ATTACHMENTS 13P, 14P AND 16P CONTAIN INFORMATION REQUESTED TO BE WITHHELD FROM PUBLIC DISCLOSURE UNDER 10 CFR 2.390

FPL.

L-2018-174
10 CFR 54.17
10 CFR 2.390

October 24, 2018

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555-0001

Re: Florida Power & Light Company
Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Turkey Point Units 3 and 4 Subsequent License Renewal Application
Safety Review Requests for Additional Information (RAI) Set 4 Responses

References:

1. FPL Letter L-2018-004 to NRC dated January 30, 2018, Turkey Point Units 3 and 4 Subsequent License Renewal Application (ADAMS Accession No. ML18037A812)
2. FPL Letter L-2018-082 to NRC dated April 10, 2018, Turkey Point Units 3 and 4 Subsequent License Renewal Application – Revision 1 (ADAMS Accession No. ML18113A134)
3. NRC RAI E-Mail to FPL dated October 1, 2018, Requests for Additional Information for the Safety Review of the Turkey Point Subsequent License Renewal Application – Set 4 (EPID No. L-2018-RNW-0002) (ADAMS Accession Nos. ML18269A209 and ML18269A210)

Florida Power & Light Company (FPL) submitted a subsequent license renewal application (SLRA) for Turkey Point Units 3 and 4 to the NRC on January 30, 2018 (Reference 1) and SLRA Revision 1 on April 10, 2018 (Reference 2).

The purpose of this letter is to provide, as attachments to this letter, public and certain non-public (proprietary) responses to the safety review RAIs issued by the NRC on October 1, 2018 (Reference 3). Each RAI response and its corresponding attachment and associated information enclosure are indexed on page 3 of this letter. The attachments identify revisions amending the SLRA (if applicable).

Attachments 13P, 14P and 16P have been placed after Attachment 34 of this submittal and contain proprietary information (enclosed within brackets and/or marked 'Withhold from Public Disclosure Under 10 CFR 2.390') that FPL requests be withheld from public disclosure under 10 CFR 2.390(a)(4). The withholding request applications for this proprietary information are enclosed with Attachments 13, 13P, 16 and 16P.

A084
NRR

As an SLRA NRC In-Office Audit follow-up item, FPL committed to revise the original 80-year CUF_{en} value using the most recent methodology prescribed in Reference 2 of Attachments 13 and 13P of this submittal. The revised 80-year CUF_{en} calculations and associated SLRA revisions supporting this methodology change will be submitted by November 19, 2018.

If you have any questions, or need additional information, please contact me at 561-691-2294.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on October 24, 2018.

Sincerely,

William Maher
Senior Licensing Director
Florida Power & Light Company

WDM/RFO

Attachments: 37 RAI Responses (refer to Letter Attachment Index)

Enclosures: 16 RAI Response Enclosures (refer to Letter Enclosures Index)

Enclosure 1

Westinghouse Letter CAW-18-4804 dated September 18, 2018, Application for Withholding Proprietary Information from Public Disclosure

Westinghouse Affidavit CAW-18-4804

Proprietary Information Notice and Copyright Notice

LTR-SDA-II-17-13-P, Rev. 4, "Environmentally Assisted Fatigue Evaluation of the Turkey Point Unit 3 and Unit 4 Pressurizer Upper Head and Shell and Reactor Vessel Core Support Blocks (Westinghouse Proprietary Class 2)" (Proprietary)

LTR-CECO-17-025-P, Rev. 3, "Environmentally Assisted Fatigue Evaluation of the Turkey Point Unit 3 and Unit 4 Replacement Steam Generators" (Proprietary)

Westinghouse Non-Proprietary Class 3

Westinghouse

Westinghouse Electric Company
1000 Westinghouse Drive
Cranberry Township, Pennsylvania 16066
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Direct tel: (412) 374-4643
Direct fax: (724) 940-8542
e-mail: greshaja@westinghouse.com

CAW-18-4804

September 18, 2018

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-SDA-II-17-13-P, Rev. 4, "Environmentally Assisted Fatigue Evaluation of the Turkey Point Unit 3 and Unit 4 Pressurizer Upper Head and Shell and Reactor Vessel Core Support Blocks (Westinghouse Proprietary Class 2)" (Proprietary)

LTR-CECO-17-025-P, Rev. 3, "Environmentally Assisted Fatigue Evaluation of the Turkey Point Unit 3 and Unit 4 Replacement Steam Generators" (Proprietary)

The Application for Withholding Proprietary Information from Public Disclosure is submitted by Westinghouse Electric Company LLC ("Westinghouse"), pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Nuclear Regulatory Commission's ("Commission's") regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-18-4804 signed by the owner of the proprietary information, Westinghouse. The Affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by Florida Power & Light Company.

Correspondence with respect to the proprietary aspects of the Application for Withholding or the Westinghouse Affidavit should reference CAW-18-4804 and should be addressed to James A. Gresham, Consulting Engineer, Licensing and Regulatory Affairs, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 2 Suite 259, Cranberry Township, Pennsylvania 16066.

James A. Gresham
Licensing and Regulatory Affairs

CAW-18-4804

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

I, James A. Gresham, am authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC ("Westinghouse") and declare that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief.

Executed on: 9/18/18

James A. Gresham
Licensing and Regulatory Affairs

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

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

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

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

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

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

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

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

(vi) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in LTR-SDA-II-17-13-P, Rev. 4, "Environmentally Assisted Fatigue Evaluation of the Turkey Point Unit 3 and Unit 4 Pressurizer Upper Head and Shell and Reactor Vessel Core Support Blocks (Westinghouse Proprietary Class 2)" (Proprietary), dated September 18, 2018 and LTR-CECO-17-025-P, Rev. 3, "Environmentally Assisted Fatigue Evaluation of the Turkey Point Unit 3 and Unit 4 Replacement Steam Generators" (Proprietary), dated September 14, 2018, for submittal to the Commission, being transmitted by Florida Power & Light Company letter. The proprietary information as submitted by Westinghouse is that associated with Westinghouse's request for NRC approval of LTR-SDA-II-17-13-P and LTR-CECO-17-025-P, and may be used only for that purpose.

(a) This information is part of that which will enable Westinghouse to provide a technical justification for acceptability of environmental assisted fatigue for various components for Turkey Point Units 3 and 4 in support of their subsequent license renewal program.

(b) Further, this information has substantial commercial value as follows:

(i) Westinghouse plans to sell the use of similar information to its customers for the purpose of supporting other subsequent license renewal programs.

(ii) Westinghouse can sell support and defense of industry guidelines and acceptance criteria for plant-specific applications.

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

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

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

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

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

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

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

COPYRIGHT NOTICE

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

Enclosure 2

Framatome Application for Withholding Proprietary Information from Public Disclosure dated October 12, 2018

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Gayle Elliott. I am Deputy Director, Licensing & Regulatory Affairs, for Framatome Inc. (Framatome) and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by Framatome to determine whether certain Framatome information is proprietary. I am familiar with the policies established by Framatome to ensure the proper application of these criteria.

3. I am familiar with the Framatome information contained in Calculation Summary Sheets 32-9279161-002, "Turkey Point SLR EAF Analysis for Reactor Vessel Flange," 32-9279174-002, "Turkey Point – 3 & 4 CRDM Nozzle to Adapter Weld Connection EAF Evaluation," 32-9279212-002, "Turkey Point 3 & 4 Replacement RVCH CRDM Nozzle EAF Analysis," 32-9279362-002, "TP Vent Nozzle Environmentally Assisted Fatigue," 32-9279367-002, "TP CRDM Latch Housing Environmentally Assisted Fatigue," all dated September 2018, and 32-9280202-003, "TP CRDM Lower Joint Environmentally Assisted Fatigue," dated October 2018 and referred to herein as "Documents." Information contained in these Documents has been classified by Framatome as proprietary in accordance with the policies established by Framatome for the control and protection of proprietary and confidential information.

4. These Documents contain information of a proprietary and confidential nature and is of the type customarily held in confidence by Framatome and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in these Documents as proprietary and confidential.

5. These Documents have been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in these Documents be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by Framatome to determine whether information should be classified as proprietary:

(a) The information reveals details of Framatome's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for Framatome.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for Framatome in product optimization or marketability.

(e) The information is vital to a competitive advantage held by Framatome, would be helpful to competitors to Framatome, and would likely cause substantial harm to the competitive position of Framatome.

The information in these Documents is considered proprietary for the reasons set forth in paragraphs 6(d) and 6(e) above.

7. In accordance with Framatome's policies governing the protection and control of information, proprietary information contained in these Documents has been made available,

on a limited basis, to others outside Framatome only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. Framatome policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 12
day of October, 2018.

Heidi H Elder

Heidi Elder
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 12/31/22
Reg. # 7777873

HEIDI HAMILTON ELDER
NOTARY PUBLIC
COMMONWEALTH OF VIRGINIA
REGISTRATION NO. 7777873
MY COMM. EXPIRES: 12/31/2022

Enclosure 1

Framatome Application for Withholding Proprietary Information from Public Disclosure dated October 12, 2018

AFFIDAVIT

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) ss.
CITY OF LYNCHBURG)

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7. In accordance with Framatome's policies governing the protection and control of information, proprietary information contained in these Documents has been made available,

on a limited basis, to others outside Framatome only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. Framatome policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 12
day of October, 2018.

Heidi H Elder

Heidi Elder
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 12/31/22
Reg. # 7777873

HEIDI HAMILTON ELDER
COMMONWEALTH OF VIRGINIA
REGISTRATION NO. 7777873
MY COMM. EXPIRES 12/31/2022
NOTARY PUBLIC

Enclosure 2

Framatome Application for Withholding Proprietary Information from Public Disclosure dated October 17, 2018

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Nathan E. Hottle. I am Manager, Product Licensing, for Framatome Inc. (Framatome) and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by Framatome to determine whether certain Framatome information is proprietary. I am familiar with the policies established by Framatome to ensure the proper application of these criteria.

3. I am familiar with the Framatome information contained in the following document: ANP-3731P Revision 0, "Response to NRC Request for Additional Information (RAI 4.3.3-5) For Turkey Point Units 3 and 4 Subsequent License Renewal Application," referred to herein as "Document." Information contained in this Document has been classified by Framatome as proprietary in accordance with the policies established by Framatome Inc. for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by Framatome and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

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The information in this Document is considered proprietary for the reasons set forth in paragraphs 6(c) and 6(d) above.

7. In accordance with Framatome's policies governing the protection and control of information, proprietary information contained in this Document has been made available, on a limited basis, to others outside Framatome only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. Framatome policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 17th
day of October, 2018.

Heidi H Elder

Heidi Hamilton Elder
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 12/31/2022
Reg. # 7777873

HEIDI HAMILTON ELDER
COMMONWEALTH
REGISTRATION NO.
7777873
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12/31/2022
OF VIRGINIA
NOTARY PUBLIC

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cc: w/o Attachments 13P, 14P and 16P

Senior Resident Inspector, USNRC, Turkey Point Plant
Regional Administrator, USNRC, Region II
Project Manager, USNRC, Turkey Point Nuclear
Plant Project Manager, USNRC, SLRA
Plant Project Manager, USNRC, SLRA Environmental
Ms. Cindy Becker, Florida Department of Health

NRC RAI Letter Nos. ML18269A209 and ML18269A210 dated October 1, 2018

- **Leak-Before-Break RCS Pipe, TLAA 4.7**

Regulatory Basis:

Title 10 of the Code of Federal Regulations (10 CFR) Section 54.21(c) requires an applicant to demonstrate that the effects of aging for structures and components will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the subsequent period of extended operation. One of the findings that the staff must make to issue a renewed license (10 CFR 54.29(a)) is that actions have been identified and have been or will be taken with respect to the managing the effects of aging during the subsequent period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21, such that there is reasonable assurance that the activities authorized by the subsequent renewed license will continue to be conducted in accordance with the current licensing basis. As described in NUREG-2192, Rev. 0, "Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants," dated July 2017 (SRP-SLR), an applicant may demonstrate compliance with 10 CFR 54.21(a)(3) by referencing the applicable aging management programs and activities in the NUREG-2191, Rev. 0, "Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report," dated July 2017. In order to complete its review and enable making a finding under 10 CFR 54.29(a), the staff requires additional information in regard to the matters described below.

Background:

The regulation in 10 CFR 54.21(c)(1)(ii) states that, for a specific time limited aging analysis (TLAA) that is dispositioned in accordance with this regulation, the applicant must demonstrate that the analysis has been projected to the subsequent end of the period of extended operation. Subsequent license renewal application (SLRA) Section 4.7.3, "Leak-Before-Break Analysis for Reactor Coolant System Piping," identifies the leak-before-break (LBB) analysis in the current licensing basis as an analysis that meets the definitions of a TLAA for the SLRA.

As part of the SLRA, the applicant evaluated the LBB analysis of reactor coolant system (RCS) primary loop piping as documented in WCAP-15354, Revision 1, "Technical Justification for Eliminating Primary Loop Pipe Rupture as a Structural Design Basis for Turkey Point Units 3 and 4 Nuclear Power Plants for the Subsequent License Renewal Time-Limited Aging Analysis Program (80 Years) Leak-Before-Break Evaluation," September 2017.

The cast austenitic stainless steel material in RCS piping may be affected by thermal embrittlement during the subsequent period of extended operation. In addition, fatigue crack growth calculation for the reactor coolant system piping is part of the TLAA.

