July 23, 2001

Mr. Mark Reddemann
Site Vice President
Kewaunee and Point Beach Nuclear Plants
Nuclear Management Company, LLC
6610 Nuclear Road
Two Rivers, WI 54241

SUBJECT: POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2 - ACCEPTANCE OF METHODOLOGY FOR REFERENCING PRESSURE TEMPERATURE LIMITS REPORT (TAC NOS. MA8459 AND MA8460)

Dear Mr. Reddemann:

The Nuclear Regulatory Commission (NRC) staff has completed its review of the pressure temperature (P/T) curves and low temperature overpressure protection (LTOP) system limits methodology and the pressure temperature limits report (PTLR) submitted by Wisconsin Electric by letter dated March 10, 2000, as supplemented July 28, November 20, 2000, and April 10, 2001. The staff finds your methodology to be acceptable for referencing in the administrative controls section of the Point Beach Nuclear Power Plant, Units 1 and 2, technical specifications to the extent specified and under the limitations delineated in your submittals and the associated NRC safety evaluation (SE), which is enclosed.

The original application was submitted by Wisconsin Electric. Wisconsin Electric was subsequently succeeded by Nuclear Management Company (NMC), as the licensed operator of Point Beach Nuclear Plants, Units 1 and 2. By letter dated October 5, 2000, NMC requested that the NRC staff continue to process and disposition licensing actions previously docketed and requested by Wisconsin Electric.

The methodology for review relating to the P/T limit curves and the LTOP system limits is provided in the references listed in Section 5.0 of the enclosed SE. WCAP-14040-NP-A, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and Reactor Coolant System Heatup and Cooldown Limit Curves,” Revision 2, January 15, 1996, provided in part, the methodology used for determining the acceptance of the Point Beach methodology.

Your current calculated limits contain fluence values that are only valid until October 30, 2003, for Unit 1, and October 1, 2008, for Unit 2. We understand that you plan to submit the FERRET code for review by the staff in order to extend the use of your PTLR. Any changes to your currently approved methodology must first be reviewed and approved by the staff.

Should our criteria or regulations change so that our conclusions as to the acceptability of the methodology is invalidated, licensees referencing these documents will be expected to revise and resubmit their respective documentation, or submit justification for the continued effective applicability of the documents without revision of their respective documentation.

If you have any questions regarding this letter or enclosed SE, please contact Beth Wetzel at (301) 415-1355.

Sincerely,

/RA/

Claudia M. Craig, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-266 and 50-301

Enclosure: Safety Evaluation

cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REVIEW OF PRESSURE TEMPERATURE LIMITS REPORT AND

METHODOLOGY FOR THE RELOCATION OF THE REACTOR COOLANT SYSTEM

PRESSURE TEMPERATURE LIMIT CURVES AND LOW TEMPERATURE OVERPRESSURE

PROTECTION SYSTEM LIMITS

NUCLEAR MANAGEMENT COMPANY, LLC

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-266 AND 50-301

1.0 INTRODUCTION

By letter dated March 10, 2000, and supplemented by letters dated July 28, November 20, 2000, and April 10, 2001, Nuclear Management Company, LLC (NMC) requested changes to the technical specifications (TSs) for Point Beach Nuclear Plant, Units 1 and 2. The requested changes included relocating reactor coolant system pressure temperature (P/T) limit curves and low temperature overpressure protection (LTOP) system limits from the TSs to a licensee-controlled Pressure Temperature Limits Report (PTLR). The P/T limit curves and LTOP system setpoints were developed, in part, using the staff-approved methodology documented in WCAP-14040-NP-A, Revision 2 (Reference 1). These changes are made in accordance with Generic Letter (GL) 96-03, “Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits,” dated January 31, 1996 (Reference 2). GL 96-03 provides licensees the option to relocate the P/T limit curves and the LTOP system setpoints to a licensee-controlled PTLR provided that the limiting curves and setpoints are developed using an Nuclear Regulatory Commission (NRC)-approved methodology. The limits can then be changed using the referenced methodologies without prior NRC review and approval. Changes to the PTLR parameters will be reported to the NRC as directed by the TS administrative controls reporting requirements.

