To Regs
Docket File
50-266

MAY 2 0 1980

Docket Nos. 50-266
and 50-301

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. 45 to Facility Operating License No. DPR-24 and Amendment No. 50 to Facility Operating License No. DPR-27 for the Point Beach Nuclear Plant, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated November 2, 1978.

These amendments add limiting conditions for operation and surveillance requirements for low temperature overpressure mitigating systems and correct some clerical inconsistencies.

The clerical changes included in the November 2, 1978 submittal included the following:

1. A-change to the last column heading in Tables 15.3.5-2, -3 and -4, correcting the reference to Columns 3 or 4 vice 3 or 5.

2. The FSAR Section Reference column was deleted from Table 15.4.1-2 since the information was unnecessary and, in some cases, incorrect.

3. The comment for water well samples in Table 15.4.10-1 was deleted since the requirements are covered by the radiological sampling program,

4. The reference to the specific year of the ASTM test procedure in Specification 15.4.15.F.2 was deleted, since this standard has been revised and the licensee has committed to test in accordance with the latest approved ASTM test procedures, and

8006170244

5. The title of the Health Physics Administrative Control Policies and Procedure Manual was corrected in Specification 15.6.11.

OFFICE ➤						
SURNAME ➤						
DATE ➤						

Since these changes are all clerical in nature, the Staff has approved them and included revised Technical Specification pages in this amendment. The conclusion contained in Section 5.0 of the enclosed Safety Evaluation Report also applies to these clerical changes.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original signed by:

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

Enclosures:
1. Amendment No. 45 to DPR-24
2. Amendment No. 50 to DPR-27
3. Safety Evaluation
4. Notice of Issuance

cc: w/enclosures
See next page

SEE PREVIOUS CONCURRENCE & DISTRIBUTION

OFFICE						
SURNAME						
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Docket 50-266

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

May 20, 1980

Docket Nos. 50-266
and 50-301

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. 45 to Facility Operating License No. DPR-24 and Amendment No. 50 to Facility Operating License No. DPR-27 for the Point Beach Nuclear Plant, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated November 2, 1978.

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The clerical changes included in the November 2, 1978 submittal included the following:

1. A change to the last column heading in Tables 15.3.5-2, -3 and -4, correcting the reference to Columns 3 or 4 vice 3 or 5.

2. The FSAR Section Reference column was deleted from Table 15.4.1-2 since the information was unnecessary and, in some cases, incorrect.

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4. The reference to the specific year of the ASTM test procedure in Specification 15.4.15.F.2 was deleted, since this standard has been revised and the licensee has committed to test in accordance with the latest approved ASTM test procedures, and

5. The title of the Health Physics Administrative Control Policies and Procedure Manual was corrected in Specification 15.6.11.

8006170 244

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Robert A. Clark

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

Enclosures:
1. Amendment No. 45 to DPR-24
2. Amendment No. 50 to DPR-27
3. Safety Evaluation
4. Notice of Issuance

cc: w/enclosures
See next page

Mr. Sol Burstein
Wisconsin Electric Power Company

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

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-266

POINT BEACH NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 45
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 November 2, 1978, 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.

8006170249

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. 45 , 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: May 20, 1980

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 45 TO FACILITY OPERATING LICENSE NO. DPR-24

DOCKET NOS. 50-266 AND 50-301

Revise Appendix A as follows: -

Remove Pages	Insert Pages
15-i	15-i
Table 15.3.5-2	Table 15.3.5-2
Table 15.3.5-2 (cont.)	Table 15.3.5-2 (cont.)
Table 15.3.5-3	Table 15.3.5-3
Table 15.3.5-4	Table 15.3.5-4
---	15.3.15-1
---	15.3.15-2
---	15.3.15-3
Table 15.4.1-1 (cont.)	Table 15.4.1-1 (cont.)
Table 15.4.1-2	Table 15.4.1-2
Table 15.4.1-2 (cont.)	Table 15.4.1-2 (cont.)
Table 15.4.10-1 (cont.)	Table 15.4.10-1 (cont.)
15.4.15-2	15.4.15-2
15.6.9-10	15.6.9-10
15.6.11-1	15.6.11-1

TABLE OF CONTENTS

TABLE 15.3.5-2

INSTRUMENT OPERATION CONDITIONS FOR REACTOR TRIP

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 ACTIO IF CONDITIONS COLUMN 3 OR 4 CANNOT BE MET
1.	Manual	2	1	1	-*		Maintain hot shutdown
2.	Nuclear Flux Power Range**					2 of 4 power range channels greater than 10% F.P. (low setting only)	Maintain hot shutdown
	low setting	4	2	3	2		
	high setting	4	2	3	2		
3.	Nuclear Flux Intermediate Range	2	1	1	-*	2 of 4 power range channels greater than 10% F.P.	Maintain hot shutdown. Not
4.	Nuclear Flux Source Range	2	1	1	-*	1 of 2 intermediate range channels greater than 10^{-10} amps	Maintain hot shutdown. Not
5.	Overtemperature ΔT	4	2	3	2		Maintain hot shutdown
6.	Overpower ΔT	4	2	3	2		Maintain hot shutdown
7.	Low Pressurizer Pressure	4	2	3	2		Maintain hot shutdown
8.	Hi Pressurizer Pressure	3	2	2	1		Maintain hot shutdown
9.	Pressurizer-Hi Water Level	3	2	2	1		Maintain hot shutdown
10.	Low Flow in one loop (>50% F.P.)	3/loop	2/loop (any loop)	2	1		Maintain hot shutdown
	Low Flow Both Loops (10-50% F.P.)	3/loop	2/loop (any loop)				

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
16.	Control rod misalignment as monitored by on-line computer	1	-	1	-		Log individual rod positions once/hour, and after a load change >10% or after >30 inches of control rod motion

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 REDUNDANCY	5 PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS O 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***

** - Must actuate 2 switches simultaneously.

*** - If minimum conditions are not met within 24 hours, steps shall be taken on the affected unit to place the unit in cold shutdown conditions.