RAI 4.7.3-1

Issue:

In Section 8 of WCAP-15354, Revision 1, the applicant analyzed fatigue crack growth for the postulated circumferential flaw. The staff understands that typically an axial flaw is not limiting in LBB analyses; therefore, fatigue crack growth would not be performed for the axial flaw. However, WCAP-15354 does not clearly state that an axial flaw is not limiting for the applicant's LBB analysis.

Request:

Discuss whether fatigue crack growth was performed for a postulated axial flaw. In addition, discuss whether an axial flaw is not limiting in terms of pipe rupture such that fatigue crack growth is not needed for the axial flaw.

FPL Response:

Fatigue crack growth was not performed for a postulated axial flaw orientation in the RCL piping of Turkey Point Units 3 and 4. Based on past evaluations of RCL piping for other operating plants, the circumferential flaw evaluations bound the axial flaws. Firstly, the loading conditions, including internal blow-off pressure axial force and the conservative combination of moment loads prescribed by SRP 3.6.3 (References 1 and 2) ensure that the circumferential flaw orientation is most limiting. In addition, LBB evaluations are generally focused on the weld locations since the fracture toughness of the weld material is weaker than the base piping material. As such, an axial flaw in the weld material would be unlikely to see considerable growth into the tougher base metal and thus, be restricted to the weld material. Circumferential growth of a flaw through the weld material represents the more realistic scenario.

Axial-oriented flaws are typically only considered for locations where Alloy 82/182 material is present, due to the susceptibility to Primary Water Stress Corrosion Cracking (PWSCC). Section 10 of WCAP-15354, Revision 1 (Reference 3) confirms that there is not Alloy 82/182 material present in the welds for the Turkey Point Units 3 and 4 primary loop piping. As such, the postulation of an axial orientation is not necessary for addressing concerns related to PWSCC susceptible materials.

The intention of an LBB evaluation is to justify that the double-ended guillotine type of pipe break is not a credible failure mode for the RCL piping system. Fatigue crack growth for an LBB evaluation is typically presented as a defense-in-depth justification to demonstrate that a small surface flaw would not develop to a through-wall flaw during the plant design life. In this demonstration of fatigue crack growth, the evaluation considers the growth of a circumferential flaw since this orientation is directly representative of a scenario that could result in a double-ended guillotine failure. The premise of an LBB evaluation is focused on the double-ended guillotine failure since these have the potential for more severe secondary damage from jet impingement and pipe whip. Therefore, the evaluation of a circumferential flaw is more appropriate and conservative than an axial flaw since an axial flaw will not result in a double-ended guillotine break.

References:

1. Standard Review Plan: Public Comments Solicited; 3.6.3 Leak-Before-Break Evaluation Procedures; Federal Register/Vol. 52, No. 167/Friday August 28, 1987/Notices, pp. 32626-32633.
2. NUREG-0800, Revision 1, Standard Review Plan: 3.6.3 Leak-Before-Break Evaluation Procedures, March 2007.
3. Westinghouse Reports, WCAP-15354-P (Proprietary) and WCAP-15354-NP (Non-proprietary), "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Turkey Point Units 3 and 4 Nuclear Power Plants for the Subsequent License Renewal Time-Limited Aging Analysis Program (80 Years) Leak-Before-Break Evaluation," August 2017.

Associated SLRA Revisions:

None

Associated Enclosures:

None

NRC RAI Letter Nos. ML18269A209 and ML18269A210 dated October 1, 2018

RAI 4.7.3-2

Issue:

WCAP-14237 is the original LBB analysis for the primary loop piping. Table 7-1 in WCAP-14237 lists fracture toughness values J_{Ic} and J_{max} of locations 2 and 11 of the primary loop piping. Table 4-5 of WCAP-15354 also lists J_{Ic} and J_{max} of location 2 (hot leg) and location11 (cold leg). The staff noted that there are some discrepancies between WCAP-14237 and WCAP-15354. Specifically, the J_{Ic} values are slightly higher in WCAP-15354 than those in WCAP-14237. The J_{max} values are much higher in WCAP-15353 than that in WCAP-14237.

Request:

Discuss why there are differences in J_{Ic} and J_{max} between WCAP-14237 and WCAP-15353. As part of the response, discuss specifically why J_{max} values are so much higher in WCAP-15353 than those in WCAP-14237.

FPL Response:

The original LBB analysis for the RCL piping, documented in WCAP-14237 (Reference 1), calculates the aged fracture toughness properties of cast stainless steel components using the Westinghouse methodology that is documented in WCAP-10931, Revision 1 (Reference 2). This methodology was based on a nominal set of material testing data and results in conservative approximations of the aged fracture toughness values; J_{Ic} and J_{max}. Specifically, for the Turkey Point analysis location 11 of WCAP-14237 (Reference 1), the chemical composition of the cast material results in the calculation of a very low value of Charpy U-notch fracture toughness (KCU). For very low values of KCU, the methodology of WCAP-10931, Revision 1 (Reference 2) required the assumption that this material be considered as fully aged, and the maximum fracture toughness (J_{max}) is conservatively taken to be equal to the crack initiation fracture toughness (J_{Ic}). This conservative assumption was not necessary for analysis location 2 of WCAP-14237 (Reference 1) since the material chemical composition was not as limiting.

The update to the LBB analysis for the RCL piping for the 80-year SLR, documented in WCAP-15354-P, Revision 1 (Reference 3), considers the updated NRC-approved and industry-adopted methodology for estimating the aged fracture toughness properties of cast stainless steel components which is established in NUREG-4513, Revision 2 (Reference 4). The methodology of NUREG-4513, Revision 2 (Reference 4) is based on a considerably more extensive set of material testing data which eliminates the need for some of the overly conservative approximations that were inherent in the methodology of WCAP-10931, Revision 1 (Reference 2). By utilizing the methodology of NUREG-4513, Revision 2 (Reference 4), the updated LBB analysis for the RCL piping for the 80-year SLR was able to establish increases to the aged fracture toughness values, J_{Ic} and J_{max}, for the most limiting material locations.

References:

1. Westinghouse Report, WCAP-14237, "Technical Justification for Eliminating Large Primary Loop Rupture as the Structural Design Basis for the Turkey Point Units 3 and 4 Nuclear Power Plants," December 1994
2. Westinghouse Report, WCAP-10931, Revision 1, "Toughness Criteria for Thermally Aged Cast Stainless Steel," July 1986.
3. Westinghouse Reports, WCAP-15354-P (Proprietary) and WCAP-15354-NP (Non-proprietary), "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Turkey Point Units 3 and 4 Nuclear Power Plants for the Subsequent License Renewal Time-Limited Aging Analysis Program (80 Years) Leak-Before-Break Evaluation," August 2017.
4. O.K. Chopra, "Estimation of Fracture Toughness of Cast Stainless Steels during Thermal Aging in LWR Systems," NUREG/CR-4513, Revision 2, U.S. Nuclear Regulatory Commission, Washington, DC, May 2016.

Associated SLRA Revisions:

None

Associated Enclosures:

None

NRC RAI Letter Nos. ML18269A209 and ML18269A210 dated October 1, 2018

RAI 4.7.3-3

Issue:

On page 8-2 of WCAP-15354, the applicant stated that the calculated fatigue crack growth for semi-elliptic surface flaws of circumferential orientation and various depths is summarized in Table 8-2. However, it is not clear from Table 8-2 whether the final flaw size is in terms of the though-wall crack depth or circumferential crack length.

Request:

Clarify whether the final flaw size derived in Table 8-2 of WCAP-15354 is the depth of the pipe wall thickness, or the length in the circumferential direction.

FPL Response:

The values shown in Table 8-2 of WCAP-15354, Revision 1 (Reference 1) represent the depth of a semi-elliptic surface in the through-wall (radial) direction. The flaw depths are measured from the inside surface of the piping through the pipe wall thickness.

References:

1. Westinghouse Reports, WCAP-15354-P (Proprietary) and WCAP-15354-NP (Non-proprietary), "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Turkey Point Units 3 and 4 Nuclear Power Plants for the Subsequent License Renewal Time-Limited Aging Analysis Program (80 Years) Leak-Before-Break Evaluation," August 2017.

Associated SLRA Revisions:

None

Associated Enclosures:

None

NRC RAI Letter Nos. ML18269A209 and ML18269A210 dated October 1, 2018

RAI 4.7.3-4

Issue:

During the staff audit of the applicant's documents, the staff noticed that document AR 01610224 is related to errors in pipe stress software.

Request:

Discuss whether errors in pipe stress software as discussed in AR 01610224 affected the applied loads and stresses used in the LBB analysis of reactor coolant piping.

FPL Response:

Error report AR 01610224 is associated with the PIPESTRESS analysis program. A review of the piping analysis inputs used in the LBB evaluation of the RCL piping has confirmed that PIPESTRESS was not used. As such, the software error discussed in AR 01610224 does not apply to the RCL piping analysis and does not impact the applied loads and stresses used in the LBB evaluation.

References:

None

Associated SLRA Revisions:

None

Associated Enclosures:

None

NRC RAI Letter Nos. ML18269A209 and ML18269A210 dated October 1, 2018

RAI 4.7.3-5

Issue:

The staff noted that Electric Power Research Institute (EPRI) topical report "Materials Reliability Program: Assessment of Residual Heat Removal Mixing Tee Thermal Fatigue in PWR [Pressurized Water Reactors] Plants (MRP [Materials Reliability Program]- MRP-192, Revision 2), August 2012," and "Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines (MRP-146, Revision 1), June 2011" are related to thermal fatigue of safety-related piping.

Request:

Discuss whether the RCS primary loop piping at Turkey Point Units 3 and 4 is subject to the thermal fatigue as discussed in MRP-146 and MRP-192. If yes, discuss whether RCS primary loop piping satisfies the LBB screening criteria as specified in Standard Review Plan 3.6.3 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition." Discuss whether thermal fatigue data are included in the fatigue crack growth calculations in WCAP-15354, Revision 1. If yes, provide examples of the thermal fatigue data (transient loading).

FPL Response:

The reactor coolant loop (RCL) piping of Turkey Point Units 3 and 4 are not subject to the thermal fatigue effects identified in Electric Power Research Institute (EPRI) reports MRP-146 and MRP-192.

MRP-146 provides guidance related to the screening and evaluation of locations in normally stagnant, non-isolable piping systems attached to the RCL piping where swirl penetration and/or valve in-leakage may cause thermal fatigue cracking. The RCL piping of Turkey Point Units 3 and 4 are not normal stagnant lines, and therefore, the screening and evaluation guidance of MRP-146 do not apply to the RCL piping analysis and do not impact the conclusion of the LBB evaluation.

MRP-192 reviews an occurrence of thermal fatigue cracking in a mixing tee component of a Residual Heat Removal (RHR) line caused by cyclic mixing of hot and cold reactor coolant in a zone where hot heat exchanger bypass flow rejoined the cold heat exchanger outlet flow. This type of cyclic thermal mixing is not able to occur at any location of the RCL piping of Turkey Point Units 3 and 4, and therefore, an assessment related to MRP-192 does not apply to the RCL piping analysis and does not impact the conclusion of the LBB evaluation.

As discussed in Section 2.3 of WCAP-15354, Revision 1 (Reference 1), the assessment of low cycle fatigue is performed in the form of a fatigue crack growth analysis which is documented in Section 8 of WCAP-15354, Revision 1 (Reference 1). The applied transients, which contribute the thermal fatigue effects considered for the 80-year SLR operating life of Turkey Point Units 3 and 4, are identified in Table 8-1 of WCAP-15354, Revision 1 (Reference 1).

References:

1. Westinghouse Reports, WCAP-15354-P (Proprietary) and WCAP-15354-NP (Non-proprietary), "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Turkey Point Units 3 and 4 Nuclear Power Plants for the Subsequent License Renewal Time-Limited Aging Analysis Program (80 Years) Leak-Before-Break Evaluation," August 2017.

Associated SLRA Revisions:

None

Associated Enclosures:

None

NRC RAI Letter Nos. ML18269A209 and ML18269A210 dated October 1, 2018

RAI 4.7.3-6

Issue:

The elbows in the RCS primary loop piping are made of cast austenitic stainless steel material which is susceptible to thermal embrittlement when the component is placed in a long term service. In SLRA Section 4.7.3, the applicant used the method in NRC document NUREG/CR-4513, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems," Revision 2, to predict the fully aged fracture toughness values for the elbows at the end of 80 years. SLRA Section 4.7.5 discusses thermal embrittlement of the reactor coolant pump casing which is made of cast austenitic stainless-steel material. In SLRA Section 4.7.5, the applicant used NUREG/CR-4513, Revision 2 and Westinghouse report, WCAP-13045, to predict the fully aged fracture toughness values for the reactor coolant pump casing.

Request:

Discuss whether the fracture toughness data in WCAP-13045 as discussed in Section 4.7.5 are applicable to the elbows in the RCS primary piping as discussed in Section 4.7.3. If yes, discuss whether the fracture toughness values used for the elbows in the RCS primary piping in the LBB analysis in Section 4.7.3 are the lowest values (i.e., most limiting) based on the data in WCAP-13045 and NUREG/CR-4513, Revision 2.

FPL Response:

The aged fracture toughness values presented in WCAP-13045 (Reference 1) are based on specific material testing composition data for RCL pump casings and are not specific to any cast RCL piping component for Turkey Point Units 3 and 4. As such, the aged fracture toughness values in WCAP-13045 (Reference 1) are not applicable to the Turkey Point Units 3 and 4 LBB evaluation of the RCL piping. Instead, for the Turkey Point Units 3 and 4 CF8M cast RCL elbows, the aged fracture toughness values were calculated for the specific elbow material compositions; first in WCAP-14237 (Reference 2), then most recently in WCAP-15354-P, Revision 1 (Reference 3) using the updated NRC-approved methodology of NUREG-4513, Revision 2 (Reference 4).