2.0 BACKGROUND

The implementation of the Point Beach Nuclear Plant (PBNP) Units 1 and 2 PTLR will be a two-step licensing process. The first step utilizes the calculated fluence values (without adjustments) to estimate material properties and will be valid for operation for Unit 1 until October 30, 2003, and Unit 2 until October 1, 2008. This corresponds to 25.59 effective full power years (EFPY) of operation for Unit 1 and 30.51 EFPY for Unit 2. The second step will occur when the FERRET code is reviewed and approved by the staff, which should extend the time limit for the fluence values. In its March 10, 2000, submittal, the licensee calculated 32.2

EFPY for Unit 1, and 34.0 EFPY for Unit 2. However, these values were calculated using the FERRET code to adjust the measured and calculated data. The FERRET code has not been reviewed and approved by the NRC; therefore, it cannot be used to calculate the data. The licensee submitted revised fluence values in its submittal dated November 20, 2000.

3.0 EVALUATION

3.1 Neutron Fluence

Material properties change with irradiation, therefore, the P/T and LTOP limits change accordingly and need to be reevaluated as a function of vessel exposure. GL 96-03 requires that the P/T and LTOP limits be determined using NRC-approved methodologies; and that such methodologies be referenced in the PTLR. With the P/T and LTOP limits in the PTLR, the licensee is not required to request a license amendment every time the limits are reevaluated. PTLRs are plant-specific.

3.1.1 Short Term Resolution and Proposed PTLR

In its submittals dated March 10 and July 28, 2000, the licensee showed how the fluence values were derived from information contained in WCAP-12794 for Unit 1 (Reference 7) and WCAP-12795 for Unit 2 (Reference 8). WCAP-12794 and WCAP-12795 integrate the surveillance capsule data with reactor cavity measurements for Units 1 and 2, respectively. Both reports use the FERRET code to adjust the measured and calculated data. However, the FERRET code has not been reviewed or approved by the NRC; therefore, neither reference can be used in the PTLR. To remedy this problem, the licensee submitted a revised two-step plan for the proposed PTLR. The first step is based on accepting the calculated fluence values and lowering the applicability period from 32.2 EFPYs of operation to 25.59 EFPYs for Unit 1, and from 34.0 to 30.51 EFPYs for Unit 2. In this manner the FERRET values are not used, therefore, all codes used in the estimation of the P/T limits have been approved by the staff and, therefore, are acceptable.

The licensee converted the remaining EFPYs to calendar years which corresponded to expiration dates of October 30, 2003, for Unit 1 and October 1, 2008, for Unit 2. The method of conversion was based on linear interpolation of the fluence values. This could be slightly conservative and is acceptable.

3.1.2 Long-Term Solution

The licensee (through the Owner's group) is pursuing an effort to obtain staff approval for the FERRET code. The FERRET code will then be incorporated into the set of codes to be used in the WCAP-14040 methodology and possibly reevaluate the end of license fluence and establish new P/T and LTOP limits.

Based on the above and considering that the requirements of GL 96-03 are satisfied, the staff finds that the proposed solution is acceptable provided that: (1) the current limits are not extended beyond October 30, 2003, for Unit 1, and October 1, 2008, for Unit 2, (2) the proposed PTLR methodology is acceptable provided that the FERRET code is excluded, (3) the

FERRET code will receive staff approval before it is referenced in the PTLR, and (4) the licensee submits for staff review PTLR modifications for the inclusion of the FERRET (or any other) code.

3.2 Pressure Temperature (P/T Limits)

3.2.1. Requirements and Methodology for Calculating P/T Limits

The proposed P/T limit calculations were based on the current American Society of Mechanical Engineers (ASME) Appendix G methodology, as modified by Code Case N-641, which provides rules in (1) the postulation of a circumferentially-oriented flaw in lieu of an axially-oriented flaw for the evaluation of reactor pressure vessel (RPV) circumferential welds, (2) the use of K_{IC} fracture toughness curve instead of K_{Ia} fracture toughness curve for RPV materials, in determining the P/T limits, and (3) the set of LTOP setpoints allowing the maximum pressure to be 110 percent of the P/T limits based on Appendix G if the K_{Ia} fracture toughness curve is used for RPV materials. The licensee's exemption request dated July 14, 2000, to apply Code Case N-641 was approved by the NRC in a letter dated September 29, 2000.

