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~~PROPRIETARY INFORMATION – WITHHOLD UNDER 10 CFR 2.390~~

10 CFR 50.90
10 CFR 2.390

October 11, 2013

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Peach Bottom Atomic Power Station, Units 2 and 3
Renewed Facility Operating License Nos. DPR-44 and DPR-56
NRC Docket Nos. 50-277 and 50-278

Subject: Extended Power Uprate License Amendment Request – Supplement 12 Response to Request for Additional Information

Reference:

1. Exelon letter to the NRC, "License Amendment Request - Extended Power Uprate," dated September 28, 2012 (ADAMS Accession No. ML122860201)
2. NRC letter to Exelon, "Request for Additional Information Regarding License Amendment Request for Extended Power Uprate (TAC Nos. ME9631 and ME9632)," dated September 11, 2013 (ADAMS Accession No. ML13253A416)

In accordance with 10 CFR 50.90, Exelon Generation Company, LLC (EGC) requested amendments to Facility Operating License Nos. DPR-44 and DPR-56 for Peach Bottom Atomic Power Station (PBAPS) Units 2 and 3, respectively (Reference 1). Specifically, the proposed changes would revise the Renewed Operating Licenses to implement an increase in rated thermal power from 3514 megawatts thermal (MWt) to 3951 MWt. During their technical review of the application, the NRC Staff identified the need for additional information. Reference 2 provided the Request for Additional Information (RAI).

This letter addresses requests from the staff of Mechanical and Civil Engineering Branch (EMCB) of the U. S. Nuclear Regulatory Commission to provide information in support of the request for amendment for the extended power uprate. Responses to these questions are provided in the attachments to this letter.

GE Hitachi Nuclear Energy America (GEH) considers portions of the information provided in the responses in Attachment 1 to be proprietary and, therefore, exempt from public disclosure pursuant to 10 CFR 2.390. The proprietary information in Attachment 1 is identified; a non-proprietary version of this information is provided in Attachment 2. In

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**Attachment 1 contains ~~Proprietary Information.~~**
**When separated from Attachment 1, this document is decontrolled.**

accordance with 10 CFR 2.390, EGC requests Attachment 1 be withheld from public disclosure. An affidavit supporting this request for withholding is included as Attachment 3.

EGC has reviewed the information supporting a finding of no significant hazards consideration and the environmental consideration provided to the U. S. Nuclear Regulatory Commission in Reference 1. The supplemental information provided in this submittal does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration. Further, the additional information provided in this submittal does not affect the bases for concluding that neither an environmental impact statement nor an environmental assessment needs to be prepared in connection with the proposed amendment.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), EGC is notifying the Commonwealth of Pennsylvania and the State of Maryland of this application by transmitting a copy of this letter along with the non-proprietary attachments to the designated State Officials.

There are no regulatory commitments contained in this letter.

Should you have any questions concerning this letter, please contact Mr. David Neff at (610) 765-5631.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 11th day of October 2013.

Respectfully,

Kevin F. Borton
Manager, Licensing – Power Uprate
Exelon Generation Company, LLC

Attachments:

1. Response to Request for Additional Information – EMCB - Proprietary
2. Response to Request for Additional Information – EMCB
3. Affidavit in Support of Request to Withhold Information

| | | |
|---|---|---|
| cc: | USNRC Region I, Regional Administrator | w/attachments |
| | USNRC Senior Resident Inspector, PBAPS | w/attachments |
| | USNRC Project Manager, PBAPS | w/attachments |
| | R. R. Janati, Commonwealth of Pennsylvania | w/o proprietary attachment |
| | S. T. Gray, State of Maryland | w/o proprietary attachment |

**Attachment 3**

**Peach Bottom Atomic Power Station Units 2 and 3**

**NRC Docket Nos. 50-277 and 50-278**

**AFFIDAVIT**

**Note**

Attachment 1 contains proprietary information as defined by 10 CFR 2.390. GEH, as the owner of the proprietary information, has executed the enclosed affidavit, which identifies that the proprietary information has been handled and classified as proprietary, is customarily held in confidence, and has been withheld from public disclosure. The proprietary information has been faithfully reproduced in the attachment such that the affidavit remains applicable.

## GE-Hitachi Nuclear Energy Americas LLC

## AFFIDAVIT

**I, Peter M. Yandow,** state as follows:

(1) I am the Vice President, Nuclear Plant Projects/Services Licensing, Regulatory Affairs, GE-Hitachi Nuclear Energy Americas LLC (GEH), and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in Enclosure 1 of GEH letter, GEH-PBAPS-EPU-428, "GEH Response to NRC EMCB RAIs 1, 7, 8, 17, 20, 22 and 23 and SRXB RAI 30," dated October 7, 2013. The GEH proprietary information in Enclosure 1, which is entitled "GEH Response to NRC EMCB RAIs 1, 7, 8, 17, 20, 22 and 23," is identified by a dark red dotted underline inside double square brackets. [[This sentence is an example.[3]]]. In each case, the superscript notation [3] refers to Paragraph (3) of this affidavit that provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner or licensee, GEH relies upon the exemption from disclosure set forth in the *Freedom of Information Act* (FOIA), 5 U.S.C. Sec. 552(b)(4), and the Trade Secrets Act, 18 U.S.C. Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.390(a)(4) for trade secrets (Exemption 4). The material for which exemption from disclosure is here sought also qualifies under the narrower definition of trade secret, within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975 F.2d 871 (D.C. Cir. 1992), and Public Citizen Health Research Group v. FDA, 704 F.2d 1280 (D.C. Cir. 1983).

(4) The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a. and (4)b. Some examples of categories of information that fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by GEH's competitors without license from GEH constitutes a competitive economic advantage over GEH or other companies.

b. Information that, if used by a competitor, would reduce their expenditure of resources or improve their competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

c. Information that reveals aspects of past, present, or future GEH customer-funded development plans and programs, that may include potential products of GEH.

d. Information that discloses trade secret or potentially patentable subject matter for which it may be desirable to obtain patent protection.

(5) To address 10 CFR 2.390(b)(4), the information sought to be withheld is being submitted to the NRC in confidence. The information is of a sort customarily held in confidence by GEH, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GEH, not been disclosed publicly, and not been made available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary or confidentiality agreements that provide for maintaining the information in confidence. The initial designation of this information as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure are as set forth in the following paragraphs (6) and (7).

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, who is the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge, or who is the person most likely to be subject to the terms under which it was licensed to GEH. Access to such documents within GEH is limited to a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist, or other equivalent authority for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GEH are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary or confidentiality agreements.

(8) The information identified in paragraph (2) above is classified as proprietary because it contains results of analyses performed using the GEH EPU methodology including proprietary technical methods and processes. Development of these methodologies and the supporting analysis techniques and information, and their application to the design, modification, and processes were achieved at a significant cost to GEH.

The development of the evaluation methodology along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GEH asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GEH's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GEH's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

## GE-Hitachi Nuclear Energy Americas LLC

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GEH. The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial. GEH's competitive advantage will be lost if its competitors are able to use the results of the GEH experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GEH would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GEH of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing and obtaining these very valuable analytical tools.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 7th day of October, 2013.

Peter M. Yandow

Peter M. Yandow
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Regulatory Affairs
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**Attachment 2**

**Peach Bottom Atomic Power Station Units 2 and 3**

**NRC Docket Nos. 50-277 and 50-278**

**Response to Request for Additional Information – EMCB**

## Response to Request for Additional Information

## Mechanical and Civil Engineering Branch

By letter dated September 28, 2012, Exelon Generation Company, LLC (Exelon) submitted a license amendment request for Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3. The proposed amendment would authorize an increase in the maximum power level from 3514 megawatts thermal (MWt) to 3951 MWt. The requested change, referred to as an extended power uprate (EPU), represents an increase of approximately 12.4 percent above the current licensed thermal power level.

The NRC staff has reviewed the information supporting the proposed amendment and by letter dated September 11, 2013, (ADAMS Accession No. ML13253A416) has requested additional information. During a conference call between Mr. Borton of EGC and Mr. Ennis of NRC, conducted on September 10, it was agreed that EMCB RAI-5 would be deleted and that EGC would provide responses to the remaining questions by October 11, 2013. The responses to those questions are provided below.

### EMCB-RAI-1

Table 1-2 of the Power Uprate Safety Analysis Report (PUSAR)[1] contains information on plant parameters for CLTP conditions and the proposed EPU conditions. The licensee is requested to update Table 1-2 by adding a column for original licensed thermal power (OLTP) conditions. Please include design and maximum temperatures and pressures for the vessel inlet and outlet reactor recirculation system (RRS) nozzles, feedwater (FW) inlet and main steam (MS) outlet, and core spray (CS) inlet.

### RESPONSE

PUSAR Table 1-2 is updated below by adding a column for OLTP conditions. Table 1-1 provides the design and maximum temperatures for the requested RPV inlet and outlet nozzles. All nozzle maximum pressures are the same as the maximum normal dome pressure and the design pressure, 1250 psig, remains unchanged from OLTP to EPU.

**PUSAR Table 1-2 with OLTP Data**

| Plant Operating Conditions | OLTP | CLTP(1) | EPU |
|---|---|---|---|
| Thermal Power (MWt) | 3293 | 3514 | 3951 |
| Vessel Steam Flow (Mlb/hr) (2) | 13.37 | 14.387 | 16.171 |
| Full Power Core Flow Range | | | |
| Mlb/hr | 76.9 to 107.63 | 84.87 to 112.75 | 101.48 to 112.75 |
| % Rated | 75.0 to 105.0 | 82.8 to 110.0 | 99.0 to 110.0 |
| Maximum Normal Dome Pressure (psia) | 1020 | 1050 | 1050 |
| Maximum Normal Dome Temperature (°F) | 547 | 550.5 | 550.5 |

[1] A proprietary (i.e., non-publicly available) version of the PUSAR is contained in Attachment 6 to the application dated September 28, 2012. A non-proprietary (i.e., publicly available) version of the PUSAR is contained in Attachment 4 to the application dated September 28, 2012.

| Plant Operating Conditions | OLTP | CLTP(1) | EPU |
|---|---|---|---|
| Pressure Upstream of TSV (psia) | 950 | 994 | 979 |
| Full Power Feedwater | | | |
| Flow (Mlb/hr) | 13.37 | 14.355 | 16.139 |
| Temperature (°F) | 376.1 | 381.5 | 381.5 |
| Core Inlet Enthalpy (Btu/lb)(3) | 521.5 | 524.3 | 521.6 |

Notes:
(1) *Based on current reactor heat balance*
(2) At normal FW heating.
(3) At 100% core flow conditions

**Table 1-1**
**RPV Inlet and Outlet Nozzle Temperatures**

| Reactor Nozzle | OLTP | CLTP | EPU |
|---|---|---|---|
| RRS Outlet Design Temperature | 575°F | 575°F | 575°F |
| RRS Outlet Maximum Temperature | 546°F | 551°F | 551°F |
| RRS Inlet Design Temperature | 575°F | 575°F | 575°F |
| RRS Inlet Maximum Temperature | 546°F | 551°F | 551°F |
| FW Inlet Nozzle Design Temperature | 575°F | 575°F | 575°F |
| FW Inlet Nozzle Maximum Temperature | 573°F | 573°F | 573°F |
| MS Outlet Nozzle Design Temperature | 575°F | 575°F | 575°F |
| MS Outlet Nozzle Maximum Temperature | 546°F | 551°F | 551°F |
| CS Nozzle Design Temperature | 575°F | 575°F | 575°F |
| CS Nozzle Maximum Temperature | 546°F | 551°F | 551°F |

**EMCB-RAI-2**

Please verify that the design calculations to demonstrate that systems, structures and components (SSCs) credited to and/or affected by the proposed EPU have been completed and that controlled documentation exists which finds that these SSCs are structurally adequate to perform their intended design functions under EPU conditions.