However, it should be noted that both WCAP-14237 (Reference 2) and WCAP-13045 (Reference 1) considers the same methodology for calculating aged fracture toughness properties of cast stainless steel components from WCAP-10931, Revision 1 (Reference 5). While the fracture toughness properties of WCAP-13045 (Reference 1) are not applicable to the LBB evaluation of the RCL piping, the corresponding methodology for the calculation of the fracture toughness properties was considered in previous revision of the LBB evaluation of the RCL piping, WCAP-14237 (Reference 2). This methodology has since been superseded by NUREG-4513, Revision 2 (Reference 4) which is utilized in the current LBB evaluation, WCAP-15354-P, Revision 1 (Reference 3).

References:

1. Westinghouse Report, WCAP-13045, "Compliance to ASME Code Case N-481 of the Primary Loop Pump Casings of Westinghouse Type Nuclear Steam Supply System," September 1991
2. Westinghouse Report, WCAP-14237, "Technical Justification for Eliminating Large Primary Loop Rupture as the Structural Design Basis for the Turkey Point Units 3 and 4 Nuclear Power Plants," December 1994
3. Westinghouse Reports, WCAP-15354-P (Proprietary) and WCAP-15354-NP (Non-proprietary), "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Turkey Point Units 3 and 4 Nuclear Power Plants for the Subsequent License Renewal Time-Limited Aging Analysis Program (80 Years) Leak-Before-Break Evaluation," August 2017
4. O.K. Chopra, "Estimation of Fracture Toughness of Cast Stainless Steels during Thermal Aging in LWR Systems," NUREG/CR-4513, Revision 2, U.S. Nuclear Regulatory Commission, Washington, DC, May 2016
5. Westinghouse Report, WCAP-10931, Revision 1, "Toughness Criteria for Thermally Aged Cast Stainless Steel," July 1986

Associated SLRA Revisions:

None

Associated Enclosures:

None

NRC RAI Letter Nos. ML18269A209 and ML18269A210 dated October 1, 2018

- **Reactor Vessel Neutron Embrittlement Analyses, TLAA 4.2**

Regulatory Basis:

Section 54.21(c)(1) of 10 CFR states that a list of time-limited aging analyses, as defined in 10 CFR 54.3, must be provided, and that the applicant shall demonstrate that:

(i) The analyses remain valid for the period of extended operation;

(ii) The analyses have been projected to the end of the period of extended operation; or

(iii) The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

In order to verify that TLAAs for pressurized thermal shock (PTS) and adjusted reference temperature (ART) have been conservatively projected in accordance with 10 CFR 54.21(c)(1)(ii), the staff requires additional information as detailed below.

RAI 4.2-1

Background:

For the pressurized thermal shock (PTS), SRP-SLR Section 4.2.2.1.3 references 10 CFR 50.61 TLAA, which requires that the RT_{PTS} values be updated when there is a change in the expiration date of a plant's operating license. Therefore, the SRP-SLR states the RT_{PTS} values must be calculated for the subsequent period of extended operation.

The applicant described its evaluation of the PTS TLAA in SLRA Section 4.2.2. The applicant dispositioned the PTS TLAA as projected through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii). Among the input data for calculation of RT_{PTS} is the unirradiated reference temperature, $RT_{NDT(u)}$, and the standard deviation of $RT_{NDT(u)}$, designated σ_U. SLRA Tables 4.2.2-1 and 4.2.2-2 provide the input data and results of the RT_{PTS} calculations for 72 EFPY.

The staff also notes that the same $RT_{NDT(u)}$ and σ_U values are used in the calculation of ART in SLRA Tables 4.2.4-1 and 4.2.4-2, and that this RAI is also applicable to the ART TLAA discussed in SLRA Section 4.2.4.

Issue:

The unirradiated reference temperature ($RT_{NDT(u)}$) and the standard deviation of $RT_{NDT(u)}$ (σ_U) values for certain reactor pressure vessel (RPV) materials, and the ART values for TP3 and TP4 reported in the SLRA have changed compared to those used in the current PTS analysis of record, both of which are contained in the Turkey Point Units 3 and 4 Extended Power Uprate (EPU) Licensing Report,[1] , which is Attachment 4 to the license amendment request for an EPU[2]. The license amendment request for EPU was approved via a license amendment dated June 15, 2012.[3] The tables below compare the changes from the EPU to the SLRA for the two parameters.

The SLRA did not provide references to source documents for the revised values of $RT_{NDT(u)}$ and σ_U. The staff noted that some of the revised values result in lower values of RT_{PTS} and ART. Therefore, it is necessary for the staff to verify the revised $RT_{NDT(u)}$ and σ_U are accurate in order to assess the applicant's disposition for the PTS and ART TLAAs in accordance with 10 CFR 54.21(c)(1)(ii).

Request:

For the materials listed in Table 1 and Table 2, the staff requests that the applicant:

a) Justify the discrepancy between the SLRA[1] (Table 4.2.2-1 for Unit 3 and Table 4.2.2-2 for Unit 4) and the EPU licensing report[2] (Table 2.1.2-3 for Unit 3 and Table 2.1.2-4 for Unit 4) for the $RT_{NDT(u)}$ values and σ_u.

b) Describe how the $RT_{NDT(u)}$ values and σ_u reported in the SLRA were determined, including a description of the data set.

FPL Response:

a) Explanation for the differences/discrepancies in the values of $RT_{NDT(u)}$ and σ_u reported in SLRA Table 4.2.2-1 for Unit 3 and Table 4.2.2-2 for Unit 4 and Table 2.1.2-3 for Unit 3 and Table 2.1.2-4 for Unit 4 in the EPU licensing report are provided in Table A (Unit 3) and Table B (Unit 4) below. Reference 1 and 2 below are the sources for each of the values provided in Table A for Unit 3 and Table B for Unit 4 below.

b) The data sets and the determination of the generic values of $RT_{NDT(u)}$ and σ_u for US supplied forgings and Linde 80 welds are reported in PROPRIETARY BAW-2313 Rev 7 and is the source report for the values discussed in Table A for Unit 3 and Table B for Unit 4.

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1. Turkey Point, Units 3 and 4 - License Amendment Request for Extended Power Uprate, Attachment 4; Licensing Report, December 14, 2010 (ADAMS Accession No. ML103560204)

2. FPL Letter No. L-2010-113 from Michael Kiley to U.S. Nuclear Regulatory Commission, "Turkey Point Units 3 and 4, Docket Nos. 50-250 and 50-251, License Amendment Request for Extended Power Uprate (LAR 205)," October 21, 2010, (ML103560167)

3. Jason C. Paige (NRC) letter to Mano Nazar (FPL), "Turkey Point Units 3 and 4 -Issuance of Amendments Regarding Extended Power Uprate," June 15, 2012 (ML11293A365)

Table A: Explanation of Differences in the Values of RT_{NDT} and σ_u Between the Turkey Point Unit 3 EPU License Report (EPU-LR) and Turkey Point Unit 3 Subsequent License Renewal Application (SLRA)

Reactor Vessel Material	Explanation of Differences in values	Data Source for $RT_{NDT(u)}$ and σ_u
Lower Head Ring (Transition)	This extended beltline (EB) material was not previously reported in the EPU-LR. The SLRA values reported generic US supplier forging data from a larger data set and are considered more representative.	Generic forging data from PROPRIETARY BAW-2313 R7 report, Section 2.2.3 and Table 2-5 for these USA supplied forgings.
Inlet Nozzle 1, 2, 3	The EPU-LR reported values of RT_{NDT} are from NRC BTP 5-3 with a generic σ_u. The SLRA values reported generic US supplier forging data from a larger data set and are considered more representative.	Generic forging data from PROPRIETARY BAW-2313 R7 report, Section 2.2.3 and Table 2-5 for these USA supplied forgings.
Outlet Nozzle 1, 2 & 3	The EPU-LR reported legacy values of RT_{NDT} are estimated from hand drawn Charpy curves with limited data and a generic σ_u. The SLRA values reported generic US supplier forging data from a larger data set and are considered more representative.	Generic data from PROPRIETARY BAW-2313 R7 report, Section 2.2.3 and Table 2-5 for these USA supplied forgings.
Inlet & Outlet Nozzle Welds	The EPU-LR reported the RT_{NDT} and σ_u for 3 Linde 80 heats of welds. The values were reported for each heat and applied to all inlet and outlet welds. The EPU-LR reported the RT_{NDT} and σ_u values from BAW-2308 R2-A which requires an exemption to use this NRC approved report. FPL submitted an exemption to use BAW-2308 R2-A (Reference 3) specifically for the Unit 3 and Unit 4, upper to intermediate shell and intermediate to lower shell circumferential welds and received NRC approval (Reference 4). Since the exemption request did not specifically identify these locations, use of the BAW-2308 R2-A was not considered appropriate for the SLRA. In preparation for the SLRA, a more detailed fabrication records review was performed which identified the specific Linde 80 heats for each unit and the inlet or outlet nozzle. During the review, one (1) additional Linde 80 weld wire heat was identified (Unit 3 only) and not all the wire heats were used in Unit 3 and Unit 4. For Unit 3, the inlet nozzle welds used three (3) Linde 80 wire heats. For Unit 3, the outlet nozzle welds used just one (1) Linde 80 wire heat. The SLRA values reported are generic values for the Linde 80 welds.	Generic data from PROPRIETARY BAW-2313 R7 report, Section 2.2.3 and Table 2-6 for welds fabricated with Linde 80 flux.

Table A: Explanation of Differences in the Values of RT_{NDT} and σ_u Between the Turkey Point Unit 3 EPU License Report (EPU-LR) and Turkey Point Unit 3 Subsequent License Renewal Application (SLRA)

Reactor Vessel Material	Explanation of Differences in values	Data Source for $RT_{NDT(u)}$ and σ_u
Lower Shell to Transition Ring Circ Weld	This extended beltline (EB) material was not previously reported in the EPU-LR. The SLRA values reported generic values for this Linde 80 weld.	Generic data from PROPRIETARY BAW-2313 R7 report, Section 2.2.3 and Table 2-6 for welds fabricated with Linde 80 flux.

Table B: Explanation of Differences in the Values of RT_{NDT} and σ_u Between the Turkey Point Unit 4 EPU License Report (EPU-LR) and Turkey Point Unit 4 Subsequent License Renewal Application (SLRA)

Reactor Vessel Material	Explanation of Differences in values	Data Source for $RT_{NDT(u)}$ and σ_u
Lower Head Ring (Transition)	This extended beltline (EB) material was not previously reported in the EPU-LR. The SLRA values reported generic US supplier forging data from a larger data set and are considered more representative.	Generic forging data from PROPRIETARY BAW-2313 R7 report, Section 2.2.3 and Table 2-5 for these USA supplied forgings.
Inlet Nozzle 1, 2 & Outlet Nozzle 3	The EPU-LR reported values of RT_{NDT} are from NRC BTP 5-3 with a generic σ_u. The SLRA values reported generic US supplier forging data from a larger data set and are considered more representative.	Generic forging data from PROPRIETARY BAW-2313 R7 report, Section 2.2.3 and Table 2-5 for these USA supplied forgings.
Inlet Nozzle 3 & Outlet Nozzle 1, 2	The EPU-LR reported legacy values of RT_{NDT} are estimated from hand drawn Charpy curves with limited data and a generic σ_u. The SLRA values reported generic US supplier forging data from a larger data set and are considered more representative.	Generic data from PROPRIETARY BAW-2313 R7 report, Section 2.2.3 and Table 2-5 for these USA supplied forgings.

Table B: Explanation of Differences in the Values of RT_{NDT} and σ_u Between the Turkey Point Unit 4 EPU License Report (EPU-LR) and Turkey Point Unit 4 Subsequent License Renewal Application (SLRA)

Reactor Vessel Material	Explanation of Differences in values	Data Source for $RT_{NDT(u)}$ and σ_u
Inlet & Outlet Nozzle Welds	The EPU-LR reported the RT_{NDT} and σ_u for 3 Linde 80 heats of welds. The values were reported for each heat and applied to all inlet and outlet welds. The EPU-LR reported the RT_{NDT} and σ_u values from BAW-2308 R2-A which requires an exemption to use this NRC approved report. FPL submitted an exemption to use BAW-2308 R2-A (Reference 3) specifically for the Unit 3 and Unit 4, upper to intermediate shell and intermediate to lower shell circumferential welds and received NRC approval (Reference 4). Since the exemption request did not specifically identify these locations, use of the BAW-2308 R2-A was not considered appropriate for the SLRA. In preparation for the SLRA, a more detailed fabrication records review was performed which identified the specific Linde 80 heats for each unit and the inlet or outlet nozzle. During the review, one additional Linde 80 heat was identified (Unit 3 only) and not all the heats were used in Unit 3 and Unit 4 For Unit 4, the inlet nozzle welds used three (3) Linde 80 wire heats. For Unit 4, the outlet nozzle welds used two (2) Linde 80 wire heats. The SLRA values reported generic values for the Linde 80 welds.	Generic data from PROPRIETARY BAW-2313 R7 report, Section 2.2.3 and Table 2-6 for welds fabricated with Linde 80 flux.
Lower Shell to Transition Ring Circ Weld	This extended beltline (EB) material was not previously reported in the EPU-LR. The SLRA values reported generic values for the Linde 80 weld.	Generic data from PROPRIETARY BAW-2313 R7 report, Section 2.2.3 and Table 2-6 for welds fabricated with Linde 80 flux.