The NRC has established requirements in 10 CFR Part 50 to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The staff evaluates the P/T limit curves based on the following NRC regulations and guidance: 10 CFR Part 50, Appendix G; GL 88-11; GL 92-01, Revision 1; GL 92-01, Revision 1, Supplement 1; Regulatory Guide (RG) 1.99, Revision 2; and NUREG 0800, "Standard Review Plan for Reviewing Safety Analysis Reports for Nuclear Power Plants, LWR Edition," Section 5.3.2. GL 88-11 advised licensees that the staff would use RG 1.99, Revision 2, to review P/T limit curves. RG 1.99, Revision 2 contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy resulting from neutron radiation. GL 92-01, Revision 1, requested that licensees submit their RPV data for their plants to the staff for review. GL 92-01, Revision 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations. These data are used by the staff as the basis for review of the P/T limit curves as well as other reviews. Appendix G to 10 CFR Part 50 requires that P/T limit curves for the RPV be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the ASME Code.

Standard Review Plan Section 5.3.2 provides an acceptable method of determining the P/T limit curves for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics methodology of Appendix G to Section XI of the ASME Code. The basic parameter of this methodology is the stress intensity factor K_I, which is a function of the stress state and flaw configuration. Appendix G requires a safety factor of 2.0 on stress intensities resulting from reactor pressure during normal and transient operating conditions, and a safety factor of 1.5 for hydrostatic testing curves. The methods of Appendix G postulate the existence of a sharp surface flaw in the RPV that is normal to the direction of the maximum stress. This flaw is postulated to have a depth that is equal to 1/4 of the RPV beltline thickness and a length equal to 1.5 times the RPV beltline thickness. The critical locations in the RPV beltline region for calculating heatup and cooldown P/T curves are the 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively.

The Appendix G ASME Code methodology requires that licensees determine the adjusted reference temperature (ART or adjusted RT_{NDT}). The ART is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin (M) term.

The ΔRT_{NDT} is a product of a chemistry factor and a fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Revision 2, or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the chemistry factor was determined using the tables in RG 1.99, Revision 2, or surveillance data. The margin term is used to account for uncertainties in the values of the initial RT_{NDT}, the copper and nickel contents, and the fluence and calculational procedures. RG 1.99, Revision 2, describes the methodology to be used in calculating the margin term.

3.2.2. <u>Licensee Evaluation of P/T Limits</u>

The licensee submitted detailed calculations for ART and P/T limit curves valid for up to 32.2 EFPYs for Unit 1 and 34.0 EFPYs for Unit 2. The applicable EFPYs for the proposed P/T limits were later modified by the licensee in its November 20, 2000, letter, to 25.59 EFPYs for Unit 1, and 30.51 EFPYs for Unit 2, in response to the staff question on its fluence calculation methodology. However, except for the applicable EFPYs, this modification will not change the licensee's P/T limits evaluation.

For PBNP Unit 1, the licensee determined that the most limiting material for cooldown curves is the lower shell axial welds that were fabricated using weld wire heat 61782. The licensee employed the methodology in RG 1.99, Revision 2 and calculated an ART of 209 °F at the 1/4T fluence of 1.049E19 n/cm^2 (25.59 EFPYs) for this limiting material based on a ΔRT_{NDT} value of 1165.42 °F, an initial RT_{NDT} of -5 °F, and a plant-specific margin term of 48.34 °F (σ_I = 19.7 °F and σ_Δ = 14 °F). The ΔRT_{NDT} value for this material was determined using surveillance data. Similarly, the licensee determined that the most limiting material for heatup curves is the intermediate shell axial welds that were fabricated using weld wire heat 1P0661. The licensee calculated an ART of 194 °F at the 3/4T fluence of 0.5431E19 n/cm^2 (25.59 EFPYs) for this limiting material based on a ΔRT_{NDT} value of 130.70 °F, an initial RT_{NDT} of -5 °F, and a plant-specific margin term of 68.45 °F (σ_I = 19.7 °F and σ_Δ = 28 °F). This time, the ΔRT_{NDT} value was determined using the tables of RG 1.99, Revision 1.

