NMC
Committed to Nuclear Excellence
Point Beach Nuclear Plant
Operated by Nuclear Management Company, LLC

NRC 2003-0091

10 CFR 50.67
10 CFR 50.90

September 26, 2003

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2
DOCKETS 50-266 AND 50-301
REQUEST FOR WITHDRAWAL, AND RESUBMITTAL, OF REQUEST FOR EXEMPTIONS TO 10 CFR 50.61, APPENDICES G AND H TO 10 CFR 50, APPROVAL OF PTS APPLICATION FOR PBNP UNIT 2 (TAC NOS. MB7926 AND MB7927) AND WITHDRAWAL OF ASSOCIATED LICENSE AMENDMENT REQUEST (LAR) 235 AND SUBMITTAL OF ASSOCIATED LAR 236

Reference:
1) Letter from NMC to NRC dated March 3, 2003 (NRC 2003-0018)
2) Letter from NMC to NRC dated June 27, 2003 (NRC 2003-0051)
3) Letter from NMC to NRC dated June 27, 2003 (NRC 2003-0058)
4) Letter from NRC to NMC dated August 8, 2001 (Amendments 201 and 206)
5) BAW-2308, Revision 1, Initial RT_{NDT} of Linde 80 Weld Materials, August 2003

In reference 1, Nuclear Management Company, LLC (NMC), submitted a request for permanent exemption from certain requirements of 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Thermal Shock Events", 10 CFR 50, Appendix G, "Fracture Toughness Requirements", and 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements", for the Unit 2 reactor vessel. Reference 1 also provided new pressure/temperature (P/T) curves at end of life (EOL) for both Units 1 and 2 (WCAP-15976, Revision 0, "Point Beach Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation", dated February 2003).

Reference 2 provided additional information in support of the exemption request. Reference 3 requested a supporting change to Point Beach Nuclear Plant (PBNP) Technical Specification (TS) 5.6.5, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)", to reference the new analysis methodology.

As discussed during a conference call between NRC staff and NMC personnel on July 9, 2003, the NRC staff is in the later stages of reviewing a new RT_{NDT} methodology contained in report BAW-2308, Revision 1 (reference 5). A PBNP specific calculation using the methodology of BAW-2308, Revision 0, was included in reference 1. We believe that the methodology contained in BAW-2308, Revision 1 is preferable to that requested by NMC in reference 1. NMC intends to pursue this new methodology and we therefore ask that the requests submitted in references 1 and 3 be withdrawn and replaced by the requests provided herein.

NMC plans to use the P/T curves contained in WCAP-15976, Revision 0, "Point Beach Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation", dated February 2003. A copy

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of WCAP-15976, which included pressure and temperature limit curves, was provided to NRC in reference 1. The P/T curves provided in reference 1 are not dependent on the new methodology submitted to NRC staff as part of that submittal. The new P/T curves contained in reference 1 were developed using the currently approved methodology referenced in PBNP TS 5.6.5.b and satisfy the requirement of PBNP TS 5.6.5.c. As such, the TS change that had been requested in reference 3 is not needed coincident with these P/T curves.

It is noted that although the Master Curve toughness application for the PBNP 2 limiting circumferential weld has been considered in the P/T curve evaluation, there is no effect on the resulting P/T curves since the limiting materials for these curves are the intermediate and lower shell axial welds from PBNP Unit 1.

The NRC safety evaluation provided in reference 4 stated that the currently calculated fluence values for operation of Unit 1 are valid until October 30, 2003 and are valid for Unit 2 until October 1, 2008. However, the revised P/T curves provided in reference 1 were developed by using the existing approved methodology to extend the time limit for the fluence values to end-of-life. Therefore, these P/T curves can be implemented without additional NRC approval.

In order to adopt the BAW-2308, Revision 1, methodology and in accordance with the provisions of 10 CFR 50.12, "Specific Exemptions", NMC is herein submitting a request for permanent exemption from certain requirements of 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Thermal Shock Events", 10 CFR 50, Appendix G, "Fracture Toughness Requirements", and 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements", for the Unit 2 reactor vessel. Attachment I to this letter provides the justification for this exemption request.

The requested exemptions would allow use of a different method, the Master Curve Methodology as described in Report BAW-2308, Revision 1, for determining the adjusted RT_{NDT} (reference nil-ductility temperature) of the Point Beach Nuclear Plant Unit 2 (PBNP 2) reactor vessel limiting circumferential weld metal. This method is used for the Pressurized Thermal Shock (PTS) screening evaluation.

In association with this request for exemptions, NMC has reassessed PBNP Unit 2 compliance with 10 CFR 50.61. The new PTS evaluation was provided in ATI Consulting Report 021-030-2003-1, "Master Curve Fracture Toughness Application for Point Beach Nuclear Plant Unit 2", dated January 2003. This evaluation contained a PTS evaluation using the methodology contained in BAW-2308, Revision 0. A copy of the ATI Consulting report was previously provided to NRC in reference 1. A plant-specific calculation of RT_{PTS} was performed using the revised methodology of BAW-2308, Revision 1. This calculation (Framatome ANP Calculation 32-5019743-01 PBNP Unit 2 power Uprate PTS Evaluation 53 EFPY) was conservatively performed for 53 EFPY and considered full power uprate conditions. This calculation is enclosed.

NMC hereby requests the NRC to review and approve the PTS evaluation and its associated methodology for Unit 2.

In association with the request for exemptions to support the associated methodology, a revised reference in PBNP Technical Specifications (TS) 5.6.5, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)", is needed. Therefore, in accordance with 10 CFR 50.90, NMC is submitting this request for an amendment to the TS for PBNP, Units 1

and 2. The proposed amendment would modify TS 5.6.5, to add a reference to the issuance date of the NRC safety evaluation accepting the new Master Curve Methodology for Unit 2.

This amendment request is consistent with changes made to NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Technical Specification Task Force TSTF- 419, Revision 0, "Revise PTLR Definition and References in ISTS 5.6.6, RCS PTLR".

Attachment II to this letter provides a description, justification, and a significant hazards determination for the proposed change. Attachment III provides the existing Technical Specifications page marked up to show the proposed change. Attachment IV provides the revised (clean) Technical Specifications page.

NMC requests approval of the above requested exemptions and the proposed license amendment by March 2004, with the amendment being implemented within 60 days.

This application contains three new commitments that are listed in Attachment I.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated Wisconsin Official.

I declare under penalty of perjury that the foregoing is true and accurate. Executed on September 26, 2003.

A. J. Cayia
Site Vice-President

LAS/kmd

Attachments:
- I Justification for Exemption
- II Description and Assessment of Change
- III Proposed Technical Specification Change
- IV Revised Technical Specification Pages

Enclosure
Framatome ANP Calculation 32-5019743-01, PBNP Unit 2 Power Uprate PTS Evaluation, 53 EFPY

cc: Regional Administrator, Region III, USNRC
Project Manager, Point Beach Nuclear Plant, NRR, USNRC
NRC Resident Inspector - Point Beach Nuclear Plant
PSCW

JUSTIFICATION FOR

EXEMPTION REQUEST

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

1.0 INTRODUCTION

In accordance with the provisions of 10 CFR 50.12, "Specific Exemptions", Nuclear Management Company, LLC (NMC) is submitting a request for exemption from certain requirements of 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Thermal Shock Events", 10 CFR 50, Appendix G, "Fracture Toughness Requirements", and 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements". The requested exemptions would allow a different method, the Master Curve (BAW-2308) Methodology, for determining the adjusted RT_{NDT} (reference nil-ductility temperature) of the Point Beach Nuclear Plant Unit 2 (PBNP 2) reactor vessel limiting circumferential weld metal. This method is used for the Pressurized Thermal Shock (PTS) screening evaluation.

2.0 BACKGROUND

10 CFR 50.60 and 10 CFR 50.61 establish criteria that ensure that each reactor vessel has adequate fracture toughness. When these rules were first promulgated fracture toughness specimens were too large to be used in reactor vessel radiation surveillance capsule programs. Therefore, smaller Charpy V-notch specimens were used to estimate and monitor fracture toughness.