TABLE 15.3.5-4

INSTRUMENT OPERATING CONDITIONS FOR ISOLATION FUNCTIONS

		1	2	3	4	5	
NO.	FUNCTIONAL UNIT	NO. OF CHANNELS	NO. OF CHANNELS TO TRIP	MIN. OPERABLE CHANNELS	MINIMUM DEGREE OF REDUNDANCY	PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 4 CANNOT BE MET
	1 CONTAINMENT ISOLATION						
a.	Safety Injection	See Item No. 1 b,c, and d of Table 15.3.5-3					Hot Shutdown***
b.	Manual	2	1	1	-		Hot Shutdown
	2 STEAM LINE ISOLATION						
a.	Hi Hi Steam Flow with Safety Injection	2/loop	1	1	-		Hot Shutdown***
b.	Hi Steam Flow and 2 of 4 Low T_{avg} with Safety Injection	2/loop	1	1	-		Hot Shutdown***
c.	Hi Containment Pressure	3	2	2	1		Hot Shutdown***
d.	Manual	1/loop	1/loop	1/loop	-		Hot Shutdown

*** - If minimum conditions are not met within 24 hours, steps shall be taken on the affected unit to place the unit in cold shutdown conditions.

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 Figures 15.3.1-1 (Unit 1) and 15.3.1-3 (Unit 2). Operability requirements are:
   a. Both pressurizer power operated relief valves operable at a setpoint of $\leq$425 psig.
   b. The upstream isolation valves to both power operated relief valves are open.
2. The requirements of 15.3.15.A.1 may be modified to allow one of the two power operated relief valves to be inoperable for a period of not more than seven days.

3. If the inoperable power operated relief valves cannot be made operable within seven days, the reactor coolant system must be depressurized and vented to the pressurizer relief tank within eight hours.

4. If both power operated relief valves are inoperable, the reactor coolant system must be depressurized and vented to the pressurizer relief tank within eight 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°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°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

   b. The secondary water temperature of each steam generator is less than 50°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.

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 reactor coolant system is not open to the atmosphere and the temperature is less than the minimum pressurization temperature for the inservice pressure test, except that one PORV may be out of service for a period of up to seven days.

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.

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 System	S	R	****	

S - Each Shift

D - Daily

W - Weekly

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, except for first core cycle)

NA - 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.

**** 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.

MINIMUM FREQUENCIES FOR EQUIPMENT AND SAMPLING TESTS

		Test	Frequency
1.	Reactor Coolant Samples	Gross Beta-gamma activity (excluding tritium)	5/week (7)
		Tritium activity	Monthly
		Radiochemical $\bar{E}$ Determination (1)	Semiannually (2)
		Chloride Concentration	5/week (8)
		Diss. Oxygen Conc.	5/week (6)
		Fluoride Conc.	Weekly
2.	Reactor Coolant Boron	Boron Concentration	Twice/week
3.	Refueling Water Storage Tank Water Sample	Boron Concentration	Weekly (6)
4.	Boric Acid Tanks	Boron Concentration	Twice/week
5.	Spray Additive Tank	NaOH Concentration	Monthly
6.	Accumulator	Boron Concentration	Monthly
7.	Spent Fuel Pit	Boron Concentration	Monthly
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	Each refueling shutdown
12.	Main Steam Safety Valves	Set point	Each refueling shutdown
13.	Containment Isolation Trip	Functioning	Each refueling shutdown

		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

---

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

Sample Type	Locations[a,c]	Frequency	Analysis	Comments
Well Water	1-Onsite Well (10)	Quarterly	Gross Beta-T.S.[b] Gamma Scan T.S. Tritium Strontium-89 Strontium-90	
Milk	1-Local dairy pool (11) 1-Dairy Farm, NNW (19) 1-Dairy Farm, SSE (21)	Monthly	Gamma Scan Radioiodine Strontium-89 Strontium-90	Radioiodine analysis done by the resin extraction technique.
Algae	1-North of Discharge (5) 1-Discharge of Flume (12)	3x/yr as available	Gross Beta Gamma Scan	
Fish	1-Travelling screens (13)	3x/yr as available	Gross Beta Gamma Scan	Analysis of edible portions only.

(a) Reference location is chosen well in excess of 10 miles from the plant in a low X/Q sector to provide an estimate of background levels.

(b) T.S. - Total Solids

(c) Numbers given under location correspond to sampling locations shown in Figure 15.4.10-1.

C. Fire Hose Stations

Test	Frequency
1. Visual Inspection	Monthly
2. Hose Hydro-Test	Yearly
3. Partially open each hose station valve to verify operability and no blockage	3 years

D. Fire Detection

Test	Frequency
1. Channel Functional Test	2 mo.