The licensee repeated the similar calculations for PBNP Unit 2 and found that the ART for the most limiting material, the intermediate to lower shell circumferential weld that was fabricated using weld wire heat 72442, is 272 °F at the 1/4T fluence of 1.764E19 n/cm^2 (30.51 EFPYs) for the cooldown, and the ART for the most limiting material is 233 °F at the 3/4T fluence of 0.8088E19 n/cm^2 (30.51 EFPYs) for the heatup. PBNP Unit 2 has the same limiting material for both the cooldown and heatup P/T limits.

Based on the limiting ART of 209 °F for the cooldown and the limiting ART of 194 °F for the heatup, the licensee used the methodology of Appendix G in the 1996 Addenda to the 1995 Edition of Section XI of the ASME Code, as modified by Code Case N-641, to calculate the P/T limits for PBNP Unit 1. The licensee found the heatup and cooldown P/T limits for Unit 1 bound the corresponding P/T limits for Unit 2, and therefore, applied the more conservative Unit 1 P/T limits to both units.

3.2.3. Staff Evaluation of P/T Limits

The staff first compared the licensee's material information from the PBNP Units 1 and 2 Reactor Cavity Neutron Measurement Program contained in Tables 1 through 6 of the March 20, 2000, submittal with that in the NRC's reactor vessel integrity database (RVID). The staff determined that, except for the fluence for Unit 1, the material data for the limiting beltline materials for both Units are consistent with that in the RVID. The staff then performed an independent calculation of the ART values for the limiting materials for both units using the methodology in RG 1.99, Revision 2. Based on these calculations, the staff has verified the licensee's identification of the limiting material for both units, and agrees with the licensee's calculated ART values.

The licensee used a computer program, "P/T Calculator for Windows," developed by Electric Power Research Institute to generate the heatup and cooldown P/T limits for both Point Beach units. As mentioned above in Section 3.2.2, the underlying methodology for the P/T limits is Appendix G in the 1996 Addenda to the 1995 Edition of Section XI of the ASME Code, as modified by Code Case N-641. The staff performed calculations using the formula based on membrane tension for the stress intensity factor due to pressure, K_{Im}, for the axial and circumferential flaws, the plane strain fracture toughness, K_{Ic}, (both are from Code Case N-641), and the formula based on heatup and cooldown rates for the stress intensity factor due to thermal gradient, K_{It} , (from the 1996 Addenda to the 1995 ASME Code), to verify the licensee's P/T limits. The licensee's approach of performing the analysis at 1/4T for the cooldown curves and at 3/4T for the heatup curves is in accordance with the ASME Code. However, the staff found that for both heatup and cooldown curves, the proposed P/T limits at high RPV pressure (e.g., 1400 psig) are more conservative than the P/T limits calculated by the staff, but the proposed P/T limits at low RPV pressure (e.g., 550 psig) are less conservative than the staff's P/T limits. The discrepancies between the licensee's and the staff's P/T limits at low pressure (the non-conservative part) are only 5 °F for cooldown curves and 8 °F for heatup curves at 550 psig. Since the licensee's K_{It} calculation was based on the more complex formula using the actual thermal stress profile calculated by the computer program, while the staff's calculation was based on the formula for a generic vessel using the heatup and cooldown rates, discrepancies of this magnitude are anticipated and the licensee's K_{It} results are acceptable. Hence, the staff determined that the licensee's proposed P/T limit curves meet the requirements of the ASME Code as modified by Code Case N-641. Although the limiting beltline material of Unit 2 has larger ART than that of Unit 1, Unit 1 P/T limits bound Unit 2 P/T limits because circumferential flaws (Unit 2) have much lower membrane stresses at the same pressure.