The latest Charpy-based toughness evaluation following current regulations for the Point Beach Nuclear Plant, Unit 2 (PBNP 2) limiting circumferential weld metal indicates that the projected value of RT_{PTS} at end-of-license (EOL) is close to (but below) the pressurized thermal shock (PTS) screening criterion of 300°F (10 CFR 50.61). For bounding and evaluation purposes, conditions at end of license extended (EOLE) were projected. For the projected EOLE fluence, which assumes removal of hafnium fluence suppression and planned power up-rates, the PTS screening criteria will be exceeded. Therefore, NMC performed an evaluation of Master Curve fracture toughness data, using the methodology in BAW-2308, Revision 1, for assuring reactor pressure vessel (RPV) integrity for PBNP 2 out to EOLE.

This document summarizes the technical basis and justifications for the exemption requests to use ASME Code Cases N-629 and N-631, ASTM E185-98, ASTM E-1921-02, and the methodology described in ATI Consulting Report 021-030-2003-1, BAW-2308, Revision 1, and Framatome ANP Calculation 32-5019743-01 for establishing reference temperature values for assessment of the integrity of the PBNP 2 reactor vessel.

3.0 PROPOSED EXEMPTIONS

The three exemptions requested by NMC address portions of the following regulations:

(1) Appendix G to 10 CFR Part 50, which sets forth fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the system may be subjected over its service lifetime;

(2) 10 CFR 50.61, which sets forth fracture toughness requirements for protection against pressurized thermal shock (PTS); and,

(3) Appendix H to 10 CFR Part 50, which requires the establishment of a RPV material surveillance program.

The exemption from Appendix G to 10 CFR 50 is to replace the required use of the existing Charpy V-notch and drop-weight-based methodology and allow the use of an alternate methodology to incorporate the use of fracture toughness test data for evaluating the integrity of the PBNP 2 circumferential beltline weld based on use of the 2002 Edition of American Society for Testing and Materials (ASTM) Standard Method E 1921 (E-1921-02) and American Society for Mechanical Engineering (ASME) Code Cases N-629 and N-631. The exemption is required since Appendix G to Section XI of the ASME Code pursuant to 10 CFR 50.55(a) requires the use of a methodology based on Charpy V-notch and drop weight data.

The exemption from 10 CFR 50.61 is to use an alternate methodology to allow the use of fracture toughness test data for evaluating the integrity of the PBNP 2 limiting circumferential beltline weld based on the use of ASTM E 1921-02 and ASME Code Case N-629. The exemption is required because the methodology for evaluating RPV material fracture toughness in 10 CFR 50.61 requires the use of Charpy V-notch and drop weight data for establishing the PTS reference temperature (RT_{PTS}).

Finally, the exemption from Appendix H to 10 CFR 50 is to modify the basis for the PBNP 2 surveillance program to allow the acquisition and use of fracture toughness data instead of the Charpy V-notch impact testing required by Appendix H to 10 CFR 50. The exemption is required because Appendix H to 10 CFR 50 does not address the testing of surveillance specimens for direct measurement of fracture toughness. A second reason for the exemption relates to a supplemental surveillance capsule. Due to the need for additional fracture toughness data for the PBNP 2 weld metal at fluence levels extending out to EOLE, a supplemental capsule has been added to the surveillance program for PBNP 2. This capsule has been installed in the highest lead factor location and includes other RPV beltline materials. The capsule is designed for Master Curve fracture toughness testing and evaluation at the projected EOLE fluence, so that the integrity of the RPV will be directly validated with the testing of this capsule. The composition of materials, specimen types, and estimated schedule for removal of this new capsule were addressed in ATI Consulting Report 021-030-2003-1.

A tabular summary of the requested exemptions and the proposed alternatives are shown below.

Description	Existing Requirement	Proposed Alternative
Determination of adjusted/indexing reference temperatures	10 CFR 50.61 and Appendix G to 10 CFR Part 50	ASME Code Case N-629, ASME Code Case N-631, ATI Consulting Report 021-030-2003-1, BAW-2308 Revision 1, and Framatome ANP Calculation 32-5019743-01
Use of the latest edition of ASTM E185-98	App H to Part 50 specifies use of ASTM E185-73,-79, -82 for testing of surveillance materials	(1) ASTM E185-98 allows use of ASTM E1921-02 for testing of surveillance capsule material; (2) Use fracture toughness surveillance data from PBNP 2 supplemental surveillance capsule for verification of EOLE toughness properties
Alternative testing methods for determination of fracture toughness	Appendices G and H to Part 50 specifies Charpy V-Notch impact and drop weight testing	ASTM E1921-02, ATI Consulting Report 021-030-2003-1

4.0 TECHNICAL ANALYSIS

ATI Consulting Report 021-030-2003-1 and BAW-2308, Revision 1, provide the detailed technical analysis and basis for the Master Curve application at PBNP 2. A general summary is presented here.

The PBNP 2 RPV limiting weld metal heat 72442 was not included in the current surveillance program for PBNP 2, but it was irradiated as part of the B&W Owners Group integrated surveillance program. The latest projections based on Charpy impact testing, when analyzed following NRC guidelines and rules, indicate that this weld will reach the PTS screening criterion limit before EOLE. Therefore, fracture toughness testing of other irradiated surveillance specimens (from two different welds fabricated using weld wire 72442) has been performed and analyzed using the Master Curve methodology following ASME Code Cases N-629 and N-631. The evaluation performed involves extrapolation to EOLE fluences and shows that the RPV limiting weld metal has more than adequate toughness for operation out to EOLE and beyond. These projections will be confirmed by additional testing of weld heat 72442 from the B&W Owners Group Master Integrated Reactor Vessel Materials Surveillance Program (MIRVP) prior to reaching the EOL fluence at PBNP 2. A supplemental surveillance program also has been designed and initiated at PBNP 2 that includes the limiting weld metal for future evaluation using the Master Curve methodology. The testing of this supplemental capsule at a fluence corresponding to EOLE will confirm the toughness condition for the PBNP 2 RPV weld at about 38 EFPY, which is well before EOLE is reached.

Conclusion

Use of the Master Curve methodology, extrapolated to EOLE fluences, shows that the RPV limiting weld metal meets PTS screening criteria out to EOLE and beyond. These projections will be confirmed by additional testing of weld heat 72442 from the B&W Owners Group MIRVP prior to reaching the EOL fluence at PBNP 2. A supplemental surveillance program has been initiated at PBNP 2 that includes the insertion of an additional surveillance capsule containing limiting weld metal for future evaluation using the Master Curve methodology. This surveillance program will be fully implemented. Testing of the supplemental capsule at a fluence corresponding to EOLE will confirm the toughness condition for the PBNP 2 RPV weld at about 38 EFPY, which is well before EOLE is reached.

5.0 REGULATORY ANALYSIS

5.1 No Significant Impact Determination and Environmental Evaluation

In accordance with the provisions of 10 CFR 50.12, "Specific Exemptions", Nuclear Management Company, LLC (NMC) is submitting a request for exemption from certain requirements of 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Thermal Shock Events", 10 CFR 50, Appendix G, "Fracture Toughness Requirements", and 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements". The requested exemptions would allow a different method, the Master Curve (BAW-2308) Methodology, for determining the adjusted RT_{NDT} (reference nil-ductility temperature) of the Point Beach Nuclear Plant Unit 2 (PBNP 2) reactor vessel limiting circumferential weld metal. This method is used for the Pressurized Thermal Shock (PTS) screening evaluation.

NMC has evaluated the proposed exemptions against the criteria in 10 CFR 51.32 and has determined that the operation of PBNP in accordance with the proposed exemptions presents no significant impact. Use of the Master Curve (BAW-2308) Methodology shows that the RPV limiting weld metal meets PTS screening criteria. The underlying purpose of 10 CFR 50.61 and Appendices G and H to 10 CFR 50, which is to establish criteria that ensure that each reactor vessel has adequate fracture toughness, continues to be achieved.

Operation of PBNP in accordance with the proposed exemptions will not significantly increase the probability or consequences of accidents, no changes are being made in the types of effluents that may be released off site, and there is no significant increase in occupational or public radiation exposure. Therefore, operation of PBNP in accordance with the proposed exemptions does not result in any significant radiological environmental impacts.