E. Fire Barrier Penetration Fire Seals

Test	Frequency
1. Visual Inspection	18 mo. and following repairs or maintenance

F. Fire Pump Diesel Engine

Test	Frequency
1. a. Verify 200 gallons of fuel in fuel storage tank	Monthly
b. Verify diesel starts from ambient conditions and operates for at least 20 minutes.	Monthly
2. Sample diesel fuel per ASTM-D270 and verify acceptable per Table 1 of ASTM-D975 with respect to viscosity, water content and sediment.	Quarterly
3. a. Inspect diesel per procedures prepared in conjunction with its manufacturer's recommendations	18 months
b. Verify diesel starts from ambient conditions and operates for ≥ 20 minutes while loaded with the fire pump	18 months

(1) The number and types of samples taken and the measurements made on the samples; e.g., gross beta gamma scan, etc.

(2) Any changes made in sample types or locations during the reporting period, and criteria for these changes.

b. A summary of survey results during the reporting period.

4. <u>Leak Testing of Source</u>

Results of required leak tests performed on seal sources if the tests reveal the presence of 0.005 microcuries or more of removable contamination.

D. <u>Poison Assembly Removal from Spent Fuel Storage Racks</u>

Plans for removal of any poison assemblies from the spent fuel storage racks shall be reported and described at least 14 days prior to the planned activity. Such report shall describe neutron attenuation testing for any replacement poison assemblies, if applicable, to confirm the presence of boron material.

E. <u>Overpressure Mitigating System Operation</u>

In the event the overpressure mitigating system is operated to relieve a pressure transient which, by licensee's evaluation, could have resulted in an overpressurization incident had the system not been operable; a special report shall be prepared and submitted to the Commission within 30 days. The report shall describe the circumstances initiating the transient, the effect of the system on the transient and any corrective action necessary to prevent recurrence.

15.6.11 RADIATION PROTECTION PROGRAM

Specification

Radiological control procedures shall be written and made available to all station personnel, and shall state permissible radiation exposure levels. The radiation protection program shall meet the requirements of 10 CFR 20, with the exception of the following:

Paragraph 20.203 - Caution signs, labels and signals

In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2), each radiation area in which the intensity of radiation is greater than 100 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area, and entrance thereto shall be controlled in accordance with the Point Beach Nuclear Plant Health Physics Administrative Control Policies and Procedure Manual, Section 2.7, Radiation Work Permit. A person or persons permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area. In addition, each High Radiation Area outside the containment building in which the intensity of radiation is greater than 1000 mrem/hr shall be provided with locked barricades to prevent unauthorized entry into these areas, and the keys to these locked barricades shall be maintained under the administrative control of the Duty Shift Supervisor.

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. 50
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 November 2, 1978, 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. 50 , 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: May 20, 1980

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 50 TO FACILITY OPERATING LICENSE NO. DPR-27

DOCKET NOS. 50-266 AND 50-301

Revise Appendix A as follows:

Remove Pages	Insert Pages
15-i	15-i
Table 15.3.5-2	Table 15.3.5-2
Table 15.3.5-2 (cont.)	Table 15.3.5-2 (cont.)
Table 15.3.5-3	Table 15.3.5-3
Table 15.3.5-4	Table 15.3.5-4
---	15.3.15-1
---	15.3.15-2
---	15.3.15-3
Table 15.4.1-1 (cont.)	Table 15.4.1-1 (cont.)
Table 15.4.1-2	Table 15.4.1-2
Table 15.4.1-2 (cont.)	Table 15.4.1-2 (cont.)
Table 15.4.10-1 (cont.)	Table 15.4.10-1 (cont.)
15.4.15-2	15.4.15-2
15.6.9-10	15.6.9-10
15.6.11-1	15.6.11-1

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TABLE 15.3.5-2

INSTRUMENT OPERATION CONDITIONS FOR REACTOR TRIP

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MIN. OPERABLE CHANNELS	4 MINIMUM DEGREE OF REDUNDANCY	5 PERMISSABLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 4 CANNOT BE MET
1.	Manual	2	1	1	-*		Maintain hot shutdown
2.	Nuclear Flux Power Range**					2 of 4 power range channels greater than 10% F.P. (low setting only)	Maintain hot shutdown
	low setting	4	2	3	2		
	high setting	4	2	3	2		
3.	Nuclear Flux Intermediate Range	2	1	1	-*	2 of 4 power range channels greater than 10% F.P.	Maintain hot shutdown. Note 1
4.	Nuclear Flux Source Range	2	1	1	-*	1 of 2 intermediate range channels greater than 10^{-10} amps	Maintain hot shutdown. Note 1
5.	Overtemperature ΔT	4	2	3	2		Maintain hot shutdown
6.	Overpower ΔT	4	2	3	2		Maintain hot shutdown
7.	Low Pressurizer Pressure	4	2	3	2		Maintain hot shutdown
8.	Hi Pressurizer Pressure	3	2	2	1		Maintain hot shutdown
9.	Pressurizer-Hi Water Level	3	2	2	1		Maintain hot shutdown
10.	Low Flow in one loop (>50% F.P.)	3/loop	2/loop (any loop)	2	1		Maintain hot shutdown
	Low Flow Both Loops (10-50% F.P.)	3/loop	2/loop (any loop)				

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
16.	Control rod misalignment as monitored by on-line computer	1	-	1	-		Log individual rod positions once/hour, and after a load change >10% or after >30 inches of control rod motion

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 REDUNDANCY	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***

** - Must actuate 2 switches simultaneously.

*** - If minimum conditions are not met within 24 hours, steps shall be taken on the affected unit to place the unit in cold shutdown conditions.