**RESPONSE**

All design calculations have been completed to demonstrate that SSCs credited in the EPU safety analyses are structurally adequate to perform their intended design functions at EPU conditions.

Exelon continues to refine the final design and analyses in order to reduce the scope of required field work and radiological dose expected for plant modifications associated with the EPU project. The EGC Configuration Change Process will ensure that any revisions to the final designs are controlled and will continue to be structurally adequate to perform their intended design functions under EPU conditions.

**EMCB-RAI-3**

Page 2-41 of the PUSAR states, "For PBAPS, HELB [high-energy line breaks] locations in MS piping inside containment are not based on stress criteria." Please clarify whether the current licensing basis (LB) and design basis (DB) require postulation of pipe failures at specific locations inside containment. If so, please discuss the methodology and criteria used to postulate pipe ruptures inside containment for CLTP and EPU. If the methodology and criteria are different for EPU and CLTP, provide a technical justification that reconciles the differences.

**RESPONSE**

Specific main steam piping break locations within containment are not postulated as part of the PBAPS design and licensing basis. The EPU methodology and criteria for HELB is unchanged from the CLTP analysis.

**EMCB-RAI-4**

Please provide the following:

a) PUSAR page 2-41 lists high energy piping inside and outside containment that could potentially be affected by the proposed EPU. Please provide a justification which demonstrates that plant SSCs (including but not limited to block walls) susceptible to differential pressures resulting from postulated high energy pipe failures inside and outside containment, are capable of maintaining their structural adequacy within DB established limits at EPU conditions.

b) PUSAR Table 2.2-1 shows pressure and temperature increases due to EPU from postulated reactor water cleanup (RWCU) line breaks. Please quantify these increases and provide a technical justification to reconcile these EPU increases on potentially affected SSCs by adding more detail to PUSAR Section 2.2.1.2.1, "RWCU Line Breaks." Specify whether the loads used in the structural calculations of the current analyses-of-record (AOR) bound the loads due to these increases. If not, discuss required structural reevaluations and reconcile any differences from the AOR methodology and criteria used.

**RESPONSE**

This response supplements the information in PUSAR Section 2.2.1.2.1.

a) As described in PUSAR Section 2.2.1, EPU has no effect on mass and energy releases or dynamic effects from postulated steam line breaks. For liquid line breaks, FW and RWCU are the only high energy systems which experience an increase in pressure with EPU.

   The increased FW pressure and temperature with EPU will have a negligible impact on FW line break mass and energy releases and dynamic effects, including differential pressures. The current dynamic effects analyses of record remain bounding. The effects of a FW line break on MS tunnel pressures and temperatures remain bounded by a MS line break. No new high energy pipe break locations are postulated with EPU. Therefore, the SSCs susceptible to differential pressures resulting from these high energy pipe failures are capable of maintaining their structural adequacy at EPU conditions.

As described in PUSAR Section 2.2.1.2.1, the design basis RWCU line break analysis has been revised for changes not related to EPU. Thus, only the re-analyzed RWCU line breaks have the potential to change the compartment pressurization loading on structures. These effects are described further in Part (b) below.

b) The design basis RWCU line break analysis has been revised for changes not related to EPU based on deficiencies found in the existing analysis of record. For the new analysis, RWCU line break mass and energy releases are calculated to bound both CLTP and EPU conditions. Compartment pressurization and environmental temperature effects resulting from the increase in RWCU mass and energy releases were also calculated. The resulting increases in pressure and temperature shown in Table 4-1 and Table 4-2 below are due to the RWCU analysis change, and any contribution due to EPU is very small and therefore negligible. RWCU breaks are postulated in four locations:

- Isolation Valve Compartment
- RWCU Pump Rooms
- Regenerative Heat Exchanger Room
- Non-Regenerative Heat Exchanger Room

The increases in compartment peak pressures for postulated RWCU breaks in each of these rooms are shown in Table 4-1 below.

**Table 4-1**
**Increased Compartment Pressures for Postulated RWCU Line Breaks [(1)]**

| Compartment | Previous AOR Compartment Peak Pressure (psia) | Revised AOR Compartment Peak Pressure (psia) | Change in Peak Pressure (psi) |
|---|---|---|---|
| Isolation Valve Compartment | 16.8 | 16.8 | 0 |
| RWCU Pump Rooms | 16.8 | 18.8 | +2.0 |
| Regenerative Heat Exchanger Room | 17.6 | 18.0 | +0.4 |
| Non-Regenerative Heat Exchanger Room | 15.0 | 16.7 | +1.7 |

(1) Bounding pressure for Unit 2 and Unit 3 analyses.

An evaluation was performed to assess the structural capability to withstand the increased peak compartment pressures. The evaluation concludes that no structural failures or penetration seal failures will result from the increase in calculated peak compartment pressures from postulated RWCU line breaks. The applicable structural calculations remain bounding.

The environmental temperatures in the Reactor Building areas which are increased as a result of the re-analyzed RWCU line breaks are summarized in Table 4-2 below. Peak room temperatures resulting from the re-analyzed RWCU line breaks for all other rooms not shown in the table below are bounded by peak temperatures reported in the existing analysis of record.

**Table 4-2**
**Increased Environmental Temperatures for Postulated RWCU Line Breaks (1)**

| Compartment | Previous AOR Peak Room Temperature (°F) | Revised AOR Peak Room Temperature (°F) | Change in Peak Room Temperature (°F) |
|---|---|---|---|
| Non-Regenerative Heat Exchanger Room | 137 | 213 | +76 |
| RWCU Backwash Transfer Pump Room | 118 | 130 | +12 |

(1) Bounding temperature of rooms for Unit 2 and Unit 3 analyses

**EMCB-RAI-5**

*Question deleted following conference call on September 10, 2013. No response needed.*

**EMCB-RAI-6**

NRC Regulatory Issue Summary (RIS) 2008-30, "Fatigue Analysis of Nuclear Power Plant Components," identified a concern with the simplified single-stress methodology used by some license renewal applicants to perform fatigue calculations and as input for on-line fatigue monitoring programs. This methodology was being used in lieu of the American Society of Mechanical Engineers (ASME) Code, Section III, Subsection NB method which requires licensees to consider all six stress components. Approval of the PBAPS license renewal application was issued on March 2003 (see NUREG-1769), prior to RIS 2008-30. Therefore, please demonstrate and/or confirm that when you are required to perform stress-based fatigue monitoring, in the current and plant licensing renewal period, the methodology used is in accordance with the ASME Section III, Subsection NB, which considers the six stress components.

**RESPONSE**

There are no components or locations that require stress-based fatigue monitoring per ASME Code Section III, Subsection NB for the current and EPU plant conditions. If future surveillance monitoring results determine that stress based monitoring becomes required for certain components, then stress-based fatigue monitoring per ASME Code Section III, Subsection NB will be implemented for the required components or locations.

**EMCB-RAI-7**

On page 2-59, PUSAR Section 2.2.2.2.2.2, "Structural Evaluation for Affected BOP [balance of plant] Piping," identifies that the design-basis loss-of-coolant-accident hydrodynamic loads are not changed for EPU conditions. Per PUSAR Section 2.6.1.2.1, "Loss-of-Coolant Accident Loads," the vent thrust loads at four locations exceeded the plant-specific vent thrust loads originally calculated during the Mark I Containment long term program by approximately 2.5 percent.

a) Please discuss this discrepancy.

b) Identify the four locations mentioned above. Discuss how the structural evaluations of the affected SSCs have been revised and provide a brief summary of the evaluation results and conclusions.

**RESPONSE**

a) There is no discrepancy. PUSAR Section 2.2.2.2.2.2 discusses the structural evaluation for piping systems and valves attached to the torus. The design basis accident LOCA loads which affect piping systems in, or attached to, the torus are pool swell, Condensation Oscillation (CO) and chugging loads which, as stated in this section, are not changed for EPU. PUSAR Section 2.6.1.2.1 discusses all of the Mark I containment program loads which include vent thrust loads in addition to pool swell, CO and chugging loads. PUSAR Section 2.6.1.2.1 confirms that pool swell, CO and chugging loads do not change and vent thrust loads are also bounded by current loads except for four locations.

b) The four load components that exceeded the plant-specific vent thrust loads originally calculated during Mark I Containment long term program, which is the PBAPS Analysis of Record (AOR), are as follows:

    1. Vertical load on main vent cap – Exceeded AOR load by 0.12%
    2. Horizontal load on main vent cap – Exceed AOR load by 0.05%
    3. Horizontal on vent header per miter bend – Exceed AOR load by 2.46%
    4. Total vertical load on main vent – Exceeded AOR load by 0.12%

   Reconciliation of the above-stated mechanical load increases with the AOR structural evaluation was performed. This reconciliation was performed by review of the margins to structural acceptance criteria in the AOR stress report for the containment vent system (Reference 7-1). A review of Reference 7-1 showed a maximum stress ratio of 0.87. This means the minimum margin to the allowable stress for the vent system is 13%. The stress analysis of the vent system components considers the contribution of all loads acting on the vent system, including vent thrust loads. All loads on the vent system, except for the increased vent thrust loads at EPU condition, remain bounded at EPU conditions by the AOR loads. Therefore, a conservative assessment was made which assumed that the vent system maximum stress would increase in direct proportion to the increase in the vent thrust loads even though other concurrent loads that contribute to component stress are unchanged at EPU conditions. Since the increase in the vent thrust load component, approximately 2.5%, was much less than the minimum 13% margin between the calculated stress and allowable stress, it was determined that allowable stresses would not be exceeded with the predicted increase in the calculated vent thrust loads at EPU conditions. Therefore, the vent system remains structurally qualified at EPU conditions.

Reference

7-1 Exelon, "Peach Bottom Atomic Power Station Units 2 & 3 Mark I Long-Term Program Plant Unique Analysis Docket Numbers 50-277 and 50-278," P-1-Q-614, Revision 2, December 1985. (PUSAR Reference 71).

**EMCB-RAI-8**

*According to the PUSAR, PBAPS meets the NRC-approved General Electric (GE) topical reports CLTR. ELTR1 and ELTR2 requirements for the disposition of the structural integrity of SSCs affected by the proposed EPU. All three topical reports require that structural integrity evaluations of SSCs for EPU show continued compliance with the construction code and standard for these SSCs (including code allowables and analytical techniques) applicable to the current plant licensing basis and that no change to comply with more recent codes and standards will be proposed due to the power uprate (ELTR1, Page 50). Based on the above, for SSCs in PUSAR Section 2.2, "Mechanical and Civil Engineering," including important to safety piping, pipe supports, pressure retaining components and their supports, reactor pressure vessel (RPV) internals and core support structures that were required to be reevaluated due to higher loads resulting at EPU conditions, please provide the following:*

a) The code of construction for the SSC installation and design.

b) Justify and discuss why it is required and acceptable to utilize a different code or code edition rather than the code of construction for SSCs that do not require repair or replacement for EPU.

c) For repair/replacement activities, where a different code or code edition rather than the code of construction has been utilized for EPU, discuss whether documented code reconciliation exists that allows the use of this code and verify that the allowable values from the code of construction have been utilized with the reconciled code. Otherwise, provide a technical justification and the supporting basis which demonstrate that the allowance to use alternate codes and allowables other than the construction codes and allowables is acceptable.