With regard to potential nonradiological impacts, the proposed action does not have a potential to affect any historic sites. It does not affect nonradiological plant effluents and has no other environmental impact. Therefore, there are no significant nonradiological environmental impacts associated with the proposed action.

Conclusion

Since there are no significant radiological or nonradiological environmental impacts associated with the proposed action, we conclude that the proposed exemptions will not have a significant effect on the quality of the human environment. Therefore, as provided in 10 CFR 51.32, an environmental impact statement need not be prepared.

The following observations and conclusions were documented in ATI Consulting Report 021-030-2003-1 for the PBNP 2 limiting beltline weld metal:

- The latest Charpy-based toughness evaluation following current regulation for the PBNP 2 limiting circumferential weld metal indicates that the PTS screening criterion of 300°F will be reached before EOLE when future plant operation is considered (removal of hafnium fluence suppression and planned power up-rates).

- Application of the Master Curve methodology for the PBNP 2 weld metal requires extrapolation (from the three available surveillance irradiations) to the RPV EOLE fluence. The extrapolation can be performed following several different approaches. Three approaches were evaluated: (1) use of measured initial RT_{To} and adding Charpy shift (BAW-2308 methodology); (2) use of measured initial RT_{To} and adding the shift in RT_{To} due to irradiation; and, (3) use of the measured irradiated RT_{To} values directly without projection from zero fluence. All methods show that the EOLE RT_{PTS} value is less than the PTS screening limit of 300°F. Method 1 somewhat follows the current regulatory practice and is conservative. Method 2 was evaluated following the Kewaunee SE, and the resulting projections in ART were substantially less than Method 1. Method 3 is the most accurate method, and the results obtained applying this direct measurement approach reveal that Method 2 is quite conservative.

- The Margin term was chosen depending upon the analysis approach discussed above. For Method 1, Margin was based on three uncertainties: material variability based on a Monte Carlo study from BAW-2308 of weld heat 72442 non-irradiated data (σ_{MC} = 9.3°F), the uncertainty in determining T_o from ASTM E 1921-02 (σ_{To} = 7.4°F), and the current regulatory value for weld metal Charpy shift (σ_{Δ} = 28°F); σ_{MC} and σ_{To} are combined to give a measure of the uncertainty in initial properties (σ_I = 11.9°F). Method 2 used the Margin specified by the NRC in the Kewaunee SE, which used a larger σ_I (14°F) and the same σ_{Δ} of 28°F. Method 3 used a more complete uncertainty analysis: material variability (σ_{MC} = 9.3°F as above), determination of irradiated T_o (σ_{To} = 10.7°F), Cu content (σ_{Cu} = 1.6-1.7°F), Ni content (σ_{Ni} = 4.1-4.2°F), irradiation temperature (σ_{Tirr} = 6.9-8.9°F), fluence ($\sigma_{\phi t}$ = 13.2-12.5°F), and fluence projection (σ_{Proj} = 1.0-1.6°F). Remaining consistent with industry practice, an approximate 95% statistical level (or two sigma) Margin was chosen, where the individual uncertainties were combined as the square root sum of the squares.

- Since there was a need to extrapolate to higher fluence levels (higher than where current fracture toughness measurements exist) to assess PTS and pressure-temperature operating curves, the current Regulatory fluence function for CVN-based predictions was used for the Master Curve approach.

- The supplemental surveillance program utilizes irradiation of the limiting weld metal heat in a new capsule that will be available for testing near the time corresponding to 38 EFPY for the RPV. The direct measurement of fracture toughness for key weld metal will be evaluated at a fluence near to the projected EOLE. Fracture toughness data from the B&W Owners Group on this same weld metal will be available around 2008. This B&W Owners Group data should correspond closely to the PBNP 2 EOL fluence for the limiting RPV weld.

5.2 Commitments

The following table identifies those actions committed to by NMC in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

COMMITMENTS	Due Date/Event
NMC will confirm the projections for the RPV limiting weld metal by additional testing of weld heat 72442 from the B&W Owners Group Master Integrated Reactor Vessel Materials Surveillance Program.	Prior to reaching the EOL fluence at PBNP Unit 2 06/2010
NMC will implement a supplemental surveillance program at PBNP Unit 2 that includes the limiting weld metal for future evaluation using the Master Curve methodology.	02/2006
Should extended operation be considered, NMC will test the supplemental capsule at a fluence corresponding to EOLE to confirm the toughness condition for the PBNP Unit 2 RPV weld at about 38 EFPY, which is well before EOLE is reached.	Should extended operation be considered

5.3 Applicable Regulatory Requirements

10 CFR 50.12(a) states that the Commission may grant exemptions from the requirements of the regulations contained in 10 CFR 50 that are:

(1) authorized by law;
(2) will not present an undue risk to the public health and safety;
(3) consistent with the common defense and security; and,
(4) special circumstances, as listed in 10 CFR 50.12(a)(2), are present.

This exemption request meets the criteria set forth in 10 CFR 50.12, as discussed herein. Additional technical bases for the proposed exemptions were provided to NRC in reference 1.

1. <u>The requested exemption is authorized by law.</u>

   No law exists which precludes the activities covered by this exemption request. 10 CFR 50.60(b) allows the use of alternatives to 10 CFR 50, Appendices G and H when an exemption is granted by the Commission under 10 CFR 50.12.

2. The requested exemption does not present an undue risk to the public health and safety.

   10 CFR 50 Appendices G and H specify that surveillance capsules shall be tested in accordance with ASTM E 185-73, -79, and -82. The latest version of ASTM E-185-98 encourages that supplemental fracture toughness testing be conducted in accordance with procedures and requirements of Practice E636, Method E-1820, or Method E-1921 when the surveillance materials exhibit marginal properties. Fracture toughness testing of weld metal heat 72442 has been performed to satisfy the requirements established in accordance with ASTM E-1921.

   The use of this proposed approach ensures that the intent of the requirements specified in 10 CFR 50.61 is satisfied. Therefore, these exemptions do not present an undue risk to the public health and safety.

3. The requested exemption is consistent with the common defense and security.

   The common defense and security are not endangered by this exemption request.

4. Special circumstances are present which necessitate the request for an exemption to the regulations of 10 CFR 50.61.

   Pursuant to 10 CFR 50.12(a)(2), the Commission will consider granting an exemption to the regulations of 10 CFR 50 if special circumstances are present. This exemption request meets the special circumstances described in 10 CFR 50.12(a)(2)(ii):

   > Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule.

   The underlying purpose of 10 CFR 50.61 and Appendix G to 10 CFR 50 is to establish criteria that ensure that each reactor vessel has adequate fracture toughness. Westinghouse Electric Corporation, Framatome ANP, and ATI Consulting prepared the reports to assess and document the integrity of the PBNP 2 reactor vessel relative to the requirements and underlying purpose of 10 CFR 50.61, and Appendices G and H to 10 CFR 50. These reports provided the technical justification for the exemption requests. ATI Consulting Report 021-030-2003-1 provided an updated PTS evaluation using Master Curve methodology for PBNP Unit 2, showing compliance with PTS screening criteria through EOLE; a supplemental surveillance program to validate EOLE fracture toughness properties was also described. Framatome ANP submitted B&W Owners Group, Reactor Vessel Working Group, Topical Report BAW-2308, Revision 1, for NRC review and approval. Together, these reports demonstrate that the alternate methodology (Master Curve methodology) achieves the underlying purposes of the regulatory rules from which exemptions are requested; therefore, the exemptions being requested are justified.

DESCRIPTION AND ASSESSMENT OF CHANGE

LICENSE AMENDMENT REQUEST 236

TECHNICAL SPECIFICATION LCO 5.6.5, "REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)"

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

1.0 DESCRIPTION

This proposed License Amendment Request (LAR) is made pursuant to 10 CFR 50.90 to modify Technical Specifications (TS) 5.6.5, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)".

2.0 BACKGROUND

NMC is requesting that the PBNP TS be modified consistent with the changes described in the staff approved Industry/TSTF Standard TS Change, TSTF-419 Revision 0 (TSTF-419), "Revise PTLR Definition and References in ITS 5.6.6, RCS PTLR". TSTF-419 was approved by the NRC on March 21, 2002.

