NMC
Committed to Nuclear Excellence

Point Beach Nuclear Plant
Operated by Nuclear Management Company, LLC

January 28, 2005

NRC 2005-0004
10 CFR 2.390
10 CFR 54

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington DC 20555

Point Beach Nuclear Plant, Units 1 and 2
Dockets 50-266 and 50-301
License Nos. DPR-24 and DPR-27

Response to Request for Additional Information
Regarding the Point Beach Nuclear Plant License Renewal Application
(TAC Nos. MC2099 and MC2100) and
Request for Withholding of Proprietary Information from Public Disclosure

By letter dated February 25, 2004, Nuclear Management Company, LLC (NMC), submitted the Point Beach Nuclear Plant (PBNP) Units 1 and 2 License Renewal Application (LRA). On November 17, 2004, the Nuclear Regulatory Commission (NRC) requested additional information regarding Time Limited Aging Analysis (TLAA) (LRA Section 4.3). Enclosure 2 to this letter contains NMC's responses to the staff's questions.

Enclosure 1 contains a Westinghouse proprietary authorization letter, CAW-04-1931; accompanying affidavit; Proprietary Information Notice; and Copyright Notice concerning information proprietary to Westinghouse Electric Company, LLC ("Westinghouse"). (Responses to RAI 4.3.2.2 and 4.3.4.3).

As Enclosure 2 contains information proprietary to Westinghouse, it is supported by an affidavit signed by Westinghouse; the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR 2.390 of the Commission's regulations.

Accordingly, it is respectfully requested that the information, which is proprietary to Westinghouse, be withheld from public disclosure in accordance with 10 CFR 2.390 of the Commission's regulations. Correspondence regarding the copyright or proprietary aspects of the items listed above, or the supporting Westinghouse affidavit, should reference CAW-04-1931 and should be addressed to J. A. Gresham, Manager,

A093

6590 Nuclear Road • Two Rivers, Wisconsin 54241
Telephone: 920.755.2321

Regulatory Compliance and Plant Licensing, Westinghouse Electric Company, LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Enclosure 3 contains the non-proprietary version of Enclosure 2.

On December 1, 2004, the NRC staff verbally provided additional time for NMC to respond to this request for additional information in order for further clarifications to be provided. The clarifications allowed the PBNP License Renewal project staff to clearly understand the information needed and for further analysis to be completed.

This letter contains no new commitments and no revisions to existing commitments.

I declare under penalty of perjury that the forgoing is true and correct. Executed on January 28, 2005.

Dennis L. Koehl

Dennis L. Koehl
Site Vice-President, Point Beach Nuclear Plant
Nuclear Management Company, LLC

Enclosures (3)

cc: (w/o enclosures)
Regional Administrator, Region III, USNRC
Project Manager, Point Beach Nuclear Plant, USNRC
Resident Inspector, Point Beach Nuclear Plant, USNRC

ENCLOSURE 1

Westinghouse Application for Withholding Proprietary Information from Public Disclosure, CAW-04-1931;
Affidavit;
Proprietary Information Notice;
Copyright Notice

7 pages follow

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

COPYRIGHT NOTICE

The reports transmitted herewith each bear a Westinghouse copyright notice. The NRC is permitted to make the number of copies of the information contained in these reports which are necessary for its internal use in connection with generic and plant-specific reviews and approvals as well as the issuance, denial, amendment, transfer, renewal, modification, suspension, revocation, or violation of a license, permit, order, or regulation subject to the requirements of 10 CFR 2.390 regarding restrictions on public disclosure to the extent such information has been identified as proprietary by Westinghouse, copyright protection notwithstanding. With respect to the non-proprietary versions of these reports, the NRC is permitted to make the number of copies beyond those necessary for its internal use which are necessary in order to have one copy available for public viewing in the appropriate docket files in the public document room in Washington, DC and in local public document rooms as may be required by NRC regulations if the number of copies submitted is insufficient for this purpose. Copies made by the NRC must include the copyright notice in all instances and the proprietary notice if the original was identified as proprietary.

Westinghouse

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Washington, DC 20555-0001

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e-mail: greshaja@westinghouse.com

Our ref: CAW-04-1931

December 14, 2004

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: "Response to Point Beach Nuclear Plant, Units 1 and 2, License Renewal Application (LRA) Request for Additional Information (RAIs)" (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-04-1931 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by Nuclear Management Company.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-04-1931, and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

J A Gresham

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Enclosures

cc: B. Benney
L. Feizollahi

A BNFL Group company

CAW-04-1931

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared J. A. Gresham, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

J. A. Gresham

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed
before me this 14th day
of December, 2004

Sharon L. Fiori

Notary Public

Notarial Seal
Sharon L. Fiori, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires January 29, 2007
Member, Pennsylvania Association Of Notaries

(1) I am Manager, Regulatory Compliance and Plant Licensing, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse "Application for Withholding" accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in "Response to Point Beach Nuclear Plant, Units 1 and 2, License Renewal Application (LRA) Request for Additional Information (RAIs)," (Proprietary) dated December 2004, being transmitted by the Nuclear Management Company letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted for use by Westinghouse for the Point Beach Nuclear Plant, Units 1 and 2 is expected to be applicable for other licensee submittals in response to certain NRC requirements for justification of continued safe operation of Point Beach Units 1 and 2.

This information is part of that which will enable Westinghouse to:

(a) Assess the technical justification for renewing the operating license.

(b) Assist the customer in obtaining NRC approval.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for purposes of meeting NRC requirements for licensing documentation.

(b) Westinghouse can sell support and defense of continued safe operation with a renewed license.

(c) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar support documentation and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

ENCLOSURE 3

RESPONSE TO POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2 REGARDING LICENSE RENEWAL APPLICATION (LRA) REQUESTS FOR ADDITIONAL INFORMATION (RAIs)

January 2005

Westinghouse Electric Company LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

RESPONSE TO POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2 REGARDING LICENSE RENEWAL APPLICATION (LRA) REQUESTS FOR ADDITIONAL INFORMATION (RAIs)

The following information is provided in response to the Nuclear Regulatory Commission (NRC) staff's request for additional information (RAI) regarding the Point Beach Nuclear Plant (PBNP) License Renewal Application (LRA).

The NRC staff's questions are restated below, with the Nuclear Management Company (NMC) response following.

4.3 Metal Fatigue

NRC Question RAI-4.3.1 - Reactor Vessel Structural Integrity:

Provide confirmation that the limiting locations of the PBNS reactor vessels evaluated for extended operation correspond to the structures and/or components listed in Table IV.A2 of NUREG-1801, Volume 2, for PWR reactor vessels structures and/or components, where cumulative fatigue damage/fatigue is the aging effect/mechanism, and which require further evaluation as TLAAs for the period of extended operation. Alternatively, provide the location in the LRA where this information is shown.

NMC Response:

As noted in the following table, the limiting locations of the PBNP reactor pressure vessel (RPVs) evaluated for extended operation correspond to the structures and/or components listed in Table IV.A2 of NUREG-1801, Volume 2, for PWR reactor vessel structures and/or components, where cumulative fatigue damage/fatigue is the aging effect/mechanism.