In addition to the beltline materials, Appendix G of 10 CFR Part 50 also imposes a minimum temperature at the closure head flange based on the reference temperature for the flange material. Section IV.A.2 of Appendix G states that when the pressure exceeds 20 percent of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120 °F for normal operation and by 90 °F for hydrostatic pressure tests and leak tests. Based on the flange RT_{NDT} of 60 °F for PBNP Units 1 and 2, the staff has determined that the vertical-line segments of the proposed P/T limits have satisfied the requirement for the closure flange region during normal operation and inservice leak and hydrostatic testing.

It should be noted that the November 20, 2000, supplement, did not revise fluence values reported in the original submittal dated March 10, 2000. It only revised the EFPY values based on the linear interpolation method using "calculated" projections of neutron fluence for the surveillance materials. This linear interpolation method is a standard approach and has been accepted by the staff for application in other P/T limit submittals.

3.3 LTOP Setpoints

The LTOP system uses the pressurizer power operated relief valves (PORVs) during startup or shutdown operation. By choosing proper PORV setpoints, the LTOP system is designed with the capability to automatically prevent the RCS from exceeding the P/T limits, and prevent the reactor vessel from being exposed to conditions of fast propagating brittle fracture. Once the system is enabled, no operator action is involved for the LTOP system to perform its intended pressure mitigation function. Modifications of the LTOP enable temperature and the PORV setpoints as a result of the changes in the P/T limits to extend reactor operation period are routinely made on nuclear plants.

The LTOP setpoint and enable temperature are determined to ensure that 10 CFR Part 50 Appendix G pressure limit is not exceeded. The Appendix G pressure limit is calculated based on the limiting projected ART at the 1/4T location for the limiting reactor vessel materials, subject to the fluence values at the projected EFPY. As discussed in Section 3.2.2., the most limiting postulated axial 1/4T flaw is in Unit 1 weld heat 61782 with a fluence of 1.049E19 n/cm^2 and ART of 209 °F; and the limiting postulated circumferential 1/4T flaw is in Unit 2 weld heat 72442 with a fluence value of 1.764E19 n/cm^2 and ART of 272 °F. The LTOP setpoints for both PBNP units are conservatively determined based on the limiting maximum allowable pressure of these two materials.

The maximum allowable pressure limits are calculated corresponding to the reactor vessel permissible membrane stress intensity, and corrected for the pressure instrument uncertainty and location bias in relation to the reactor vessel beltline. The maximum allowable pressure limit calculation is based on ASME Code Case N-641. Details of this Code Case are discussed in Section 3.2.1 of this safety evaluation (SE).

The LTOP pressure setpoint is selected to ensure that the maximum allowable pressure limits are not exceeded by the maximum attainable pressure during design-basis transients. The setpoint is determined in accordance with the Westinghouse Owners Group report, WCAP 14040-NP-A (Reference 1), with consideration of two types of design-basis transients: mass addition and heat addition transients. The setpoint calculation is performed assuming that the plant is operating at a minimum reactor pressure vessel metal temperature of 60 °F with two operating residual heat removal pumps, up to two operating reactor coolant pumps, and up to three operating charging pumps. For both PBNP units, an LTOP setpoint of 500 psig is evaluated to show that with the addition of pressure overshoots during the design-basis mass and heat addition transients, and the consideration of the instrumentation uncertainties and measurement location bias, the maximum attainable reactor coolant system (RCS) pressure during the transients would not exceed the maximum allowable pressure limits based on the most limiting reactor vessel materials.