NRC Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits", dated January 31, 1996, allows licensees to relocate the pressure temperature (P/T) limit curves from their plant TS to a PTLR or a similar document. The methodology used to determine the P/T and LTOP system limit parameters must comply with the specific requirements of Appendices G and H to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR), be documented in an NRC approved topical report or an NRC approved plant-specific submittal, and be incorporated by reference into the TS. Subsequent changes in the methodology must be approved by a license amendment.

3.0 DESCRIPTION OF CHANGE

Summary of Proposed Technical Specifications Change:

NMC proposes to modify TS 5.6.5, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)", to add a reference to the issuance date of the NRC safety evaluation accepting the new Master Curve Methodology for PBNP Unit 2. The formatting of the specification is also being revised to comport to TSTF-419R0.

4.0 ASSESSMENT

This amendment request is consistent with TSTF-419R0, meeting the requirements in TS 5.6.5, to identify either the Topical Report(s) by number and title or the NRC Safety Evaluation for a plant specific methodology by NRC letter and date. The NRC letters currently referenced in TS 5.6.5 are those dated October 6, 2000 and July 23, 2001. For PBNP Unit 2 only, this proposed amendment would add a reference to the NRC letter that will be issued accepting the new Master Curve Methodology. The NRC exemption dated October 6, 2000 and NRC safety evaluation dated July 23, 2001, will continue to remain applicable to both PBNP Units.

The details of the proposed change and the justification for the finding of no significant hazards are provided in this attachment. Attachments III and IV provide the revisions to the affected TS pages.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Determination

In accordance with the requirements of 10 CFR 50.90, Nuclear Management Company, LLC (NMC) (licensee) hereby requests amendments to facility operating licenses DPR-24 and DPR-27, for Point Beach Nuclear Plant (PBNP), Units 1 and 2, respectively. The purpose of the proposed amendments is to revise TS 5.6.5, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)", to add the reference of the NRC Safety Evaluation letter accepting the new Master Curve Methodology for PBNP Unit 2.

NMC has evaluated the proposed amendment in accordance with 10 CFR 50.91 against the standards in 10 CFR 50.92 and has determined that the operation of PBNP in accordance with the proposed amendments presents no significant hazards. Our evaluation against each of the criteria in 10 CFR 50.92 follows.

1. **Operation of PBNP in accordance with the proposed amendments does not result in a significant increase in the probability or consequences of any accident previously evaluated.**

   The proposed change references the NRC safety evaluation accepting the new Master Curve Methodology used in the evaluation of the revised P/T limits and LTOP setpoints. Implementation of revisions to Topical Reports would still be reviewed in accordance with 10 CFR 50.59 and, where required, receive NRC review and approval. The proposed change does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility or the manner in which the plant is operated and maintained. The proposed change does not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed change does not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Further, the proposed change does not increase the types or amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposures. The proposed change is consistent with safety analysis assumptions and resultant consequences. Therefore, it is concluded that this change does not increase the probability of occurrence of an accident previously evaluated.

2. **Operation of PBNP in accordance with the proposed amendments does not result in a new or different kind of accident from any accident previously evaluated.**

   The proposed change references the NRC safety evaluation accepting the new Master Curve Methodology used in the evaluation of the revised P/T limits and LTOP setpoints. Implementation of revisions to Topical Reports would still be reviewed in accordance with 10 CFR 50.59 and, where required, receive NRC review and approval. The change does not involve a physical alteration of the plant (i.e., no new

or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements or eliminate any existing requirements. The changes do not alter assumptions made in the safety analysis. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Operation of PBNP in accordance with the proposed amendments does not result in a significant reduction in a margin of safety.**

   The proposed change references the NRC safety evaluation accepting the new Master Curve Methodology used in the evaluation of the revised P/T limits and LTOP setpoints. Implementation of revisions to Topical Reports would still be reviewed in accordance with 10 CFR 50.59 and, where required, receive NRC review and approval. The proposed change does not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The setpoints at which protective actions are initiated are not altered by the proposed changes. Sufficient equipment remains available to actuate upon demand for the purpose of mitigating an analyzed event.

Conclusion

Operation of PBNP in accordance with the proposed amendment will not result in a significant increase in the probability or consequences of any accident previously analyzed; will not result in a new or different kind of accident from any accident previously analyzed; and, does not result in a significant reduction in any margin of safety. Therefore, operation of PBNP in accordance with the proposed amendment does not result in a significant hazards determination.

5.2 Applicable Regulatory Requirements

Administrative Controls, per 10 CFR 50.36(c)(5), are "...the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner." The technical analysis performed by NMC concludes that the proposed changes to TS 5.6.5 will continue to assure operation of the facility in a safe manner.

NMC concludes that the proposed changes are in accordance with 10 CFR 50.36(c)(5) with regards to use of approved analytical methods and providing the necessary reporting requirements. The proposed changes thus continue to be compliant with the above regulatory requirements.

5.3 Commitments

There are no actions committed to by NMC in this LAR. Any statements in this submittal are provided for information purposes and are not considered to be commitments.

6.0 ENVIRONMENTAL EVALUATION

NMC has determined that the information for the proposed amendment does not involve a significant hazards consideration, authorize a significant change in the types or total amounts of effluent release, or result in any significant increase in individual or cumulative occupational radiation exposure.

Accordingly, this proposed amendment meets 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 this proposed amendment.

PROPOSED TECHNICAL SPECIFICATION CHANGES

LICENSE AMENDMENT REQUEST 236

TECHNICAL SPECIFICATION LCO 5.6.5, "REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)"

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

MARKUP TS PAGE:

TS 5.6-5

5.6 Reporting Requirements

5.6.5 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, hydrostatic testing, LTOP enabling, and PORV lift settings as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

(1) LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits"
(2) LCO 3.4.6, "RCS Loops-MODE 4"
(3) LCO 3.4.7, "RCS Loops-MODE 5, Loops Filled"
(4) LCO 3.4.10, "Pressurizer Safety Valves"
(5) LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP)"

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

(1) NRC Letter~~s~~ dated October 6, 2000 ~~and~~
(2) NRC Letter dated July 23, 2001
(3) NRC Letter dated [, 2004] (Unit 2 only)

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

5.6.6 PAM Report

When a report is required by Condition B or G of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Tendon Surveillance Report

Abnormal conditions observed during testing will be evaluated to determine the effect of such conditions on containment structural integrity. This evaluation should be completed within 30 days of the identification of the condition. Any condition which is determined in this evaluation to have a significant adverse effect on containment structural integrity will be considered an abnormal degradation of the containment structure.

PROPOSED TECHNICAL SPECIFICATION CHANGES

LICENSE AMENDMENT REQUEST 236

TECHNICAL SPECIFICATION LCO 5.6.5, "REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)"

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

REVISED TS PAGE:

TS 5.6-5

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ENCLOSURE

TO

NRC 2003-0091

REQUEST FOR EXEMPTIONS TO 10 CFR 50.61, APPENDICES G AND H TO 10 CFR 50, APPROVAL OF PTS APPLICATION FOR PBNP UNIT 2 AND SUBMITTAL OF ASSOCIATED LICENSE AMENDMENT REQUEST 236

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

Framatome ANP Calculation 32-5019743-01,
PBNP Unit 2 Power Uprate PTS Evaluation, 53 EFPY

FRAMATOME ANP

CALCULATION SUMMARY SHEET (CSS)

Document Identifier 32 - 5019743 - 01

Title POINT BEACH UNIT 2 POWER UPRATE PTS EVALUATION 53 EFPY

PREPARED BY:		REVIEWED BY:	
		METHOD: ☒ DETAILED CHECK ☐ INDEPENDENT CALCULATION	
NAME J. B. HALL		NAME K. K. YOON	
SIGNATURE JB Hall		SIGNATURE K.K. Yoon	
TITLE PRINCIPAL ENG	DATE 8-19-03	TITLE TECH CONSULTANT	DATE 8-19-03
COST CENTER 41628	REF. PAGE(S) 12	TM STATEMENT: REVIEWER INDEPENDENCE	JFS for ADM

PURPOSE AND SUMMARY OF RESULTS:

Purpose:

Nuclear Management Company is considering plant life extension, power uprate and removal of hafnium power suppression assemblies from the core for Point Beach Unit 2. The purpose of this analysis is to evaluate the Point Beach Unit 2 reactor vessel pressurized thermal shock reference temperature (RT_{PTS}) values applicable to the projected end-of-life extension period, 53 effective full power years (EFPY).