TABLE 15.3.5-4

INSTRUMENT OPERATING CONDITIONS FOR ISOLATION FUNCTIONS

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MIN. OPERABLE CHANNELS	4 MINIMUM DEGREE OF REDU-DANCY	5 PERMISSABLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 4 CANNOT BE MET
	1 CONTAINMENT ISOLATION						
a.	Safety Injection	See Item No. 1 b,c, and d of Table 15.3.5-3					Hot Shutdown***
b.	Manual	2	1	1	-		Hot Shutdown
	2 STEAM LINE ISOLATION						
a.	Hi Hi Steam Flow with Safety Injection	2/loop	1	1	-		Hot Shutdown***
b.	Hi Steam Flow and 2 of 4 Low T_{ave} with Safety Injection	2/loop	1	1	-		Hot Shutdown***
c.	Hi Containment Pressure	3	2	2	1		Hot Shutdown***
d.	Manual	1/loop	1/loop	1/loop	-		Hot Shutdown

*** - If minimum conditions are not met within 24 hours, steps shall be taken on the affected unit to place the unit in cold shutdown conditions.

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 Figures 15.3.1-1 (Unit 1) and 15.3.1-3 (Unit 2). Operability requirements are:

   a. Both pressurizer power operated relief valves operable at a setpoint of ≤425 psig.

   b. The upstream isolation valves to both power operated relief valves are open.

2. The requirements of 15.3.15.A.1 may be modified to allow one of the two power operated relief valves to be inoperable for a period of not more than seven days.

3. If the inoperable power operated relief valves cannot be made operable within seven days, the reactor coolant system must be depressurized and vented to the pressurizer relief tank within eight hours.

4. If both power operated relief valves are inoperable, the reactor coolant system must be depressurized and vented to the pressurizer relief tank within eight 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 ≤275°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 ≤275°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

   b. The secondary water temperature of each steam generator is less than 50°F above the temperature of the reactor coolant system.

Basis

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 set-point 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.

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 reactor coolant system is not open to the atmosphere and the temperature is less than the minimum pressurization temperature for the inservice pressure test, except that one PORV may be out of service for a period of up to seven days.

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.

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 System	S	R	****	

S - Each Shift

D - Daily

W - Weekly

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, except for first core cycle)

NA - 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.

**** 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

MINIMUM FREQUENCIES FOR EQUIPMENT AND SAMPLING TESTS

		Test	Frequency
1.	Reactor Coolant Samples	Gross Beta-gamma activity (excluding tritium)	5/week (7)
		Tritium activity	Monthly
		Radiochemical $\overline{E}$ Determination (1)	Semiannually (2)
		Chloride Concentration	5/week (8)
		Diss. Oxygen Conc.	5/week (6)
		Fluoride Conc.	Weekly
2.	Reactor Coolant Boron	Boron Concentration	Twice/week
3.	Refueling Water Storage Tank Water Sample	Boron Concentration	Weekly (6)
4.	Boric Acid Tanks	Boron Concentration	Twice/week
5.	Spray Additive Tank	NaOH Concentration	Monthly
6.	Accumulator	Boron Concentration	Monthly
7.	Spent Fuel Pit	Boron Concentration	Monthly
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	Each refueling shutdown
12.	Main Steam Safety Valves	Set point	Each refueling shutdown
13.	Containment Isolation Trip	Functioning	Each refueling shutdown

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

---

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

Sample Type	Locations[a,c]	Frequency	Analysis	Comments
Well Water	1-Onsite Well (10)	Quarterly	Gross Beta- T.S.[b] Gamma Scan T.S. Tritium Strontium-89 Strontium-90	
Milk	1-Local dairy pool (11) 1-Dairy Farm, NNW (19) 1-Dairy Farm, SSE (21)	Monthly	Gamma Scan Radioiodine Strontium-89 Strontium-90	Radioiodine analysis done by the resin extraction technique.
Algae	1-North of Discharge (5) 1-Discharge of Plume (12)	3x/yr as available	Gross Beta Gamma Scan	
Fish	1-Travelling screens (13)	3x/yr as available	Gross Beta Gamma Scan	Analysis of edible portions only.

(a) Reference location is chosen well in excess of 10 miles from the plant in a low X/Q sector to provide an estimate of background levels.

(b) T.S. - Total Solids

(c) Numbers given under location correspond to sampling locations shown in Figure 15.4.10-1.

C. Fire Hose Station

Test	Frequency
1. Visual Inspection	Monthly
2. Hose Hydro-Test	Yearly
3. Partially open each hose station valve to verify operability and no blockage	3 years

D. Fire Detection

Test	Frequency
1. Channel Functional Test	2 mo.

E. Fire Barrier Penetration Fire Seals

Test	Frequency
1. Visual Inspection	18 mo. and following repairs or maintenance

F. Fire Pump Diesel Engine

Test	Frequency
1. a. Verify 200 gallons of fuel in fuel storage tank	Monthly
b. Verify diesel starts from ambient conditions and operates for at least 20 minutes.	Monthly
2. Sample diesel fuel per ASTM-D270 and verify acceptable per Table 1 of ASTM-D975 with respect to viscosity, water content and sediment.	Quarterly
3. a. Inspect diesel per procedures prepared in conjunction with its manufacturer's recommendations	18 months
b. Verify diesel starts from ambient conditions and operates for $\geq$20 minutes while loaded with the fire pump	18 months

(1) The number and types of samples taken and the measurements made on the samples; e.g., gross beta gamma scan, etc.