References:

1. Areva Report BAW-2313 Revision 7, "B&W Fabricated Reactor Vessel Materials and Surveillance Data Information," September 2016 (PROPRIETARY).
2. Areva Report BAW-2308 Revision 2-A, "Initial RT_{NDT} of Linde 80 Weld Materials," March 2008. (NRC Safety Evaluation, March 24, 2008, ML080770349).
3. FPL letter to NRC L-2009-023, "Update to NRC Reactor Vessel Integrity Database and Exemption Request for Alternate Material Properties Bases Per 10 CFR 50.12 and 10 CFR 50.60 (b)," dated March 18, 2009 (ML090920408).
4. NRC letter to FPL, "Turkey Point Units 3 and 4 - Exemption from the Requirements of 10 CFR Part 50, Appendix G and 10 CFR Part 50, Section 50.61 (TAC NOS. ME1007 and ME1008)," dated March 11, 2010 (ML100150599).

Associated SLRA Revisions:

None

Associated Enclosures:

None

NRC RAI Letter Nos. ML18269A209 and ML18269A210 dated October 1, 2018

3. Metal Fatigue of Class 1 Components, TLAA 4.3.1

Regulatory Basis:

Section 54.21(c)(1) of 10 CFR states a list of time-limited aging analyses, as defined in Section 54.3, must be provided. The applicant shall demonstrate that (i) The analyses remain valid for the period of extended operation; (ii) The analyses have been projected to the end of the period of extended operation; or (iii) The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

RAI 4.3.1-1

Background:

As discussed in SLRA Section 4.3.1, the applicant dispositioned the TLAA for ASME Boiler and Pressure Vessel Code, Section III, Class 1 fatigue calculations, in accordance with 10 CFR 54.21(c)(1)(i), that the analyses remain valid for the subsequent period of extended operation (SPEO). The applicant stated that the results demonstrate that the number of assumed design cycles will not be exceeded in 80 years of plant operation. The applicant also stated that it will monitor design cycles using the Fatigue Monitoring Program and assure that corrective action specified in the program is taken if any of the actual design cycles approach 80 percent of their analyzed numbers during the SPEO.

Issue:

The staff noted that the applicant is relying on its Fatigue Monitoring Program to monitor design cycles to ensure that these fatigue TLAAs remain valid for the SPEO in accordance with 10 CFR 54.21(c)(1)(i). Thus, the staff identified that the applicant's disposition for these TLAAs (i.e., 10 CFR 54.21(c)(1)(i)) is not applicable because the Fatigue Monitoring Program is managing fatigue by ensuring these analyses to continually remain valid and that the ASME Code design limit will not be exceeded during the SPEO.

Request:

- If the disposition for these fatigue TLAAs remains in accordance with 10 CFR 54.21(c)(1)(i) – the staff requests the following:
    - Based on the weighted projection methodology discussed in the SLRA and Calc 1700109.402P.R4, the staff noted that the weighted projection method was not applied to all transients to determine 80-year cycles.
        - Discuss the method used for these transients and justify that it is conservative for determining 80-year projected cycles and supports the disposition of 10 CFR 54.21(c)(1)(i).

- SLRA Table 4.3-2 and SLRA Table 4.3-3 – foot note 12 – "no additional design cycles expected"
    - Provide the basis that the "Hydrostatic pressure tests (pressurized to 1356 psig)" transient in the secondary Coolant system is no longer expected during the SPEO.
- Otherwise, if the Fatigue Monitoring Program is managing fatigue during the SPEO, justify that the disposition for these TLAAs in accordance with 10 CFR 54.21(c)(1)(i) is appropriate when compared to 10 CFR 54.21(c)(1)(iii).

FPL Response:

The PTN Fatigue Monitoring program is credited with managing fatigue of Class 1 components by ensuring that the number of occurrences and severity of each design transient remains within the limits of the component fatigue analyses during the SPEO. Therefore, PTN will manage the effects of aging due to metal fatigue in accordance with 10 CFR 54.21(c)(1)(iii). The PTN Fatigue Monitoring program provides for corrective actions when any applicable transient cycle count comes within 80 percent of the design or projected cycle limit.

Table 4.1-2 and Sections 4.3.1 and 17.3.3.1 of the PTN SLRA are revised as indicated below to address this RAI.

References:

None

Associated SLRA Revisions:

The SLRA is amended as indicated by the following text deletion (strikethrough) and text addition (red underlined font) revisions:

Revise SLRA Table 4.1-2 as follows:

Table 4.1-2
Summary of Results — Turkey Point TLAAs

METAL FATIGUE		4.3
Metal Fatigue of Class 1 Components	~~(i) remains valid for the SPEO~~ - **(iii) the effects of aging on the intended function will be adequately managed for the SPEO**	4.3.1

Revise SLRA Section 4.3.1 as follows:

TLAA Disposition: 10 CFR 54.21(c)(1)(iii)

~~The ASME Boiler and Pressure Vessel Code, Section III, Class 1 fatigue calculations remain valid for the SPEO. The results demonstrate that the number of assumed design cycles will not be exceeded in 80 years of plant operation. PTN will monitor design cycles using the Fatigue Monitoring AMP described in Section B.2.2.1 and assure that corrective action specified in the program is taken if any of the actual design cycles approach 80 percent of their analyzed numbers during the SPEO.~~

Metal fatigue of the PTN Class 1 reactor vessels, reactor vessel internals, pressurizers, steam generators, reactor coolant pumps and pressurizer surge lines will be managed using the Fatigue Monitoring AMP (Section B.2.2.1). The AMP provides for corrective actions when any applicable transient cycle count comes within 80 percent of the design or projected cycle limit.

Revise the second paragraph of SLRA Section 17.3.3.1 as follows:

The ASME Code Section III, Class 1, allowable stress calculations remain valid for the SPEO. The results demonstrate that the number of assumed thermal cycles will not be exceeded in 80 years of plant operation. PTN will monitor transient cycles using the PTN Fatigue Monitoring AMP and assure that the corrective action specified in the program is taken if any **applicable transient cycle count comes within 80 percent of the design or projected cycle limit.**~~of the actual cycles approach their analyzed numbers. However, not all the components can pass the environmentally assisted fatigue analysis using the original number of design transients, and the containment liner has taken credit for restricting the number of RCS heatup and cooldown events.~~ Therefore, this TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

Associated Enclosures:

None

NRC RAI Letter Nos. ML18269A209 and ML18269A210 dated October 1, 2018

4. Metal Fatigue of Non-Class 1 Components, TLAA 4.3.2

Regulatory Basis:

Section 54.21(c)(1) of 10 CFR states a list of time-limited aging analyses, as defined in § 54.3, must be provided. The applicant shall demonstrate that (i) The analyses remain valid for the period of extended operation; (ii) The analyses have been projected to the end of the period of extended operation; or (iii) The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

RAI 4.3.2-1

Background:

SLRA Section 4.3.2 states a review of the ANSI B31.1 piping within the scope of SLR was performed to identify those systems that operate at elevated temperature and to establish a conservative number of projected cycles based on 80 years of operation. The applicant cited EPRI Report TR-104534, "Fatigue Material Handbook" Volume 2, Section 4, and indicated that carbon steel systems or portions of systems with operating temperatures less than 220°F and stainless-steel systems or portions of systems with operating temperatures less than 270°F may generally be excluded from such concerns, since room temperature represents a practical minimum exposure temperature for most plant systems.

SLRA Section 17.3.3.2 states that any system or portions of systems with operating temperatures less than 220°F *were conservatively excluded* from further consideration for fatigue. *[emphasis added]*

Issue:

Analyses that meet the definition of a TLAA in 10 CFR 54.3 are required to be identified in the SLRA in accordance with 10 CFR 54.21(c)(1). Based on SLRA Section 4.3.2 and specifically on Section 17.3.3.2, it appears that the applicant may have implemented the methodology described in EPRI Report TR-104534 to exclude systems or portions of systems from consideration in the SLRA as a TLAA.

Request:

- Confirm that this screening criteria was not used to exclude systems or components *designed for fatigue* from consideration as a TLAA in the SLRA.
    - If it was not used in this way, discuss how the screening criteria was used in the SLRA and explain how this is in accordance with 10 CFR 54.21(c)(1).
    - If it was used in this way, justify that exclusion of these systems and components from consideration in the SLRA is in accordance with 10 CFR 54.21(c)(1). Otherwise, identify the systems or portions of systems *designed for fatigue* that were excluded and evaluate them in accordance with 10 CFR 54.21(c)(1).

FPL Response:

- The non-class 1 fatigue TLAA calculations performed for the original PTN license renewal application and the subject SLRA used the methodology described in the EPRI Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools (Ref. 1). Appendix H of the EPRI report describes the non-Class 1 fatigue screening criteria. Figures 4-1 and 4-2 of Appendix H are fatigue screening flow charts for non-Class 1 components. The flow charts show that carbon steel and stainless steel components with a maximum fluid temperature ≤ 220°F and ≤ 270°F, respectively, are acceptable for the period of extended operation and no additional evaluation is required. PTN SLR mechanical systems with maximum fluid temperatures below these limits are not considered to be susceptible to metal fatigue and are not a TLAA. Therefore, these mechanical systems are not considered to be in the scope of 10 CFR 54.21(c)(1).

  All mechanical systems within the scope of SLR were screened for metal fatigue using this criterion and the screening results are included in the PTN SLR non-Class-1 metal fatigue report. Sections 4.3.2 and 17.3.3.2 of the PTN SLRA are revised as indicated below to address this RAI.

References:

1. EPRI Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools, Revision 4, January 2006 (ML12335A508)

Associated SLRA Revisions:

The SLRA is amended as indicated by the following text deletion (strikethrough) and text addition (red underlined font) revisions:

Revise the third and fourth paragraph of the "TLAA Evaluation" portion of SLRA Section 4.3.2 as follows:

From the EPRI Report TR-104534, "Fatigue Material Handbook" Volume 2, Section 4 (Reference 4.3.6.3) **and the EPRI Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools (Reference 4.3.6.32)**, piping and tubing systems subject to thermal fatigue due to temperature cycling are described as, "For initial screening, systems in which the fluid temperature can vary more than 150°F in austenitic steel components and more than 200°F in carbon and low alloy steel components are potentially of concern for fatigue due to thermal transients. Thus, carbon steel systems or portions of systems with operating temperatures less than 220°F and stainless steel systems or portions of systems with operating temperatures less than 270°F may generally be excluded from such concerns, since room temperature represents a practical minimum exposure temperature for most plant systems."

All non-Class 1 mechanical systems within the scope of the PTN SLRA were initially screened for the TLAA associated with metal fatigue. Appendix H of Reference 4.3.6.32 describes the non-Class 1 fatigue screening criteria used at PTN. Figures 4-1 and 4-2 of Appendix H are fatigue screening flow charts for non-Class 1 components. The flow charts show that carbon steel and stainless steel

components with a maximum fluid temperature of ≤ 220°F and ≤ 270°F, respectively, are acceptable for the period of extended operation and no additional evaluation is required. Therefore, PTN SLR mechanical systems with maximum fluid temperatures below these limits are not considered to be susceptible metal fatigue and are not a TLAA. However, ~~For PTN,~~ any **PTN non-Class 1 mechanical** system or portions of systems with operating temperatures above 220°F are conservatively evaluated for **metal** fatigue. The **non-Class 1** piping and tubing systems requiring evaluation for **the metal fatigue TLAA for** SLR are listed in Table 4.3.2-2 below.

Revise SLRA Section 4.3.6 to include the following additional reference:

4.3.6.32 EPRI Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools, Revision 4, January 2006 (ML12335A508)

Revise the fifth paragraph of Section 17.3.3.2 as follows:

Conservatively, based on this assessment, any system or portions of systems with operating temperatures less than 220°F were **determined to be acceptable for the subsequent period of operation and no additional evaluation is required.** ~~excluded from further consideration.~~ Once a system is established to operate at a temperature above 220°F, the next step is to determine the system operating characteristics. For example, it is determined when the system is in heatup and cooldown mode, such as during testing reactor trips, sampling or swapping trains. The operating characteristics of a pipe segment are established by reviewing system operations and conducting interviews with appropriate operations personnel. With these operating characteristics defined, a determination can be made regarding whether a system is expected to exceed 7,000 full temperature cycles in 80 years of operation. In order to exceed 7,000 cycles, a system would be required to heatup and cooldown approximately once every four days. Systems that may exceed 7,000 cycles in 80 years were evaluated further.

Associated Enclosures:

None

NRC RAI Letter Nos. ML18269A209 and ML18269A210 dated October 1, 2018

RAI 4.3.2-2

Background:

SLRA Section 4.3.2 states that for the systems that are subjected to elevated temperatures above the fatigue threshold, a calculation was performed to determine a conservative number of projected full temperature cycles for 80 years of plant operation for the piping, tubing and in-line components. SLRA Table 4.3.2-2 provides the 80-year projected number of full temperature cycles for each of the systems evaluated in SLRA Section 4.3.2.

Issue:

The staff noted that the applicant provided the total number of projected transients through the end of the SPEO but did not provide the details on how these projections were determined (e.g., which transients are represented by the projections and how the projections were determined). The applicant did not explain or provide the basis for how these 80-year projections are conservative for 80 years of plant operation and how it supports the disposition of this TLAA in accordance with 10 CFR 54.21(c)(1)(i).