Revision 01 updates the RT_{PTS} values due to the revision of BAW-2308 Revision 00 to BAW-2308 Revision 01.

Results:

The RT_{PTS} values (in support of plant life extension, power uprate and removal of hafnium from the core) applicable to the projected end-of-life extension period (53 EFPY) for the Point Beach Unit 2 reactor vessel beltline materials are listed in Table 5-1. These values were calculated in accordance with the requirements in the Code of Federal Regulations, Title 10, Part 50.61. The controlling beltline material for the Point Beach Unit 2 reactor vessel is the intermediate shell to lower shell circumferential weld, SA-1484, with a RT_{PTS} value of 276.0°F. The initial RT_{NDT} of SA-1484 was taken from the BAW-2308 Revision 01 topical report, which is based on ASME Boiler and Pressure Vessel Code Case N-629. The screening criterion for this weld is 300°F.

NON-PROPRIETARY

THE FOLLOWING COMPUTER CODES HAVE BEEN USED IN THIS DOCUMENT:		THE DOCUMENT CONTAINS ASSUMPTIONS THAT MUST BE VERIFIED PRIOR TO USE ON SAFETY-RELATED WORK
CODE/VERSION/REV	CODE/VERSION/REV	
N/A		☐ YES ☒ NO

Record of Revisions

REVISION	DESCRIPTION	Date
00	Original Release	10/02
01	Updated Ref. 7 to Rev. 1; updated Tables 4-3 and 5-1; updated Section 2.0 SA-1484 RT_{PTS} value.	8/03

Table of Contents

1.0 Introduction

Nuclear Management Company is considering plant life extension, power uprate and removal of hafnium power suppression assemblies from the core for Point Beach Unit 2. The purpose of this analysis is to evaluate the Point Beach Unit 2 reactor vessel pressurized thermal shock reference temperature (RT_{PTS}) values applicable to the projected end-of-life extension period, 53 effective full power years (EFPY).

2.0 Summary of Results

The RT_{PTS} values (in support of plant life extension, power uprate and removal of hafnium from the core) applicable to the projected end-of-life extension period (53 EFPY) for the Point Beach Unit 2 reactor vessel beltline materials are listed in Table 5-1. These values were calculated in accordance with the requirements in the Code of Federal Regulations, Title 10, Part 50.61 (10 CFR 50.61).[1] The controlling beltline material for the Point Beach Unit 2 reactor vessel is the intermediate shell to lower shell circumferential weld, SA-1484, with a RT_{PTS} value of 276.0°F. The initial RT_{NDT} of SA-1484 was taken from the BAW-2308 topical report, which is based on ASME Boiler and Pressure Vessel (B&PV) Code Case N-629. The screening criterion for this weld is 300°F.

3.0 Assumptions

No major assumptions are contained in this report.

4.0 Basis of Input Data

In the evaluations for predicting radiation damage of reactor vessel beltline materials, the data used in the specific calculations must be based on the best available source. The following sections describe the sources for all data used in the evaluations.

4.1 Material Identification

The reactor vessel beltline materials (both base metal and weld metals) included in this report conform to the beltline definition specified in Code of Federal Regulations, Title 10, Part 50 (10 CFR 50), Appendix G.[2]

4.2 Chemical Compositions

The copper and nickel chemical compositions for the Point Beach Unit 2 reactor vessel beltline materials are presented in Table 4-1. These values are taken from BAW-2325, Revision 1.[3]

Table 4-1. Chemical Composition of Point Beach Unit 2 Reactor Vessel Beltline Materials[3]

Matl. Ident.	Chemical Composition, wt%	
	Cu wt%	Ni wt%
123V352	0.11	0.73
123V500	0.09	0.70
122W195	0.05	0.72
21935	0.18	0.70
SA-1484	0.26	0.60

4.3 Neutron Fluence Estimates

The inside surface fluence values extrapolated to 53 EFPY for the Point Beach Unit 2 reactor vessel beltline materials are presented in Table 4-2[4]. The fluence projections were determined from neutron transport calculations without adjustment for the dosimetry bias factor. The fluence values include a power uprate and removal of hafnium rods.

Table 4-2. 53 EFPY Inside Surface Fluence Predictions for Reactor Vessel Beltline Materials in Point Beach Unit 2

Material Ident.	53 EFPY Inside Surface Fluence[4] (a) E > 1.0 MeV, n/cm^2
123V352	0.550E+19
123V500	5.385E+19
122W195	5.315E+19
21935	0.550E+19
SA-1484	5.085E+19

(a) The 53 EFPY inside surface fluence values were determined by linear interpolation between 52 EFPY and 54 EFPY.

4.4 Initial Reference Nil-Ductility Temperature

The method for determining the initial reference nil-ductility temperature (RT_{NDT}) is specified in the ASME B&PV Code, Section III, Paragraph NB-2331.[5] The initial RT_{NDT} is the greater of the drop weight nil-ductility transition temperature (per ASTM Standard E 208-81[6]) or the temperature that is 60°F below that at which the material exhibits 50 ft-lbs and 35 mils lateral expansion. If measured values of initial RT_{NDT} for the material in question are not available, generic values for that class of material may be used if there are sufficient test results to establish a mean and standard deviation for the class. The initial RT_{NDT} of Linde 80 weld SA-1484, heat 72442, was determined using ASME B&PV Code Case N-629.[7] The initial margin of SA-1484 was determined based on an analysis of the heat variability and the certainty of the measured reference temperature due to the number of specimens tested.[7]

Table 4-3 lists the initial RT_{NDT} values for the Point Beach Unit 2 reactor vessel beltline materials and the method used for the determination of the initial RT_{NDT}.

Table 4-3. Initial Reference Temperatures for Reactor Vessel Beltline Materials in Point Beach Unit 2

Matl. Ident.	Initial RT_{NDT}, F	Method for Determining RT_{NDT}
123V352	+40	Measured (σ_I = 0°F)[3]
123V500	+40	Measured (σ_I = 0°F)[3]
122W195	+40	Measured (σ_I = 0°F)[3]
21935	-56	Generic (σ_I = 17°F)[3]
SA-1484	-37.8	BAW-2308, Rev. 1 (σ_I = 11.9°F)[7]

5.0 Pressurized Thermal Shock Reference Temperature

The reference temperature for pressurized thermal shock (RT_{PTS}) is the nil ductility temperature of the material as defined by 10 CFR 50.61, Paragraph (b)(2). It is compared against screening criteria for protection against severe overcooling (thermal shock) of the reactor vessel at high pressure. The pressurized thermal shock (PTS) screening criteria are defined as 270°F for plates, forgings, and axial welds, and 300°F for circumferential welds. Therefore, if the calculated RT_{PTS} values for the reactor

vessel beltline materials are less than the specified screening criteria, the vessel is acceptable with regard to the risk of vessel failure for PTS transients.

In accordance with 10 CFR 50.61, the RT_{PTS} for each material in the beltline region is determined by the following expression:

$$RT_{PTS} = Initial\ RT_{NDT} + \Delta RT_{NDT} + Margin \qquad (1)$$

where: Initial RT_{NDT} = Initial nil-ductility reference temperature

ΔRT_{NDT} = Irradiation-induced change in reference temperature

Margin = a combination of the standard deviation of the initial RT_{NDT} and ΔRT_{NDT}.

The PTS data for the Point Beach Unit 2 reactor vessel beltline materials are presented in Table 5-1. The results were calculated in accordance with 10 CFR 50.61, and the calculational methods are briefly discussed below.

5.1 Initial Nil-Ductility Reference Temperature

The method for determining the initial nil-ductility reference temperature (RT_{NDT}) is described in Section 4.4.

5.2 Irradiation Induced Change in Reference Temperature

The irradiation-induced change in reference temperature (ΔRT_{PTS}) is defined as the mean value of the shift in reference temperature caused by irradiation and is calculated as follows:

$$\Delta RT_{NDT} = CF * ff \qquad (2)$$

where: CF = Chemistry Factor

ff = Fluence Factor

5.2.1 Chemistry Factor

The chemistry factor (CF) is a function of the material's copper and nickel content. The CF is determined from Table 1 (for weld metals) and Table 2 (for base metals) in 10 CFR 50.61. When determining the CF, the weight percent copper and nickel are the best estimate values for the material; these values are normally taken as the mean of the measured values.

