Mr. Richard R. Grigg
Chief Nuclear Officer
Wisconsin Electric Power Company
231 West Michigan Street, Room P379
Milwaukee, WI 53201

bruary 20, 1997

SUBJECT: POINT BEACH NUCLEAR PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS RE: LOW TEMPERATURE OVERPRESSURE PROTECTION SYSTEM (TAC NOS. M96571 AND M96572)

Dear Mr. Grigg:

The Commission has issued the enclosed Amendment Nos. 172 and 176 to Facility Operating License Nos. DPR-24 and 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 original application dated September 19, 1996, as supplemented November 18, 1996, revised on January 13 and supplemented on January 27, 1997.

These amendments revise the reactor coolant system temperature below which the low temperature overpressure protection (LTOP) system and pressurizer power-operated relief valves (PORVs) shall be operable, modify the requirement to limit operation of the high pressure safety injection pump from reactor coolant system cold leg temperature of $\leq 275\ ^{\circ}F$ to whenever the LTOP is required to be operable, change the name of the system from the overpressure mitigation system to the LTOP system, and revise the PORV setpoint from 425 psig to 440 psig.

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

Sincerely,

Orig. signed by
Linda L. Gundrum, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-266
and 50-301

Enclosures: 1. Amendment No. 172 to DPR-24
2. Amendment No. 176 to DPR-27
3. Safety Evaluation

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DATE	2/18/97		2/19/97		2/11/97		2/12/97		2/18/97		2/19/97	

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DATED: February 20, 1997

AMENDMENT NO. 172 TO FACILITY OPERATING LICENSE NO. DPR-24 - POINT BEACH UNIT NO. 1
AMENDMENT NO. 176 TO FACILITY OPERATING LICENSE NO. DPR-27 - POINT BEACH UNIT NO. 2

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Mr. Richard R. Grigg
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Wisconsin Electric Power Company
231 West Michigan Street, Room P379
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DATE	2/11/97		2/11/97		2/ /97		2/ /97		2/18/97		2/ /97	

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

February 20, 1997

Mr. Richard R. Grigg
Chief Nuclear Officer
Wisconsin Electric Power Company
231 West Michigan Street, Room P379
Milwaukee, WI 53201

SUBJECT: POINT BEACH NUCLEAR PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS RE: LOW TEMPERATURE OVERPRESSURE PROTECTION SYSTEM (TAC NOS. M96571 AND M96572)

Dear Mr. Grigg:

The Commission has issued the enclosed Amendment Nos. 172 and 176 to Facility Operating License Nos. DPR-24 and 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 original application dated September 19, 1996, as supplemented November 18, 1996, revised on January 13 and supplemented on January 27, 1997.

These amendments revise the reactor coolant system temperature below which the low temperature overpressure protection (LTOP) system and pressurizer power-operated relief valves (PORVs) shall be operable, modify the requirement to limit operation of the high pressure safety injection pump from reactor coolant system cold leg temperature of $\leq 275\ ^{\circ}F$ to whenever the LTOP is required to be operable, change the name of the system from the overpressure mitigation system to the LTOP system, and revise the PORV setpoint from 425 psig to 440 psig.

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Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-266
and 50-301

Enclosures: 1. Amendment No. 172 to DPR-24
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3. Safety Evaluation

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Mr. Richard R. Grigg
Wisconsin Electric Power Company

Point Beach Nuclear Plant
Unit Nos. 1 and 2

cc:

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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. 172
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 September 19, 1996, as supplemented November 18, 1996, revised on January 13 and supplemented on January 27, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-24 is hereby amended to read as follows:

   B. <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 172 , 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

Linda L. Gundrum

Linda L. Gundrum, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of issuance: February 20, 1997

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. 176
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 September 19, 1996, as supplemented November 18, 1996, revised on January 13 and supplemented on January 27, 1997, 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. 176, 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

Linda L. Gundrum

Linda L. Gundrum, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of issuance: February 20, 1997

ATTACHMENT TO LICENSE AMENDMENT NOS. 172 AND 176

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 vertical lines indicating the area of change.

