.	Each refueling shutdown
28.	120 VAC Vital Instr. Bus Power	Verify Energized[12]	Shiftly
29.	Power Operated Relief Valves (PORVs), PORV Solenoid Air Control Valves, and Air System Check	Operate[16]	Each shutdown[15]

(1) Required only during periods of power operation.
(2) $\bar{E}$ determination will be started when the gross activity analysis of a filtered sample indicates $\geq 10\mu Ci/cc$ and will be redetermined if the primary coolant gross radioactivity of a filtered sample increases by more than $10\mu Ci/cc$.
(3) Drop test shall be conducted at rated reactor coolant flow. Rods shall be dropped under both cold and hot condition, 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.

TABLE 15.4.1-2 (Continued)

(6) Not required during periods of refueling shutdown.
(7) At least once per week during periods of refueling shutdown.
(8) At least three times per week (with maximum time of 72 hours between samples) during periods of refueling shutdown.
(9) Not required during periods of cold or refueling shutdown.
(10) Sample to be taken after a minimum of 2 EFPD and 20 days power operation since the reactor was last subcritical for 48 hours or longer.
(11) An approximately equal number of valves shall be tested each refueling outage such that all valves will be tested within a five year period. If any valve fails its tests, an additional number of valves equal to the number originally tested shall be tested. If any of the additional tested valves fail, all remaining valves shall be tested.
(12) The specified buses shall be determined energized in the required manner at least once per shift by verifying correct static transfer switch alignment and indicated voltage on the buses.
(13) Not required if the block valve is shut to isolate a PORV that is inoperable for reasons other than excessive seat leakage.
(14) Only applicable when the overpressure mitigation system is in service.
(15) Required to be performed only if conditions will be established, as defined in Specification 15.3.15, where the PORVs are used for low temperature overpressure protection. The test must be performed prior to establishing these conditions.
(16) Test valve operation in accordance with the inservice test requirements of the ASME Boiler and Pressure Vessel Code, Section XI.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO AMENDMENT NOS.155 AND 159 TO FACILITY OPERATING LICENSE NOS. DPR-24 AND DPR-27

WISCONSIN ELECTRIC POWER COMPANY POINT BEACH NUCLEAR PLANT, UNIT NOS. 1 AND 2 DOCKET NOS. 50-266 AND 50-301

1.0 INTRODUCTION

By letter dated November 14, 1990, the Wisconsin Electric Power Company, the licensee, responded to Generic Letter (GL) 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,' Pursuant to 10 CFR 50.54(f)," dated June 25, 1990. The licensee's response stated its position on the issues raised in the GL, and provided a commitment to implement listed actions and submit proposed Technical Specifications (TS) as appropriate.

As per its commitment, by letter dated May 30, 1991, the licensee submitted a request for amendments to the Point Beach Nuclear Plant (PBNP) TS. By letter dated January 25, 1993, the staff responded to the licensee stating that portions of its submittal did not adequately respond to the GL. The licensee then submitted supplemental information and revised proposed TSs on May 7, 1993, and April 28, 1994. This supplemental information did not change the initial proposed determination in the notice published on March 25, 1993 (58 FR 16233) that no significant hazards consideration is involved.

These amendments would revise Technical Specifications 15.3.1.A.5 and 15.3.15, and Tables 15.4.1-1 and 15.4.1-2. The changes would specify more stringent limiting conditions for operation and surveillance requirements for pressurizer power-operated relief valves and block valves. These changes were proposed to conform to the NRC's plan for resolution of Generic Issue 70, "Power-Operated Relief Valve and Block Valve Reliability," and Generic Issue 94, "Additional Low-Temperature Overpressure Protection for Light Water Reactors," as conveyed in Generic Letter 90-06. Other related changes were also proposed.

2.0 BACKGROUND

Generic Issue 70, "Power-Operated Relief Valve and Block Valve Reliability," involves the evaluation of the reliability of power-operated relief valves (PORVs) and block valves and their safety significance in PWR plants. The generic letter discussed how PORVs are increasingly being relied on to perform safety-related functions and the corresponding need to improve the reliability of both PORVs and their associated block valves. Proposed staff positions and improvements to the plant's technical specifications were recommended to be implemented at all affected facilities.

Generic Issue 94, "Additional Low-Temperature Overpressure Protection for Light-Water Reactors," addresses concerns with the implementation of the requirements set forth in the resolution of Unresolved Safety Issue A-26, "Reactor Vessel Pressure Transient Protection (Overpressure Protection)." The generic letter discussed the continuing occurrence of overpressure events and the need to further restrict the allowed outage time for a low-temperature overpressure protection channel in operating modes 4, 5, and 6.

3.0 EVALUATION

The actions proposed by the NRC staff to improve the reliability of PORVs and block valves and the availability of the low-temperature overpressure protection (LTOP) system represent a substantial increase in overall protection of the public health and safety. Based on this, a determination has been made that the attendant costs are justified in view of this increased protection. The technical findings and the regulatory analysis are discussed in NUREG-1316, "Technical Findings and Regulatory Analysis Related to Generic Issue 70 - Evaluation of Power-Operated Relief Valve Reliability in PWR Nuclear Power Plants" and NUREG-1326, "Regulatory Analysis for the Resolution of Generic Issue 94, Additional Low-Temperature Overpressure Protection for Light-Water Reactors."

In its November 14, 1990 submittal, the licensee stated that the PORVs and block valves are on the operational QA list, that maintenance is based on the vendor's recommendations, and that replacement parts are procured in accordance with the original construction code. In addition, the licensee agreed to place the PORVs, block valves, and air control and check valves in the inservice testing (IST) program, in addition to adding the block valves to the GL 89-10 (motor-operated valve test and evaluation) program. In its May 30, 1991 submittal, the licensee stated that these valves would all be tested each refueling outage (except the block valves, which would be tested quarterly).

In its May 7, 1993 submittal, the licensee stated that it meets the intent of Mode 3 or 4 testing of the PORVs by testing the valves prior to placing them in LTOP mode, at temperature conditions which are at least as great as Mode 4. In addition, the proposed amendments to TS 15.3.1.A.5 and 15.3.15, and Tables 15.4.1-1 and 15.4.1-2 included with the May 7, 1993, and the April 28, 1994 submittals, specified more stringent limiting conditions for operation and surveillance requirements for pressurizer power-operated relief valves and block valves.

The staff has reviewed the licensee's positions and its proposed TS and TS bases changes. Since these positions and changes are consistent with the guidance stated in the GL, and adequately address the concerns noted by the staff in the GL and in the letter from the staff to the licensee dated January 25, 1993, the staff finds the proposed modifications acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Wisconsin State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

These amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously published a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR §51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

6.0 CONCLUSION

The staff has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: A. G. Hansen

Date: September 30, 1994