5.2.2 Fluence Factor

The fluence factor (ff) is determined as follows:

$$ff = f^{(0.28 - 0.10 \log f)} \quad (3)$$

where: f = fluence x 10^{-19} (n/cm^2, E > 1.0 MeV)

5.3 Margin

The "margin" term is the quantity that is added to obtain conservative, upper bound values of the RT_{PTS} for the calculations. The margin is determined by the following expression:

$$Margin = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2} \quad (4)$$

where: σ_I = standard deviation for the initial RT_{NDT}

σ_Δ = standard deviation for the ΔRT_{NDT}

If a measured value of initial RT_{NDT} for the material in question is available, σ_I is zero because the measured initial value is an absolute value and it is assumed to have no error. If generic values of initial RT_{NDT} are used, σ_I is the standard deviation obtained from the set of data used to establish the mean value.

The standard deviation for ΔRT_{NDT}, σ_Δ, is established in 10 CFR 50.61, as 28°F for welds and 17°F for base metals, except that σ_Δ need not exceed 0.50 times the mean value of ΔRT_{NDT}.

5.4 Use of Surveillance Data

To verify that the RT_{PTS} for each vessel beltline material is a bounding value for the reactor vessel, plant-specific information shall be considered. This information includes, but is not limited to, the reactor vessel operating temperature and surveillance program results.

The results from the plant-specific surveillance program must be integrated into the RT_{PTS} estimate if the plant-specific surveillance data has been deemed credible as judged by the following criteria:

1. The materials in the surveillance capsules must be those which are the controlling materials with regard to radiation embrittlement.

2. Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions must be small enough to permit the determination of the 30 ft-lb temperature unambiguously.

3. Where there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values must be less than 28°F for welds and 17°F for base metal. Even if the range in the capsule fluences is large (two or more orders of magnitude); the scatter may not exceed twice those values.

4. The irradiation temperature of the Charpy specimens in the capsule must equal the vessel wall temperature at the cladding/base metal interface within ±25°F.

5. The surveillance data for the correlation monitor material in the capsule must fall within the scatter band of the data for the material.

The surveillance data deemed credible according to the criteria specified above must be used to determine a material-specific value of CF for use in the following equation:

$$\Delta RT_{NDT} = CF * ff \tag{5}$$

A material-specific value of CF is determined from the following equation:

$$CF = \frac{\sum_{i=1}^{n}[A_i * ff_i]}{\sum_{i=1}^{n} ff_i^2} \tag{6}$$

where: n = number of surveillance data points
A_i = measured value of ΔRT_{NDT}
ff_i = fluence factor for each surveillance data point.

For cases in which the results from a credible plant-specific surveillance program are used, the value of σ_Δ to be used is 14°F for welds and 8.5°F for base metals; the value of σ_Δ may not exceed one-half ΔRT_{NDT}.

The credibility assessment for the base metal heats 123V500 and 122W195 are performed in Appendix A. The credibility assessment for the weld metal SA-1484 is performed in Appendix B. The surveillance data for the base metal heat

123V500 and weld metal SA-1484 are not credible, therefore the table chemistry factors are used. The surveillance data for the base metal heat 122W195 are credible, therefore the surveillance data are used to determine the chemistry factor.

Table 5-1. Point Beach Unit 2 Reactor Vessel Beltline Material Pressurized Thermal Shock Reference Temperature Summary Applicable to 53 EFPY Under Power Uprate Conditions with Hafnium Rods Removed

Reactor Vessel Beltline Region Material	Material Ident.	Heat Number	Type	Cu wt%[3]	Ni wt%[3]	53 EFPY Fluence at Inside Surface[4], E > 1.0 MeV n/cm^2	Chemistry Factor	ΔRT_{PTS} °F	Initial RT_{NDT} °F	σ_I, °F	σ_Δ, °F	Margin, °F	RT_{PTS}, °F	Screen Criteria
10 CFR 50.61 (Tables)														
Nozzle Belt Forging	123V352	123V352	A 508 Cl. 2	0.11	0.73	0.550E+19	76.0	63.3	40	0	17	34.0	137.3	270
Intermediate Shell Forging	123V500	123V500	A 508 Cl. 2	0.09	0.70	5.385E+19	58.0	82.2	40	0	17	34.0	156.2	270
Lower Shell Forging	122W195	122W195	A 508 Cl. 2	0.05	0.72	5.315E+19	31.0	43.8	40	0	17	34.0	117.8	270
NB to IS Circ. Weld (100%)	21935	21935	Linde 1092	0.18	0.70	0.550E+19	170.5	142.0	-56	17	28	65.5	151.5	300
IS to LS Circ. Weld (100%)	SA-1484	72442	Linde 80	0.26	0.60	5.085E+19	180.0	253.0	-37.8	11.9	28	60.8	276.0	300
10 CFR 50.61 (Surveillance Data)														
Lower Shell Forging	122W195	122W195	A 508 Cl. 2	0.05	0.72	5.315E+19	42.8	60.5	40	0	8.5	17.0	117.5	270

6.0 References

1. Code of Federal Regulations, Title 10, *"Domestic Licensing of Production and Utilization Facilities,"* Part 50.61, *"Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock,"* Effective Date: August 28, 1996.

2. Code of Federal Regulation, Title 10, Part 50, *"Domestic Licensing of Production and Utilization Facilities,"* Appendix G, Fracture Toughness Requirements.

3. M. J. DeVan, "B&WOG Reactor Vessel Working Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity," BAW-2325, Revision 1, Framatome Technologies, Inc., Lynchburg, Virginia, January 1999.

4. Brian Kemp, *"PNB Fluence Values,"* Nuclear Management Company Letter To Brian Hall Framatome ANP, Dated October 17, 2002 (Attachment 1).

5. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Subsection NB, Class 1 Components.

6. ASTM Standard E 208-81, *"Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels,"* American Society for Testing and Materials, Philadelphia, Pennsylvania.

7. K K. Yoon, *"Initial RT_{NDT} of Linde 80 Weld Materials,"* BAW-2308, Rev. 1, Framatome ANP, Inc., August 2003.

APPENDIX A

Base Metal Evaluation and Use of Surveillance Data

The following tables provide the surveillance data evaluation of the base metal heats used in the Point Beach Unit 2 reactor vessel beltline region.

NOTE: The original Charpy V-notch impact data are based on hand-fit Charpy curves using engineering judgment; these data were re-evaluated using a hyperbolic tangent curve fitting program[a] to achieve consistency in the interpretation of the available surveillance test data.

A.1 Base Metal 123V500

Table A.1-1. Surveillance Data for Base Metal Heat Number 123V500[3]

Capsule ID (including source)	Cu wt%	Ni wt%	Irradiation Temperature (°F)	Fluence E > 1.0 MeV ($x10^{19}$ n/cm^2)	Measured ΔRT_{NDT} (°F)	Data Used in Assessing Vessel (Y or N)
PB-2: Capsule V Plant-Specific RVSP Material (LT)	0.09	0.70	542	0.650	39	Y
PB-2: Capsule T Plant-Specific RVSP Material (LT)	0.09	0.70	542	0.861	62	Y
PB-2: Capsule R Plant-Specific RVSP Material (LT)	0.09	0.70	542	2.20	88	Y
PB-2: Capsule S Plant-Specific RVSP Material (LT)	0.09	0.70	542	3.10	101	Y

[a] E. D. Eason, J. E. Wright, and G. R. Odette, *"Improved Embrittlement Correlations for Reactor Pressure Vessels,"* NUREG/CR-6551, U. S. Nuclear Regulatory Commission, Washington DC, August 1998.