**RESPONSE**

Reactor Pressure Vessel

a) As stated in PUSAR Section 2.2.2.3, the Code of construction for the reactor pressure vessel evaluation is the ASME Boiler and Pressure Vessel Code, Section III, 1965 Code with addenda to and including Winter 1965.

As stated in PUSAR Section 2.2.2.3, the following reactor pressure vessel (RPV) components were modified since original construction and [[

]] due to EPU:

- Feedwater Nozzle: This component was modified and the governing Code for the modification is the ASME Boiler and Pressure Vessel Code, Section III, 1974 Edition with Addenda to and including Summer 1976.
- Recirculation Inlet Nozzle: This component was modified and the governing Code for the modification is the ASME Boiler and Pressure Vessel Code, Section III, 1989 Edition with Addenda to and including Winter 1990.
- Recirculation Outlet Nozzle Unit 3: This component was modified and the governing Code for the modification is the ASME Boiler and Pressure Vessel Code, Section III, 1980 Edition with Addenda to and including Winter 1981."

These above governing Codes in the PBAPS current licensing basis were used in the evaluation of the three above stated components for EPU conditions.

b) There are no RPV components that require repair or replacement for EPU. The RPV stress reconciliation for EPU used the same governing Codes as used in the current licensing basis. The code of construction/modification in the stress report of record specified in part a) above remains the code of construction/modification for the EPU evaluation.

c) There are no RPV components that require repair or replacement for EPU.

Reactor Internals and Core Support Structure

a) PBAPS is a BWR 4 design. The reactor internals and the core support structure are not ASME code components for a BWR 4 design. As stated in PUSAR Section 2.2.3.2.2, the RPV internals consist of the core support structure components and non-core support structure components. The RPV internals are not ASME Code components; however, the requirements of the ASME Code are used as guidelines in their design/analysis.

b) Not applicable

c) Not applicable

Important to safety piping, pipe supports, and pressure retaining components and their supports

a) The original code of construction for important to safety piping, portions of pipe supports, and pressure retaining components and their supports for stress analysis is USA Standard Code for Pressure Piping, "Power Piping," USAS B31.1.0-1967.

   Static and seismic analysis of piping utilizes the ANSI B31.1-1973 Edition with addenda thru Summer 1973, along with the ANSI B31.1-1977 Editions for the Stress Intensification Factors (SIF).

   The code of construction for portions of piping supports is the AISC Manual of Steel Construction – Allowable Stress Design 6$^{th}$ Edition (with reconciliation up to the 9$^{th}$ Edition) as delineated by the pipe support and structural jurisdictional boundaries described in the PBAPS specification for design of piping systems evaluation.

b) Main Steam piping inside containment does not require repair or replacement for EPU. The current calculations for MS piping inside containment utilize the USAS B31.1.0-1967 code from construction.

The EPU calculations for MS piping inside containment utilize the ANSI B31.1 1973 with summer 1973 addenda and Stress Intensity Factors (SIFs) from the 1977 edition as defined in PBAPS specifications for all non-Mark I load affected piping systems. The change in code editions is consistent with the application of the Normal, Upset, Emergency and Faulted loads.

Allowable stress values for MS piping inside containment and associated branch lines are taken from ANSI B31.1 2004 Edition, including the 2005 Addendum. A formally documented code reconciliation provides justification for this use consistent with ASME Section XI. Refer to the EMCB RAI-9 response for further discussion on this code reconciliation.

c) *Modifications to the MS piping supports for EPU utilize the current PBAPS applicable* codes as discussed in parts (a) and (b) above.

**EMCB-RAI-9**

The PUSAR, starting on page 2-50, states the following:

> The MS and associated branch piping inside and RCPB [reactor coolant pressure boundary] piping outside containment was evaluated for compliance with ANSI B31.1, "Power Piping," 1973 Edition including Summer 1973 Addenda stress criteria (Reference 39), including the effects of EPU on piping stresses, piping supports including the associated building structure, penetrations, piping interfaces with the RPV nozzles, flanges, and valves. Allowable stress values for MS piping inside containment and associated branch lines were taken from ANSI B31.1 2004 Edition including the 2005 Addendum (Reference 40).

The above statement indicates that the piping was analyzed using the criteria of the 1973 Edition of B31.1, but the allowable stress values were from the 2004 Edition, including the 2005 Addendum of B31.1. Provide a technical justification which demonstrates acceptability of this method.

**RESPONSE**

The technical justification which demonstrates the acceptability of the use of higher allowable stress values from later editions of the ANSI B31.1 Code is documented in a PBAPS Code Reconciliation Report. The Code Reconciliation describes regulatory and Code requirements for performance of analyses using later Editions and Addenda of Construction Codes such as USAS B31.1 and ASME Section III. Within the requirements of ASME Section XI, PBAPS Units 2 and 3 may use later Editions and Addenda of Construction Codes, such as B31.1, including increased allowable stresses over the original Code of Construction.

In 1995, Subarticle IWA-4300 was added to Section XI to provide specific requirements for design work. Section XI permits use of later Editions and Addenda of Construction Codes, such as B31.1, including increased allowable stresses for design work affecting existing materials, parts, and components. These allowable stresses may be used for design re-analysis with earlier materials, fabrication, examination, etc. Section XI has been endorsed by the NRC, through the 2008 Addenda, with no conditions imposed on use of the provisions utilized in the PBAPS Code Reconciliation.

The changes to allowable stresses were made in the 2005 Addenda of B31.1, consistent with the changes in the 1999 Addenda of ASME Section II Part D, which apply to design of Section VIII Div. 1 pressure vessels and Section III Class 2 & 3 pressure vessels and piping. The changes were made by the ASME Committees through a reduction in the material design safety factor which increased the allowable stresses for most ferritic steels by approximately 14%. The NRC has accepted the use of later allowable stresses for design re-analysis, with earlier materials, fabrication, examination, etc., through its endorsement of ASME Section XI in 10 CFR 50.55a.

**EMCB-RAI-10**

PUSAR, Table 2.2-4a presents maximum stress summaries for the MS piping due to the proposed EPU, which, as identified in the PUSAR, has been reanalyzed to include the turbine stop valve closure (TSVC) transient loading at EPU conditions.

a) The allowable value of 22,500 pounds per square inch (psi) shown for the "sustained + thermal" loading case for the "Units 2/3 Loop D" piping, shown in PUSAR Table 2.2-4a, does not appear to be correct. Please review the allowable values in Table 2.2-4a and verify that the calculated values in that table represent maximum pipe stresses.

b) Provide an explanation for node designation/location similar to that found on Updated Final Safety Analysis Report (UFSAR), Revision 24, Appendix C, Table C.5.7, "Main Steam Piping." Include information or a note that identifies equation loadings (dead weight, pressure, maximum pressure, design earthquake, maximum earthquake, thermal loads, TSVC load, etc.) for each equation used in the summaries, similar to the loadings column found in the UFSAR table. In addition, identify piping inside and outside containment and which service level or equation in the summaries contains the TSVC loading.

c) PUSAR Table 2.2-4a uses the designations "Eq. 13" and "Eq. 14" interchangeably. Please verify whether "Eq. 14" is a typo and is meant to be "Eq. 13". Discuss whether the Node 78 "Sustained + Thermal" allowable of 43,750 psi is for a different material (i.e., A106 GR C) than the material (i.e., A106 GR B) on the remainder of the nodes listed in the table.

d) MS allowable values at all critical locations shown on UFSAR, Appendix C, Table C.5.7, with the exemption of the section of piping to the high pressure coolant injection turbine, are for a material having a hot allowable stress value ($S_h$) of 17,500 psi (i.e., A106 GR C). The MS allowable values at all locations but one, shown on PUSAR Table 2.2-4a, are for a material having $S_h$ of 15,000 psi (i.e., A106 Gr B). Please provide the MS material designation(s) corresponding to the various piping locations.

e) Not all of the licensing basis criteria identified in the UFSAR, Appendix C. Table C.5.7 are shown to be satisfied in the EPU stress summaries shown in PUSAR Table 2.2-4a. For instance, the maximum stress summaries in the PUSAR use the $2.4x(S_h)$ stress allowable criterion for Level D loadings, while the LB criterion found in UFSAR Table C.5.7 is $2.0x(S_h)$. Using the LB criterion for the level D stress calculated at node 83V of Loop B would produce an EPU interaction ratio of approximately 1.354 greater than the allowed value of 1.00. Please review the LB MS structural criteria found in the UFSAR and demonstrate how these criteria will be satisfied for the proposed EPU for continued compliance with the current licensing basis.

f) For exceeding the calculated-over-allowable interaction ratio of 1.00, the PUSAR Table 2.2-4a provides the following note:

> Note 2: B31.1 2004 Code reconciliation (Reference 40) allows for interaction ratio of up to 1.14.

Please identify where this interaction ratio is discussed in the B31.1-2004 code and justify its applicability to the PBAPS.

**RESPONSE**

a) The Service Level "sustained + thermal" notation shown in the "Units 2/3 Loop D" PUSAR Table 2.2-4a is a typographical error in the PUSAR table. The Service Level for "Units 2/3 Loop D" Table should state "Thermal Expansion" (as is shown for "Units 2/3 Loop A" in PUSAR Table 2.2-4a, corresponding to Code Equation 13). The remaining information in the "Units 2/3 Loop D" PUSAR Table 2.2-4a is correct, including the allowable value of 22,500 psi for Equation 13. It is verified that the calculated values in PUSAR Table 2.2-4a represent maximum pipe stresses.

An expanded version of PUSAR Table 2.2-4a is provided below, which corrects the above typographical error, includes pipe material designations at each identified node point, and describes the location of each node point.

**PUSAR Table 2.2-4a**
**Main Steam Pipe Stresses at EPU Conditions**

**Maximum Stress Interaction Summary: Units 2/3 Loop A**

| Service Level (Operating Condition) | Code Equation | Node | Pipe Material at Node Location | EPU Stress (psi) | Code Allowable (psi) | Interaction Ratio |
|---|---|---|---|---|---|---|
| Service Level A (Sustained) | Eq. 11 | 32E (1) | A106, Gr. B | 9,165 | 15,000 | 0.611 |
| Service Level B (Upset) | Eq. 12B | 24B (2) | A106, Gr. B | 18,763 | 18,000 | 1.042 (4) |
| Service Level C (Emergency) | Eq. 12C | 24B (2) | A106, Gr. B | 20,073 | 27,000 | 0.743 |
| Service Level D (Faulted) | Eq. 12D | 24B (2) | A106, Gr. B | 23,419 | 36,000 | 0.651 |
| Thermal Expansion | Eq. 13 | 24V (3) | A106, Gr. B | 12,599 | 22,500 | 0.560 |

(1) Node 32E – 10" outlet flange of Safety Valve RV-70A
(2) Node 24B – Interface between sweepolet and 6" inlet pipe to Safety Relief Valve RV-71A
(3) Node 24V – Interface between 6" inlet pipe and flange of Safety Relief Valve RV-71A
(4) Reported stress interaction ratio based on allowable stress values from B31.1 1973. With Code reconciliation to B31.1 2004 including 2005 Addenda, up to 14% higher allowable stress values may be credited. Thus, a reported stress interaction ratio of up to 1.14 is acceptable.