(2) Any changes made in sample types or locations during the reporting period, and criteria for these changes.

b. A summary of survey results during the reporting period.

4. Leak Testing of Source

Results of required leak tests performed on seal sources if the tests reveal the presence of 0.005 microcuries or more of removable contamination.

D. Poison Assembly Removal from Spent Fuel Storage Racks

Plans for removal of any poison assemblies from the spent fuel storage racks shall be reported and described at least 14 days prior to the planned activity. Such report shall describe neutron attenuation testing for any replacement poison assemblies, if applicable, to confirm the presence of boron material.

E. Overpressure Mitigating System Operation

In the event the overpressure mitigating system is operated to relieve a pressure transient which, by licensee's evaluation, could have resulted in an overpressurization incident had the system not been operable; a special report shall be prepared and submitted to the Commission within 30 days. The report shall describe the circumstances initiating the transient, the effect of the system on the transient and any corrective action necessary to prevent recurrence.

Specification

Radiological control procedures shall be written and made available to all station personnel, and shall state permissible radiation exposure levels. The radiation protection program shall meet the requirements of 10 CFR 20, with the exception of the following:

Paragraph 20.203 - Caution signs, labels and signals

In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2), each radiation area in which the intensity of radiation is greater than 100 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area, and entrance thereto shall be controlled in accordance with the Point Beach Nuclear Plant Health Physics Administrative Control Policies and Procedure Manual, Section 2.7, Radiation Work Permit. A person or persons permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area. In addition, each High Radiation Area outside the containment building in which the intensity of radiation is greater than 1000 mrem/hr shall be provided with locked barricades to prevent unauthorized entry into these areas, and the keys to these locked barricades shall be maintained under the administrative control of the Duty Shift Supervisor.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 45 TO FACILITY OPERATING LICENSE NO. DPR-24

AND AMENDMENT NO. 50 TO FACILITY OPERATING LICENSE NO. DPR-27

WISCONSIN ELECTRIC POWER COMPANY

POINT BEACH NUCLEAR PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-266 AND 50-301

1.0 Introduction

By letter dated July 28, 1977 (Reference 1) Wisconsin Electric Power Company (WEPCO) submitted to the NRC plant specific analyses in support of the reactor vessel overpressure mitigating system (OMS) for Point Beach Units 1 and 2. The analyses were supplemented by letter dated October 28, 1977 (Reference 2) and other documentation submitted by WEPCO (References 3-6).

Staff review of all information submitted by WEPCO in support of the proposed overpressure mitigating system is complete and has found that the system provides adequate protection from overpressure transients. A detailed safety evaluation follows.

2.0 Background

Over the last few years, incidents identified as pressure transients have occurred in pressurized water reactors. This term "pressure transients," as used in this report, refers to events during which the temperature pressure limits of the reactor vessel, as shown in the facility Technical Specifications, are exceeded. All of these incidents occurred at relatively low temperature (less than 200 degrees F) where the reactor vessel material toughness (resistance to brittle failure) is reduced.

The "Technical Report on Reactor Vessel Pressure Transients" in NUREG 0138 (Reference 7) summarizes the technical considerations relevant to this matter, discusses the safety concerns and existing safety margins of operating reactors, and describes the regulatory actions taken to resolve this issue by reducing the likelihood of future pressure transient events at operating reactors. A brief discussion is presented here.

2.1 Vessel Characteristics

Reactor vessels are constructed of high quality steel made to rigid specifications, and fabricated and inspected in accordance with the time-proven rules of the ASME Boiler and Pressure Vessel Code. Steels used are particularly tough at reactor operating conditions. However, since reactor vessel steels are less tough and could possibly fail in a brittle manner if subjected to high pressures at low temperatures, power reactors have always operated with restrictions on the pressure allowed during startup and shutdown operations.

At operating temperatures, the pressure allowed by Apendix G limits is in excess of the setpoint of currently installed pressurizer code safety valves. However, most operating PWRs did not have pressure relief devices to prevent pressure transients during cold conditions from exceeding the Appendix G limit.

2.2 Regulatory Actions

By letter dated August 11, 1976, (Reference 8) the NRC requested that WEPCO begin efforts to design and install plant systems to mitigate the consequences of pressure transients at low temperatures. It was also requested that operating procedures be examined and administrative changes be made to guard against initiating overpressure events. It was felt by the staff that proper administrative controls were required to assure safe operation for the period of time prior to installation of the proposed overpressure mitigating hardware.

WEPCO responded (References 5 and 6) with preliminary information describing interim measures to prevent these transients along with some discussion of proposed hardware. The proposed hardware change was to install a low pressure actuation setpoint on the pressurizer air operated relief valves.

WEPCO participated as a member of a Westinghouse user's group which was formed to support the analysis effort required to verify the adequacy of the proposed system to prevent overpressure transients. Using input data generated by the user's group, Westinghouse performed transient analyses (References 9 and 10) which are used as the basis for plant specific analysis.

Plant specific analyses for Point Beach Units 1 and 2 were submitted by WEPCO by letter dated July 18, 1977 (Reference 1) and supplemented by letter dated October 28, 1977 (Reference 2).

2.3.1 Design Criteria

Through this series of meetings and correspondence with PWR vendors and licensees, the staff developed a set of criteria for an acceptable overpressure mitigating system. The basic criterion is that the mitigating system will prevent reactor vessel pressures in excess of these allowed by Appendix G. Specific criteria for system performance are:

1) Operator Action: No credit can be taken for operator action for ten minutes after the operator is aware of a transient.