Table A.1-2. Credibility Assessment for Base Metal Heat Number 123V500

Capsule Designation	Cu wt%	Ni wt%	Chem. Factor	Irrad. Temp. (°F)	Fluence Factor	Meas. ΔRT_{NDT} (°F)	Predicted ΔRT_{NDT} from Best Fit Line (°F)	(Measured - Predicted) ΔRT_{NDT} (°F)
PB-2: Capsule V Plant-Specific RVSP Material (LT)	0.09	0.70	58.0	542	0.879	39	60	-21
PB-2: Capsule T Plant-Specific RVSP Material (LT)	0.09	0.70	58.0	542	0.958	62	66	-4
PB-2: Capsule R Plant-Specific RVSP Material (LT)	0.09	0.70	58.0	542	1.214	88	83	5
PB-2: Capsule S Plant-Specific RVSP Material (LT)	0.09	0.70	58.0	542	1.298	101	89	12

where ***Predicted ΔRT_{NDT} = ($Slope_{best\ fit}$) * (Fluence Factor)*** and

$Slope_{best\ fit}$ = best fit line relating Measured ΔRT_{NDT} to the Fluence Factor **(i.e., 68.4)**

These data are not credible since the scatter is greater than ±17°F for one surveillance capsule data point.

Table A.1.3. Table Chemistry Factor Non-Conservatism Assessment for Base Metal Heat Number 123V500

Capsule Designation	Table Chem. Factor (Surv. Avg.)	Capsule Fluence Factor	Measured ΔRT_{NDT} (°F)	Predicted ΔRT_{NDT} (°F)	(Measured - Predicted) ΔRT_{NDT} (°F)
PB-2: Capsule V Plant-Specific RVSP Material (LT)	58.0	0.879	39	51	-12
PB-2: Capsule T Plant-Specific RVSP Material (LT)	58.0	0.958	62	56	6
PB-2: Capsule R Plant-Specific RVSP Material (LT)	58.0	1.214	88	70	18
PB-2: Capsule S Plant-Specific RVSP Material (LT)	58.0	1.298	101	75	26

where ***Predicted ΔRT_{NDT} = (Table Chem. Factor) * (Capsule Fluence Factor)***

Since the scatter for all data points is less than 2 standard deviations (34°F), the Table chemistry factor is conservative, therefore the Table chemistry factor based on the base metal best-estimate copper and nickel contents is used in the assessment of reactor vessel integrity.

A.2 Base Metal 122W195

Table A.2-1. Surveillance Data for Base Metal Heat Number 122W195 [3]

Capsule ID (including source)	Cu wt%	Ni wt%	Irradiation Temperature (°F)	Fluence E > 1.0 MeV ($x10^{19}$ n/cm^2)	Measured ΔRT_{NDT} (°F)	Data Used in Assessing Vessel (Y or N)
PB-2: Capsule V Plant-Specific RVSP Material (LT)	0.05	0.72	542	0.650	39	Y
PB-2: Capsule T Plant-Specific RVSP Material (LT)	0.05	0.72	542	0.861	35	Y
PB-2: Capsule R Plant-Specific RVSP Material (LT)	0.05	0.72	542	2.20	50	Y
PB-2: Capsule S Plant-Specific RVSP Material (LT)	0.05	0.72	542	3.10	61	Y

Table A.2-2. Credibility Assessment for Base Metal Heat Number 122W195

Capsule Designation	Cu wt%	Ni wt%	Chem. Factor	Irrad. Temp. (°F)	Fluence Factor	Meas. ΔRT_{NDT} (°F)	Predicted ΔRT_{NDT} from Best Fit Line (°F)	(Measured - Predicted) ΔRT_{NDT} (°F)
PB-2: Capsule V Plant-Specific RVSP Material (LT)	0.05	0.72	31.0	542	0.879	39	38	1
PB-2: Capsule T Plant-Specific RVSP Material (LT)	0.05	0.72	31.0	542	0.958	35	41	-6
PB-2: Capsule R Plant-Specific RVSP Material (LT)	0.05	0.72	31.0	542	1.214	50	52	-2
PB-2: Capsule S Plant-Specific RVSP Material (LT)	0.05	0.72	31.0	542	1.298	61	56	5

where ***Predicted ΔRT_{NDT} = (Slope$_{best\ fit}$) * (Fluence Factor)*** and

Slope$_{best\ fit}$ = best fit line relating Measured ΔRT_{NDT} to the Fluence Factor **(i.e., 42.8)**

These data are credible since the scatter is less than ±17°F for all the surveillance capsule data point.

Table A.2.3. Base Metal Heat Number 122W195 Chemistry Factor Calculation

Capsule Designation	Cu wt%	Ni wt%	Chem. Factor	Irrad. Temp. (°F)	Fluence Factor	Meas. ΔRT_{NDT} (°F)
PB-2: Capsule V Plant-Specific RVSP Material (LT)	0.05	0.72	31.0	542	0.879	39
PB-2: Capsule T Plant-Specific RVSP Material (LT)	0.05	0.72	31.0	542	0.958	35
PB-2: Capsule R Plant-Specific RVSP Material (LT)	0.05	0.72	31.0	542	1.214	50
PB-2: Capsule S Plant-Specific RVSP Material (LT)	0.05	0.72	31.0	542	1.298	61
Vessel Best-Estimate	0.05	0.72	31.0	542		

$CF_{Surv.\ data}$ = best fit line relating Measured ΔRT_{NDT} to the Fluence Factor

(i.e., $CF_{Surv.\ data} = 42.8$)

APPENDIX B

Evaluation and Use of Surveillance Data for Weld Wire Heat 72442

Both Regulatory Guide 1.99, Revision 2 and 10 CFR 50.61 require that surveillance data (if available) be considered in evaluating reactor vessel integrity. The best-estimate copper and nickel chemical compositions for both the weld wire heats and their weld metal sources are used in the evaluation of the surveillance data. The process of evaluating surveillance data includes a credibility assessment against five criteria and the calculation of the chemistry factor based on the surveillance data.

B.1 Surveillance Data Credibility Assessment

Point Beach Unit 2 participates in the B&WOG Master Integrated Reactor Vessel Surveillance Program (MIRVP) established in 1989 from which two surveillance data sets are available for evaluation of irradiation embrittlement.

When assessing credibility for surveillance data from several sources, the capsule data may have to be *"adjusted"* to account for the irradiation environment and chemical composition differences. All the available surveillance data for the weld wire heat 72442 were irradiated in a B&W plant. Therefore, no temperature adjustments are necessary. However, the surveillance data are from multiple sources, therefore it is necessary to adjust the capsule data for chemical composition (copper and nickel contents) differences. For the credibility determination, the surveillance data are *"normalized"* to the mean copper and nickel contents of the surveillance materials using the following equation:

$$Ratio\ Adjusted\ \Delta RT_{NDT,\,normalized} = \left(\frac{CF_{Table,\,Surv.\,Avg.\,Chem.}}{CF_{Table,\,Surv.\,Chem.}}\right) * \Delta RT_{NDT,\,measured}$$

A best-fit line (least squares regression) is then determined from the adjusted ΔRT_{NDT} capsule surveillance data as a function of the capsule fluence factor.

The data are considered credible if the difference between the adjusted ΔRT_{NDT} (i.e., chemistry adjusted) and the predicted ΔRT_{NDT} (from the best-fit line) for all the data are within ±28°F for weld metals and ±17°F for base metals.

B.2 Credible Surveillance Data

In accordance with Regulatory Guide 1.99, Revision 2 and 10 CFR 50.61, credible surveillance data are used to determine material-specific chemistry factor values for use in reactor vessel integrity assessments. The chemistry factor is determined from a best-fit line through the surveillance data adjusted to account for differences in chemical composition (i.e., copper and nickel contents) and irradiation environment (i.e., irradiation temperature) between the capsules and the vessel. The surveillance data are adjusted in the same manner as for the credibility determination except that the 30 ft-lb transition temperature values are *"normalized"* to the best estimate copper and nickel contents and the irradiation temperature of the vessel being assessed.