NUREG-1801 RPV Components (1)		
Table IV.A2 Item	Component Description	Evaluated for Fatigue at PBNP
A2.1-b A2.1.1	Closure Head: - Dome	 Yes
A2.1-e A2.1.3	Closure Head: - Stud Assembly	 Yes
A2.2-c A2.2.1 A2.2.2	Control Rod Drive Head Penetration: - Nozzle - Pressure Housing	 Yes Yes
A2.3-c A2.3.1 A2.3.2 A2.3.3	Nozzles: - Inlet - Outlet - Safety Injection	 Yes Yes Yes
A2.4-a A2.4.1 A2.4.2 A2.4.3	Nozzle Safe Ends: - Inlet - Outlet - Safety Injection	 Yes (2) Yes (2) Yes (2)
A2.5-d A2.5.1 A2.5.2 A2.5.3 A2.5.4	Vessel shell: - Upper (nozzle) Shell - Intermediate and Lower Shell - Vessel Flange - Bottom Head	 Yes Yes Yes Yes
A2.8-a A2.8.1	Pressure Vessel Support: - Skirt Support	 Yes (3)

(1) - Reactor Vessel structures and/or components listed in Table IV.A2 of NUREG-1801, Volume 2, for PWR reactor vessels structures and/or components.

(2) - Included with specific nozzle analyses.

(3) - PBNP RPVs are supported off of external support brackets at the nozzle elevation.

<u>NRC Question RAI-4.3.2.1 Reactor Vessel Internals Structural Integrity:</u>

Provide confirmation that the limiting locations of the PBNS reactor vessel internals evaluated for extended operation correspond to the structures and/or components listed in Table IV.B2 of NUREG-1801, Volume 2, for PWR reactor vessel internals structures and/or components, where cumulative fatigue damage/fatigue is the aging effect/mechanism, and which require further evaluation as TLAAs for the period of extended operation. Alternatively, provide the location in the LRA where this information is shown.

NMC Response:

The following table identifies the structures and/or components listed in Table IV.B2 of NUREG-1801, Volume 2, for PWR reactor vessel internals structures and/or components where cumulative fatigue damage/fatigue is the aging effect/mechanism. The table also identifies whether the component locations for PBNP reactor vessel internals were evaluated for fatigue for extended operation. The major components have been evaluated for fatigue. Since the PBNP reactor internals were designed and manufactured prior to the release of Subsection NG of the ASME Code Section III, fatigue evaluations were not performed nor were they required for all of the locations noted in Table IV.B2 of NUREG-1801, Volume 2, for PWR reactor vessel internals structures and/or components, where cumulative fatigue damage/fatigue is the aging effect/mechanism.

NUREG-1801 Reactor Vessel Internals (PWR) – Westinghouse (1)		
Table IV.B2 Item	Component Description	Evaluated for Fatigue at PBNP
B2.1-c B2.1.1 B2.1.4 B2.1.7	Upper Internals Assembly: - Upper Support Plate - Upper Core Plate - Hold-Down Spring	 Yes Yes No (2)
B2.1-h B2.1.2	Upper Internals Assembly: - Upper Support Column	 Yes
B2.1-m B2.1.6	Upper Internals Assembly: - Fuel Alignment Pins	 No (2)
B2.2-c B2.2.1	RCCA Guide Tube Assemblies: - RCCA Guide Tubes	 Yes
B2.2-f B2.2.2 B2.2.3	RCCA Guide Tube Assemblies: - RCCA Guide Tube Bolts - RCCA Guide Tube Support Pins	 No (2) Yes
B2.3-d B2.3.1 B2.3.2 B2.3.3 B2.3.4	Core Barrel: - Core Barrel (CB) - CB Flange (upper) - CB Outlet Nozzles - Thermal Shield	 Yes Yes Yes Yes
B2.4-g B2.4.1 B2.4.2	Baffle/Former Assembly: - Baffle and Former Plates - Baffle/Former Bolts	 No (2) Yes (Bolt Qualification by Testing)
B2.5-d B2.5.1 B2.5.4	Lower Internal Assembly: - Lower Core Plate - Lower Support Plate Columns	 Yes Yes

NUREG-1801 Reactor Vessel Internals (PWR) – Westinghouse (1)		
Table IV.B2 Item	Component Description	Evaluated for Fatigue at PBNP
B2.5-j B2.5.2 B2.5.5	Lower Internal Assembly: - Fuel Alignment Pins - Lower Support Plate Column Bolts	 No (2) No (2)
B2.5-p B2.5.6 B2.5.7	Lower Internal Assembly: - Radial Keys and Clevis Inserts - Clevis Insert Bolts	 Yes No (2)

(1) - Reactor Vessel Internals (PWR) – Westinghouse, structures and/or components listed in Table IV.B2 of NUREG-1801, Volume 2, for PWR reactor vessels structures and/or components.

(2) - Prior to the release of Subsection NG of the ASME Code Section III, the reactor vessel internals were designed to the intent of Subsection NB which was a pressure vessel code. The PBNP reactor internals were designed and manufactured prior to the release of Subsection NG of the ASME Code Section III. During this period, several sets of internals were being designed and manufactured at or about the same time. The reactor internals designs were segregated as 2-loop, 3-loop and 4-loop, but plant specific design packages and calculations were not produced. There were no ASME code design specifications or stress reports developed for the internal packages manufactured prior to Subsection NG. Typically, hand calculations were performed for the various subcomponents of the internals. Many of the individual subcomponent calculations were assembled into a single document, which is identified as the Westinghouse 2-Loop Design Manual. When a particular design feature was changed, new calculations would be performed to demonstrate the adequacy of the changed component. For components that were not changed, the original design and drawings continue to apply.

NRC Question RAI-4.3.2.2 Reactor Vessel Internals Structural Integrity:

Provide a summary of 60-year primary-plus-secondary stress intensities and cumulative fatigue usage factors (similar to revised Tables 4.3-1 and 4.3-2 in Appendix A of the LRA for components of the reactor vessel) for the key reactor internal components listed on page 4-41 of the LRA.

NMC Response:

In accordance with discussions with NRC staff on October 28, 2004, it was agreed that a summary of 60-year primary-plus-secondary stress intensities for the PBNP reactor vessel internals need not be provided since the NRC staff indicated that the summary was not critical for the review and it is not a part of the PBNP current licensing basis (CLB).

A summary of the PBNP key reactor vessel internal component cumulative fatigue evaluation results is included in the following table.

PBNP Key Reactor Vessel Internals Component Design Basis Cumulative Fatigue Results	
Component Description	Cumulative Usage Factor (CUF)
Upper Support Plate Perforations	[ ]b,c (1)
Upper Core Plate alignment Pins	[ ]b,c (1)
Upper Core Plate	[ ]b,c (2)
Upper Support Column / Base Weld	[ ]b,c (1)
RCCA Guide Tube Sheath Weld	[ ]b,c (1)
Guide Tube Flange Weld	[ ]b,c (2)
Lower Support Plate/ CB Weld	[ ]b,c (2)
CB / Flange Weld	[ ]b,c (2)
Outlet Nozzle Inner Weld	[ ]b,c (1)
Thermal Shield Flexures	[ ]b,c (1)
Thermal Shield Flexure Bolts	[ ]b,c (1)
Lower Core Plate SC Bolt Holes	[ ]b,c (2)
Lower Support Column Extensions	[ ]b,c (2)
Lower Radial Restraint Dowel Pin	[ ]b,c (1)
Flexureless Insert	[ ]b,c

BRACKETED NUMBERS ARE WESTINGHOUSE PROPRIETARY

(1) - WCAP-14459 "Reactor Pressure Vessel and Internals Evaluations for the Point Beach Units 1 and 2 Power Uprating / Replacement Steam Generator Program," April 1996.

(2) - Westinghouse, "Power Uprate Project, Point Beach Nuclear Plant Units 1 and 2, Volume 1, NSSS Engineering Report," April 2002.

NRC Question RAI-4.3.3 Control Rod Drive Mechanism Structural Integrity:

Provide a comparison of the CLB set of transient conditions and design cycles, and the revised set of full power uprate transient conditions and design cycles, that were used in the CRDM fatigue TLAAs to show conformance with the CLB fatigue limits to the end of the period of extended operation.