**Maximum Stress Interaction Summary: Units 2/3 Loop B**

| Service Level (Operating Condition) | Code Equation | Node | Pipe Material at Node Location | EPU Stress (psi) | Code Allowable (psi) | Interaction Ratio |
|---|---|---|---|---|---|---|
| Service Level A (Sustained) | Eq. 11 | 67 (1) | A106, Gr. B (7) | 7,951 | 15,000 | 0.530 |
| Service Level B (Upset) | Eq. 12B | C75 (2) | A106, Gr. B | 19,656 | 18,000 | 1.092 (6) |
| Service Level C (Emergency) | Eq. 12C | 83B (3) | A106, Gr. B | 21,795 | 27,000 | 0.807 |
| Service Level D (Faulted) | Eq. 12D | 83V (4) | A106, Gr. B | 40,615 | 36,000 | 1.128 (6) |
| Sustained + Thermal Expansion | Eq. 14 | 78 (5) | A155, Gr. KC70, CL. 1 | 35,898 | 43,750 | 0.821 |

(1) Node 67 – MS line at location of 6" sweepolet to Safety Relief Valve RV-71D
(2) Node C75 – 10" discharge pipe from Safety Relief Valve RV-71C
(3) Node 83B – Interface between sweepolet and 6" inlet pipe to Safety Relief Valve RV-71F
(4) Node 83V – Interface between 6" inlet pipe and flange of Safety Relief Valve RV-71F
(5) Node 78 – MS line 26" x 26" tee coming from the RPV
(6) Reported stress interaction ratio based on allowable stress values from B31.1 1973. With Code reconciliation to B31.1 2004 including 2005 Addenda, up to 14% higher allowable stress values may be credited. Thus, a reported stress interaction ratio of up to 1.14 is acceptable.
(7) Node 67 material is A155 Gr. KC70, Cl. 1 with 17,500 psi Code allowable stress, but is conservatively modeled as A106 Gr. B with 15,000 psi allowable.

**Maximum Stress Interaction Summary: Units 2/3 Loop C**

| Service Level (Operating Condition) | Code Equation | Node | Pipe Material at Node Location | EPU Stress (psi) | Code Allowable (psi) | Interaction Ratio |
|---|---|---|---|---|---|---|
| Service Level A (Sustained) | Eq. 11 | 72E (1) | A106, Gr. B | 9,165 | 15,000 | 0.611 |
| Service Level B (Upset) | Eq. 12B | 67B (2) | A106, Gr. B | 11,806 | 18,000 | 0.656 |
| Service Level C (Emergency) | Eq. 12C | 67B (2) | A106, Gr. B | 11,897 | 27,000 | 0.441 |
| Service Level D (Faulted) | Eq. 12D | 67B (2) | A106, Gr. B | 12,219 | 36,000 | 0.339 |
| Sustained + Thermal Expansion | Eq. 14 | 78 (4) | A155, Gr. KC70, CL. 1 | 35,516 | 43,750 | 0.812 |

(1) Node 72E – 10" outlet flange of Safety Valve RV-70C
(2) Node 67B – Interface between sweepolet and 6" inlet pipe to Safety Relief Valve RV-71H
(3) Node 69B – Interface between sweepolet and 6" inlet pipe to Safety Relief Valve RV-71G
(4) Node 78 – MS line 26" x 26" tee coming from the RPV

**Maximum Stress Interaction Summary: Units 2/3 Loop D**

| Service Level (Operating Condition) | Code Equation | Node | Pipe Material at Node Location | EPU Stress (psi) | Code Allowable (psi) | Interaction Ratio |
|---|---|---|---|---|---|---|
| Service Level A (Sustained) | Eq. 11 | 32E (1) | A106, Gr. B | 9,165 | 15,000 | 0.611 |
| Service Level B (Upset) | Eq. 12B | 24B (2) | A106, Gr. B | 20,321 | 18,000 | 1.129 (4) |
| Service Level C (Emergency) | Eq. 12C | 24B (2) | A106, Gr. B | 21,576 | 27,000 | 0.799 |
| Service Level D (Faulted) | Eq. 12D | 24B (2) | A106, Gr. B | 21,757 | 36,000 | 0.604 |
| Thermal Expansion | Eq. 13 | 24V (3) | A106, Gr. B | 12,536 | 22,500 | 0.557 |

(1) Node 32E – 10" outlet flange of Safety Valve RV-70B
(2) Node 24B – Interface between sweepolet and 6" inlet pipe to Safety Relief Valve RV-71K
(3) Node 24V – Interface between 6" inlet pipe and flange of Safety Relief Valve RV-71K
(4) Reported stress interaction ratio based on allowable stress values from B31.1 1973. With Code reconciliation to B31.1 2004 including 2005 Addenda, up to 14% higher allowable stress values may be credited. Thus, a reported stress interaction ratio of up to 1.14 is acceptable.

b) The expanded PUSAR Table 2.2-4a for MS piping inside primary containment shown in the above response to EMCB-RAI-10(a) provides descriptions of each identified node location where the maximum stress interaction occurs. Table 10-1 below defines the design basis load combination cases for each Service Level and Code equation, including applicable deadweight, pressure, seismic, transient (TSV closure transient, or SRV/SV actuation transient), and thermal expansion loads, used for MS piping analyses inside primary containment. The same load combination cases are analyzed for MS piping outside primary containment except for SV/RV transients which are not applicable.

**Table 10-1**
**MS Piping Analysis Design Basis Load Combinations**

| Service Level (Operating Condition) | Design Basis Load Case Combinations | Code Equation | Code Allowable |
|---|---|---|---|
| Service Level A (Sustained) | Peak Pressure + Deadweight | Eqn. 11 | $1.0 \times S_h$ (*Note 1d*) |
| Service Level B (Upset) | Peak Pressure + Deadweight + Design Earthquake (DE) | Eqn. 12 | $1.2 \times S_h$ (*Note 1d*) |
| Service Level B (Upset) | Peak Pressure + Deadweight + TSV Transient | Eqn. 12 | $1.2 \times S_h$ (*Note 1d*) |
| Service Level B (Upset) | Peak Pressure + Deadweight + SV/RV Transient | Eqn. 12 | $1.2 \times S_h$ (*Note 1d*) |
| Service Level C (Emergency) | Peak Pressure + Deadweight + SRSS [Design Earthquake (DE) and TSV Transient] | Eqn. 12 | $1.8 \times S_h$ (*Note 1d*) |
| Service Level C (Emergency) | Peak Pressure + Deadweight + SRSS [Design Earthquake (DE) and SV/RV Transient] | Eqn. 12 | $1.8 \times S_h$ (*Note 1d*) |
| Service Level D (Faulted) | Peak Pressure + Deadweight + SRSS [Maximum Credible Earthquake (MCE) and TSV Transient and SV/RV Transient] | Eqn. 12 | $2.4 \times S_h$ (*Note 1d*) |
| Thermal Expansion | Thermal Stress Range + DE + Seismic Anchor Movements (SAM) | Eqn. 13 | $S_A$ $\{S_A = f*(1.25*S_C + 0.25*S_h)\}$ (*Notes 1a, 1b, 1c*) |
| Sustained + Thermal Expansion | Peak Pressure + Deadweight + Thermal Stress Range + SAM | Eqn. 14 | $S_h + S_A$ (*Notes 1a, 1b, 1c, 1d*) |

Notes:

(1a) f is the stress range reduction factor from Table 102.3.2(c) of Power Piping Code – ANSI B31.1, 1973 Edition. $S_h$ is the allowable stress for the design temperature of the pipe and $S_C$ is the allowable stress for the ambient temperature of the pipe.

(1b) The number of cycles for this piping is considered not to exceed 7000; therefore, f=1.0.

(1c) If the secondary pipe stresses (Eqn. 13) exceed the allowable $S_A$, the margin between the allowable $S_h$ and the sustained load stress may be added to the 0.25 $S_h$ term in the $S_A$ formulation. Therefore, if only one of the two equations (EQN 13 or EQN 14) meets the code allowable, then the stresses are acceptable.

(1d) The pressure stresses are calculated in the pipe longitudinal direction.

c) The use of Equation 14 is not a typographical error. In each MS pipe stress analysis, thermal expansion stresses are first checked using B31.1 Code Equation 13. If the Code allowable stress $S_A$ is met with Equation 13 (thermal expansion), then Equation 14 need not be checked. If the Code allowable stress $S_A$ is not met with Equation 13, then Equation 14 (sustained + thermal expansion) is checked against Code allowable stress $S_h + S_A$. Calculated stresses are acceptable if one of the two Equations 13 or 14 meets the Code allowable.

Node 78 (sustained + thermal expansion) allowable stress of 43,750 psi is for a different material (A155, Gr. KC70, CL. 1) than the material (A106 Gr. B) for the remainder of the nodes listed in the table.

d) The 26-inch main steam piping headers are fabricated from A155, Gr. KC70, CL. 1 material, with a hot allowable stress value ($S_h$) of 17,500 psi. For most node points shown on PUSAR Table 2.2-4a, the maximum stress interactions occur at the inlets or outlets of Safety Relief Valves (SRV) and Safety Valves (SV), fabricated from A106 Gr. B material with $S_h$ of 15,000 psi. See expanded PUSAR Table 2.2-4a provided in the response to EMCB-RAI-10(a), with material designations.

e) Load combinations and results from the original design basis analysis of the MS piping performed in 1970 are presented in UFSAR Table C.5.7. The footnote to UFSAR Table C.5.7 cites other references for analysis of power rerate (current) conditions. The power rerate analysis cites by reference a revised MS piping analysis from 1986, which is the current, pre-EPU design and licensing basis MS piping analysis. The 1986 MS analysis evaluated the effects of newly defined SRV loads on MS branch piping at the inlet to the SRVs, which considered the evolution of the B31.1 Code and applied a Service Level D faulted condition stress limit of 2.4 x $S_h$ for the SRV inlet piping.

The new MS piping analyses for EPU are consistent with the current (pre-EPU) design basis MS piping analysis in the application of a Service Level D faulted condition stress limit of 2.4 x $S_h$. This stress limit is also defined in PBAPS station piping analysis design criteria.

Design basis load combinations evaluated in the new EPU MS piping analyses are shown in Table 10-1 above. These include new load combinations for TSV closure transients which were not included in the original MS design analysis as shown in UFSAR Table C.5.7. In the original design analysis performed in 1970, general criteria were developed for evaluating combinations of loads which have a low probability of occurrence, which included a stress allowable criterion of 2.0 x $S_h$ for the combined loads of deadweight + maximum pressure + MCE. At that time, the ASME Code did not specify stress criteria for Emergency and Faulted service level conditions. Later editions of the ASME Code provided clear guidance for Faulted service level conditions with the allowable stress criterion of 2.4 x $S_h$.

The MS piping original design Code of record is USAS B31.1.0 – 1967, including B31 Case 70. Since the establishment of the original design criteria, additional loading conditions resulting from TSV closure transients have been identified. Because these transient dynamic loads have not been considered in the original MS piping design, criteria had to be developed to provide a consistent and uniform basis for acceptability. For example, historically, new loads were identified which led to the re-evaluation of Mark I containments and piping. The Torus Attached Piping and SRV discharge piping at PBAPS, while built originally to B31.1 – 1967, was re-analyzed to the Class 2/3 piping rules of ASME Section III, 1977 Edition through S77 Addenda. Since the TSV closure load is also a newly defined load and is due to Normal Operating and Design Basis Accident conditions, it is consistent to use a similar, and already accepted, set of criteria for the revised MS piping analysis for EPU.

f) The Note 2 under PUSAR Table 2.2-4a requires clarification. This clarification is provided in the footnotes to the expanded PUSAR Table 2.2-4a shown in the response to EMCB-RAI-10(a). The interaction ratio of 1.14 is not found directly in the B31.1 Code. Rather, it is derived from the increase in material allowable stress values found in later Editions of the Code. For example, $S_h$ = 15,000 psi for A106 Gr. B material per ANSI B31.1, 1973 Edition. In the 2005 Addenda to the 2004 Edition of B31.1, $S_h$ is increased to 17,100 psi for A106 Gr. B material, resulting in an allowable stress increase of 17.1 ksi / 15 ksi = 1.14.