Request:

- Explain how the number of projected full temperature cycles for 80 years of plant operation for the piping, tubing and in-line components identified in SLRA Table 4.3-2 were determined.
- Justify that these methods are conservative for demonstrating that the ASME Section III, Class 3 and ANSI B31.1 allowable stress calculations remain valid for the SPEO and that the number of assumed thermal cycles will not be exceeded in 80 years of plant operation in accordance with 10 CFR 54.21(c)(1)(i).

FPL Response:

- The methodology used to determine the number of projected full temperature cycles for 80 years of plant operation for the piping, tubing and in-line components identified in SLRA Table 4.3-2 is described below. As stated in Section 4.3.2 of the SLRA, the non-Class 1 piping systems identified in Table 4.3-2 would have to experience a full thermal cycle approximately once every 4 days for the 7000 thermal fatigue cycle limit to be reached during 80 years of operation.

  a) Reactor coolant system (RCS)

  The RCS piping and piping components that exceed the fatigue screening temperature of 220°F operate continuously during plant operation and would only typically be cycled during plant startup and shutdown for refueling outages and forced outages. As noted in SLRA Tables 4.3.2 and 4.3.3, the PTN RCS is designed for a maximum of 200 heatup and cooldown cycles (400 total thermal cycles), and therefore would not approach the 7000 thermal fatigue cycle limit.

Therefore, there is significant margin in the thermal fatigue design of the RCS piping and piping components for the SPEO.

b) Residual heat removal (RHR) system

The RHR system is normally in standby service and operates at elevated temperatures only in support of RCS heatup and cooldowns. Recognizing the RCS heatup and cooldown design cycles are limited to 200 cycles, the RHR system is limited to 400 cycles which provides significant margin to the 7000 thermal fatigue cycle limit. Therefore, the RHR system piping design for metal fatigue remains valid for the SPEO.

c) Chemical and volume control system (CVCS)

Letdown piping from the CVCS regenerative heat exchanger operates at temperatures in excess of the thermal fatigue screening temperature of 220°F. During normal operation, there is continuous letdown flow through the letdown piping to the demineralizers for normal RCS cleanup. For this reason, thermal cycles for this piping will be significantly less than the approximately once every four (4) days required to achieve 7000 thermal cycles in 80 years of operation.

Outlet piping for the CVCS regenerative heat exchanger and the shell side inlet of the heat exchanger are also subjected to operating temperatures above the 220°F screening temperature. On the shell side of the heat exchanger there is continuous flow with steady-state operating temperatures and thermal cycles would occur only during plant heatup and cooldown. In addition, a monthly flow path verification is performed by cycling CVCS system valves. This monthly cycle, in addition to the cycles during unit heatup and cooldown, would conservatively result in 80 x 12 + 400 (RCS cycles) or 1360 cycles over the 80-year SPEO. This is well below the 7000 thermal fatigue cycle limit.

Piping to the excess letdown heat exchanger is subjected to temperatures above the 220°F screening temperature during a unit heatup (200 cycles) when drawing a steam bubble in the pressurizer. This portion of CVCS piping is not subjected to cyclic operation. Therefore, the CVCS system piping design for thermal fatigue is conservative and remains valid for the SPEO.

d) Primary sampling (only tubing is exposed to the temperature cycles)

The primary sample path from the pressurizer steam space is only cycled during pressurizer heatup and cooldown. Since there are only 400 cycles allowed for heatup and cooldown of the RCS, it can be concluded that the pressurizer steam space sample piping has significant margin to the 7000 thermal fatigue cycle limit for the SPEO.

The primary sample path from the pressurizer liquid space consists of piping and tubing that is cycled on a weekly basis. Periodic sampling of the pressurizer liquid space is also performed during RCS heatup activities. The fatigue calculation for the original license renewal determined there are no more than 10 pressurizer liquid space samples (cycles) taken per heatup/cooldown. For SLR, this would represent a maximum of (200 x 10) plus

(52 x 80) or 6160 cycles over an 80-year period. This is well below the 14000 cycle screening criteria for tubing and below the 7000 thermal cycle limit for the short section of sample piping. Based on this assessment, the pressurizer liquid space sample path fatigue design remains valid for the SPEO.

The sample paths for the post accident sampling system (PASS) are utilized only when the RHR system is placed in service during plant heatup and cooldown. As discussed for the RHR system above, these sample paths will experience significantly less than 7000 thermal fatigue cycles during the 80-year SPEO. Therefore, the PASS and RHR sample path piping and tubing design remains valid for the SPEO.

The two sample paths from the RCS hot legs originate in the "A" and "B" hot leg piping. The sample path from the "A" RCS hot leg is considered an alternative path and is not used for periodic sampling. Therefore, this primary sample path will not experience 7000 thermal cycles during the 80-year SPEO.

As described in PTN SLRA Section 4.3.2 and RAI 4.3.2-3 below, the "B" hot leg RCS sample path is expected to reach the 14,000 thermal cycle limit for tubing during the current 60-year PEO. In accordance with Fatigue Monitoring program, this condition has been entered into the PTN corrective action program requiring resolution prior to the end of 2018.

e) Secondary sampling system

The section of secondary sampling system piping from the steam generators to the containment isolation valves is within the scope of license renewal and operates at temperatures above the metal fatigue screening temperature of 220°F. This system is normally in continuous service and would essentially only be cycled once every refueling or forced outage. There are additional cycles for unforeseen incidents (i.e. when steam generator blowdown is suspended); however, these are infrequent operations that will not approach the frequency of one thermal cycle every four (4) days required to approach the 7000 thermal cycle fatigue limit. Therefore, the secondary sampling system piping fatigue design remains valid for the SPEO.

f) Emergency diesel generator (EDG) air (diesel exhaust)

The exhaust system for the EDG diesel engines has been identified as operating at a temperature above the fatigue screening temperature of 220°F. Based on the evaluation performed for the 60-year PEO, the EDG engines are conservatively assumed to start 21 times each year. Therefore, for an 80-year life, the EDG engine exhaust system will experience a total of 1680 cycles. This represents considerable margin compared to the 7000 thermal fatigue cycle limit. Therefore, the EDG engine exhaust system design remains valid for the SPEO.

g) Main steam and turbine

The main steam and turbine system piping is typically in continuous operation while the PTN nuclear units are producing electrical power. Since the PTN nuclear units are baseloaded to the grid, the main steam and turbine system will be subjected to thermal cycles significantly less than the approximate once every four (4) days required to approach the 7000 thermal fatigue cycle limit. Therefore, the main steam and turbine system piping fatigue design remains valid for the SPEO.

h) Feedwater and blowdown

Similar to the main steam and turbine systems, the feedwater and blowdown system piping within the scope of SLR is in continuous operation while the PTN units are producing electrical power and will be subjected to thermal cycles significantly less than the approximate once every four (4) days required to approach the 7000 thermal fatigue cycle limit. Therefore, the feedwater and blowdown system piping fatigue design remains valid for the SPEO.

i) Auxiliary feedwater and condensate storage (steam supply)

Steam supply piping to the auxiliary feedwater pump turbines experience temperatures above the fatigue screening temperature of 220°F. The evaluation performed for this piping for the 60-year PEO determined that the piping would conservatively experience 4320 cycles over the 60-year period. Since this PEO evaluation was completed, PTN Technical Specification Section 4.7.1.2.1 for testing of the auxiliary feedwater pumps has increased the interval for pump operability testing to 92 days from the previous 31 day testing interval. For this reason, extrapolating the 31 day PEO pump thermal cycles of 4320 to 80 years would result in a conservative estimate of 5760 cycles over the 80-year SPEO. This conservative estimate of thermal cycles remains below the 7000 fatigue cycle limit. Therefore, the auxiliary feedwater system fatigue design remains valid for the SPEO.

j) Auxiliary steam

Piping for the auxiliary steam system within the subsequent license renewal boundary scope is confined to piping between the main steam system and the turbine gland seal system, and to the air ejectors. Piping to the turbine gland seal system and the steam jet air ejectors is in continuous operation while PTN Units 3 and 4 are producing electrical power. Piping to the priming and hogging air ejectors is only used during startup conditions. Similar to the main steam, turbine, feedwater and blowdown systems above, this piping would not approach the 7000 thermal cycle screening limit in 80 years of operation. Therefore, the auxiliary steam system fatigue design remains valid for the SPEO.

k) Condensate

Condensate system piping within the license renewal boundary is in continuous operation while PTN Units 3 and 4 are producing electrical power. Therefore, this piping would not experience thermal cycling approaching the 7000 fatigue cycle limit. Based on this assessment, the condensate system fatigue design remains valid for the SPEO.

l) Feedwater heater drains and vents

Feedwater heater drains and vents piping within the license renewal boundary is confined to the drain piping from FWH 6A/B to FWH 5A/B and portions of the piping from the reheater drain tanks to the condensers. This piping is in continuous operation while PTN Units 3 and 4 are producing electrical power. Therefore, this piping would not experience thermal cycling approaching the 7000 fatigue cycle Limit. Based on this assessment, the feedwater heater drains and vents System fatigue design remains valid for the SPEO.

- With the exception of the "B" hot leg RCS sample path, the allowable stress calculations for the PTN Unit 3 and 4 non-Class 1 piping systems described above are conservative for the SPEO and the number of assumed thermal cycles will not be exceeded in 80 years of plant operation in accordance with 10 CFR 54.21(c)(1)(i).

References:

None

Associated SLRA Revisions:

The SLRA is amended as indicated by the following text deletion (strikethrough) and text addition (red underlined font) revisions:

Revise SLRA Table 4.3.2-2 as follows:

Table 4.3.2-2
Projected Number of Full Temperature Cycles

System	Number of Projected Cycles
Chemical and volume control	~~200~~ **1360**

Associated Enclosures:

None

NRC RAI Letter Nos. ML18269A209 and ML18269A210 dated October 1, 2018

RAI 4.3.2-2 (-3)

Background:

SLRA Section 4.3.2 states that the design thermal cycle limit for the reactor coolant system B hot leg tubing (i.e., less than 14,000 full temperature cycles) could be reached at the end of 2018 based on current operation. In order to ensure that the tubing can continue to perform its function for the current PEO as well as the SPEO, the applicant identified that one of the four actions in the SLRA can be completed.

The applicant dispositioned the TLAA for the reactor coolant system B hot leg tubing in accordance with 10 CFR 54.21(c)(1)(i) and stated that the results demonstrate that the number of assumed thermal cycles will not be exceeded in 80-years of plant operation.

Issue:

As noted in the SLRA, based on the cycle accumulation for the reactor coolant system B hot leg tubing the applicant expects that the design thermal cycle limit can be reached at the end of 2018. Thus, the staff noted that the applicant has not demonstrated that the TLAA for the RCS B hot leg sample tubing will remain valid for the SPEO in accordance with 10 CFR 54.21(c)(1)(i) since the number of design cycles will be exceeded prior to the end of the SPEO.

Request:

- Considering that the design cycle limit is expected to be reached by the end of 2018, justify the disposition for the RCS B hot leg sample tubing TLAA in accordance with 10 CFR 54.21(c)(1)(i).
- Otherwise, disposition the TLAA for the RCS B hot leg sample tubing in accordance with 10 CFR 54.21(c)(1)(ii) or (iii) and demonstrate accordingly:
    - If 10 CFR 54.21(c)(1)(ii) is selected – Provide the details of the design code allowable stress range and stress range reduction factor. Furthermore, justify that the projected number of cycles through the SPEO will be less than the revised allowable number of equivalent full temperature cycles.
    - If 10 CFR 54.21(c)(1)(iii) is selected – Describe and justify the method that will be used to manage fatigue of the RCS B hot leg sample tubing during the SPEO.

FPL Response:

Per Section 4.3.2 of the PTN SLRA, the RCS B hot leg sample tubing could potentially reach its design thermal cycle limit at the end of 2018 based on current operation. As part of current license renewal, actions have been included in the PTN corrective action program and require resolution prior to the end of 2018. Thus, the TLAA for 60 years will be resolved under the current license.

The disposition of the metal fatigue TLAA for the PTN Unit 3 and 4 RCS B hot leg sample tubing for SLR will be in accordance with 10 CFR 54.21(c)(1)(iii) regardless of the ultimate disposition of the TLAA for 60 years. The Fatigue Monitoring program will be credited during the SPEO to manage the aging effects associated with cracking.

References:

None

Associated SLRA Revisions:

The SLRA is amended as indicated by the following text deletion (strikethrough) and text addition (red underlined font) revisions:

Revise SLRA Table 4.1-2 as follows:

METAL FATIGUE		4.3
Metal Fatigue of Piping Components	(i) remains valid for the SPEO **(iii) the effects of aging on the intended function will be adequately managed for the SPEO**	4.3.2

Revise the concluding TLAA disposition statement in SLRA Section 4.3.2 as follows:

TLAA Disposition: 10 CFR 54.21(c)(1)(i)

The ASME Section III, Class 3 and ANSI B31.1 allowable stress calculations remain valid for the SPEO. The results demonstrate that the number of assumed thermal cycles will not be exceeded in 80 years of plant operation. Note that ~~if~~ aging management ~~is selected for addressing fatigue~~ of the **RCS hot leg** primary sampling tubing~~, the~~ TLAA ~~disposition~~ will be **dispositioned in accordance with**~~modified to~~ 10 CFR 54.21(c)(1)(iii).

Revise the last paragraph of SLRA Section 17.3.3.2 as follows:

The ASME Code Section III, Class 3, and ANSI B31.1 allowable stress calculations remain valid for the SPEO. The results demonstrate that the number of assumed thermal cycles will not be exceeded in 80 years of plant operation. Therefore, this TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(i) **with the exception of the RCS hot leg primary sample tubing which will be dispositioned in accordance with 10 CFR 54.21(c)(1)(iii)**.