NMC Response:

The revised set of full power uprate transient conditions and design cycles is PBNP's CLB set of transient conditions and design cycles. These are shown in the revised Table 4.1-8, "Thermal and Loading Cycles" in Appendix A "FSAR Supplement" of the PBNP LRA. This set of transient conditions and design cycles was used in the evaluation of all PBNP TLAAs that required the use of transient conditions and design cycles. This set of transient conditions and design cycles was used in the CRDM fatigue TLAA evaluation to show conformance with the CLB fatigue limits to the end of extended life (EOEL).

NRC Question RAI-4.3.4.1 Steam Generator Structural Integrity:

Provide confirmation that the limiting locations of the PBNS steam generators evaluated for extended operation correspond to the structures and/or components listed in Table IV.D1 of NUREG-1801, Volume 2, for PWR reactor vessels structures and/or components, where cumulative fatigue damage/fatigue is the aging effect/mechanism, and which require further evaluation as TLAAs for the period of extended operation. Alternatively, state the location in the LRA where this information has been provided.

NMC Response:

As noted in the following table, the locations of the PBNP steam generators evaluated for extended operation correspond to the structures and/or components listed in Table IV.D1 of NUREG-1801, Volume 2, for PWR reactor vessel internals structures and/or components, where cumulative fatigue damage/fatigue is the aging effect/mechanism.

NUREG-1801 Steam Generator Components (1)			
Table IV.D1 Item	Component Description	Evaluated for Fatigue at PBNP	
		Unit 1 (44F)	Unit 2 (Δ47)
D1.1-a	Pressure Boundary and Structural:		
D1.1.1	- Top Head	Yes	Yes
D1.1.2	- Steam Nozzle and	Yes	Yes
	- Safe End	Yes	Yes
D1.1-b	Pressure Boundary and Structural:		
D1.1.3	- Upper and Lower Shell	Yes	Yes
D1.1.4	- Transition Cone	Yes	Yes
D1.1.5	- FW Nozzle and	Yes	Yes
	- Safe End	Yes (5)	Yes (5)
D1.1.6	- FW Impingement Plate and Support	N/A (2)	N/A (2)
D1.1-h	Pressure Boundary and Structural:		
D1.1.8	- Lower Head	Yes	Yes
D1.1.9	- Primary Nozzles and	Yes	Yes
	- Safe Ends	Yes (4)	Yes (4)
D1.2-d	Tube Bundle:		
D1.2.1	- Tubes and	Yes	Yes
	- Sleeves	N/A (3)	N/A (3)

(1) - Steam Generator structures and/or components listed in Table IV.D1 of NUREG-1801, Volume 2, for PWR reactor vessels structures and/or components.

(2) - The 44F and Δ47 replacement steam generators are feed ring designs and have no impingement plates.

(3) - There are no sleeved tubes in the PBNP steam generators.

(4) - The Unit 1 SGs' safe ends are stainless steel weld buildup, Unit 2 steam generators contain stainless steel safe ends. The Unit 1 and Unit 2 safe ends were analyzed with the nozzle.

(5) - The Unit 1 and Unit 2 feedwater nozzles do not have a nozzle-to-piping safe end. The feedwater nozzles do have a safe end in the nozzle-to-thermal sleeve. The thermal sleeve safe ends were included in the nozzle analysis.

NRC Question RAI-4.3.4.2 Steam Generator Structural Integrity:

Provide a comparison of the CLB set of transient conditions and design cycles, and the revised set of Steam Generator Replacement and Full Power Uprate transient conditions and design cycles, that were used in the Units 1 and 2 steam generator fatigue TLAAs to show conformance with the CLB fatigue CUF limit to the end of the period of extended operation. Alternatively, provide clarification stating that the applicable transient conditions and design cycles are those stated in Table 4.1-8 of Appendix A to the LRA.

NMC Response:

The revised set of full power uprate transient conditions and design cycles is PBNP's CLB set of transient conditions and design cycles. These are shown in the revised Table 4.1-8, "Thermal and Loading Cycles," in Appendix A "FSAR Supplement" of the PBNP LRA. This set of transient conditions and design cycles was used in the evaluation of all PBNP TLAAs that required the use of transient conditions and design cycles. This set of transient conditions and design cycles was used in the steam generator fatigue TLAA evaluation to show conformance with the CLB fatigue limits to the EOEL.

NRC Question RAI-4.3.4.3 Steam Generator Structural Integrity:

List the key Units 1 and 2 steam generator components, and provide for each a summary of 60-year primary-plus-secondary stress intensities and cumulative fatigue usage factors (similar to revised Tables 4.3-1 and 4.3-2 in Appendix A of the LRA for components of the reactor vessel) for these components.

NMC Response:

In accordance with discussions with NRC staff on October 28, 2004, it was agreed that a summary of 60-year primary-plus-secondary stress intensities for the PBNP steam generators need not be provided since the secondary stress intensities were not critical for the review and are not a part of the PBNP CLB.

A summary of the PBNP key steam generator components fatigue evaluation results is included in the following table.

It should be noted that the design of the steam generators for the two PBNP units are not identical. The Unit 1 steam generators are an early 80's vintage, incorporating bounding generic analyses based on the Westinghouse 41 series steam generator design. The Unit 2 steam generators are a mid 90's vintage, incorporating a PBNP specific design analysis.

PBNP Key Steam Generator Components Design Basis Cumulative Fatigue Results		
Component Description	Unit 1 (44F) CUF	Unit 2 (Δ47) CUF
Tube(s)	[ ]b,c (1)	[ ]b,c (1)
Tube-to-Tubesheet Weld	[ ]b,c (2)	[ ]b,c (4)
Primary Chamber, Tubesheet, and Stub Barrel	[ ]b,c (1)	[ ]b,c (4)
Primary Nozzle(s)	[ ]b,c (3)	[ ]b,c (4)
Primary Manway Openings	[ ]b,c (3)	[ ]b,c (4) (6)
Divider Plate	[ ]b,c (2)	[ ]b,c (4)
Steam Nozzle, Upper Head, Upper Shell	[ ]b,c (3)	[ ]b,c (1)
Steam Nozzle Venturi	[ ]b,c (3)	[ ]b,c (4)
Feedwater Nozzle	[ ]b,c (1)	[ ]b,c (1)
Transition Cone	[ ]b,c (7)	[ ]b,c (1)
Secondary Manway Opening	[ ]b,c (8)	[ ]b,c (4)
Secondary Handholes / Access Openings	[ ]b,c (1)	[ ]b,c (4)
Secondary Inspection Ports	[ ]b,c (1)(5)	[ ]b,c (4)
Minor Penetrations	[ ]b,c (3)	[ ]b,c (4)

BRACKETED NUMBERS ARE WESTINGHOUSE PROPRIETARY

(1) - Westinghouse, "Power Uprate Project, Point Beach Nuclear Plant Units 1 and 2, Volume 1, NSSS Engineering Report," April 2002.

(2) - WCAP-14602, Volume 1, "Point Beach Nuclear Plant Unit 2 Steam Generator Replacement Engineering Report," March 1996.

(3) - Westinghouse WNEP-8393, "Model 44F Replacement Steam Generator Stress Report for Wisconsin Electric Power Company Point Beach Unit 1," Revision 0, October 1983.

(4) - WNEP-9513, "Delta 47 Steam Generator Stress Report Summary Wisconsin Electric Power Company Point Beach Unit 2," Revision 1, December 1996.

(5) - The limiting location shown is the bolts. These are managed by replacement on a periodic basis. The next limiting location is the manway pad with a CUF of [ ]b,c.