   Stress interaction ratios reported in PUSAR Table 2.2-4a are based on allowable stress values from B31.1 1973 Edition. With Code reconciliation to B31.1 2004 including 2005 Addenda, up to 14% higher allowable stress values may be credited. Thus, a reported stress interaction ratio of up to 1.14 is acceptable.

   The applicability of the use of later B31.1 Code Editions and Addenda to PBAPS for piping design analysis is justified in a PBAPS Code Reconciliation (see response to EMCB-RAI-9 for further details).

**EMCB-RAI-11**

In addition to the MS locations listed in the PUSAR Table 2.2-4a, supplement the table with quantitative maximum result summaries inside and outside the primary containment, which show calculated values compared to the LB allowable values, with the highest resulting interaction ratios at critical locations (such as the MS nozzles, MS relief valve flanges, MS supports, MS flued head anchor penetrations, etc.). For the MS containment penetrations, show the maximum results from the penetration structural qualifications, which include loads from both sides of the penetration.

**RESPONSE**

<u>MS Pipe Stress Analysis</u>
The following supplemental tables show calculated MS pipe stresses (bounding maximum of all four MS loops on both Units 2 and 3) at critical locations both inside and outside of primary containment, compared to the Code allowable values. Critical locations are selected at piping anchor points (RPV nozzles and containment penetration anchors) and also at SRVs and TSVs where high stresses occur with transient loading. See response to EMCB-RAI-10(b) for explanation of design basis load combinations for each Service Level. The calculated pipe stresses reflect bounding EPU conditions with Turbine Stop Valve closure transient loading and MS pipe support modifications.

**Table 11-1**
**MS Maximum Pipe Stress Summary: MS RPV Nozzle Location**

| Service Level (Operating Condition) | Code Equation | Node | EPU Stress (psi) | Code Allowable (1) (psi) | Interaction Ratio |
|---|---|---|---|---|---|
| Service Level A (Sustained) | Eq. 11 | 56 (2) | 8,903 | 17,500 | 0.509 |
| Service Level B (Upset) | Eq. 12B | 4 (3) | 11,813 | 21,000 | 0.563 |
| Service Level C (Emergency) | Eq. 12C | 4 (3) | 12,050 | 31,500 | 0.383 |
| Service Level D (Faulted) | Eq. 12D | 4 (3) | 13,515 | 42,000 | 0.322 |
| Thermal Expansion | Eq. 13 | 56 (4) | 15,859 | 26,250 | 0.604 |

(1) Allowable stress for A-155 Gr. KC70, Class 1 material, from B31.1 1973.
(2) Node 56 – RPV nozzle N-03B on MS line B
(3) Node 4 – RPV nozzle N-03D on MS line D
(4) Node 56 – RPV nozzle N-03C on MS line C

**Table 11-2**
**MS Maximum Pipe Stress Summary: MS Safety Relief Valve Inlet Location**

| Service Level (Operating Condition) | Code Equation | Node | EPU Stress (psi) | Code Allowable (1) (psi) | Interaction Ratio |
|---|---|---|---|---|---|
| Service Level A (Sustained) | Eq. 11 | 67V (2) | 6,502 | 15,000 | 0.433 |
| Service Level B (Upset) | Eq. 12B | 24V (3) | 17,883 | 18,000 | 0.994 |
| Service Level C (Emergency) | Eq. 12C | 83V (4) | 20,693 | 27,000 | 0.766 |
| Service Level D (Faulted) | Eq. 12D | 83V (4) | 40,615 | 36,000 | 1.128 (6) |
| Sustained + Thermal Expansion | Eq. 14 | 69V (5) | 29,720 | 37,500 | 0.793 |

(1) Allowable stress for A-106 Gr. B material, from B31.1 1973.
(2) Node 67V – Interface between 6” inlet pipe and flange to Unit 3 RV-71H
(3) Node 24V – Interface between 6” inlet pipe and flange to Units 2/3 RV-71K
(4) Node 83V – Interface between 6” inlet pipe and flange to Unit 3 RV-71F
(5) Node 69V – Interface between 6” inlet pipe and flange to Unit 3 RV-71G
(6) Reported stress interaction ratio based on allowable stress values from B31.1 1973. With Code reconciliation to B31.1 2004 including 2005 Addenda, up to 14% higher allowable stress values may be credited. Thus, a reported stress interaction ratio of up to 1.14 is acceptable.

**Table 11-3**
**MS Maximum Pipe Stress Summary: Primary Containment Anchor Location (Inside Containment)**

| Service Level (Operating Condition) | Code Equation | Node | EPU Stress (psi) | Code Allowable (1) (psi) | Interaction Ratio |
|---|---|---|---|---|---|
| Service Level A (Sustained) | Eq. 11 | 50 (2) | 8,729 | 17,500 | 0.499 |
| Service Level B (Upset) | Eq. 12B | 96 (3) | 9,939 | 21,000 | 0.473 |
| Service Level C (Emergency) | Eq. 12C | 96 (3) | 9,969 | 31,500 | 0.316 |
| Service Level D (Faulted) | Eq. 12D | 96 (3) | 10,215 | 42,000 | 0.243 |
| Thermal Expansion | Eq. 13 | 96 (3) | 7,070 | 26,250 | 0.269 |

(1) Allowable stress for A-155 Gr. KC70, Class 1 material, from B31.1 1973.
(2) Node 50 – Containment Penetration N-7D
(3) Node 96 – Containment Penetration N-7B

**Table 11-4**
**MS Maximum Pipe Stress Summary: Primary Containment Anchor Location (Outside Containment)**

| Service Level (Operating Condition) | Code Equation | Node | EPU Stress (psi) | Code Allowable (1) (psi) | Interaction Ratio |
|---|---|---|---|---|---|
| Service Level A (Sustained) | Eq. 11 | D05 (2) | 8,257 | 17,500 | 0.472 |
| Service Level B (Upset) | Eq. 12B | B05 (3) | 9,668 | 21,000 | 0.460 |
| Service Level C (Emergency) | Eq. 12C | D05 (2) | 9,865 | 31,500 | 0.313 |
| Service Level D (Faulted) | Eq. 12D | D05 (2) | 11,286 | 42,000 | 0.269 |
| Thermal Expansion | Eq. 13 | D05 (2) | 1,541 | 26,250 | 0.059 |

(1) Allowable stress for A-155 Gr. KC70, Class 1 material, from B31.1 1973.
(2) Node D05 – Containment Penetration X-7A (Unit 2) and X-7D (Unit 3)
(3) Node B05 – Containment Penetration X-7C (Unit 2) and X-7B (Unit 3)

**Table 11-5**
**MS Maximum Pipe Stress Summary: Turbine Stop Valve Location**

| Service Level (Operating Condition) | Code Equation | Node | EPU Stress (psi) | Code Allowable (1) (psi) | Interaction Ratio |
|---|---|---|---|---|---|
| Service Level A (Sustained) | Eq. 11 | A97 (2) | 10,643 | 17,500 | 0.608 |
| Service Level B (Upset) | Eq. 12B | CSV (3) | 19,632 | 21,000 | 0.935 |
| Service Level C (Emergency) | Eq. 12C | CSV (3) | 21,832 | 31,500 | 0.693 |
| Service Level D (Faulted) | Eq. 12D | CSV (3) | 29,136 | 42,000 | 0.694 |
| Thermal Expansion | Eq. 13 | ASV (2) | 2,635 | 26,250 | 0.100 |

(1) Allowable stress for A-155 Gr. KC70, Class 1 material, from B31.1 1973.
(2) Node A97/ASV – MSV-4 Stop Valve (Unit 2) / MSV-1 Stop Valve (Unit 3)
(3) Node CSV – MSV-2 Stop Valve (Unit 2) / MSV-3 Stop Valve (Unit 3)

The following table shows calculated pipe stresses compared to the Code allowable values at locations with the highest stress interaction ratio for all MS piping loops outside the primary containment. The calculated pipe stresses reflect bounding EPU conditions with Turbine Stop Valve closure transient loading and MS pipe support modifications.

**Table 11-6**
**Maximum Stress Ratios for MS Piping Outside Primary Containment**

| Service Level (Operating Condition) | Code Equation | Node | Pipe Material at Node Location | EPU Stress (psi) | Code Allowable (psi) | Interaction Ratio |
|---|---|---|---|---|---|---|
| Service Level A (Sustained) | Eq. 11 | 257 (1) | A106 Gr. B | 9,710 | 15,000 | 0.647 |
| Service Level B (Upset) | Eq. 12B | C96 (2) | A155, Gr. KC70, CL. 1 | 20,114 | 21,000 | 0.958 |
| Service Level C (Emergency) | Eq. 12C | GC5 (3) | A106 Gr. A | 15,499 | 21,600 | 0.718 |
| Service Level D (Faulted) | Eq. 12D | GC5 (3) | A106 Gr. A | 23,568 | 28,800 | 0.818 |
| Thermal Expansion | Eq. 13 | GC5 (3) | A106 Gr. A | 5,926 | 18,000 | 0.329 |

(1) Node 257 - 14" bypass line to the turbine bypass valve inlet (tee)
(2) Node C96 - 6" branch stub off the turbine stop valve inlet header on MS line B (Unit 2) / MS line C (Unit 3)
(3) Node GC5 - HP turbine inlet

<u>MS Relief Valve Flanges</u>
Per PBAPS piping design criteria, flanges which are manufactured to ANSI B16.5 standards are acceptable if the piping attached to the flanges is within Code stress allowables. Therefore, no specific analysis of MS relief valve flanges is required since they are manufactured to ANSI B16.5 and the attached pipe stresses at EPU conditions are within Code allowables.

MS Pipe Supports
MS pipe supports have been analyzed for the new loads resulting from revised pipe stress analyses with Turbine Stop Valve closure transient loading at EPU conditions. Descriptions of new and modified MS pipe supports required to meet piping and structural Code allowables are provided in the response to EMCB-RAI-14. Each existing, new, and modified MS pipe support is analyzed to meet Code stress allowables with interaction ratios of less than 1.0 for the most limiting component of each support.

MS Containment Penetration Anchor Structural Evaluation
For the MS containment penetrations, the maximum results from the penetration structural qualifications, which include loads from both sides of the penetration, are shown in Table 11-7 below. The structural anchor frame at the MS primary containment penetrations was conservatively evaluated for the changes in combined MS piping loading from both sides of the penetrations, with the maximum increase in load applied to the most critically loaded structural member and connection.