The design-basis mass addition transient is analyzed assuming one high pressure injection pump and three charging pumps discharging to the RCS while the system is solid with pressure relieved by one PORV. The minimum RCS temperature is assumed to be 60 °F. The transient pressure overshoot calculation follows the methods described in a July 1977 Westinghouse Report (Reference 9). The pressure overshoot is calculated with a simplified interpolation equation, which was developed from the analysis of a reference plant using the LOFTRAN code for a range of relief valve setpoint, relief valve opening time, mass input rate, and RCS volume. The LOFTRAN calculations of coolant pressure transients assumed that the coolant was enclosed by a rigid, non-yielding boundary and that the pressure change was a direct result of the inability of the coolant to expand into a larger volume. The resulting pressure overshoot from the interpolation equation is corrected for the effect of the RCS volumetric expansion due to increasing RCS pressure. This method of analysis and the correction value had been reviewed and accepted in the NRC SE supporting License Amendments 45 and 50 for Units 1 and 2, respectively, dated May 20, 1980 (Reference 10). The results in Section V of Reference 11 show that, with the PORV setpoint of 500 psig, the pressure overshoot for the mass input transient is 107.5 psi, which results in the maximum attained pressure of 607.5 psig. The maximum allowable pressure is calculated to be 721.5 psig. Corrected for the instrumentation uncertainty and the measurement location bias, the maximum allowable indicated pressure is 629.2 psig. Since the maximum attained pressure for the mass addition transient is less than the maximum allowable indicated pressure, the LTOP setpoint of 500 psig is acceptable.

The design-basis heat addition transient assumes the starting of the first reactor coolant pump (RCP) during water solid conditions with a temperature difference between the RCS and the steam generator (SG) of 50 °F. Pressure is relieved by a single PORV. The pressure overshoot for the heat input transient is calculated with the information provided in the Supplement to the July 1977 Westinghouse Report (Reference 9). The pressure overshoot calculation is conservatively performed with the larger SG heat transfer area of Unit 2, a bounding RCS volume of 6000 ft^3 compared to the actual volume of 6148 ft^3, and a 3-second relief valve opening time. The analysis was performed for cases with initial RCS temperature of 100 °F, 180 °F and 250 °F. The results in Section VI of Reference 11 show the 100 °F initial RCS temperature case has the limiting maximum allowable pressure of 725.0 psig. Corrected for the location bias and instrument uncertainty, the maximum allowable indicated pressure is 641.7 psig. With the PORV setpoint of 500 psig, the pressure overshoot is determined to be 24 psi, which results in the maximum attained RCS pressure of 524 psig. Since the maximum attained pressure is below the 641.7 psig, the LTOP setpoint of 500 psig is acceptable.

The actuation of a PORV during an increasing pressure transient will result in the RCS pressure increase being slowed and reversed. The RCS pressure will then decrease, and continue to undershoot below the PORV reset pressure as the valve recloses. This pressure undershoot is typically considered when establishing the PORV setpoint for the protection of the RCP #1 seal, i.e., to ensure that, as a result of pressure undershoot, the RCS boundary remains above the minimum pressure required to maintain a nominal differential pressure across the seal faces for proper film-riding performance. Since the proposed LTOP setpoint of 500 psig is higher than the current setpoint of 440 psig, it is acceptable with respect to the aspect of pump seal operability.

3.4 <u>Enable Temperature</u>

The LTOP enable temperature is determined based on the limiting axial and circumferential ART for the two units. According to ASME Code Case N-641, the LTOP enable temperature may be determined as the greater of a RCS temperature of 200 °F or a coolant temperature corresponding to a reactor vessel metal temperature of at least ART + tau at the 1/4T beltline location, when tau can be calculated on a plant-specific basis by a formula delineated in Code Case N-641. The enable temperature is also adjusted for the maximum 1/4T temperature lag of 20.1 °F at a 100 °F/hr heatup rate and a temperature instrument uncertainty of 17.8 °F. The calculation results in Section VII of Reference 11 show that the enable temperature is 270 °F for Unit 1, and 224.9 °F for Unit 2. Therefore, the licensee proposed LTOP enable temperature of 270 °F for both units is the higher value and is acceptable.

4.0 CONCLUSIONS

Based upon the staff evaluations, as discussed in Section 3.0 above, the NRC staff concludes that it is acceptable for NMC to relocate the P/T limits and LTOP setpoints from the PBNP Units 1 and 2 TS to a licensee-controlled PTLR. The relocation of the limits from TS and the associated TS changes will be reviewed under a separate action, a future license amendment. The proposed heatup, cooldown and hydrostatic testing curves will expire at 25.59 EFPYs for Unit 1, and at 30.51 EFPYs for Unit 2.