(6) - The bolts and drain hole are qualified for fatigue based on tests. The CUF shown is for the cover and pad knuckle.

(7) - WTD-EM-79-039, "Model 44F Steam Generator Replacement Units Shell / Cone / Lower Shell Analysis," April 1, 1979.

(8) - Westinghouse WNEP-8393, "Model 44F Replacement Steam Generator Stress Report for Wisconsin Electric Power Company Point Beach Unit 1," Revision 0, October 1983. - The value for the inside radius of the steam nozzle is bounding.

NRC Question RAI-4.3.5.1 Pressurizer Structural Integrity:

Provide confirmation that the limiting fatigue locations of the PBNS pressurizers evaluated for extended operation correspond to the pressurizer structures and/or components listed in Table IV.C2.5 of NUREG-1801, Volume 2, for PWR reactor vessels structures and/or components, where cumulative fatigue damage/fatigue is the aging effect/mechanism, and which require further evaluation as TLAAs for the period of extended operation. Alternatively, state the location in the LRA where this information has been provided.

NMC Response:

As noted in the following table, the fatigue locations of the PBNP pressurizers evaluated for extended operation correspond to the pressurizer structures and/or components listed in Table IV.C2.5 of NUREG-1801, Volume 2, for PWR reactor vessels structures and/or components, where cumulative fatigue damage/fatigue is the aging effect/mechanism.

NUREG-1801 Pressurizer Components (1)		
Table IV.C2.5 Item	Component Description	Evaluated for Fatigue at PBNP
C2.5-a C2.5.1	Pressurizer: - Shell and - Heads	 Yes Yes
C2.5-d C2.5.2 C2.5.4	Pressurizer: - Spray Line Nozzle - Spray Head	 Yes No (2)
C2.5-e C2.5.3	Pressurizer: - Surge line nozzle	 Yes
C2.5-f C2.5.5 C2.5.6 C2.5.7	Pressurizer: - Thermal sleeves - Instrument penetrations - Safe ends	 Yes Yes Yes
C2.5-q C2.5.10	Pressurizer: - Heater Sheaths and - Heater Sleeves	 Yes Yes
C2.5-t C2.5.11	Pressurizer: - Support keys, - Skirt and - Shear Lugs	 No (N/A for PBNP) (3) Yes No (N/A for PBNP) (3)
C2.5-w C2.5.12	Pressurizer: - Integral Support	 No (N/A for PBNP) (3)

(1) - Pressurizer structures and/or components listed in Table IV.C2.5 of NUREG-1801, Volume 2, for PWR reactor vessels structures and/or components.

(2) - The spray head is a non-structural or pressure retaining component and is not in the scope of License Renewal.

(3) - The PBNP pressurizers do not have support keys, shear lugs, or integral supports.

NRC Question RAI-4.3.5.2 Pressurizer Structural Integrity:

Provide a comparison of the CLB set of transient conditions and design cycles, and the revised set of Steam Generator Replacement and Full Power Uprate transient conditions and design cycles, that were used in the Units 1 and 2 pressurizers fatigue TLAAs to show conformance with the CLB fatigue limit to the end of the period of extended operation.

NMC Response:

The revised set of full power uprate transient conditions and design cycles is PBNP's CLB set of transient conditions and design cycles. These are shown in the revised Table 4.1-8, "Thermal and Loading Cycles," in Appendix A, "FSAR Supplement," of the PBNP LRA. This set of transient conditions and design cycles was used in the evaluation of all PBNP TLAAs that required the use of transient conditions and design cycles. This set of transient conditions and design cycles was used in the pressurizer fatigue TLAA evaluation to show conformance with the CLB fatigue limits to the EOEL.

NRC Question RAI-4.3.5.3 Pressurizer Structural Integrity:

Provide clarification that the "plant-specific insurge/outsurge" fatigue analyses are based on the combination of the insurge/outsurge transient condition and the transients listed in the revised set of Steam Generator Replacement and Full Power Uprate transient conditions.

NMC Response:

The plant-specific insurge/outsurge fatigue analysis was based on a combination of the actual insurge/outsurge transients and other loadings experienced by the pressurizer and surge line components. Projections were made backward and forward in time to estimate the cumulative fatigue usage for these components for the entire 60-year operating life.

Extensive experience with fatigue monitoring has demonstrated that a significant system temperature differential (i.e., the difference between pressurizer water temperature and RCS hot leg temperature) is required to produce thermal fatigue in the surge line and lower head. This effect occurs during plant heatups and cooldowns. Other transients, such as a reactor trip, do not produce stresses above the minimum fatigue threshold.

Several types of loadings contribute to heatup and cooldown transients, including:

- Internal pressure
- Surge line piping thermal expansion
- Surge line piping thermal stratification
- Thermal shock, or "insurge/outsurge" temperature transients from flow reversals

The FatiguePro software installed at Point Beach Unit 1 and Unit 2 was used to evaluate these effects on the pressurizer locations affected by the insurge/outsurge transients.

FatiguePro computes stresses in various fatigue-sensitive components based on real plant data. A computational scheme was devised to compute the water temperature at various zones in the surge line and pressurizer lower head based on available temperatures, flows and other applicable instruments to capture any insurge/outsurge effect that the plant may experience during operation.

The following locations in the surge line and pressurizer lower head are monitored:

- Hot Leg Surge Nozzle
- Pressurizer Surge Nozzle
- Pressurizer Heater Penetration Weld
- Pressurizer Water Temperature Instrument Nozzle

The pressurizer heater penetration weld was determined to be the bounding location for fatigue usage in the surge line and pressurizer lower head. Plant data was available for Point Beach Units 1 and 2 from 1994 to present. The data was screened for heatup and cooldown transients to be analyzed by FatiguePro software. A cooldown followed by a heatup was assumed to represent a transient cycle.

Fatigue Usage Projections

Fatigue usage projections were based on the assumed number of future cycles that the plant will experience. For the purpose of projecting future fatigue usage, the incremental usage for a cooldown/heatup cycle was assumed to be the average incremental fatigue from the template periods.

The projected number of pressurizer heatups for the life of the plant is 100 for Unit 1 and 90 for Unit 2. Using these values and the average incremental fatigue usage for the periods in question, the projected fatigue usage for each location of interest were computed.

Backward Projections

Because the Point Beach plants may have historically operated at a higher system ΔT than in the template period of available plant data, a sensitivity analysis was performed to account for the possibly higher average incremental fatigue usage in the earlier time period.

For the time period before data was available, some Point Beach heatups and cooldowns were assumed to be at higher maximum system ΔTs than current operation for conservatism.

This sensitivity analysis was performed by running simulated data with higher ΔTs (by lowering the hot leg temperature) to determine a correlation between maximum ΔT and an increased fatigue usage factor.

This analysis demonstrated that increases in fatigue usage were small. The hot leg temperature was reduced by 100°F (which has the effect of increasing the maximum ΔT by the same amount). This resulted in a relatively small increase in average incremental fatigue usage for the highest usage location, the pressurizer heater weld.

In most cases the fatigue usage was dependent primarily on the rate of temperature change of the pressurizer water temperature, which is not expected to change significantly during operations with different system ΔTs and relatively few significant insurges and outsurges.

However, it was considered conservative to assume that on the average, increased system ΔTs during the earlier time frame resulted in a maximum 50% increase from the current operation's average incremental fatigue usage. For Unit 1, 53 RCS cooldown cycles occurred before 1994. For Unit 2, 39 RCS cooldown cycles occurred before 1994.

<u>Projections</u>

The 50% increase was assumed to apply to the first 53 cycles for Unit 1 (39 for Unit 2). In reality, only approximately 100 heatup/cooldown cycles for Unit 1 are expected for the plant based on the frequency of past heatup/cooldown occurrences (90 cycles for Unit 2).