Associated Enclosures:

None

NRC RAI Letter Nos. ML18269A209 and ML18269A210 dated October 1, 2018

5. Environmentally-Assisted Fatigue, TLAA 4.3.3

Regulatory Basis:

Section 54.21(c)(1) of 10 CFR states a list of time-limited aging analyses, as defined in § 54.3, must be provided. The applicant shall demonstrate that (i) The analyses remain valid for the period of extended operation; (ii) The analyses have been projected to the end of the period of extended operation; or (iii) The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

RAI 4.3.3-1

Background:

During its audit, the staff reviewed the licensee's refined calculations for environmentally assisted fatigue (EAF) documented in Enclosure 4 and 5 of the SLRA. The staff noted that the licensee used the methodology in the Draft Report for Comment version of NUREG/CR-6909, Rev. 1, dated March 2014.

GALL-SLR AMP X.M1 and the SRP-SLR states, in part, that environmental effects on fatigue for these critical components may be evaluated using the guidance in Regulatory Guide (RG) 1.207, Revision 1. RG 1.207, Revision 1, which was issued in June 2018, recommends the use of NUREG/CR–6909, Revision 1 (May 2018).

Issue:

The staff noted that due to the timing of the SLRA, the applicant used the most up-to-date methodology available, which was documented in the Draft Report for Comment version of NUREG/CR-6909, Rev. 1, dated March 2014. However, the use of this draft report is not consistent with the recommendations in the GALL-SLR and SRP-SLR.

Request:

- Qualitatively, discuss and justify the impacts, if any, to the refined calculations for EAF due to the recent issuance of RG 1.207, Revision 1.
- Otherwise, justify that the use of the Draft Report for Comment version of NUREG/CR- 6909, Rev. 1, dated March 2014, is more conservative when compared to RG 1.207, Revision 1.

FPL Response:

- NUREG/CR-6909 Draft Revision 1 and final Revision 1 (endorsed by RG 1.207, Revision 1) differ only in the application of strain rate for wrought and cast stainless steel materials when calculating F_{en} values. Therefore, the impact to the refined calculations are dependent on the impact of the F_{en} value changes on CUF_{en} results.
- An assessment was performed to illustrate the range of conservatisms between the draft and final version of NUREG/CR-6909 Revision 1. The

strain rate parameter was revised to be equal to zero at a strain rate of 7%/sec, rather than at 10%/sec as in the draft version. The draft version of NUREG/CR-6909, Revision 1 is conservative by up to 9% for pressurized water reactors (PWRs) (see table below) as compared to the final version as temperature increases. The table presents a comparison of the F_{en} results for different combinations of temperature and strain rate computed by the draft and final versions of NUREG/CR-6909, Revision 1.

Therefore, the refined EAF evaluations performed using the draft version of NUREG/CR-6909, Revision 1 in the SLRA are conservative for stainless steel materials as compared to an evaluation using the final version of NUREG/CR-6909, Revision 1.

Demonstration of F_{en} Differences Between Draft and Final NUREG/CR-6909, Rev. 1

Temperature Effect	
T (°C)	Ratio of Final / Draft F_{en} Values
325	91%
300	92%
275	93%
250	94%
225	95%
200	96%
175	97%
150	98%
125	99%
100	100%

Strain Rate Effect at 325 °C	
Strain Rate (/sec)	Ratio of Final / Draft F_{en} Values
0.4	91%
0.04	91%
0.004	91%
0.0004	91%

Strain Rate Effect at 125 °C	
Strain Rate (/sec)	Ratio of Final / Draft F_{en} Values
0.4	99%
0.04	99%
0.004	99%
0.0004	99%

References:

1. NUREG/CR-6909 (ANL-12/60), Draft Revision 1, Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials Draft Report for Comment, March 2014.
2. NUREG/CR-6909, Revision 1, Effect of LWR Water Environments on the Fatigue Life of Reactor Materials Final Report, May 2018.
3. Regulatory Guide 1.207, Revision 1, Guidelines for Evaluating the Effects of Light-Water Reactor Coolant Environments in Fatigue Analyses of Metal Components, June 2018.

Associated SLRA Revisions:

The SLRA is amended as indicated by the following text addition (red underlined font) revisions:

Add reference 4.3.6.11 and add new references 4.6.3.32 and 4.6.3.33 to SLRA Section 4.3.6 as follows:

4.3.6.11 NUREG/CR-6909 (ANL-06/08), **Draft** Revision 1, Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials, March 2014.

4.3.6.32 NUREG/CR-6909, Revision 1, Effect of LWR Water Environments on the Fatigue Life of Reactor Materials Final Report, May 2018.

4.3.6.33 Regulatory Guide 1.207, Revision 1, Guidelines for Evaluating the Effects of Light-Water Reactor Coolant Environments in Fatigue Analyses of Metal Components, June 2018. Regulatory Guide 1.207, Revision 1

Associated Enclosures:

None

NRC RAI Letter Nos. ML18269A209 and ML18269A210 dated October 1, 2018

RAI 4.3.3-2

Background:

Section 3.4 of Report No. 1700109.401P.R5 indicates that 14 locations have a CUF_{en} value greater than 1.0 when using the ASME Code fatigue curves of record for each location. Of these 14 locations, the following were not addressed in SLRA 4.3.3: the SG Tube to Tubesheet weld, RPV Head Flange and S/G Primary Chamber, Tubesheet and Stub Barrel Complex.

Per the SLRA, the SG Tube to Tubesheet weld is no longer part of the reactor coolant pressure boundary since the applicant has a permanently approved H* alternate repair criteria for both the hot- and cold-leg side of the steam generator; thus, the staff noted this component would not be subject to further EAF assessment consistent with the SRP-SLR.

Issue:

The rationale for the inconsistency between Report No. 1700109.401P.R5 and the SLRA is not clear for the RPV Head Flange and S/G Primary Chamber, Tubesheet and Stub Barrel Complex.

Request:

- Explain and justify the inconsistency between SLRA Section 4.3.3 and Report No. 1700109.401P.R5 for the RPV Head Flange and S/G Primary Chamber, Tubesheet and Stub Barrel Complex.

FPL Response:

There was an error in Structural Integrity Associates (SI) Report No. 1700109.401P.R5. Accordingly, the report has been revised (Reference 1) to be consistent with information in SLRA section 4.3.3 and the text that discusses locations where CUF_{en} screening was greater than 1.

Note that Table 3-4 of Structural Integrity Associates (SI) Report No. 1700109.401P.R7 (Reference 1 and Enclosure 3) includes revised 80-year CUF_{en} values due to the use of the methodology described in NUREG-6909, Revision 1 (Reference 2). As indicated in RAI 4.3.3-1 above, FPL originally used the most up-to-date methodology available in the PTN SLRA to calculate 80-year CUF_{en} values (Reference 3).

References:

1. Structural Integrity Associates Engineering Report No. 1700109.401P, Revision 7 – REDACTED, "Evaluation of Environmentally-Assisted Fatigue for Turkey Point Units 3 and 4 for Subsequent License Renewal", October 2018.
2. NUREG/CR-6909, Revision 1, Effect of LWR Water Environments on the Fatigue Life of Reactor Materials Final Report, May 2018.
3. NUREG/CR-6909 (ANL-12/60), Draft Revision 1, Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials Draft Report for Comment, March 2014.

Associated SLRA Revisions:

The SLRA is amended as indicated by the following text deletion (strikethrough text) and text addition (red underlined font) revisions:

Revise reference 4.3.6.8 of SLRA Section 4.3.6 as follows:

4.3.6.8 Structural Integrity Associates Engineering Report No. 1700109.401 P/NP, Revision ~~5~~7, "Evaluation of Environmentally-Assisted Fatigue for Turkey Point Units 3 and 4 for Subsequent License Renewal", ~~April~~ **October** 2018 (Enclosure 4, Attachment 5).

Associated Enclosures:

Enclosure 1. Westinghouse Letter CAW-18-4804 dated September 18, 2018, Application for Withholding Proprietary Information from Public Disclosure

Enclosure 2. Framatome Application for Withholding Proprietary Information from Public Disclosure dated October 12, 2018

Enclosure 3. Structural Integrity Associates Engineering Report No. 1700109.401P, Revision 7 – REDACTED, "Evaluation of Environmentally-Assisted Fatigue for Turkey Point Units 3 and 4 for Subsequent License Renewal," October 2018

NRC RAI Letter Nos. ML18269A209 and ML18269A210 dated October 1, 2018

RAI 4.3.3-3

Background:

SLRA Section 4.3.3 states the following with regard to the refined CUF_{en} calculations:

- Reactor Vessel Shell at Core Support Pads – A revised CUF_{en} was calculated by crediting 80-year projected design cycles for the hydrostatic test at 2485 psig pressure and 400°F temperature. CUF_{en} Final = 0.910
- Pressurizer Upper Head – A revised CUFen was calculated by crediting 80-year projected design cycles for plant loading, unloading, and boron concentration equalization transients. CUF_{en} Final = 0.974

The applicant dispositioned these refined CUF_{en} calculations in accordance with 10 CFR 54.21(c)(1)(iii), such that the effects of aging on the intended function(s) of these components will be adequately managed for the period of extended operation.

Issue:

SLRA Section 4.3.3 and Report No. LTR-SDA-II-17-13-P, Revision 2, indicates that the CUF_{en} Final is applicable for both Units 3 and 4, and that 80-year projected cycles were used for certain transients.

Based on SLRA Table 4.3-2 and 4.3-3, the 80-year projected cycles are different between Units 3 and 4; thus, it's not clear which 80-year cycles were used or whether the revised CUF_{en} results are applicable to both units. This information is necessary to ensure that the Fatigue Monitoring Program incorporates the appropriate cycle limits and can adequately manage environmentally assisted fatigue during the SPEO.

Request:

- Confirm that the 80-year projected cycles used for the transients in the refined CUF_{en} analyses for the Reactor Vessel Shell at Core Support Pads and, the Pressurizer Upper Head is the larger number of cycles between the two units. If not, justify that the CUF_{en} value is applicable to Units 3 and 4.

FPL Response:

Where 80-year projected cycles were used for the transients in the refined CUF_{en} analyses of the reactor vessel shell at the core support pads and the pressurizer upper head, the larger number of cycles between the two units was used. Text has been added to Revision 4 of LTR-SDA-II-17-13-P (Enclosure) for clarification.

References:

None

Associated SLRA Revisions:

The SLRA is amended as indicated by the following text deletion (strikethrough text) and text addition (red underlined font) revisions:

Revise reference 4.3.6.19 of SLRA Section 4.3.6 as follows:

4.3.6.19 Westinghouse LTR-SDA-II-17-13-P/NP, ~~Revision 2,~~ Environmentally Assisted Fatigue Evaluation of the Turkey Point Unit 3 and Unit 4 Pressurizer Upper Head and Shell and Reactor Vessel Core Support Blocks, **Revision 2 dated** November 30, 2017**, and Revision 4 dated September 2018** (Enclosure 4, Attachment 7).

Associated Enclosures:

(Refer to) Attachment 13 Enclosure 1. Westinghouse Letter CAW-4804 dated September 18, 2018, Application for Withholding Proprietary Information from Public Disclosure

Enclosure. Westinghouse Letter, LTR-SDA-II-17-13-NP (Nonproprietary), "Environmentally Assisted Fatigue Evaluation of the Turkey Point Unit 3 and Unit 4 Pressurizer Upper Head and Shell and Reactor Vessel Core Support Blocks," Revision 4 final approved September 18, 2018

NRC RAI Letter Nos. ML18269A209 and ML18269A210 dated October 1, 2018

RAI 4.3.3-4

Background:

SLRA Section 4.3.3 states that for the Pressurizer Spray Nozzle, a revised CUF_{en} was calculated by performing a finite element fatigue calculation using the methodology of Subarticle NB-3200 of Section III of the ASME Code and projected design cycles for plant heatup and cooldown. The applicant dispositioned this refined CUFen calculation in accordance with 10 CFR 54.21(c)(1)(iii), such that the effects of aging on the intended function(s) of these components will be adequately managed for the period of extended operation.

Issue:

During its review of Calculation 1700804.315P it appears that the 80-year projected number of cycles for the inadvertent auxiliary spray was incorporated into the refined CUF_{en} calculation for the pressurizer spray nozzle. This information is necessary to ensure that the Fatigue Monitoring Program incorporates the appropriate cycle limits and can adequately manage environmentally assisted fatigue during the SPEO.

Request:

- Clarify the discrepancy between Calculation 1700804.315P and the SLRA. Identify the transients that used 80-year projected number of cycles in the refined environmentally assisted fatigue evaluation for the Pressurizer Spray Nozzle and confirm that the Fatigue Monitoring Program manages the appropriate number of cycles to ensure the analysis remains valid for the SPEO.

FPL Response:

- SLRA Section 4.3.3 indicates that a revised CUF_{en} was calculated for the pressurizer spray nozzle using projected design cycles for plant heatup and cooldown. The SLRA inadvertently omitted that the other transient that used 80-year projected cycles in calculation 1700804.315P is the inadvertent auxiliary spray (IAS) cycle.

  One IAS cycle was used consistent with the projected number of IAS cycles from SLRA Table 4.3-2 and 4.3-3. In addition, calculation 1700804.315P Revision 2 evaluated the impact of additional IAS cycles as a buffer in case more than 1 was experienced. Up to four additional IAS cycles can be experienced without exceeding a CUF_{en} of 1.0.