B.3 Non-Credible Surveillance Data

If the surveillance data are determined to be non-credible, the chemistry factor value is calculated from the generic Tables in 10 CFR 50.61 and Regulatory Guide 1.99, Revision 2 unless the chemistry factor determined from the surveillance data is significantly greater than that from the generic Tables, indicating that the Table chemistry factor is non-conservative. To determine if the generic Table chemistry factor is non-conservative, the following steps are performed:

1. Determine the chemistry factor from the generic Tables based on the surveillance specimen chemical composition; use this chemistry factor to determine the predicted ΔRT_{NDT} for each capsule:

   $$(Predicted\ \Delta RT_{NDT} = CF_{Table,\ Surv.\ Avg.\ Chem.} * ff_{capsule})$$

2. Determine difference between the predicted ΔRT_{NDT} and the measured ΔRT_{NDT}.

If the difference between the predicted ΔRT_{NDT} and the measured ΔRT_{NDT} values exceeds 2 standard deviations (i.e., 56°F for weld metals and 34°F for base metals), the Table chemistry factor is considered non-conservative. When the Table chemistry factor is determined to be non-conservative, the chemistry factor determined from the "non-credible" surveillance data is used in the assessment of reactor vessel integrity using the "full" value of σ_Δ in calculating the Margin term.

B.4 Assessment of Weld Wire Heat Surveillance Data

The following tables provide the surveillance data evaluation of the weld wire heat 72442 used in the Point Beach Unit 2 reactor vessel beltline region.

NOTE: The original Charpy V-notch impact data are based on hand-fit Charpy curves using engineering judgment; these data were re-evaluated using a hyperbolic tangent curve fitting program to achieve consistency in the interpretation of the available surveillance test data.

Table B.1-1. Surveillance Data for Weld Wire Heat Number 72442[3]

Capsule ID (including source)	Cu wt%	Ni wt%	Irradiation Temperature (°F)	Fluence E > 1.0 MeV ($x10^{19}$ n/cm^2)	Measured ΔRT_{NDT} (°F)	Data Used in Assessing Vessel (Y or N)
B&WOG: Capsule CR3-LG1 WF-67: MD1 Nozzle Belt Dropout Matl.	0.22	0.60	556	0.609	167	Y
B&WOG: Capsule CR3-LG2 WF-67: MD1 Nozzle Belt Dropout Matl.	0.22	0.60	556	1.95	138	Y

Table B.1-2. Credibility Assessment for Weld Wire Heat Number 72442

Capsule Designation	Cu wt%	Ni wt%	Chem. Factor	Irrad. Temp. (°F)	Fluence Factor	Meas. ΔRT_{NDT} (°F)	Adjusted ΔRT_{NDT} (°F)	Predicted ΔRT_{NDT} from Best Fit Line (°F)	(Adjusted - Predicted) ΔRT_{NDT} (°F)
B&WOG: Capsule CR3-LG1 WF-67: MD1 Nozzle Belt Dropout Matl.	0.22	0.60	167.0	556	0.861	167	—	124	43
B&WOG: Capsule CR3-LG2 WF-67: MD1 Nozzle Belt Dropout Matl.	0.22	0.60	167.0	556	1.182	138	—	170	-32
Surv. Avg.	0.22	0.60	167.0	556					

where ***Predicted*** ΔRT_{NDT} ***= (Slope***$_{best\ fit}$***) * (Fluence Factor)*** and
Slope$_{best\ fit}$ ***= best fit line relating Adjusted*** ΔRT_{NDT} ***to the Fluence Factor (i.e., 143.5)***

These data are not credible since the scatter is greater than ±28°F for two surveillance capsule data points.

Table B.1-3. Table Chemistry Factor Non-Conservatism Assessment for Weld Wire Heat Number 72442

Capsule Designation	Cu wt%	Ni wt%	Chem. Factor	Irrad. Temp. (°F)	Fluence Factor	Meas. ΔRT_{NDT} (°F)	Adjusted ΔRT_{NDT} (°F)	Predicted ΔRT_{NDT} (°F)	(Adjusted - Predicted) ΔRT_{NDT} (°F)
B&WOG: Capsule CR3-LG1 WF-67: MD1 Nozzle Belt Dropout Matl.	0.22	0.60	—	556	0.861	167	167	143.8	23.2
B&WOG: Capsule CR3-LG2 WF-67: MD1 Nozzle Belt Dropout Matl.	0.22	0.60	—	556	1.182	138	138	197.4	-59.4
Surv. Avg.	0.22	0.60	167.0	556					

where ***Predicted ΔRT_{NDT} = (Average Table Chem. Factor) * (Fluence Factor)***

The above assessment results indicates that the generic Table chemistry factor for the surveillance data over-predicts the adjusted measured data for one data point while the other data point is within 2 standard deviations (i.e., 56°F). Therefore, the Table chemistry factor calculated using the weld wire heat best-estimate copper and nickel contents is considered conservative.

ATTACHMENT 1

PNB Fluence Values

NMC
Committed to Nuclear Excellence

Point Beach Nuclear Plant
Operated by Nuclear Management Company, LLC

October 17, 2002

Brian Hall
Framatome ANP, Inc.
3315 Old Forest Rd
Lynchburg, VA 24501

Dear Brian:

Subject: PBNP Fluence Values

Please use the attached fluence values for the PBNP RT_{pts} evaluations being performed by Framatome under NMC Purchase Order P002267. These values have been taken from Reference 1. The fluence projections were determined from neutron transport calculations as reported in Section 2 of Reference 1 without adjustment for the dosimetry bias factor. The fluence values include a power uprate and removal of hafnium rods. Use 53 EFPY fluence values determined by linear interpolation between 52 EFPY and 54 EFPY.

Regards,

Brian Kemp
Senior Engineer
Point Beach Nuclear

6610 Nuclear Road • Two Rivers, Wisconsin 54241-9516
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Partial Table 2.3-11 from Ref. 1

Intermediate Shell to Lower Shell Circumferential Weld (SA-1484)
Calculated Neutron Fluence (E > 1.0 MeV) Projections With Uprate
Point Beach Unit 2

Without Hafnium Suppression Rods

Operating Time [efpy]	Neutron Fluence [n/cm²]			
	0 Deg.	15 deg.	30 Deg.	45 Deg.
52.0	4.98E+19	3.25E+19	2.53E+19	2.25E+19
54.0	5.19E+19	3.39E+19	2.64E+19	2.34E+19

Partial Table 2.3-12 from Ref. 1

Lower Shell Forging (122W195)
Calculated Neutron Fluence (E > 1.0 MeV) Projections With Uprate
Point Beach Unit 2

Without Hafnium Suppression Rods

Operating Time [efpy]	Neutron Fluence [n/cm²]			
	0 Deg.	15 deg.	30 Deg.	45 Deg.
52.0	5.21E+19	3.39E+19	2.63E+19	2.34E+19
54.0	5.42E+19	3.53E+19	2.74E+19	2.43E+19

Partial Table 2.3-13 from Ref. 1

Intermediate Shell Forging (123V500)
Calculated Neutron Fluence (E > 1.0 MeV) Projections With Uprate
Point Beach Unit 2

Without Hafnium Suppression Rods

Operating Time [efpy]	Neutron Fluence [n/cm²]			
	0 Deg.	15 deg.	30 Deg.	45 Deg.
52.0	5.28E+19	3.41E+19	2.66E+19	2.37E+19
54.0	5.49E+19	3.55E+19	2.77E+19	2.46E+19

Partial Table 2.3-14 from Ref. 1

Intermediate Shell to Upper Shell Circumferential Weld (21935)
Calculated Neutron Fluence (E > 1.0 MeV) Projections With Uprate
Point Beach Unit 2

Without Hafnium Suppression Rods

Operating Time [efpy]	Neutron Fluence [n/cm²]			
	0 Deg.	15 deg.	30 Deg.	45 Deg.
52.0	5.39E+18	3.51E+18	2.71E+18	2.40E+18
54.0	5.61E+18	3.66E+18	2.83E+18	2.50E+18

Partial Table 2.3-15 from Ref. 1

Upper Shell Forging
Calculated Neutron Fluence (E > 1.0 MeV) Projections With Uprate
Point Beach Unit 2

Without Hafnium Suppression Rods

Operating Time [efpy]	Neutron Fluence [n/cm²]			
	0 Deg.	15 deg.	30 Deg.	45 Deg.
52.0	5.39E+18	3.51E+18	2.71E+18	2.40E+18
54.0	5.61E+18	3.66E+18	2.83E+18	2.50E+18

Reference:

1. S. L. Anderson, "*Pressure Vessel Neutron Exposure Evaluations Point Beach Units 1 and 2*," LTR-REA-02-23, Westinghouse Electric Company, February 2002.