The Point Beach operation during heatup and cooldown is relatively benign because large system temperature differentials (ΔTs) do not occur. In most cases the ΔT is less than 150°F. Therefore, any amount of insurge/outsurge due to flow reversal is unlikely to contribute significantly to future fatigue usage.

<u>Environmental Effects</u>

The fatigue usage projections discussed above were adjusted for the maximum effect of environmental fatigue by multiplying by 15.35. The resulting environmental cumulative usage factors are acceptable (less than 1.0), and are shown in the LRA Table 4.3.10.2.

<u>NRC Question RAI-4.3.5.4 Pressurizer Structural Integrity:</u>

Provide a description of the "Modified Operating Procedures" (page 4-45) that were used to minimize or eliminate in-surge/out-surge cycling.

NMC Response:

PBNP follows a water solid heatup/cooldown method for both units. The modified operating procedures set a maximum allowable ΔT limit of 210°F between the RCS hot leg and the pressurizer liquid space. This ensures that operation of the plant is within the ΔT limit assumed in the surge line thermal stratification analyses.

NRC Question RAI-4.3.5.5 Pressurizer Structural Integrity:

List the key Units 1 and 2 pressurizer components, and provide for each a summary of 60-year primary-plus-secondary stress intensities and cumulative fatigue usage factors (similar to revised Tables 4.3-1 and 4.3-2 in Appendix A of the LRA for components of the reactor vessel) for these components.

NMC Response:

In accordance with discussions with NRC staff on October 28, 2004, it was agreed that a summary of 60-year primary-plus-secondary stress intensities for the PBNP pressurizers need not be provided since the stress intensities are not critical for the review and are not a part of the PBNP CLB.

A summary of the PBNP pressurizers key component fatigue evaluation results is included in the following table.

PBNP Key Pressurizer Component Design Basis Cumulative Fatigue Results	
Component Description	CUF
Upper Head and Shell	[ ]b,c (1)
Lower Head Perforation	[ ]b,c (1)
Spray Nozzle	[ ]b,c (1)
Surge Nozzle	[ ]b,c (1)
Instrument Nozzle	[ ]b,c (1)
Lower Head Heater Well	[ ]b,c (1)
Immersion Heater	[ ]b,c (1)
Support Skirt and Flange	[ ]b,c (1)
Safety and Relief Nozzle	[ ]b,c (1)
Manway Pad	[ ]b,c (1)
Manway Cover	[ ]b,c (1)
Manway Bolts	[ ]b,c (1)

BRACKETED NUMBERS ARE WESTINGHOUSE PROPRIETARY

(1) - Westinghouse, "Power Uprate Project, Point Beach Nuclear Plant Units 1 and 2, Volume 1, NSSS Engineering Report," April 2002.

NRC Question RAI-4.3.7 - Pressurizer Surge Line Structural Integrity:

Provide a comparison of the CLB set of transient conditions and design cycles, and the revised set of Steam Generator Replacement and Full Power Uprate transient conditions and design cycles, that were used in the Units 1 and 2 pressurizer surge line fatigue TLAAs to show conformance with the CLB fatigue limit to the end of the period of extended operation.

NMC Response:

Westinghouse performed the original PBNP surge line thermal stratification evaluations. The analysis results are documented in WCAP-13509 and WCAP-13510, "Structural Evaluation of the Point Beach Units 1 & 2 Pressurizer Surge Lines, Considering the Effects of Thermal Stratification," dated October 1992. WCAP-13509 is the Westinghouse Proprietary Class 2 version of the analysis, and WCAP-13510 is the Westinghouse Proprietary Class 3 version. Copies of both WCAP-13509 and WCAP-13510 were transmitted to the NRC in Wisconsin Electric letter VPNPD-92-360, NRC-92-139, dated November 24, 1992, "Docket 50-266 and 50-301, Completion of the Reporting Requirements for Action Item 1.d of NRC IE Bulletin 88-11." The specific analysis transient conditions and design cycles are detailed in the noted WCAPs.

Westinghouse evaluated the impacts of the changes in the RCS conditions, thermal design transients, and a 60-year life on the PBNP surge line thermal stratification analyses. The impact of changes in the revised RCS conditions, thermal design transients, and the 60-year life extension were factored into determining the ASME stress levels and allowable stress levels for the surge line. This evaluation included a review of the fatigue analysis and the stratification loadings that were transmitted to the pressurizer nozzle from the surge line piping. The changes and the percent increases for the thermal design transients were tabulated and the impact on the cumulative fatigue usage factor was calculated. The forces and moments that were generated by the stratified conditions in the surge line also exist at the pressurizer nozzle. The power uprate conditions were reviewed to determine if the old enveloping loads on the nozzle changed significantly. Temperature differences between the hot leg and pressurizer were used to calculate stratified moments in the surge line piping. The difference between the old T_{hot} (hot leg temperature) and the new T_{hot} was determined and used in the determination of new nozzle loads.

The results of this evaluation for the pressurizer surge line stratification showed that the power uprate conditions changed the cumulative fatigue usage factor at the location of highest usage factor by a negligible amount. The calculated change in loadings on the pressurizer nozzle due to stratification for the power uprate conditions was not considered significant. The results presented in WCAP-13509 and WCAP-13510 remain unchanged.

NRC Question RAI-4.3.8:

Pressurizer Spray Header Piping Structural Integrity

This section states that: "In view of the lack of margin with the Unit 1 piping system analysis results for end of life extension (EOLE), additional analysis investigations were pursued. The original 88-08 analysis incorporated simplified analysis techniques and assumptions. It was not clear that the analysis was in fact conservative. The 88-08 analyses were re-performed using the original temperature monitoring data, and refined analysis techniques and assumptions." Provide a detailed description and basis of the "refined analyses techniques and assumptions" that were used in the 88-08 re-evaluation to reduce the 60-year CUF of 0.99 for the Unit 1 piping system to a 60-year CUF of 0.277.

NMC Response:

The refined analysis techniques and assumptions referred to evaluation of actual thermocouple data and to the use of special purpose programs for performing piping analysis. Details are discussed below.

In late 1989, as a response to NRC Bulletin 88-08 issues, PBNP installed two sets of three thermocouples on the horizontal section of the Unit 2 main and auxiliary spray piping to detect leakage and thermal stratification in the lines. These thermocouple sets were installed near the tee joining the auxiliary pressurizer spray line to one of the main spray lines. Thermally stratified conditions were discovered during heatup and normal operation. Using this data, Sargent and Lundy (S&L) performed a 40-year fatigue calculation for both units for the main and auxiliary spray piping, resulting in a fatigue usage in Unit 1 of 0.66 and 0.30 for Unit 2. The Unit 1 60-year fatigue usage projections yielded a value of 0.99. The S&L calculation included a simplified hand calculation of the thermal stratification stresses. The simplifying assumptions used may or may not have been conservative. Therefore, a further review was performed.

Review of the S&L work to determine a more accurate fatigue usage for a 60-year design life for Unit 1 consisted of:

1. Reviewing the thermocouple data to verify the basis for the stresses and contributions to fatigue of the operating parameters. The rationale for extrapolating the results of the data collection sample period to the entire plant life was revisited. Contributions to fatigue usage attributed to heatup, cooldown, auxiliary spray actuation, and thermal stratification due to valve leakage were identified.

2. Reviewing the simplified hand calculation of the thermal stratification stresses done by S&L. These calculations contained estimates of global bowing moment, radial local stress, and axial local stress. Some of the simplifying assumptions used were conservative and some were possibly not conservative. The correct thermal stratification stresses are determined using the Structural Integrity (SI) program TOPBOT.