REMOVE	INSERT
15-i	15-i
15.3.1-3	15.3.1-3
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 6)	Table 15.4.1-1 (Page 3 of 6)
15.6.9-4	15.6.9-4

TABLE OF CONTENTS

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 355 °F. 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.

15.3.15 LOW TEMPERATURE OVERPRESSURE PROTECTION SYSTEM

Applicability

Applies to operability of the low temperature overpressure protection (LTOP) system when the reactor coolant system temperature is < 355 °F.

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 LTOP system and to specify further limiting conditions for operation when the reactor coolant system is operated at low temperatures.

Specification

A. System Operability

1. Except as specified in 15.3.15.A.2 below, the LTOP system shall be operable whenever the reactor coolant system is not open to the atmosphere and the temperature is < 355 °F. Operability requirements are:

    a. Both pressurizer power operated relief valves operable at a setpoint of ≤ 440 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 < 355 °F, 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 < 355 °F, 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 LTOP is required to be enabled by Specification 15.3.15.A.1, no more than one high pressure safety injection pump shall be operable. The second high pressure safety injection pump shall be rendered inoperable whenever LTOP is required to be enabled 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 < 355 °F 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 °F above the temperature of the reactor coolant system.

<u>Basis</u>

The Low Temperature Overpressure Protection System consists of a redundant means of relieving pressure during periods of water solid operation and when the reactor coolant system temperature is < 355 °F. 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. The LTOP setpoint of 440 psig is valid through an inside surface neutron fluence of the limiting reactor vessel material of less than or equal to 2.05×10^{19} n/cm^2 (E > 1.0 Mev).

During plant cooldown prior to reducing reactor coolant system temperature below 355 °F, the operator under administrative procedures shall place the keylock switch in the "Low Pressure" position. This action enables the Low Temperature Overpressure Protection System. The redundant PORV channels shall remain enabled and operable while the LTOP 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 assumes 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 LTOP is required to be enabled, 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 ensure 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.

TABLE 15.4.1-1 (continued)

NO.	CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	PLANT CONDITIONS WHEN REQUIRED
20.	Auxiliary Feedwater Flowrate	(13)	R	-	ALL
21.	Boric Acid Control System	-	R	-	ALL
22.	Boric Acid Tank Level	D	R	-	ALL
23.	Charging Flow	-	R	-	ALL
24.	Condensate Storage Tank Level	S(1)	R	-	ALL
25.	Containment High Range Radiation	S(1)	R(14)	M(1)	ALL
26.	Containment Hydrogen Monitor	D	-	-	ALL
	-Gas Calibration	-	Q(15)	-	ALL
	-Electronic Calibration	-	R	-	ALL
27.	Containment Pressure	S	R	Q(1,3,9)	ALL
28.	Containment Water Level	M	R	-	ALL
29.	Emergency Plan Radiation Survey Instruments	Q	R	Q	ALL
30.	Environmental Monitors	M	-	-	ALL
31.	In-Core Thermocouples	M	R(14)	-	ALL
32.	Low Temperature Overpressure Protection System	S(12)	R	(10)	ALL
33.	PORV Block Valve Position Indicator	Q	R	-	ALL
34.	PORV Operability	-	R	Q(11)	ALL
35.	PORV Position Indicator	S(21)	R	R	ALL

15.6.9.2 Unique Reporting Requirements

The following written reports shall be submitted to the Director, Office of Nuclear Reactor Regulation, USNRC:

A. Deleted

B. 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.

C. Low Temperature Overpressure Protection System Operation

In the event the low temperature overpressure protection system (power operated relief valves in the low temperature overpressure protection mode) or residual heat removal system relief valves are 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 172 AND 176 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 September 19, 1996, as supplemented November 18, 1996, revised on January 13 and supplemented on January 27, 1997, the Wisconsin Electric Power Company (the licensee) requested amendments to the Technical Specifications (TS) appended to Facility Operating License Nos. DPR-24 and DPR-27 for the Point Beach Nuclear Plant, Unit Nos. 1 and 2. The proposed amendments would redesignate the overpressure mitigation system to the low temperature overpressure protection (LTOP) system in TS 15.3.15, Table 15.4.1-1, TS 15.6.9, and Table of Contents entry for TS 15.3.15. Additionally, the proposed amendments would revise TS 15.3.1, "Reactor Coolant System," and TS 15.3.15, "Low Temperature Overpressure Protection System," to specify that below a reactor coolant system (RCS) temperature of 355 °F the pressurizer power-operated relief valves (PORVs) and the LTOP system are operable, modify TS 15.3.15 requirement to limit operation of the high pressure safety injection pump at an RCS cold leg temperature of ≤ 275 °F to whenever the LTOP system is required to be operable, and revise the PORV setpoint from 425 psig to 440 psig.

2.0 EVALUATION

2.1 LTOP Setpoint

The purpose of the LTOP system is to control the RCS pressure at low temperature so that the integrity of the reactor coolant pressure boundary is not compromised by violating the pressure temperature (P-T) limits of 10 CFR Part 50, Appendix G, which is based on Appendix G, Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). Currently, Point Beach Nuclear Plant (PBNP) TS state that the PORV lift setting be less than or equal to 425 psig. The licensee is proposing to increase the PORV lift setting to less than or equal 440 psig. On January 27, 1997, the Commission exempted PBNP from the requirements of 10 CFR 50.60, thereby, permitting the use of ASME Code Case N-514, "Low Temperature Overpressure Protection," for use at PBNP to establish the lift setpoint of the PORV for overpressure protection during low temperature conditions. As delineated in the Code Case, the LTOP system shall limit the maximum pressure in the vessel to 110% (1.1) of the pressure determined to satisfy Appendix G,Section XI of the ASME Code.

The licensee has performed an evaluation that shows the PORV setpoint of less than or equal to 440 psig is sufficient to ensure that the peak pressure for the most limiting reactor vessel weld in both units is less than 1.1 times the ASME Section III, Appendix G, for both the limiting mass addition transient and the limiting energy addition transient assuming the RCS is water solid. The setpoint calculation includes temperature and pressure instrument uncertainties, in addition to a pressure correction to account for the elevation static pressure difference and the dynamic pressure difference, created from the flow of both reactor coolant pumps and both residual heat removal (RHR) pumps, between the pressure sensor and the limiting weld. The licensee is using the same methodology to verify that the current PORV setpoint is adequate that was used to verify that the previous setpoints were adequate. The previous setpoints were approved by the staff and documented in a safety evaluation issued on May 20, 1980.

The most limiting pressure transient for Point Beach is the mass addition transient which assumes that a high head safety injection pump is started while the RCS is water solid and the temperature is at the coldest allowable temperature. The Appendix G limits are most limiting at the coldest possible temperature and the use of this point will provide limiting results. The results indicate that the calculated peak pressure will not exceed the limits. The energy addition transient is analyzed at four different temperatures and assumes that the first reactor coolant pump inadvertently starts with the steam generator temperature 50 °F higher than the RCS temperature. The start of a reactor coolant pump causes the water-solid RCS to heat up and results in an increase in RCS pressure. This transient is evaluated based on initial RCS temperatures of 100 °F, 140 °F, 180 °F, and 250 °F with bias correction for two reactor coolant pumps operating and two RHR pumps operating, which is conservative because the analysis assumes only one reactor coolant pump is started. The range of temperatures chosen provides enough information to show that the results will be representative. The results indicate that the PORV setpoint of 440 psi is acceptable and prevents the RCS pressures from exceeding the limits. The staff review therefore concludes that the PORV setpoint is acceptable.

2.2 LTOP Enable Temperature

Code Case N-514 indicates that the LTOP system is required to be operable at a water temperature corresponding to a metal temperature of RT_{NDT} + 50 °F at the beltline location that is controlling in the Appendix G limit calculations. Based on the limiting RT_{NDT} of 262.4 °F for Unit 1 (See section 2.4) the metal temperature would be 312.4 °F. The LTOP enable temperature of 355 °F provides a 93 °F margin between the water temperature and the limiting RT_{NDT} (355 °F minus 262 °F = 93 °F). The margin includes uncertainties resulting from instrument error and the difference between the metal temperature and fluid temperature at the maximum heatup rate (100 °F/hr). The staff finds that the licensee's proposed TS LTOP enable temperature of 355 °F is acceptable.