2) Single Failure: The system must be designed to relieve the pressure transient given a single failure in addition to the failure that initiated the pressure transient.

3) Testability: The system must be testable on a periodic basis consistent with the system's employment.

4) Seismic and IEEE 279 Criteria: Ideally, the system should meet seismic Category I and IEEE 279 criteria. The basic objective is that the system should not be vulnerable to a common failure that would both initiate a pressure transient and disable the overpressure mitigating system. Such events as loss of instrument air and loss of offsite power must be considered.

The staff also instructed the licensee to provide an alarm which monitors the position of the pressurizer relief valve isolation valves, along with the low setpoint enabling switch, to assure that the overpressure mitigating system is properly aligned for shutdown conditions.

2.4 Design Basis Events

The incidents that have occurred to date have been the result of operator errors or equipment failures. Two varieties of pressure transients can be identified: a mass input type from charging pumps, safety injection pumps, safety injection accumulators; and a heat addition type which causes thermal expansion from sources such as steam generators or decay heat.

On Westinghouse designed plants, the most common cause of the overpressure transients to date has been isolation of the letdown path. Letdown during low pressure operations is via a flowpath through the RHR system. Thus, isolation of RHR can initiate a pressure transient if a charging pump is left running. Although other transients occur with lower frequency, those which result in the most rapid pressure increases were identified by the staff for analysis. The most limiting mass input transient identified by the staff is inadvertent injection by the largest safety injection pump. The most limiting thermal expansion transient is the start of a reactor coolant pump with a 50 degree F temperature difference between the water in the reactor vessel and the water in the steam generator.

Based on the historical record of overpressure transients and the imposition of more effective administrative controls, the staff believes that the limiting events identified above form an acceptable bases for analyses of the proposed overpressure mitigating system.

3.0 System Description and Evaluation

WEPCO adopted the "Reference Mitigating System" developed by Westinghouse and the user's group. The licensee proposed to modify the actuation circuitry of the existing air operated pressurizer relief valves to provide a low pressure setpoint at 425 psig during startup and shutdown conditions. When the reactor vessel is at low temperatures, with the low pressure setpoint selected, a pressure transient is terminated below the Appendix G limit by automatic opening of these relief valves. A manual switch is used to enable and disable the low setpoint of each relief valve. The OMS will remain in service during heatup until the RCS temperature reaches a level corresponding to the value (approximately 370 degrees F) at which the inservice pressure test may be performed. Conversely during cooldown the OMS will be enabled when the RCS is depressurized to a pressure less than 425 psig (the OMS setpoint) and before the RCS temperature drops below the temperature at which the inservice pressure test may be performed (~370 degrees F). The staff finds the pressurizer relief valves with a manually enabled low pressure setpoint to be an acceptable concept for an overpressure mitigating system. Discussion and evaluation of the system proposed by WEPCO follows.

3.1 Air Supply

The power operated relief valves (PORVs) are spring-loaded-closed, air required to open valves, which are supplied by a control air source. To assure operability of the valves upon loss of control air, a backup air supply is provided. The backup air supply consists of a compressed gas bottle for each PORV. Each tank contains enough air for approximately 139 valve openings. The staff finds the backup air supply to be acceptable.

3.2 Electrical Controls

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, depending on plant conditions, by the operator with a key-lock switch. 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 relief valves on the RHR system are available for pressure relief whenever the RHR system is connected to the RCS. The RHR system is normally connected to the RCS during plant conditions when overpressurization events have been most prevalent, i.e., during low-temperature and low-pressure conditions. The RHR-system relief valves can be considered a diverse relief system at Point Beach because the RHR system isolation valves do not automatically isolate the RHR system during a pressure transient, thereby making the relief valves available throughout the transient.

During plant cooldown and prior to the collapse of the steam bubble in the pressurizer, the operator acting under administrative procedure places the keylock switch in the "low pressure" position and connects the RHR system to the RCS. An alarm will alert the operator when the pressure is sufficiently low so that activation of the low pressure setpoint circuit will not inadvertently open a PORV. Placing the keylock switch in the "low pressure" position blocks the alarm indicating low pressure and enables the low-temperature high-pressure alarm.

During plant heatup, the operating procedure will identify the plant conditions for which low-pressure protection is no longer needed. The operator places the keylock switch in the "normal" position, thereby returning control of the PORV's to the operating high-pressure condition and avoiding inadvertent opening of the PORV's. Placing the keylock switch in the "normal" position removes or blocks the low-temperature high-pressure alarms and enables the low-pressure operation alert alarm.

We find the above design features acceptable.

3.3. Testability

Testability will be provided. WEPCO has stated that verification of operability of the OMS control system will be performed prior to entering water solid conditions. PORV testing will be performed during each refueling outage. Testing requirements will be incorporated in the Technical Specifications as discussed in Section 4.2 of this evaluation.

3.4 Appendix G

The Appendix G curve submitted by WEPCO for purposes of overpressure transient analysis is based on 32 effective full power years irradiation. The zero degree heatup curve is allowed since most pressure transients occur during isothermal metal conditions. Margins of 30 psig and 10 degrees F are included for possible instrument errors. The staff finds that use of this curve is acceptable as a basis for overpressure mitigating system performance.

3.5 Setpoint Analysis

The one loop version of LOFTRAN (Reference WCAP 7907) was used to perform the mass input analyses. The four loop version was used for the heat input analysis. Both versions require some input modeling and initialization changes were required. LOFTRAN is currently under review by the staff and is judged to be an acceptable code for treating problems of this type.