**Table 11-7**
**Containment Penetration Anchor Structural Evaluation**

| Structural Component | Stress Type | Current Stress (ksi) | EPU Stress (ksi) | Stress Allowable (ksi) | Current Interaction Ratio | EPU Interaction Ratio |
|---|---|---|---|---|---|---|
| Beam 1 (Node 80) | Axial/Bending | --- | --- | --- | 0.75 | 0.85 |
| | Shear/Torsion | 6.83 | 7.78 | 18.4 | 0.37 | 0.42 |
| Shear Key Connection (Node 9) | Bearing | 3.33 | 3.79 | 4.76 | 0.70 | 0.80 |
| End Plate | Bending | 21.8 | 24.8 | 27 | 0.81 | 0.92 |

**EMCB-RAI-12**

For piping affected by the EPU, which is important to safety or is required to withstand a seismic event, please provide the following:

a) A list of piping systems (or portions of piping systems) inside and outside containment, which contain parameters (including temperature, pressure and flow) that have increased at EPU conditions and list the values of these parameters for OLTP, CLTP and EPU. Also show whether the resulting piping loads from these parameters at EPU conditions (including but not limited to loads due to temperature, pressure, flow, transient and mechanical loads) are bounded in the current DB analyses.

b) For piping with loads that are not bounded by the DB AOR, discuss the method of evaluation. If the methods utilized are different than methods in the current LB and DB, provide a technical justification and the supporting basis which demonstrate acceptability of these methods. Please include quantitative summaries of maximum stresses or loads and fatigue usage factors (if applicable) for EPU and OLTP or CLTP (whichever governs for the EPU derived values) with a comparison to the code of construction or LB (UFSAR) allowable values (whichever is applicable). If scaling factors have been used in determining EPU results from OLTP or CLTP analyses, show the scaling factors and discuss the basis of their development. Include only maximum stresses with the highest interaction ratios and data at critical locations (i.e., anchors and flued head anchors, nozzles, penetrations, flanged connections, valve connections, branching pipe connections, pipe supports, etc.). For containment penetrations, show the maximum results from the penetration structural qualifications which include loads from both sides of the penetration.

c) For any piping loads (such as the ones discussed in part (a) and (b) above), that are not included in the current DB analyses, but could potentially occur at EPU conditions and are not included in the EPU piping evaluations, provide an explanation which justifies exclusion of these loads in the EPU evaluations. Also, provide a technical justification which demonstrates that occurrence of these loads on the piping will not challenge the piping system (including all system components such as inline components, supports, anchors, penetrations, nozzles, etc.) and connected SSCs beyond their structural design limits.

**RESPONSE**

a) The following table provides a listing of PBAPS piping systems important to safety or required to withstand a seismic event which contain operating parameters (pressure, temperature, or flow) that will increase at EPU conditions.

**Table 12-1**
**PBAPS EPU Operating Conditions for Safety-Related and/or Seismic Piping Systems [1,5]**

| System | Parameter | OLTP | CLTP | EPU | EPU Change |
|---|---|---|---|---|---|
| Main Steam (MS) | Pressure (psia) | 1020 | 1050 | 1050 | 0 |
| | Temperature (°F) | 547 | 550.5 | 550.5 | 0 |
| | Flow (Mlbm/hr) | 13.37 | 14.39 | 16.25 | +1.86 |
| Feedwater (FW) | Pressure (psig) [2] | 1070 | 1098 | 1115 | +17 |
| | Temperature (°F) | 376.1 | 381.5 | 385.0 | +3.5 |
| | Flow (Mlbm/hr) | 13.37 | 14.36 | 16.22 | +1.86 |
| Reactor Recirculation (RR) [3] | Pump Discharge Pressure (psia) | 1255 | 1284.8 | 1285.6 | +0.8 |
| | Suction Temperature (°F) | 528 | 530.1 | 527.9 | -2.2 |
| | Core Flow (Mlbm/hr) | 102.5 | 105 | 104.5 | -0.5 |
| | Pump Mass Flow (Mlbm/hr) | 17.1 | 18.44 | 18.48 | +0.04 |
| Reactor Water Cleanup (RWCU) | Inlet Pressure (psia) | 1020 | 1050 | 1050 | 0 |
| | Discharge Pressure (psig) [4] | 1070 | 1098 | 1115 | +17 |
| | Suction Temperature (°F) | 528 | 530.1 | 527.9 | -2.2 |
| | Flow (Mlbm/hr) | 0.133 | 0.147 | 0.147 | 0 |

(1) 100% Rated EPU Conditions @ 1050 psia maximum nominal reactor dome pressure
(2) FW system RFP discharge pressure at normal operating condition - increases with increased flow at EPU.
(3) RR conditions at rated core flow – bounded by ICF.
(4) RWCU discharge pressure – increases at tie-in to FW system
(5) EPU parameter values are from Turbine Cycle Heat Balance analysis in order to provide bounding values for piping analysis.

The EPU operating conditions for each system above are bounded by the piping system design pressures and temperatures. Current design basis RR system and RWCU system pipe stress calculations use pressures and temperatures which bound the EPU conditions.

FW system pressure, temperature, and flow rate will increase with EPU. The current design basis FW system pipe stress calculations use pressures and temperatures which bound the EPU conditions.

MS system pressures and temperatures do not increase with the constant pressure EPU; however the MS flow rate will increase. MS Turbine Stop Valve (TSV) closure transient dynamic loads will increase with increased MS flow at EPU. The MS piping has been re-analyzed to include TSV closure transient loads at bounding EPU conditions, and resultant pipe stresses are within Code allowables as described below.

b) The current design analysis for MS piping inside primary containment does not include a TSV closure transient load definition. The current design analysis for MS piping outside containment does include a TSV closure transient load based on MS flow rates at CLTP conditions.

   The MS piping both inside and outside primary containment has been re-analyzed for EPU conditions, and includes TSV closure transient loading. Scaling factors are not utilized in the revised MS piping analyses. For EPU, a new fluid transient loading analysis was performed to develop MS line force time histories for TSV closure events, based on bounding turbine Valves Wide Open (VWO) steam flow conditions at EPU (see PUSAR Section 2.5.1.2.2 for discussion of HP turbine steam path modifications which will increase the VWO steam flow at EPU). Structural analyses for the MS containment penetration anchors and pipe supports were also revised to qualify the piping system for the EPU loading conditions. The methods utilized for the revised MS piping analyses are consistent with (i) the current PBAPS piping design specification, and (ii) the design basis analysis of record methods, and include new load combinations to address TSV transient loads.

   Changes to the MS piping design analyses are described in the response to EMCB-RAI-14. Design basis load combinations for evaluated Service Level conditions are described in the response to EMCB-RAI-10(b). Quantitative summaries of maximum stress interaction ratios for re-analyzed MS piping are shown in PUSAR Table 2.2-4a (updated in the response to EMCB-RAI-10(a)). Supplemental tables of MS piping stress interaction ratios at other critical MS piping locations are provided in the response to EMCB-RAI-11. Maximum stress interaction results from structural analyses of the MS line containment penetration anchors are also provided in the response to EMCB-RAI-11.

c) For the systems listed above, the only piping loads that are not included in the current design basis analyses and are also not included in the EPU piping analyses, but could potentially occur at EPU conditions, are FW system fluid transient loads. Technical evaluations show that fluid transient loads associated with valve closures or pump trips were considered and are insignificant to the FW piping system design. The occurrence of these loads does not challenge the piping system and connected SSCs beyond their structural design limits. See the response to EMCB-RAI-16 for a detailed discussion.

**EMCB-RAI-13**

For safety-related and non-safety-related piping, consider the piping loads discussed in part (c) of EMCB-RAI-12 and demonstrate that additional postulated piping failure locations, in accordance with the LB for the plant, are not required due to these loads that were omitted from the piping analysis.

**RESPONSE**

As described in the response to EMCB-RAI-12(c), the only system piping loads that are not included in the current design basis analyses and are also not included in the EPU piping analyses, but could potentially occur at EPU conditions, are FW system fluid transient loads. Technical evaluations show that fluid transient loads associated with valve closures or pump trips were considered and are insignificant to the FW piping system design. The occurrence of these loads does not challenge the piping system and connected SSCs beyond their structural design limits. See the response to EMCB-RAI-16 for a detailed discussion. These minor transient loads also do not influence FW pipe break locations. EPU evaluations conclude there are no new postulated pipe failure locations in any system at PBAPS as a result of EPU conditions.

**EMCB-RAI-14**

Please verify whether pipe support additions and modifications, for the MS inside and outside of containment, are required only due to the added spring safety valve (SSV) on the MS line "C" and the added TSVC load case in the MS piping analyses for the EPU (which was not considered in the AOR) and that all other pipe loading cases at EPU conditions are bounded by the current AOR. In addition, clarify whether or not MS piping modifications are required due to the EPU, other than the addition of a new SSV. Please provide a discussion which shows the type and number of added supports and briefly discuss the extent of modifications and repairs to the existing supports.

**RESPONSE**

Pipe support additions and modifications for the MS inside and outside of containment are required not only due to the added SSV on the MS line "C" and the added TSV closure load case, but also due to other changes to the reconstituted MS piping analysis for EPU. The new MS line pipe stress analyses incorporate modifications to pipe supports as required to maintain pipe stresses within Code allowables due to both EPU and non-EPU changes. MS piping deadweight, pressure, and thermal expansion loads are not changed with EPU and are bounded by the current analysis of record. No modifications to MS piping are required for EPU, other than the addition of a new SSV. The EGC Configuration Change Process will ensure that any revisions to the final designs are controlled and will continue to be structurally adequate to perform their intended design functions under EPU conditions.

New and revised analyses for MS line pipe stress, pipe supports, and structural anchors have been performed for the EPU condition. These analyses demonstrate that calculated stresses are within Code allowable stresses for all applicable load cases.

As part of the MS piping reanalysis for EPU, the following changes unrelated to EPU were made to the piping model which contributed to an increase in analyzed pipe support loads:

- MS pipe support configurations were modeled to match the as-installed field configuration.
- Seismic response spectra were regenerated. The original MS piping analysis utilized seismic spectra that were not PBAPS plant specific and not retrievable. The revised analysis for EPU incorporates reconstituted PBAPS plant specific seismic response spectra.

New and Modified MS Pipe Supports – Inside Primary Containment
The following new and modified supports will be installed for the MS lines inside primary containment:

- Four new snubbers per Unit will be added (two on MS line "A", and two on MS line "D") inside primary containment. The new snubbers are required to maintain pipe stresses within Code allowables in the revised piping analysis.
- Four snubbers on Unit 2, and two snubbers on Unit 3 will be replaced to accommodate the increased loading.
- Twelve existing supports per Unit require stiffeners and/or side plates will be added to auxiliary steel at the rear bracket attachment to snubbers (five on each MS lines "B" and "C", and one on each MS line "A" and "D").
- One snubber setting on Unit 3 will be modified to accommodate thermal movement.
- Three spring can hangers per Unit will be replaced to accommodate seismic movement.
- One spring can hanger per Unit will have a pipe clamp replaced with a U-bolt.

New and Modified MS Pipe Supports – Outside Primary Containment
The following new and modified supports will be installed for the MS lines outside primary containment:

- Four new snubbers per Unit will be added (one on each of four MS line risers in MS tunnel).
- Eight existing pipe whip restraints per Unit (two on each of four MS lines) will be modified (added angle restraints) to be active in north-south direction.
- Four existing restraints per Unit (one on each of four MS lines) will be modified to take load in vertical upward direction (added channel restraints to existing base plates for uplift loads).
- Four spring can supports per Unit will be replaced with vertical rigid struts with clamps and end brackets.
- Two snubbers per Unit will be replaced with higher capacity snubbers.

- Four snubbers per Unit will require replacement of supporting frames and/or baseplates and anchor bolts.

- Nine spring cans will be replaced on Unit 2 and eight replaced on Unit 3, along with hardware for change in loads. Auxiliary steel for two spring can hangers on Unit 2 and one on Unit 3 also will be modified.

- One rigid support per Unit will require increased weld size of end brackets.

- One clamp support per Unit will be modified by replacing anchor rods with higher strength rods.