The staff has reviewed the proposed fluence values for PBNP and finds that the requirements of GL 96-03 are satisfied, and the proposed values are acceptable. The staff finds the licensee’s short-term solution for fluence values acceptable provided that: (1) the current limits are not extended beyond October 30, 2003, for Unit 1, and October 1, 2008, for Unit 2, (2) the proposed PTLR methodology is acceptable provided that the FERRET code is excluded, (3) the FERRET code will receive staff approval before it is referenced in the PTLR, and (4) the licensee submits for staff review PTLR modifications for the inclusion of the FERRET (or any other) code.

The staff has determined that the proposed P/T limits satisfy the requirements in Appendix G to Section XI of the ASME Code, as modified by Code Case N-641, and Appendix G to 10 CFR Part 50 for 25.59 EFPYs for Unit 1, and 30.51 EFPYs for Unit 2. The proposed P/T limits also satisfy GL 88-11, because the method in RG 1.99, Revision 2 was used to calculate the ART.

The staff has determined that the proposed PTLR meets the criteria of GL 96-03, and is acceptable to the staff.

5.0 REFERENCES

1. WCAP-14040-NP-A, Revision 2, Westinghouse Electric Corporation, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,” January 15, 1996.

2. NRC Generic Letter 96-03, “Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits,” January 31, 1996.

3. Letter from Mark E. Reddemann, Wisconsin Electric, to NRC Document Control Desk, “Technical Specification Change Request 219, Adoption of Pressure and Temperature

Limits Report and Revised P/T and LTOP Limits, Point Beach Nuclear Plant, Units 1 and 2," March 10, 2000.

4. Letter from Mark E. Reddemann, Wisconsin Electric, to NRC Document Control Desk, "Response to Request for Additional Information; Technical Specification Change Request 219, Adoption of Pressure and Temperature Limits Report and Revised P/T and LTOP Limits (Tac Nos. MA8459 and MA8460), Point Beach Nuclear Plant, Units 1 and 2," July 28, 2000.

5. Letter from Mark E. Reddemann, Nuclear Management Company, LLC, to NRC Document Control Desk, "Response to Request for Additional Information; Supplement 1 to Technical Specification Change Request 219, Adoption of Pressure and Temperature Limits Report and Revised P/T and LTOP Limits, Point Beach Nuclear Plant, Units 1 and 2 (Tac Nos. MA8459 and MA8460)," November 20, 2000.

6. Letter from Mark E. Reddemann, Nuclear Management Company, LLC, to NRC Document Control Desk, "Response to Request for Additional Information; Supplement 2 to Technical Specification Change Request 219, Adoption of Pressure and Temperature Limits Report and Revised P/T and LTOP Limits, Point Beach Nuclear Plant, Units 1 and 2 (Tac Nos. MA8459 and MA8460)," April 10, 2001.

7. WCAP-12794, Revision 4, "Reactor Cavity Neutron Measurement Program for Wisconsin Electric Power Company Point Beach Unit 1," by S. Anderson, Westinghouse Electric Corporation, February, 2000.

8. WCAP-12795, Revision 3, "Reactor Cavity Neutron Measurement Program for Wisconsin Electric Power Company Point Beach Unit 2," by S. Anderson, Westinghouse Electric Corporation, August 1995.

9. Westinghouse Report, "Pressure Mitigating Systems Transient Analysis Results," July 1977, and Supplement to the July 1977 Report, September 1977.

10. Letter from R.A. Clark (USNRC) to S. Burnstein (Wisconsin Electric Power Company), Docket Nos. 50-266 and 50-301, May 20, 1980.

11. Calculation 2000-0001-00, "RCS Pressure-Temperature Limits and LTOP Setpoints Applicable Through 32.2 EFPY - Unit 1 and 34.0 EFPY - Unit 2," Attachment 5 to March 10, 2000, letter from Mark E. Reddemann.

12. NRC Letter, "Point Beach Nuclear Plant, Units 1 and 2 - Exemption from the Requirements of 10 CFR 50.60 (TAC NOS. MA9680 and MA9681)," dated October 6, 2000.