The results of this analysis are provided in terms of PORV setpoint overshoot. The predicted maximum transient pressure is simply the sum of the overshoot magnitude and the setpoint magnitude. The PORV setpoint is adjusted so that given the setpoint overshoot, the resultant pressure is still below that allowed by Appendix G limits.

WEPCO presented the following Point Beach Units 1 and 2 plant characteristics to determine the pressure reached for the design basis pressure transients:

SI Pump Flowrate	Flowrate versus pressure used in generic analysis (Ref. 10)
RCS Volume	6,900 ft^3
PORV Opening Time	2 sec
S G Heat Transfer area	44,000 ft^2
Relief Valve setpoint	425 psig

Westinghouse identified certain assumptions and input parameters as conservative with respect to the analysis. These include one PORV assumed to fail, conservative heat transfer coefficients, conservative modeling of stored energy of steam generators, and conservative interpolation schemes to obtain plant specific results from generic analyses.

The relief capacity of the RHR (990 gpm at 500 psig) has been conservatively neglected. In fact the relief capacity of the RHR will accommodate hypothesized mass injection from a single safety injection pump. It is a diverse as well as redundant subsystem.

3.5.1 Mass Input Case

The inadvertent start of a safety injection pump with the plant in a cold shutdown condition was selected as the limiting mass input case.

Westinghouse provided the licensee with a series of curves based on the LOFTRAN analysis of a generic plant design which indicates PORV setpoint overshoot for this transient as a function of system volume, relief valve opening time and relief valve setpoint. These sensitivity analyses were then applied to the Point Beach Units 1 and 2 plant parameters to obtain a conservative estimate of the PORV setpoint overshoot. WEPCO has taken credit for volumetric expansion of the RCS with increasing RCS pressure. This is calculated to reduce the pressure overshoot (P max - P setpoint) to 74% of the value calculated assuming no expansion. The calculations are documented in References 1 and 2. The staff finds this method of analysis to be acceptable.

The calculated pressure overshoot assuming inadvertent mass addition from a single safety injection pump is less than or equal to 94.5 psi. With an OMS low pressure setpoint of 425 psig, the highest predicted pressure for the worst case mass input is 519.5 psig. It has been assumed that only one PORV opens. No credit has been taken for the RHR system relief capacity. The 32 EFPY Appendix G limit of Units 1 and 2 at RCS temperatures greater than or equal to 109 F and 136 F respectively is greater than or equal to 520 psig. Use of an Appendix G curve applicable for less than 32 EFPY's would show additional conservatism. Hence the OMS setpoint is considered acceptable.

3.5.2 Heat Input Case

Inadvertent startup of a reactor coolant pump with a primary to secondary temperature differential across the steam generator, and with the plant in a water solid condition, was selected as the limiting heat input case. For the heat input case, Westinghouse provided the licensee with a series of curves based on the LOFTRAN analysis of a generic plant design to determine the PORV setpoint overshoot as a function of RCS volume, steam generator UA, initial RCS temperature, reactor coolant/steam generator ΔT, relief valve setpoint, and relief valve opening time.

WEPCO calculated the following values of the maximum pressure for the heat input transient for a fixed ΔT of 50 degrees F as a function of the initial RCS temperature.

RCS Temperature	Maximum Pressure
100	441
140	454
180	465
250	490

In all these cases, for the given RCS temperature, the Appendix G limits are not exceeded.

The staff finds that the analyses of the limiting mass input and heat input cases show a maximum pressure transient below that allowed by Appendix G limits and is therefore acceptable.

3.6 Implementation

Unit 1

WEPCO installed interim hardware protection (a single control system, without redundant air supply) during the fall of 1976. Redundant control channels were installed during the October 1977 refueling. Redundant air supplies were installed on Unit 1 during the November 1979 refueling outage.

Unit 2

WEPCO installed interim hardware protection during the spring 1977 refueling. Redundant control channels and backup air supplies were installed on Unit 2 during the March 1979 refueling outage.

4.0 Administrative Controls

To supplement the hardware modifications and to limit the magnitude of postulated pressure transients to within the bounds of the analysis provided by the licensee, a defense in depth approach is adopted using procedural and administrative controls. Those specific conditions required to assure that the plant is operated within the bounds of the analysis will be enumerated in forthcoming Technical Specifications.

4.1 Procedures

A number of provisions for prevention of pressure transients have been incorporated in the plant operating procedures.

With respect to hypothesized mass addition transients, HPSI pump, HPSI isolation valve motors, and accumulator isolation valve motors are electrically isolated with circuit beakers locked in the open position.

Of particular concern is the conduct of the loss of A.C. simultaneous with Safety Injection test which may be performed with the RCS in a water solid condition. During this test which simulates loss of A.C. power and startup of emergency diesel generators both safety injection pumps and safety injection isolation valves receive command signals. The OMS is not designed to mitigate mass addition from both safety injection trains. Additional steps will be incorporated in the plant operating procedures to check that Safety Injection Isolation Valves, MOV866A and B are closed, and circuit breakers providing power to the valve motor operators are open and tagged, prior to performing this test.

With respect to heat addition hypothesized transients, RCP starts are minimized, RCS/SG differential temperatures are checked prior to RCP starts. In addition RCP starts during solid water conditions are performed with minimized charging flow rates and maximized letdown flow rates.

With respect to any scenario, additional, redundant, relief capacity is provided by insuring that the RHR system is aligned prior to taking the plant solid.