  As discussed in SLRA Section B.2.2.1, the FPL Fatigue Monitoring Program monitors accumulated cycles against projected values in SLRA Table 4.3-2 and Table 4.3-3 to ensure component fatigue usage, including environmental effects, remains within allowable values.

References:

None

Associated SLRA Revisions:

(Refer to) FPL letter L-2018-166, dated October 16, 2018, Attachment 1 Associated SLRA Revisions in the response to NRC RAI No. RAI B.2.2.1-1. That revision added the inadvertent auxiliary spray cycle for the pressurizer spray nozzle to SLRA Section 4.3.3 on page 4.3-23.

Associated Enclosures:

None

NRC RAI Letter Nos. ML18269A209 and ML18269A210 dated October 1, 2018

RAI 4.3.3-5

Background:

SLRA Section 4.3.3 states the following as it relates to the CRDM Lower Joint:

- A revised CUF_{en} was calculated by performing a more refined analysis and crediting 80- year projected design cycles for plant heat up, 10% step load increases, 50% step load decreases, loss of load, loss of AC power, and reactor trips (Unit 4 only). CUF_{en} Final = 0.749

Issue:

During its review of Areva Calculation No. 32-9280202, Revision 1, TP CRDM Lower Joint Environmentally Assisted Fatigue, December 15, 2017, the staff noted that a unique CUF_{en} value for both components was calculated for Units 3 and 4, which may indicate a possible variation in, but not limited to, component design or geometry, and assumed transients and number of cycles used in the calculation. However, SLRA Section 4.3.3 only indicates one refined CUF_{en} value for the component.

In order to determine if the Fatigue Monitoring Program will adequately manage environmentally assisted fatigue for the CRDM Lower Joint it is necessary to understand whether the calculation for one unit bounds the other or whether the assumptions for both Units 3 and 4 will be incorporated into the Fatigue Monitoring Program.

In addition, as noted in the SLRA and Areva Calculation No. 32-9280202, Revision 1, the transient "Rod Trips" is only applicable to the refined CUF_{en} calculation for Unit 4; however, the basis for this was not clear.

Request:

- Clarify if a bounding CUF_{en} represents the CRDM Lower Joint for Units 3 and 4.
  - If so, justify that the CUF_{en} value selected is applicable and appropriate for both units. Aspects such as, but not limited to transient selection, assumed number of cycles, design loading, material fabrication and geometry, should be addressed, if applicable.
  - If not, confirm that the Fatigue Monitoring Program incorporates the appropriate transient cycle limits used in the respective calculation for Units 3 and 4.
- Explain and justify why the refined CUF_{en} calculation for Unit 3 does not incorporate the transient, "Rod Trips."

FPL Response:

- Revision 003 of the CRDM Lower Joint calculation (32-9280202) (Reference 3) adds the calculation of a bounding CUF_{en} that is applicable to both units. A comparison of the bounding CUF_{en} table with the CUF_{en} tables of the individual units shows that each of the individual CUF_{en} tables are subsets of the bounding CUF_{en} table. All transient pairs and cycles from both individual unit CUF_{en} calculations are included in the bounding CUF_{en} calculation. A comparison of the CUF_{en} values shows that the CUF_{en} for Unit 3 (0.415) and the CUF_{en} for Unit 4 (0.539) are both less than the bounding CUF_{en} of 0.597.

  The only difference between Unit 3 and Unit 4, with respect to the CUF_{en} calculation, is the number of [ ] cycles, so a discussion of design loading, material fabrication and geometry differences is not applicable.

  - The bounding CUF_{en} does represent the CRDM Lower Joint for Units 3 and 4.

- The same 16 transients are considered in the original in-air fatigue analysis for this location (CRDM Lower Joint) for both units. The in-air fatigue analysis shows that the only transient pairs that result in partial fatigue usage are those that include Transient 1 [ ] in the pair. The fatigue usage at this location is dictated by the number of Transient 1 cycles: when all Transient 1 cycles have been considered (in ranges with the other transients), all remaining transient pairs result in an alternating stress that is below the endurance limit and therefore they do not contribute to the cumulative usage.

  For Unit 3, there are [ ] cycles of Transient 1 [ ]. As Table 6-7 in 32-9280202-001 (Reference 1) indicated, these [ ] cycles of Transient 1 are consumed when paired with Transient 10 [ ] Transient 11 [ ] and Transient 7 [ ] listed in order of decreasing alternating stress. Since at this point all Transient 1 cycles have been consumed, other transient pairs (with Transient 5, Transient 16, or other transients) result in no additional fatigue usage and therefore do not need to be considered.

  For Unit 4, there are [ ] cycles of Transient 1 [ ]. As Table 6-8 in 32-9280202-001 (Reference 1) indicated, all but [ ] of the [ ] cycles are consumed when Transient 1 is paired with Transient 10 [,] Transient 11 [ ] and Transient 7 [ ] listed in order of decreasing alternating stress. 32-9280202-001 (Reference 1) indicated that the remaining [ ] cycles of Transient 1 are consumed by Transient 5 [ ] and Transient 16 [ ]. In researching this RAI, Framatome discovered that there are [ ] cycles, not just [ ] of Transient 5 available for combination with the remaining cycles of Transient 1. Revision 002 of the calculation (Reference 2) corrected the transient combination for the remaining [ ] cycles of Transient 1 (all [ ] remaining cycles are credited to the Transient 1 – Transient 5 pair), and therefore the Unit 4 CUF_{en} calculation no longer relies on the Transient 1 – Transient 16 pair, meaning that Transient 16 [ ] is no longer considered in the Unit 4 CUF_{en} calculation. This revision resulted in a CUF_{en} of 0.539 for Unit 4.

With the correction to the CUF_{en} calculation for Unit 4 it can be seen that the transient 16 [ ] is not applicable to the CUF_{en} calculation for either unit.

References:

1. Framatome Calculation No. 32-9280712, Rev. 002, TP CRDM Lower Joint Environmentally Assisted Fatigue, October 12, 2018.

Associated SLRA Revisions:

The SLRA is amended as indicated by the following text deletion (strikethrough) and text addition (red underlined font) revisions:

Revise SLRA Section 4.3.3, page 4.3-23 for the analysis summary of the CRDM lower joint as follows:

A revised CUF_{en} was calculated by performing a more refined analysis and crediting 80-year projected design cycles **as presented in Reference 4.3.6.26.** ~~for plant heatup, 10% step load increases, 50% step load decreases, loss of load, loss of AC power, and reactor trips (Unit 4 only).~~

CUF_{en} Final = 0.**597**

Associated Enclosures:

Enclosure 1. Framatome Application for Withholding Proprietary Information from Public Disclosure dated October 12, 2018

Enclosure 2. Framatome Application for Withholding Proprietary Information from Public Disclosure dated October 17, 2018

Enclosure 3. Framatome Calculation No. 32-9280712, Rev. 002, TP CRDM Lower Joint Environmentally Assisted Fatigue dated October 12, 2018 (Non-Proprietary)

NRC RAI Letter Nos. ML18269A209 and ML18269A210 dated October 1, 2018

RAI 4.7.4-4

Issue:

During the staff audit of the applicant's documents, the staff noticed that applicant's document AR 01610224 is related to errors in pipe stress software.

Request:

Discuss whether errors in pipe stress software as discussed in AR 01610224 affected the loading calculations used in the LBB analysis of Class 1 auxiliary piping in SIA Report No. 0901350.401, Revision 3 and SIA Report No. 0901350.304, Revision 2.

FPL Response:

The software error identified in AR 01610224 is related to the thermal stratification calculation module in PIPESTRSS. For the loading calculation used in the LBB analysis of Class 1 auxiliary piping in SIA Report No. 0901350.401, Revision 3 and SIA Report No. 0901350.304, Revision 2, the PIPESTRESS software use was limited to the evaluation of design loads such as deadweight + pressure, thermal expansion, seismic inertia, transient loading and wind loading. Therefore, the AR 01610224 software error notice is not applicable to PTN 3 & 4 LBB piping evaluations of the Class 1 auxiliary piping.

References:

None

Associated SLRA Revisions:

None

Associated Enclosures:

None

NRC RAI Letter Nos. ML18269A209 and ML18269A210 dated October 1, 2018

RAI 4.7.4-5

Issue:

The following issues are related to the LBB analysis in SIA Report No. 0901350.401, Revision 3.

(a) Section 3 of SIA Report states that the subject piping has no active degradation mechanisms such as water hammer, corrosion, and high cycle fatigue. However, the applicant did not provide inspection history of the subject piping.

(b) Page v of the SIA report states that "The LBB evaluation was performed in accordance with the 10 CFR 50, Appendix A GDC-4 and NUREG-1061, Vol. 3 [6] as supplemented by NUREG-0800, Standard Review Plan 3.6.3 [7]..." The NRC staff notes that SRP 3.6.3 has been revised and the latest edition is March 2007.

(c) Page viii of the LBB analysis states that "...Limit load analysis as outlined in NUREG-0800, SRP 3.6.3, was utilized in this evaluation in order to determine the critical flaw sizes since the materials involved in this evaluation are stainless steel piping..." Page 1-3 of the SIA report stated that "Critical flaw size evaluation, based on elastic-plastic fracture mechanics techniques, is used to determine the length and depth of defects that would be predicted to cause pipe rupture under specific design basis loading conditions, including abnormal conditions such as a seismic event and including appropriate safety margins for each loading condition..." Section 5-1, Page 5-1, of the SIA report stated that the limit load method was used to calculate the critical crack sizes. It is not clear why the limit load method is discussed on pages viii and 5-1 but page 1-3 discusses the elastic-plastic fracture mechanics method.

(d) Figures 5-1 to 5-6 of the LBB analysis present leakage flaw size versus normal operating stress. However, the leakage flaw size on the Y axis and the normal operating stress values on the X axis are not identified or marked in the figures.

Request:

(a) Describe the inspection history of the subject piping including results from previously performed inspections and inspection frequency since the commercial operation.

(b) Discuss whether the latest SRP 3.6.3 dated March 2007 was used in the LBB analysis.

(c) Clarify why pages viii and 5-1 stated that critical crack size was calculated based on the limited load method whereas page 1-3 stated that critical flaw size was calculated based on the elastic-plastic fracture mechanics analysis.

(d) Provide values of leakage flaw size and normal operating stress in Figures 5-1 to 5-6. Identify the limiting leakage flaw size for each subject piping in Figures 5-1 to 5-6.

FPL Response:

(a) Appendix B, paragraph B.2.3.1 of the SLR application provides details of the inspection history of the subject piping since commercial operation. The history includes ASME Code, Section XI in-service inspections results to the requirements of subsections IWB, IWC and IWD of the Code as well as industry and site specific operating experience, QA audits and NRC inspections. It is concluded that the inspection history provides reasonable assurance that the subject piping has no active degradation mechanisms such as water hammer, corrosion, and high cycle fatigue, and provides a representative inspection history of the subject piping.

(b) The latest SRP 3.6.3 dated March 2007 was used in the LBB analysis. Reference 7 in the SIA Report No. 0901350.401, Revision 4 has been revised.

(c) The elastic-plastic fracture mechanics analysis (involving J-integral/tearing modulus analysis) was not used in current analysis as stated in the last sentence on page 1-3.

(d) The values of leakage flaw size and normal operating stress have been added to the figures and are provided below. These figures have been used to revise Figures 5-1 to 5-6 in the SIA Report No. 0901350.401, Revision 4.

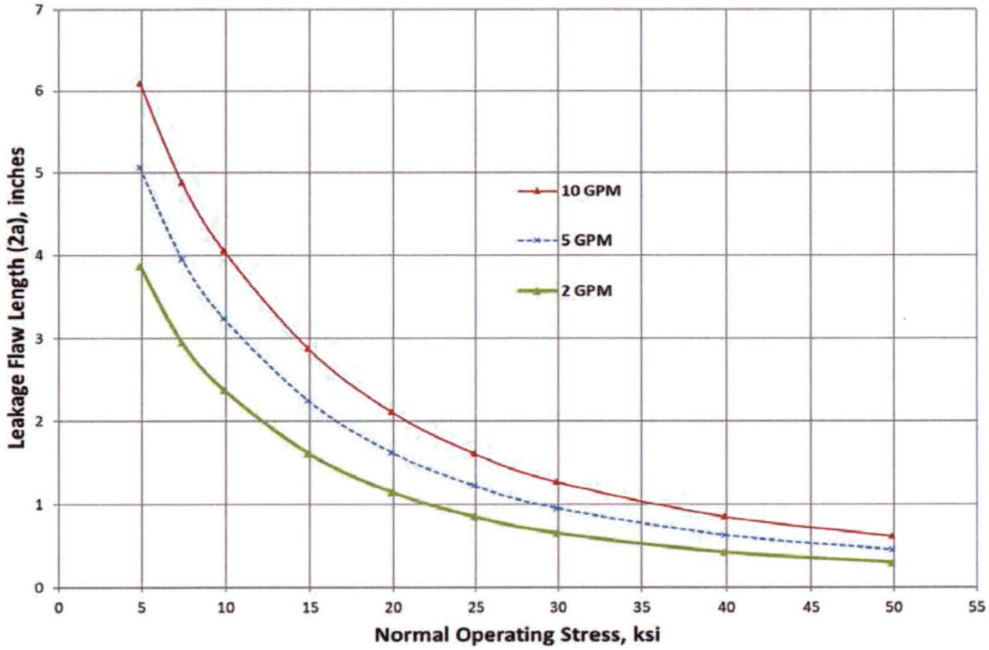


Figure 5-1. Leakage Flaw Size versus Normal Operating Stress of Accumulator Lines