Thermal stratification data was collected for a 153-day period, beginning with a plant heatup. The thermocouple data indicated that thermal stratification was present during most of the monitoring period, and the magnitude of the top-to-bottom gradient varied over time. The midlevel temperature, assumed to be indicative of the pipe average temperature also varied over time, not only due to plant heatup but also due to variations in spray demand. Thus, two types of thermal cycling occur: global thermal cycling, based on the mid-pipe temperature variations, which affects the thermal expansion moments in the pipe and the global stratification bowing effects; and thermal gradient cycling, based on the variation of the difference between the top and bottom pipe temperatures, which affects the local and global thermal stratification stresses.

In the S&L analysis, node point 210 in Unit 1 was identified as the limiting location for stress and fatigue usage. Node point 210 is located on the 3-inch line between the first main spray tee and the reducer before the second tee. Because the calculations affect all locations proportionally, point 210 remained limiting and was used for determining the 60-year fatigue usage.

The thermocouple data was reviewed in detail and temperature cycles were constructed. Two types of cycles were constructed: Type A cycles, which are thermal expansion moment cycles based on the midlevel pipe temperature variations; and Type B, local and global thermal stratification cycles, based on the top-to-bottom thermal gradient magnitude variations. Type A cycles of less than 100°F and type B cycles of less than 50°F are neglected because of small ΔTs. The peaks and valleys of these cycles were paired according to the ASME method of matching highest peak with lowest valley, second highest peak with second lowest valley, etc. This methodology was conservative because higher cyclic ranges produce exponentially higher fatigue usage.

The most severe top-to-bottom thermal stratification temperature profile is determined and modeled in the SI program TOPBOT. TOPBOT calculates the fixed-end thermal bowing moment and the local peak stresses due to the nonlinear, non-axisymmetric thermal gradient. The bowing moment was compared to that determined by S&L, and estimated piping stresses were scaled according to the more accurate moment determined by TOPBOT. These results and the local peak stresses determined by TOPBOT, were then scaled to the varied cycle amplitudes to develop stress cycles. These results were used in a revised fatigue usage calculation to determine the projected fatigue usage for a 60-year plant life.

The thermal stratification appears to have been caused by leakage past auxiliary spray isolation valve CV-296. Although leakage past CV-296 may initially be hot as it is taken from the charging system, the flow is sufficiently small and the valve is far enough away from the main spray tee (i.e., 80 feet) that the leakage mixes with the stagnant fluid in the auxiliary spray line and arrives at the tee at roughly containment ambient temperature. Although preventive maintenance, including changes to the internals, was performed on valve CV-296 after measurement of the stratification, and it was unlikely that leakage continued to occur to the degree that was measured, for purposes of bounding analysis it was conservatively assumed that the same amount of leakage and consequent magnitude of stratification continued to exist for the remainder of plant life.

It was also assumed that the 153-day period of data was representative of all plant operation. This assumption was conservative because this 153-day period included a heatup, which contained many more stress cycles than did normal steady state operation. The heatup was also considered representative of a cooldown, because the mean pipe temperature range is the same, and there are multiple main spray and auxiliary spray actuations, as well as numerous variations in spray flow rate. Heatups tend to contain more thermal cycles than cooldowns, as there are typically more procedural steps, hold points, and tests conducted than during cooldown.

To extrapolate the 153-days of cycles to 60 years of operation, the following assumptions were used:

- The cycles are considered to be repeated every 153 days of operation, despite the fact that plant heatups do not occur this often
- The plant was not operated for one month per two years due to refueling outages

Using these assumptions, S&L calculated the global thermal stratification fixed-end moment, and the local thermal stratification stresses due to the nonlinear top-to-bottom thermal gradient. These calculations were done by hand and were somewhat simplified in that they assumed that the thermocouple temperatures measured at the outside of the pipe were representative of the inside fluid temperatures, and did not account for the through-wall thermal gradients. A more accurate thermal stratification stress analysis was done using the SI program TOPBOT.

TOPBOT solves the transient thermal and stress response within a pipe subjected to a step or ramp change in boundary temperatures and heat transfer coefficients. Initial temperature conditions are specified, and then a temperature change is applied to either the top or bottom fluid in the pipe, or both. The pipe is considered to be two dimensional at a pipe cross-section (or assumed to be extremely long in the axial direction). Symmetry about the pipe vertical centerline is assumed. The pipe is modeled with rectilinear elements within the R-theta coordinate system. Stresses and temperatures are computed at the center of each element (mean radius and mean angular locations). In addition, temperatures and stresses are computed at the inside and outside surfaces of the pipe, based on the "steady state" temperature distribution between the surface elements and external boundary temperatures. Required thermal input parameters include thermal conductivity and the product of the density and specific heat, modulus of elasticity, coefficient of thermal expansion, and Poisson's ratio. The thermal boundary conditions are input as internal and external temperature distributions and heat transfer distributions. For most problems, the initial pipe temperature is uniform: a uniform temperature and heat transfer coefficient is specified on the outside of the pipe, and two sets of temperature and heat transfer coefficients are specified inside the pipe, representing the top and bottom temperatures and flow rates. These two sets of temperatures, heat transfer coefficients, and their interface level can be varied linearly by specifying different values at specific points in time. The pipe temperature distribution is determined using a classical finite difference method. An energy balance is written for each element of the model.

For this analysis, the maximum top-to-bottom temperature distribution was modeled. The stress results were scaled for smaller thermal gradients. The outside pipe temperatures at the top, midlevel and bottom of the pipe were observed. In order to determine accurate thermal stratification stresses, the inside fluid temperatures that produce the temperature distribution measured at the outside of the pipe were determined. This required some trial and error, as it is a function of the hot-cold fluid interface level and the convective heat transfer coefficients at the top and bottom inside surface of the pipe; these coefficients in turn depend on the flow rates and temperatures of the fluid levels.

The inside surface forced convection heat transfer coefficients were determined using the following relation for turbulent flow:

$$h = 0.023\, Re^{0.8}\, Pr^{0.4}\, k / D$$

where

Re = Reynolds number = $\rho VD/\mu > 4000$ for turbulent flow
Pr = Prandtl number = $\mu c_p k$
k = thermal conductivity, BTU-hr-ft/°F
D = hydraulic diameter = $4A/P$
A = flow area
P = flow perimeter
V = flow velocity, ft/sec
ρ = density, lbm/ft^3
μ = dynamic viscosity, lbm /ft-sec
c_p = specific heat at constant pressure

After trial and error, the outside pipe temperature distribution was replicated if the inside fluid temperatures were 530°F at the top (some heat loss from RCS cold leg temperature) and 100°F at the bottom (containment ambient temperature), the interface level was at 148° from the top of the pipe, and the top fluid velocity was 0.516 ft/sec, or 3.1 gpm (bypass flow circulation) and the bottom fluid velocity was 0.19 ft/sec, or 0.3 gpm (leakage past auxiliary spray control valve). The actual valve leakage and the other parameters may slightly deviate, but as long as the temperature distribution is established in the analytical model, the stress results remain representative.

<u>Thermal Stratification Analysis Results</u>

The results of the TOPBOT thermal stratification analysis were the following:

Fixed-End Moment: 108.19 in-kip (for a 300°F top-to-bottom thermal gradient)

Local Stress (maximum location): 40.31 ksi (for a 300°F top-to-bottom thermal gradient) = 134.37 psi/°F

Stress Combinations and Fatigue Usage

The stress and fatigue usage were computed at the limiting location, node point 210. This location is classified as ASME Code Class 2. ASME Section III, subsection NC-3600, does not provide explicit fatigue usage criteria; however, the approach used by Markl in his fatigue testing of piping components was used, which has been implicitly adopted in NC-3600:

$$i\,S\,(N)^{0.2} = 280{,}000$$

where

S = stress range, psi
i = stress intensification factor = 1.0 at node point 210 (straight pipe)
N = number of allowable full range stress cycles at stress S

The stress is the total of the contributions from thermal expansion moments, global thermal stratification, and local thermal stratification.