**EMCB-RAI-15**

For the EPU required additions or modifications that involve SSCs important to safety or required to withstand a seismic event, provide quantitative summaries of the highest resulting interaction ratios (calculated over the DB allowable values) at critical locations. Include structural analysis results, due to structural modifications, which install the safety-related lines for the residual heat removal and high pressure service water cross-ties and the condensate storage tank modification.

**RESPONSE**

The EPU modifications important to safety or required to withstand a seismic event which involve piping or pipe support changes include the following. The EGC Configuration Change Process will ensure that any revisions to the final designs are controlled and will continue to be structurally adequate to perform their intended design functions under EPU conditions:

- MS pipe support modifications and addition of third Spring Safety Valve (SV)
- Residual Heat Removal (RHR) Cross-Tie modification
- High Pressure Service Water (HPSW) Cross-Tie modification
- Condensate Storage Tank (CST) standpipe modification

<u>MS Pipe Support and SV Modifications</u>
For MS pipe support and SV modifications, quantitative summaries of calculated interaction ratios from analyses of pipe stresses, RPV nozzles, and structural anchor frames are shown in the supplemental tables included in the response to EMCB-RAI-11, as well as PUSAR Table 2.2-4a. MS pipe support modifications are described in the response to EMCB-RAI-14.

<u>RHR Cross-Tie Modification</u>
*RHR Pipe Stress Analysis*
The following table shows calculated RHR pipe stresses compared to the Code allowable stress values (per ANSI B31.1-1973) with the highest stress interaction ratio for the modified sections of RHR pump discharge piping (bounding results for all loops on both Units 2 and 3) , including the new cross-tie lines.

**Table 15-1**
**Maximum Stress Ratios for Modified RHR Piping**

| Service Level Description | Equation | EPU Stress (psi) | EPU Allowable (psi) | EPU Max Interaction Ratio |
|---|---|---|---|---|
| Service Level A (Sustained) | Eq. 11 | 9,041 | 15,000 | 0.603 |
| Service Level B (Upset) | Eq. 12B | 16,258 | 18,000 | 0.903 |
| Service Level D (Faulted) | Eq. 12D | 26,376 | 36,000 | 0.733 |
| Sustained + Thermal Expansion | Eq. 14 | 28,925 | 37,500 | 0.771 |

*RHR Pipe Supports*
The RHR Cross-Tie modification includes design of the following new or modified pipe supports:

- Four (4) new rigid struts per Unit
- Ten (10) new snubbers per Unit
- Eight (8) new spring hangers per Unit
- Relocated/modified two (2) spring hangers for Unit 2 and four (4) for Unit 3
- Relocated two (2) snubbers per Unit

All component stresses for new, modified, and re-analyzed RHR pipe supports are less than Code allowables (i.e., stress interaction ratio less than 1.0).

<u>HPSW Cross-Tie Modification</u>
*HPSW Pipe Stress Analysis*
The following table shows calculated HPSW pipe stresses compared to the Code allowable stress values at critical locations with the highest stress interaction ratio for the modified sections of HPSW cross-tie piping. The calculated pipe stresses reflect the final modified configuration with HPSW piping and support modifications.

**Table 15-2**
**Maximum Stress Ratios for Final Modified HPSW Piping**

| Service Level | Equation | EPU Stress (psi) | EPU Allowable (psi) | EPU Max Interaction Ratio |
|---|---|---|---|---|
| Service Level A (Sustained) | Eq. 11 | 5,208 | 15,000 | 0.347 |
| Service Level B (Upset) | Eq. 12B | 10,426 | 18,000 | 0.579 |
| Service Level D (Faulted) | Eq. 12D | 18,419 | 36,000 | 0.512 |
| Thermal Expansion | Eq. 13 | N/A | N/A | N/A |

*<u>HPSW Pipe Support Analysis</u>*
The HPSW Cross-Tie modification includes design of the following new or modified pipe supports:

- One (1) new rigid strut per Unit
- Three (3) new rod hangers per Unit

All component stresses for new and re-analyzed HPSW pipe supports are less than Code allowables (i.e., stress interaction ratio less than 1.0).

<u>CST Standpipe Modification</u>

*CST Standpipe and Shell Stress Analysis*

The following table shows the calculated CST standpipe and shell combined interaction ratios for the modified CST standpipe configuration. This stress analysis for the CST standpipe and shell was evaluated in accordance with the AISC Manual of Steel Construction 9th Edition and the Welding Research Council Bulletin for Local Stresses in Spherical and Cylindrical Shells due to External Loadings (WRC-107, 3rd Edition), utilizing allowable stress in accordance with the API Standard (API-650, 3rd Edition). The analysis combines dead weight, thermal and seismic loads. The interaction ratios listed in the table combine the hoop, bending, axial and local stresses compared to the allowable stress as applicable for each component.

**Table 15-3**
**Interaction Ratios for CST Standpipe**

| Component | Combined Interaction Ratio |
|---|---|
| Upper Standpipe Support Bar | 0.209 |
| Upper Standpipe Support Bar Weld | 0.134 |
| CST Shell @ Upper Support Bar | 0.972 |
| CST Shell @ Nozzle J | 0.713 |
| CST Shell @ Nozzle K | 0.642 |
| CST Stanchion Support | 0.025 |
| CST Bottom Shell | 0.405 |

<u>CST Cross Connect Modification</u>

*CST Cross Connect Pipe Stress Analysis*

The following table shows the calculated pipe stresses compared to the Code allowable stress values at critical locations with the highest stress interaction ratio for the modified CST cross connect piping. The calculated pipe stresses reflect the final modified configuration for the CST cross connect modification.

**Table 15-4**
**Max. Stress Ratios for Final Modified CST Cross Connect Piping**

| Service Level Description | Equation | EPU Stress (psi) | EPU Allowable (psi) | EPU Max Interaction Ratio |
|---|---|---|---|---|
| Service Level A (Sustained) | Eq. 11 | 2,828 | 18,700 | 0.151 |
| Service Level B (Upset) | Eq. 12B | 3,009 | 22,440 | 0.134 |
| Service Level D (Faulted) | Eq. 12D | 3,264 | 44,880 | 0.073 |
| Thermal Expansion | Eq. 13 | 13,158 | 28,050 | 0.469 |

**EMCB-RAI-16**

PUSAR, pages 2-47 and 2-52, show that for EPU, the MS and FW system flows increased by approximately 15 percent over the CLTP values. The PUSAR indicates that in the current DB, neither the MS nor the FW piping structural analyses contained load cases due to flow transients.

For the EPU, the MS piping structural analyses were reconciled for loads due to flow transients by reanalysis, which included a load case due to the TSVC transient that bounds the other MS transients. As a result of accounting for flow transient loads in the MS piping, additional supports and modifications to existing supports were required to structurally qualify the MS piping and maintain it within its LB and DB established allowable limits.

In the case of the FW system, the only justification offered in the PUSAR for the effect of the flow transient loads, at the higher EPU flow rates, on the structural integrity of piping and connecting SSCs, is that these flow transient loads have no effect on the piping because they were not included in the original piping evaluations. The NRC staff considers that this justification does not provide reasonable assurance that potentially affected SSCs will be able to maintain their structural integrity under EPU operation with the increased FW flow rates. Please evaluate the effect of water hammer loads, due to events such as control valve closures and feed pump trips, at EPU flows, on pipe stress, pipe breaks, and pipe supports and demonstrate that the structural integrity of the SSCs that could potentially be impacted by water hammer loads, including piping, pipe supports, nozzles, penetrations and connecting SSCs, will be maintained within LB and DB established allowable limits when operating under EPU conditions.

**RESPONSE**

There is reasonable assurance that FW system SSCs will be able to maintain their structural integrity under EPU operations with increased FW flow rates. This conclusion is based on a detailed evaluation of fluid transients in the FW System for current and EPU operations. The evaluation includes reviews in the following areas:

- FW System Design
- Operating Experience
- EPU Flow Changes Evaluation

These review areas are discussed below.

<u>FW System Design</u> – The PBAPS FW System is designed such that fluid transient loads associated with normal and upset plant conditions are minimized and thus are not a part of the design analysis of this system.

The FW System at PBAPS does not include control valves in the main flow path. Therefore, rapid closing of control valves is not a source of fluid transient loading for the FW piping and supports. The only control valves in the system are for minimum pump flow control. These valves are in small lines (6-in) and the design duty for these valves is not changed under EPU.

Each of the three Reactor Feed Pumps (RFPs) at PBAPS is provided with a discharge check valve to prevent reverse flow in the event that one or more RFPs trips while at power. These valves are swing check valves with limited travel and a spring assist. The spring assist is activated in the event of a RFP Turbine (RFPT) trip by a control signal to release the air pressure within the control actuator. This feature effectively closes the check valve disc as the flow is coasting down to remove the potential for reverse flow and check valve slam. In the absence of check valve slam, fluid transient loads are associated with pump coastdown which is not considered to generate significant loading.

The pump discharge check valves, spring assist feature, and control system are regularly inspected and tested. These components have a demonstrated reliable operating history.

<u>Operating Experience</u> – RFP trip at full power is an infrequent event. Most recently, this event occurred at Unit 3 on November 4, 2002. Based on a detailed evaluation of plant data from this event and a simulation of the event using fluid transient analysis computer modeling, the check valves functioned as intended. The check valve disc closed as designed within ~2 sec following pump trip before reverse flow could be established. The corresponding fluid transient loads were minimized.

<u>EPU Flow Change Evaluation</u> – Under EPU, FW flow and momentum in the piping system will increase by up to 15%. A simulation of the single pump trip event extrapolated for EPU increased flow operations indicates the system will continue to close the pump discharge check valve before reverse flow can become established. Projected loads show little change from current operations.

Due to the rapidly lowering FW system pressure immediately following a simultaneous trip of all RFPs, there is less potential for check valve slam with three (3) pump trip than for the single pump trip.

Based on the above evaluation, it is concluded that there is reasonable assurance that FW system SSCs will be able to maintain their structural integrity under EPU operations with increased FW flow rates.

**EMCB-RAI-17**

The PUSAR provides a summary of the EPU evaluation for the safety-related thermowells and probes, including a quantitative summary of the calculated vibratory stresses compared to ASME allowable values. The discussion in the PUSAR indicates that it has followed guidance provided in ASME Section III, Appendix N.

c) Please verify that the allowable values have been derived using guidance from Part 3 of the ASME operation and maintenance (O&M) standards and guidelines.

d) With regard to the safety-related thermowells and probes, the PUSAR makes the following statements on page 2-47.

> To calculate the structural response, a non-dimensional parameter, termed reduced damping (Reference 38, N-1324.1 Equation 76), was calculated:
>
> For off resonance (non lock-in) condition, the structural response is ordinarily small and was calculated using the standard method (Reference 38, N-1324.2, first paragraph)

Provide a quantitative summary of reduced velocity and reduced damping values, in accordance with the requirements from ASME Appendix N, subparagraph N-1324.1, to demonstrate that synchronization (lock-in) of the periodic vortex shedding frequencies to the structural natural frequency of the instrument can be prevented.

**RESPONSE**

c) The allowable values have been derived using guidance from ASME operation and maintenance (O&M) standards and guidelines Part 3.

d) Table 17-1 provides a quantitative summary of reduced velocity and reduced damping values, in accordance with the requirements from ASME Appendix N, subparagraph N-1324.1, to demonstrate that synchronization (lock-in) of the periodic vortex shedding frequencies to the structural natural frequency of the instrument can be prevented. The term "fn/fs" in the table below is component natural frequency/vortex shedding frequency. The allowable criteria based on ASME Appendix N, subparagraph N-1324.1 are: Reduced damping > 1.2, Reduced velocity < 3.3 and fn/fs < 0.7 or fn/fs >1.3.