The staff finds that the procedural and administrative controls described are acceptable.

4.2 Technical Specifications

To assure proper operation of the overpressure mitigating system, WEPCO submitted, by letter dated November 2, 1978, proposed Technical Specifications for Staff review. The proposed specifications were reviewed against the following criteria.

1. The OMS is to be operable when the RCS temperature is below the value at which inservice pressure testing may be performed. The OMS setpoint is to be incorporated in the Technical Specifications. Operability requires that the system is enabled, upstream isolation valves open and backup air supply charged. Should one redundant train (control circuitry and associated relief valve) be inoperable for more than seven days either a vapor bubble is to be established in the pressurizer or the primary system depressurized and vented to the atmosphere within eight hours. Should both redundant trains be found inoperable either a vapor bubble is to be established in the pressurizer or the primary system depressurized within eight hours.

2. Suitable surveillance requirements are to be proposed consistent with the need for use of the OMS.

3. Electrical isolation of HPSI pump and isolation valve motors and the reinstation of electrical power to these components is to be incorporated in the Technical Specifications.

4. Surveillance requirements consistent with the assumption that the RCS/SG differential temperature is less than or equal to 50 degrees F are to be proposed. It is noted that calculations based on a RCS/SG ΔT of 50 degrees F result in ample margins to the Appendix G curves. Should WEPCO chose to demonstrate that larger values of RCS/SG ΔT are acceptable with respect to violations of the Appendix G curves corresponding relaxation of surveillance requirements will be accepted.

5. Operation of the OMS (PORV's and/or RHR relief valves) to relieve a pressure transient is to be reported.

Our review of WEPCO's submittal indicated that some modifications and additions were required to ensure compliance with the Staff's criteria. These changes have been discussed with and agreed to by the WEPCO Staff. With the inclusion of these changes, the Staff finds the proposed Technical Specifications conform to our criteria and are, therefore, acceptable.

5.0 Conclusions

The administrative controls and hardware changes proposed by Wisconsin Electric Power Company provide protection for Point Beach Units 1 and 2 from pressure transients at low temperatures by reducing the probability of initiation of a transient and by limiting the pressure of such a transient to below the limits set by Appendix G. The staff finds that the overpressure mitigating system meets the criteria established by the NRC and is acceptable as a long term solution to the problem of overpressure transients. However, any future revisions of Appendix G limits for Point Beach Units 1 and 2 must be considered and the overpressure mitigating system setpoint adjusted accordingly with corresponding adjustments in the license.

Environmental Consideration

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

Conclusion

We have concluded, based on the considerations discussed above, 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.

Date: MAY 20 1980

REFERENCES

1. W.E. letter (Burstein) to NRC (Case), July 18, 1977.
2. W.E. letter (Burstein) to NRC (Case), October 28, 1977.
3. W.E. letter (Burstein) to NRC (Rusche), September 3, 1976.
4. W.E. letter (Burstein) to NRC (Rusche), October 14, 1976.
5. W.E. letter (Burstein) to NRC (Rusche), March 2, 1977.
6. W.E. letter (Burstein) to NRC (Rusche), April 18, 1977.
7. "Staff Discussion of Fifteen Technical Issues listed in Attachment G November 3, 1976 Memorandum from Director NRR to NRR Staff." NUREG-0138, November 1976.
8. NRC letter (Lear) to W.E., (Burstein), August 11, 1976.
9. "Pressure Mitigating System Transient Analysis Results" prepared by Westinghouse for the Westinghouse user's group on reactor coolant system overpressurization, July 1977.
10. "Supplement to the July 1977 Report, Pressure Mitigating Systems Transient Analysis Results," prepared by Westinghouse for the Westinghouse user's group on reactor coolant system overpressurization, September 1977.
11. W.E. letter (Burstein) to NRC (Denton), November 2, 1978.

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-266 AND 50-301

WISCONSIN ELECTRIC POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 45 to Facility Operating License No. DPR-24, and Amendment No. 50 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 add limiting conditions for operation and surveillance requirements for the low temperature overpressure mitigating systems and correct some clerical inconsistencies.

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 November 2, 1978, (2) Amendment Nos. 45 and 50 to License Nos. DPR-24 and DPR-27, and (3) 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 54451. 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 20th day of May 1980

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

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

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

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May 20, 1980

Docket No. 50-266/301

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: POINT BEACH UNITS 1 and 2

Two signed originals of the Federal Register Notice identified below are enclosed for your transmittal to the Office of the Federal Register for publication. Additional conformed copies (12) of the Notice are enclosed for your use.

- ☐ Notice of Receipt of Application for Construction Permit(s) and Operating License(s).
- ☐ Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s): Time for Submission of Views on Antitrust Matters.
- ☐ Notice of Availability of Applicant's Environmental Report.
- ☐ Notice of Proposed Issuance of Amendment to Facility Operating License.
- ☐ Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.
- ☐ Notice of Availability of NRC Draft/Final Environmental Statement.
- ☐ Notice of Limited Work Authorization.
- ☐ Notice of Availability of Safety Evaluation Report.
- ☐ Notice of Issuance of Construction Permit(s).
- ☒ Notice of Issuance of Facility Operating License(s) or Amendment(s).
- ☒ Other: Amendments Nos. 45 and 50
  Referenced documents have been provided PDR

Division of Licensing, ORB#3
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

OFFICE→	ORB#3:DL					
SURNAME→	PKreutzer/jdc					
DATE→	5/21/80					

NRC FORM 102 (1-76)