The thermal expansion moment stress at point 210 was (32113 – 6960) = 25,153 psi.

The global thermal stratification stress calculated was 6960 psi. This stress was obtained by applying the stratification fixed-end moment to the piping model. A more accurate fixed-end moment was calculated by TOPBOT as described above. Therefore, the stress adjusted by the ratio of the fixed-end moments is:

Global stratification stress = 6960 (108.19/104.34) = 7,217 psi.

Per the S&L analysis, the magnitude of stratification at point 210 is 0.93 of that measured at the instrumented point. The local stress is:

= 134.37 psi/°F (300) (.93) = 37,489 psi.

The stress cycles are grouped in the following manner:

1. 1x140=140 cycles of 455°F thermal expansion moment range + 300°F global stratification + 300°F local stratification.

2. 2x140=280 cycles of 150°F thermal expansion moment range + 300°F global stratification + 300°F local stratification.

3. 1x140=140 cycles of 150°F thermal expansion moment range + 230°F global stratification + 230°F local stratification.

4. 16x140=2240 cycles of 230°F global stratification + 230°F local stratification.

5. 11x140=1540 cycles of 110°F global stratification + 110°F local stratification.

The thermal expansion moment stress is scaled according to the temperature range of the cycle. For example, for a 150°F range, the thermal expansion range is:

= 25,153 (150/455) = 8,292 psi.

The stratification stresses, global and local, are scaled according to the top-to-bottom gradient. Thus for a 230°F gradient, the global stratification stress is:

= 7,217 (230/300) = 5,533 psi.

and the local stratification stress is:

= 37,489 (230/300) = 28,742 psi.

The number of stress cycles for groups two through five were then converted into an equivalent number of full range stress cycles using equation (2) in paragraph NC-3611.2 (e) (3) of the ASME B&PV Code. This relationship was developed by Markl in his fatigue testing of piping components and has been incorporated into the ASME Class 2/3 piping Code:

$$N = N_1 + (S_2/S_1)^5 N_2 + (S_3/S_1)^5 N_3 + (S_4/S_1)^5 N_4 + (S_5/S_1)^5 N_5$$

Thus, following this approach, 280 cycles of group 2 are equivalent to 70.36 full range cycles, for example. The total equivalent full range cycles of all five groups is 287. Using Markl's equation, the allowable number of cycles at a stress of 69,859 psi is 1,034. Therefore, the total fatigue usage is 0.277.

<u>Conclusions</u>

The result of the stress and fatigue analysis is that for the limiting location on the Unit 1 pressurizer spray line, the calculated fatigue usage for a 60-year plant design life is 0.277. This is well below the allowable of 1.0. The stresses conservatively assume that a significant amount of leakage past the auxiliary spray control valve continues to exist. Thus the stress and fatigue usage for this line is acceptable for a 60-year design life.

NRC Question RAI- 4.3.10.1 Environmental Effects on Fatigue:

For the USAS B31.1 locations, provide a description of the PBNP-specific simplified ASME Section III fatigue analyses that were used to calculate environmentally based cumulative usage factors.

NMC Response:

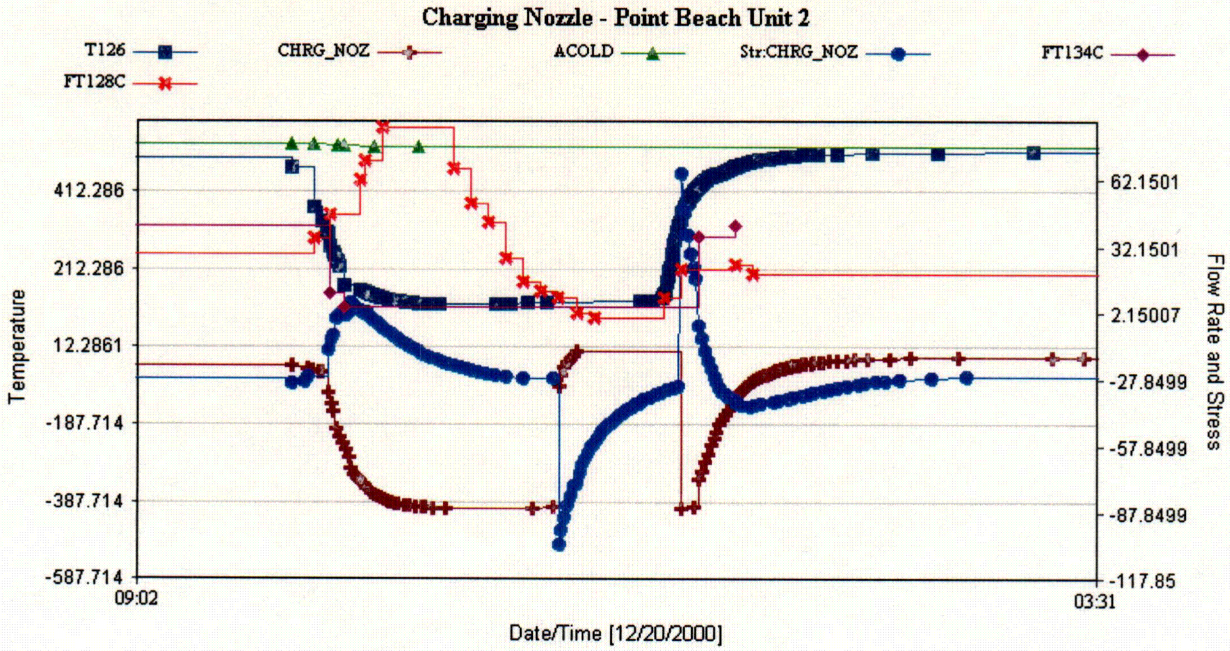
Charging Nozzle

This location was analyzed using actual plant data and the projected number of cycles in the charging nozzle model included in a FatiguePro application for PBNP and the tensile strain-integrated environmental correction factor (F_{en}) factor.

The PBNP design basis includes several design transients that affect the charging nozzles. The most severe of these events is the loss of charging and loss of letdown with delayed return to service. Examination of actual PBNP Units 1 and 2 plant data since 1994 revealed one actual event that accurately represents this design transient. This event was used to represent a bounding loss of charging/loss of letdown event.

Normal fatigue usage for this event was computed. An F_{en} was then determined to account for environmental effects. The resulting total usage was applied over the expected number of occurrences for this event (17 loss of charging and loss of letdown events projected for the 60-year operating life of each units). The incremental fatigue usage attributed to the loss of charging/loss of letdown event was determined to be 0.00695. The bounding transient is depicted in the following figure.