**Table 17-1**
**Reduced Velocity and Damping Values for Thermowells and Probes**

| Component | Reduced damping | Reduced velocity | fn/fs |
|---|---|---|---|
| MS Thermowell (TW-142) | 9.21 > 1.2[1] | 4.28 > 3.3[1] | 0.53 < 0.7[1] |
| FW Thermowell (TW-140) | 1.62 > 1.2 | 0.47 < 3.3 | 6.01 > 1.3 |
| FW Thermowell (TW-54) | 1.63 > 1.2 | 0.97 < 3.3 | 2.40 > 1.3 |
| FW Sample Probe (SE-16) | 1.72 > 1.2 | 1.21 < 3.3 | 2.11 > 1.3 |
| RRS Thermowell (TW-107) | 1.96 > 1.2 | 0.43 < 3.3 | 6.45 > 1.3 |

Note 1: The lock-in condition for MS Thermowell (TW-142) occurs at less than PBAPS current licensed thermal power. Evaluation of this lock-in condition, following ASME N1324.2 table N-1324.2(a)-1, resulted in a calculated FIV stress of 6881 psi, which is less than the 7690 psi acceptance criterion for carbon steel. At EPU conditions, lock-in does not occur and the calculated FIV stress is 1025 psi.

**EMCB-RAI-18**

With regard to the vibration of piping due to MS and FW increased flow rates and flow velocities, Section 2.2.2.1.2 of the PUSAR states that EPU vibration levels may increase by up to 54 percent of OLTP. In previous power uprates reviewed by the NRC staff, the licensees reviewed existing measured vibration data from plant startups. Vibration data were also collected at several power levels during the power ascension following outages, prior to EPU. Measured vibration data from initial startup and prior to EPU outages (such as stretch power uprate or other refueling outages) formed a baseline from which vibration data were projected to EPU conditions. This projected data was used to determine piping vibration susceptibility and to prepare any needed modifications prior to EPU power ascension. The modifications ranged from new supports and modifications to existing supports, as well as piping modifications.

Please discuss in detail the work that has been performed at PBAPS Units 2 and 3 to evaluate the EPU-impacted small bore and large bore piping systems for vibration susceptibility in order to identify whether piping and pipe support modification are required prior to EPU power ascension. Please supplement Attachment 13, "Flow Induced Vibration," Tables 3-1, 3-2, 3-3 and 3-4 of the EPU application with EPU projected vibrations and the existing measured vibrations (that were utilized for the EPU projections) compared to acceptance criteria. Otherwise, provide a justification for not having completed a baseline vibration monitoring for selected systems and components and for not having identified piping vibration vulnerability prior to EPU ascension.

**RESPONSE**

Vibration susceptibility of small bore and large bore piping due to implementation of EPU is addressed based on industry and PBAPS experience. Since historical vibration data applicable to the establishment of baseline vibration levels for the EPU vibration monitoring program is not available, the following approach has been taken.

Large bore piping, based on industry experience, is not as susceptible to flow-induced vibration fatigue failures as small bore piping. For the large bore piping at PBAPS, a confirmatory vibration monitoring program is being implemented to demonstrate that vibration amplitudes remain within acceptable limits at EPU flow conditions. Baseline vibration measurements corresponding to 100% of current licensed thermal power (CLTP) will be obtained as part of EPU power ascension. Projected EPU vibration levels will be determined based on the measured CLTP vibration levels. Adjustments to the vibration acceptance criteria will be made, if necessary, based on the measured baseline vibration amplitudes and frequencies. In order to ensure that unacceptable vibrations will not occur during EPU power ascension above CLTP, vibration data will be obtained at predetermined power plateaus (see EPU Startup Test Plan, EPU LAR Attachment 10) and the measured vibrations will be compared to Level 1 and Level 2 vibration limits. The Level 1 limits are based on the endurance stress values and the guidance of 2009 ASME O&M-S/G, part 3 (OM-3), as discussed in EPU LAR Attachment 13. These limits may be modified based on the CLTP baseline vibration measurements. The Level 2 limits will be 80% of the Level 1 limits.

Vibration susceptibility assessments of small bore branch piping in systems experiencing significant flow increases due to EPU are being completed prior to EPU power ascension. These assessments identify potentially susceptible configurations using various screening criteria and are supplemented by walkdowns to confirm the small bore line configurations. Initial assessments and confirmatory walkdowns have been completed for the Unit 2 small bore piping. Based on the results of these assessments and walkdowns, a large majority of the small bore lines were determined to be not susceptible to increased header-induced vibrations based on the established screening criteria and require no further action. For the remaining small bore lines, further evaluations are now being performed to better determine vibration susceptibility and the need for any support and/or piping modifications. The additional evaluations and development of any required modification designs will be completed in time to support installation of the modifications prior to EPU power ascension. The initial assessments and confirmatory walkdowns for the Unit 3 small bore lines are currently in progress. Based on the work completed to date, the Unit 3 results are expected to be similar to those for Unit 2.

The necessary monitoring, analyses and modifications to address potential piping vibration vulnerabilities are being performed to ensure there will be no adverse effects at EPU operating conditions.

**EMCB-RAI-19**

Please provide the following:

c) In the current design basis of the plant, are there any piping analyses that contain stratification and is there any CLTP stratification monitoring currently in place? Please list these stratification locations.

d) Explain how these stratification locations have been evaluated and accepted for the EPU conditions and provide a summary of their evaluation results.

**RESPONSE**

PBAPS has not experienced any physical manifestations of thermal stratification, therefore, current design basis does not include thermal stratification. PBAPS has no requirement to monitor temperatures specifically for evidence of the phenomenon. Scheduled piping and piping support inspections, combined with routine plant walk-downs and monitoring, have not identified indication of thermal stratification. Global thermal stratification is not expected to occur in feedwater lines during EPU conditions for the following reasons:

1. EPU results in relatively minor changes to the feedwater system temperatures;
2. EPU changes do not elongate horizontal feedwater piping run configurations;
3. EPU changes do not modify connected vertical feedwater piping;
4. EPU will not significantly change how the feedwater system is operated during any plant mode.

**EMCB-RAI-20**

Discuss the structural evaluation of the vessel supporting structure and its components due to potentially higher EPU loads.

**RESPONSE**

The vessel supporting structure and its components were evaluated as acceptable for EPU conditions based on the [[ ]] screening criteria stated in Section 3.2.2 of Reference 20-1.

[[

]]

In the PBAPS evaluation, the [[ ]] to account for the plant license being extended from a 40-year to 60-year plant license, this [[ ]]

The PBAPS vessel supporting structure and its components at EPU conditions do not experience an increase in flow, temperature, pressure difference or other mechanical load from current operating conditions. Therefore, these components are acceptable for EPU conditions.

Note (1): Reference 1 is GE Nuclear Energy, Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate, NEDC-32424P-A, February 1999.

Reference

20-1 GE Nuclear Energy, "Constant Pressure Power Uprate," NEDC-33004P-A, Revision 4, Class III (Proprietary), July 2003; and NEDO-33004, July 2003.

**EMCB-RAI-21**

In PUSAR Table 2.2-7 for FW and recirculation nozzles, clarify whether the column labeled as "EPU with Environmental Fatigue $U_{en}$)" is for 60-year plant life. Discuss why these EPU fatigue cumulative usage factor values are less than the values in column marked "EPU/(4030 MWt)" which do not contain environmental effects.

**RESPONSE**

The column labeled as "EPU with Environmental Fatigue $U_{en}$" is for 60-year plant life. The EPU fatigue cumulative usage factors are less than the values in the column marked "EPU/(4030 MWt)" for the following reasons:

- Actual cycle counts for PB Units 2 and 3 were used to project cycles out to 60 years of operation instead of using the original conservative cycle projections.
- A more refined analysis was performed for the Feedwater nozzle utilizing detailed modeling in accordance with ANSYS code.

**EMCB-RAI-22**

GE Hitachi Nuclear Energy (GEH) issued 10 CFR Part 21 Safety Information Communication (SC) 09-03 on the subject of Shroud Screening Criteria Reports. SC09-03 lists PBAPS (Unit 2 and Unit 3) as two of the affected plants for the shroud screening criteria flaw evaluations, due to the omission of the postulated Recirculation Line Break loads (transient acoustic or steady state flow-induced loads) in the DB evaluation for the shroud screening criteria reports, which could potentially result in allowable flaw lengths of shroud welds to be smaller than those provided in the shroud screening criteria reports. Please discuss the impact of GEH Safety Information Communication SC09-03 on the PBAPS Units 2 and 3 core shrouds.

**RESPONSE**

The transient Acoustic Load (AC) and steady state Flow-Induced Load (FIL) due to the postulated Recirculation Line Break, as discussed in SC 09-03, were incorporated into revised shroud weld loads for PBAPS. These revised shroud weld loads include the reactor internal pressure difference loads, AC loads, FIL loads, seismic loads and deadweight loads at EPU conditions. The current Unit 2 shroud weld evaluation includes EPU fluence and these revised loads, and supports the ten-year inspection interval with safety factors significantly greater than the minimum requirements. The Unit 3 EPU shroud weld evaluation is scheduled for 2015 (post-refueling weld inspection) before Unit 3 EPU implementation.

The program controlling shroud weld flaw evaluations and screening criteria at PBAPS follows BWRVIP-76-A guidelines and includes Recirculation Line Break AC and FIL loads. Therefore, impacts of SC 09-03 have been addressed at PBAPS.

**EMCB-RAI-23**

Provide a summary of the evaluation that qualified the core plate plugs for operation at the EPU conditions.

**RESPONSE**

PBAPS Unit 3 installed replacement extended core plate plugs in 2001. PBAPS Unit 2 installed replacement extended core plate plugs in 2012. The governing load characteristic of the extended core plate plug is the plug differential pressure. The extended core plate plug design [[ ]] which is higher than the EPU [[ ]] Hence, the replacement extended life core plate plugs remain qualified for operation at the EPU conditions.

**EMCB-RAI-24**

Provide a justification to demonstrate that the increased radiation exposure due to EPU is within the radiation damage threshold of the nonmetallic parts such as in valves, hydraulic snubbers and nonmetallic flexible joints

**RESPONSE**

PBAPS has active and formal programs in place to properly manage the slight increase in radiation expected for EPU. The key elements that will mitigate the increased radiation exposure are design control, procurement evaluations, testing/preventative maintenance and equipment monitoring in accordance with the equipment reliability process. The equipment reliability process also includes incorporation of industry-wide operating experience.

The design process ensures that mechanical components are adequately designed and procured for the service environment and that appropriate preventative maintenance is established. Plant processes will continue to ensure components remain acceptable after EPU LAR Implementation. Preventative maintenance frequencies for susceptible non-metallic components will be reviewed during EPU LAR Implementation to determine if the replacement frequency should be increased.

Plant programs, such as the snubber, check valve, and equipment reliability programs, provide additional controls and monitoring to ensure equipment remains capable of performing its intended function.

Components with non-metallic parts that fall outside of specialized component programs such as the check valve and snubber programs are maintained through the equipment reliability program. The equipment reliability program involves periodic maintenance and testing as well as incorporating plant operating experience, vendor recommendations, and industry experience. The integrated effect of these existing programs and component reviews provides assurance that important systems, structures, and components are capable of fulfilling their intended functions and will be acceptable following EPU LAR Implementation. Normal and accident radiation doses will increase by approximately 14% due to EPU. This change in radiation dose will not impact the ability of plant programs to manage component service life for non-metallic parts that are outside of the EQ program.