2.3 LTOP TS Changes

The licensee is proposing changes to TS 15.3.15 to:

retitle the overpressure mitigating system to the LTOP system,

restate the TS's objective as restricting RCS operation when the RCS is operated at low temperatures rather than without a pressure-absorbing volume in the pressurizer,

replace RCS temperature limits for LTOP and PORV limiting conditions for operation from "less than the minimum pressurization temperature for the inservice pressure test as specified in Figure 15.3..1-1" to less than 355 °F,

revise the restriction on having no more than one high pressure safety injection pump operable from "the temperature of one or both RCS cold legs is less than or equal to 275 °F" to when "LTOP is required to be enabled," and

revise the TS basis to reflect the above changes and correct a typographical error "insure" to "ensure."

These proposed changes modify the condition for overpressure protection during low temperature operation by ensuring temperature requirements for LTOP system operability are consistent with the LTOP event analysis and Appendix G. Based on a review of the applicability, objective, specifications, and the basis for the affected TS, the staff finds these changes acceptable.

The licensee is proposing to change to TS 15.3.1.A.5 to replace the requirement to maintain RCS temperature greater than "the minimum pressurization temperature for the inservice pressure test as defined in Figure 15.3.1-1" to 355 °F. This change will ensure consistency with the operability requirements for the TS 15.3.15 as described above and are acceptable to the staff.

The licensee is proposing to change the system designation from the overtemperature mitigating system to LTOP system in Table 15.4.1-1 and TS 15.6.9.2. These changes are administrative and are acceptable to the staff.

2.4 Reactor Vessel Material Data

The staff reviewed the applicability of reactor vessel material data included in calculations supporting the proposed change in PORV setpoint for LTOP system operability. An adjusted reference temperature (ART) was calculated based on the methods in Regulatory Guide (RG) 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2. The ART is defined as the sum of initial nil-ductility transition reference temperature (RT_{NDT}) of the material, the increase in the RT_{NDT} caused by neutron irradiation, and a margin to account for uncertainties in the prediction method. The increase in RT_{NDT} is calculated from the product of a chemistry factor (CF) and a fluence factor. The chemistry factor is calculated depending upon the amount of copper and nickel in the vessel material as specified in Table 1 of RG 1.99, Rev. 2. The staff found a typographical error for the initial Unit 1 RT_{NDT} which was listed as 0 °F rather than 10 °F. The licensee's calculation used the correct value.

The most limiting weld for Unit 1 is the intermediate-to-lower shell circumferential weld, SA-1101. The most limiting material for Unit 2 is the intermediate-to-lower shell circumferential weld, SA-1484. The licensee's calculation used the 1/4T locations for determining the ART. For weld SA-1101 an ART equal to 262.4 °F was determined based on a copper content of 0.26 weight percent (wt.%), nickel content of 0.60 wt.%, CF of 180 °F, initial RT_{NDT} of 10 °F, a margin of 56 °F, and a neutron fluence at 1/4T of 1.39 x 10^{19} neutrons per square centimeter (n/cm^2). For SA-1484 an ART equal to 249.7 °F was determined based on a copper content of 0.24 wt.%, nickel content of 0.60 wt.%, CF of 173 °F, initial RT_{NDT} of -5 °F, a margin of 66 °F, and a neutron fluence at 1/4T of 1.39 x 10^{19} n/cm^2.

The licensee calculated the limiting temperature for the closure flange region using a reference temperature of 60 °F for the limiting material in the closure flange region that is highly stressed by bolt preload. The licensee determined that the minimum allowable temperature of 77.8 °F (60 °F + instrument uncertainty of 17.8 °F) is required prior to pressurizing the reactor vessel.