Transient Instrument Values and Stress for Bounding Transient



C01

Using these results, the 60-year projection for these events (no environmental effects) is:

CUF = (17 events) ∗ (0.00695 per event) = 0.11815

Environmental Effects – F_{en} Approach

The fatigue usage analysis above does not consider environmental effects on the fatigue curve. An environmental fatigue factor (F_{en}) will be determined based on the equations provided in NUREG/CR-5704. Using this methodology, the F_{en} factor is computed as a function of three parameters using the following equation:

$$F_{en} = \exp(0.935 - T^* \cdot \varepsilon dot^* \cdot O^*)$$

where:

$T^* = 0$	$T < 200C$
$T^* = 1$	$T \geq 200C$

$\varepsilon dot^* = 0$	$\varepsilon dot > 0.4\%/sec$
$\varepsilon dot^* = \ln(\varepsilon dot/0.4)$	$0.0004 \leq \varepsilon dot \leq 0.4\%/sec$
$\varepsilon dot^* = \ln(0.0004/0.4)$	$\varepsilon dot \leq 0.0004\%/sec$

$O^* = 0.260$	$DO < 0.05$ ppm
$O^* = 0.172$	$DO \geq 0.05$ ppm

For this transient, the factors are:

- Strain range – over the tensile portions of the plant transient
- Strain rate – computed over the range of tensile strain range
- Temperature - conservatively assumed to be greater than 200°C
- Dissolved oxygen – conservatively assumed to be less than 0.05 ppm

Individual F_{en} values were integrated over the tensile strain range of the transient stress cycle(s) being analyzed. For the purpose of determining strain rate, the stress changes for each time step with increasing stress were converted to strain by dividing by the modulus of elasticity for stainless steel (28.3 x 10^3 ksi) from the ASME fatigue curve. The strain difference was divided by the length of the time step, then converted into units of (% strain/sec).

For each time step, incremental F_{en} was computed using the equation shown above. An effective F_{en} for the entire transient was computed by integrating F_{enk} for each time step with the strain step associated with that time step over the entire tensile strain range. The effective F_{en} was computed to be 6.994.

The environmentally-assisted fatigue of less than 1.0 projected to 60 years of operation is thus:

$$CUF_{EAF} = 0.11815 * 6.994 = 0.8264$$

Safety Injection Nozzle and RHR Tee

These two locations were analyzed using design transients and the design number of cycles in a combined ASME Code Section III NB-3600 Class 1 plant-specific piping model of the safety injection (SI) piping system and RHR system.

The SI piping, including the RHR tee, was modeled using the computer program, PIPESTRESS. To perform an ASME Section III Class 1 piping fatigue analysis with PIPESTRESS, thermal transients and thermal expansion cases were defined. To evaluate the fatigue usage with PIPESTRESS, the thermal expansion cases correspond to the final temperature of each analyzed transient plus the steady state operating case.

Typically, the governing contribution to fatigue usage is from thermal stresses, not pressure stresses. Thus, the pressure for each case will be the operating pressure. Use of the operating pressure is conservative when used to calculate pressure stresses. This will also be conservative when used in the fatigue evaluation.

Modes of operation were defined. The forces and moments due to differential thermal expansion as analyzed by the piping program, PIPESTRESS, were included in the fatigue evaluation.

The number of design transients that will occur through the end of the extended operation (60 years) were assumed to be less than or equal to the design limit (40 years) for each design transient. Seismic loading and thermal anchor movements were considered for fatigue analysis.

The results of the fatigue evaluation show that the fatigue usage at the SI to cold leg branch connection (SI Nozzle) is 0.0013 and the tee from the RHR to the SI (RHR Tee) is 0.0146. Application of the maximum possible F_{en} of 15.35 produced environmentally assisted fatigue usage values of 0.02 and 0.224, for the SI Nozzle and RHR Tee, respectively result in values well below 1.0.

NRC Question RAI- 4.3.10.2:

Environmental Effects on Fatigue

The Pressurizer CUFs are determined based on EPRI MRP-47 methodology. The staff has not endorsed MRP-47. Provide the environmentally assisted CUFs for the Pressurizer locations, based on the staff-accepted methodology as stated in Sections 4.3.2.2 and 4.3.3.2 of NUREG-1800.

NMC Response:

The pressurizer components were evaluated for environmental effects on fatigue per the direction of Applicant Action Item 3.3.1.1-1 of the NRC SER of Westinghouse GTR WCAP-14574-A. Applicant Action Item 3.3.1.1-1 of the NRC SER of Westinghouse GTR WCAP-14574-A does not specify a requisite method for addressing the environmental effects on fatigue. NUREG-1800 does not require that pressurizer components be evaluated for environmental effects on fatigue.

Sections 4.3.2.2 and 4.3.3.2 of NUREG-1800 note that formulas for calculating the environmental life correction factors are those contained in NUREG/CR-6583 for carbon and low-alloy steels, and in NUREG/CR-5704 for austenitic stainless steels.

Application of the NUREG/CR-6583 and 5704 formulas for calculating the environmental life correction factors for the PBNP pressurizers, versus the EPRI MRP-47 Appendix B formulas will not have an significant impact on the pressurizer components evaluations. This is because the EPRI MRP-47 "Z" factor was not used in the evaluations, the evaluation temperature of 345 °C results in the stainless steel formulas being identical between the two methods, and the carbon or low alloy steel formulas were not applied since the dissolved oxygen (DO) values were well below the threshold values.

The environmentally adjusted fatigue (EAF) CUF for the spray nozzle safe end is not affected since this location is stainless steel and the formulas are identical in the analyzed temperature range.

The 60-year EAF CUF for the surge nozzle safe end is not affected since this location is stainless steel and the formulas are identical in the analyzed temperature range.

The 60-year EAF CUF for the limiting carbon steel portion of the surge nozzle was noted to be the same as the design basis value since the DO level is well below the threshold level. Using the NUREG/CR-6583 formula, the design basis value would be multiplied by an F_{en} of 1.17 as a result of the temperature, regardless of the low DO level. The resulting 60-year EAF CUF would be 0.73, which is acceptable since it is less than 1.0.

The 60-year EAF CUF for the carbon steel / low alloy steel junction of the upper head and shell was noted to be the same as the design basis value since the DO level is well below the threshold level. Using the NUREG/CR-6583 formula, the design basis value would be multiplied by an F_{en} of 1.65 as a result of the temperature, regardless of the low DO level. The resulting 60-year EAF CUF would be 1.28. Since the evaluation for the carbon steel / low alloy steel junction of the upper head and shell is based on the design transient set, the results of the evaluation are extremely conservative in both the severity and numbers of transients. Significant reductions in the estimates are possible if adjustments are made to remove the operational transients that are not experienced or practiced at PBNP. The EPRI FatiguePro software program was customized to monitor the carbon steel to low alloy steel junction of the upper head and shell at PBNP. An analysis was performed based on available template sets of real plant data to determine the incremental fatigue usage factor for known plant transients. A cumulative usage factor for the operating life of the plant was computed based on the results of real plant data and expected future usage was computed using projections of expected plant cycles. The 60-year CUF for the Unit 1 pressurizer's carbon steel / low alloy steel junction of the upper head and shell bounds the 2 units and is projected to be 0.156. Applying the maximum environmental fatigue correction factor of 2.53 to the projected CUF of the carbon steel / low alloy steel junction of the upper head and shell location, results in a conservative 60-year EAF CUF of 0.39. This demonstrates adequate structural integrity, including the effects of environmental conditions, for a projected 60-year operational period.

The 60-year EAF CUF for the safety and relief nozzle safe ends was noted to be the same as the design basis value since the CUF of these components is zero.

The 60-year EAF CUF for the limiting carbon steel portion of the safety and relief nozzle was noted to be the same as the design basis value since the DO level is well below the threshold level. Using the NUREG/CR-6583 formula, the design basis value would be multiplied by an F_{en} of 1.17 as a result of the temperature, regardless of the low DO level. The resulting 60-year EAF CUF would be 0.174, which is acceptable since it is less than 1.0.

The environmentally adjusted fatigue (EAF) CUF for the instrument nozzle is not affected since this location is stainless steel and the formulas are identical in the analyzed temperature range.

The environmentally adjusted fatigue (EAF) CUF for the heater well is not affected since this location is stainless steel and the formulas are identical in the analyzed temperature range.