The staff verified that the copper and nickel content and initial RT_{NDT} agreed with the NRC reactor vessel material database as reported by the licensee in response to Generic Letter 92-01, "Reactor Vessel Structural Integrity." The staff also verified that the licensee evaluated the most limiting material locations for both units. The staff reviewed the material properties used in the calculation of the ART values for the limiting materials using RG 1.99, Revision 2, methodology. The staff reviewed the licensee's analysis of minimum temperature at the closure head flange performed in accordance with 10 CFR Part 50, Appendix G, Section IV.A.2. The staff verified that the properties used for the most limiting materials and conditions were used to verify the new calculated LTOP setpoint. The staff concluded that appropriate material data was used in the licensee's calculations to verify meeting the applicable requirements of 10 CFR Part 50, Appendix G, based on the use of ASME Code Case N-514.

2.5 Instrument Error

The proposed TS change required determination of new LTOP setpoints. The new setpoints include pressure instrument error, associated with existing instrumentation, a bias value associated with the elevation difference of the wide range pressure transmitters and the mid-plane of the reactor vessel, and a bias value for flow effects for the operation of both reactor coolant pumps and two RHR pumps. The total pressure instrument location bias values for each unit based on one reactor coolant pump operating and with two reactor coolant pumps running are: -41.3 psig and -70.3 psig for Unit 1, and -44.6 psig and -74.4 psig for Unit 2. Instrument uncertainties for pressure and temperature of ± 13 psi and ± 17.8 °F are included in the licensee's calculation.

The licensee inclusion of instrument uncertainties was found acceptable by the staff.

3.0 EXIGENT CIRCUMSTANCES

The Commission's regulations, 10 CFR 50.91, contain provisions for issuance of amendments where the Commission finds that exigent circumstances exist, in that a licensee and the Commission must act quickly and that time does not permit the Commission to publish a Federal Register notice allowing 30 days for prior public comment. The exigency exists in that the proposed amendments are needed prior to the start of reactor vessel head tensioning and time does not permit the Commission to publish a notice allowing 30 days for prior public comment. The licensee revised its original application on January 13, 1997, in anticipation of receiving an exemption from the requirements of 10 CFR 50.60, which the Commission granted on January 27, 1997. After submitting the revised application, the licensee determined that additional calculations were required, which the licensee promptly submitted on January 27, 1997. At the time of the licensee's January 27, 1997, submittal, reactor vessel head tensioning was scheduled for February 10, 1997, and startup of Point Beach Unit 2 was scheduled for February 25, 1997. The staff has determined that the licensee used its best efforts to make a timely application.

Accordingly, the Commission has determined that exigent circumstances exist pursuant to 10 CFR 50.91(a)(6), the submittal of information was timely and could not have been avoided, and that the licensee did not create the exigency.

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATIONS DETERMINATION

The Commission's regulations in 10 CFR 50.92(c) state that the Commission may make a final determination that a license amendment involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) result in a significant reduction in the margin of safety. The NRC staff has made a final determination that no significant hazards consideration is involved for the proposed amendments and that the amendments should be issued as allowed by the criteria contained in 10 CFR 50.91. The NRC staff's final determination is presented below.

(1) The proposed changes would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes will explicitly define the temperature at which LTOP is required to be enabled, raise the temperature at which one high pressure safety injection pump is required to be rendered inoperable, and increase the setpoint of the PORVs. The changes do not affect any accident analyses since the LTOP is required only when RCS temperatures are low. LTOP is not required during power operation. The consequences or probability of a previously evaluated accident will, therefore, not significantly be increased.

(2) The proposed changes would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes will still meet the requirements for fracture toughness requirements required by 10 CFR 50.60 as modified by the use of ASME Code Case N-514 which was developed by the ASME as an alternative to describe requirements in 10 CFR Part 50, Appendix G.. Therefore, a new or different kind of accident is not created.

(3) The proposed changes would not result in a significant reduction in the margin of safety.

The proposed changes increase the range of the temperature region where the LTOP system is needed, while increasing the allowed setpoint pressure by only 3.5 percent. Therefore, these changes do not involve a significant reduction in a margin of safety.

5.0 STATE CONSULTATION

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

6.0 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. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluent that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission made a final finding that the amendments involve no significant hazards consideration. 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.

7.0 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.

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