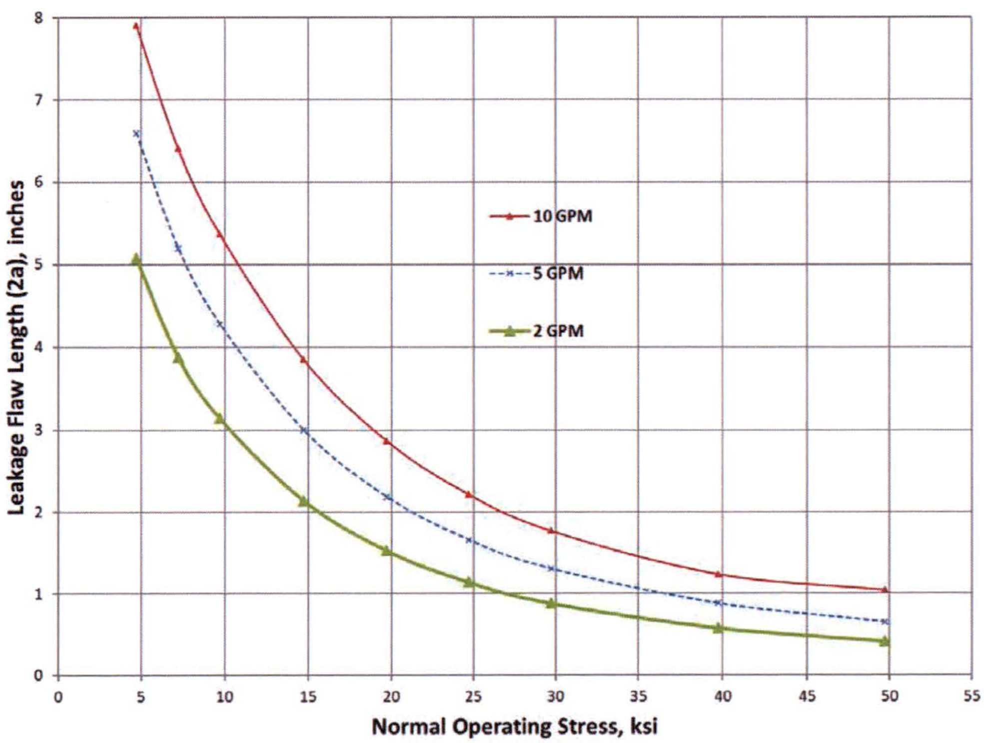


Figure 5-2. Leakage Flaw Size versus Normal Operating Stress of Pressurizer Surge Lines (Pipe Side at Pressurizer End)

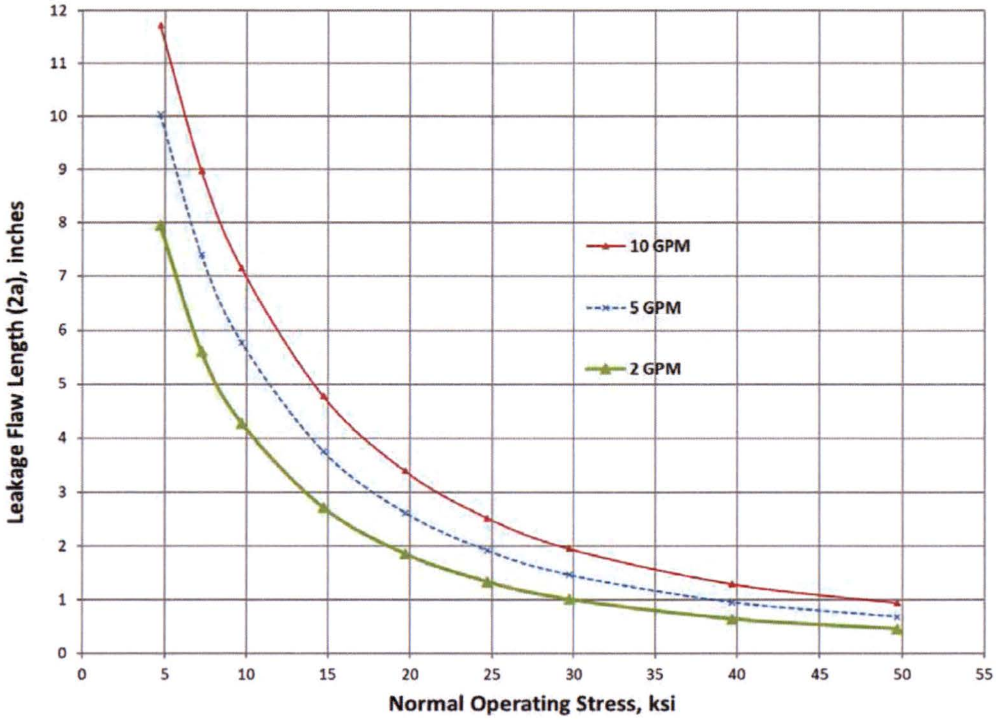


Figure 5-3. Leakage Flaw Size versus Normal Operating Stress of Pressurizer Surge Lines (Nozzle Side at Pressurizer End)

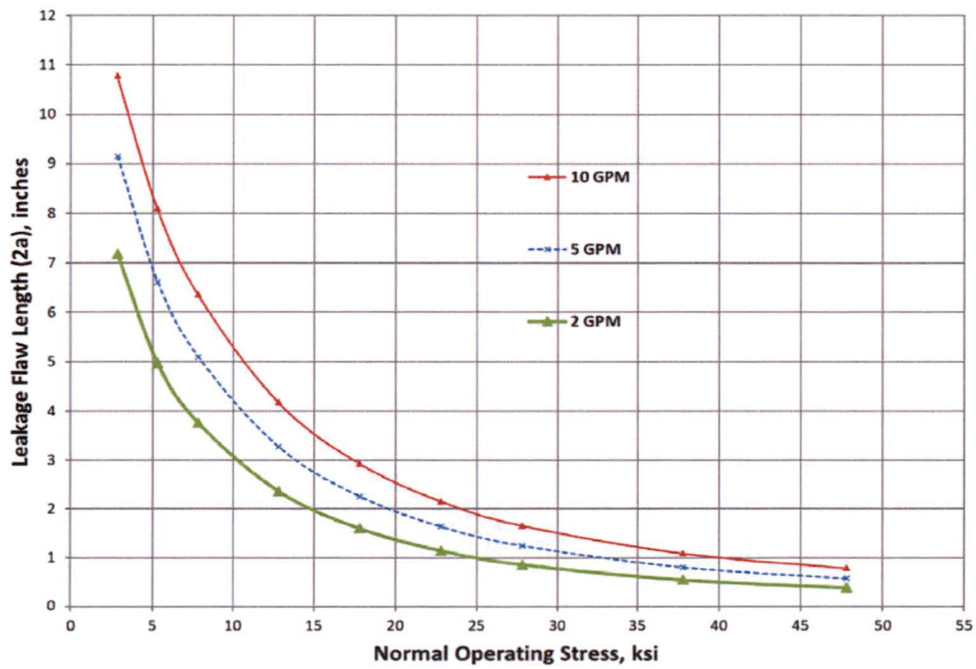


Figure 5-4. Leakage Flaw Size versus Normal Operating Stress of Pressurizer Surge Line (Nozzle Side at Hot Leg End)

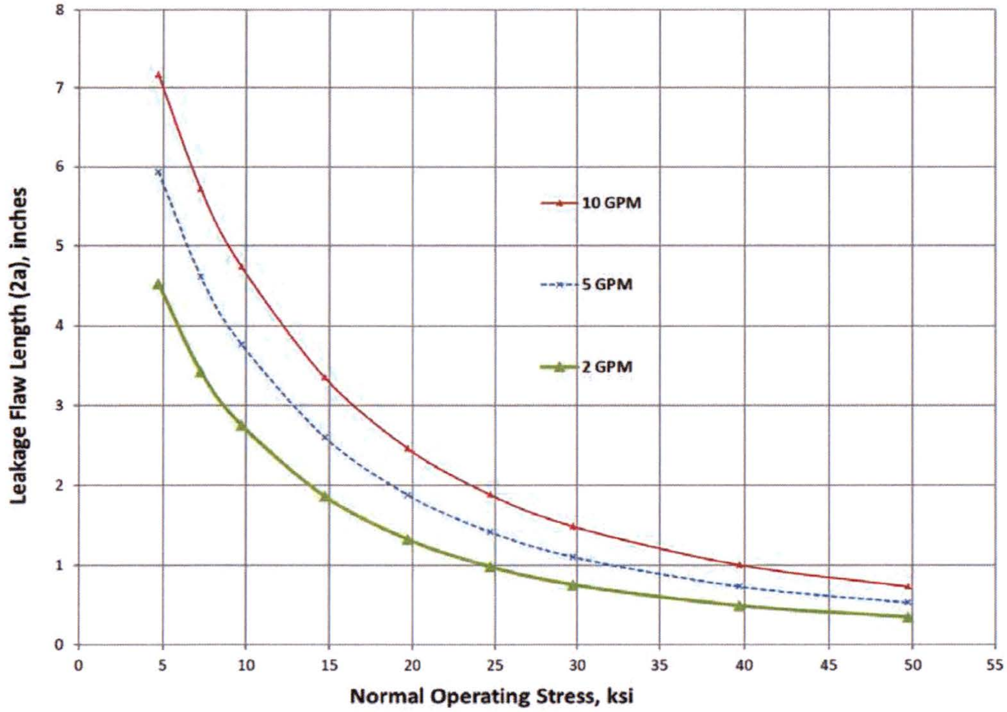


Figure 5-5. Leakage Flaw Size versus Normal Operating Stress of Pressurizer Surge Line (Pipe Side at Hot Leg End)

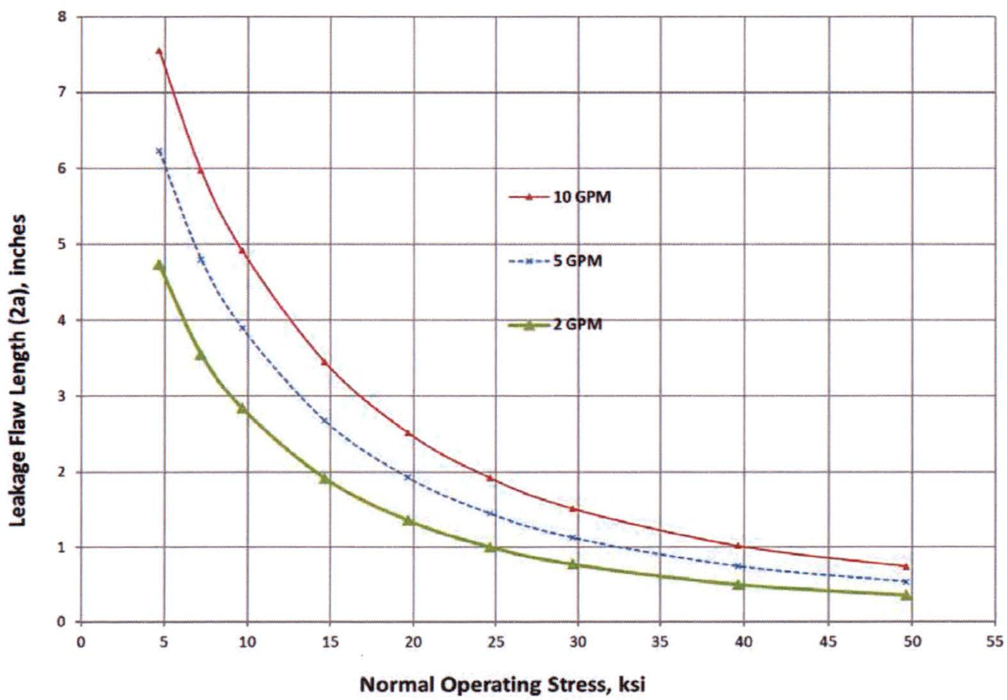


Figure 5-6. Leakage Flaw Size versus Normal Operating Stress of RHR Line at Hot Leg End

References:

None

Associated SLRA Revisions:

(Refer to) Attachment 19 FPL Response to NRC RAI No. 4.7.4-3, Associated SLRA Revisions that are applicable to NRC RAI No. 4.7.4-5

Associated Enclosures:

(Refer to) Attachment 19 FPL Response to NRC RAI No. 4.7.4-3, Enclosure 2

NRC RAI Letter Nos. ML18269A209 and ML18269A210 dated October 1, 2018

RAI 4.7.4-6

Issue:

SRP 3.6.3 specifies that piping qualified for LBB be evaluated to determine whether degradation mechanisms of wall thinning, creep and cleavage exist. Section 3 of SIA Report No. 0901350.401, Revision 3, does not discuss these degradation mechanisms.

Request:

Discuss whether wall thinning, creep and cleavage have occurred or will occur in the accumulator lines, RHR lines and safety injection lines.

FPL Response:

Wall thinning

Wall thinning is not expected to occur in the piping systems under consideration since the piping of these systems is fabricated from stainless steel which is not susceptible to wall thinning. This was covered in SIA Report No. 0901350.401, Revision 3 under Section 3.2 as flow accelerated corrosion (FAC). In addition, wall thinning is not an aging effect requiring management for RCS piping as indicated in SLRA Table 3.1.2-1.

Creep

Creep is not expected to occur in the piping systems under consideration since these are stainless steel piping systems that operate at temperatures below 800°F.

Cleavage

Cleavage is not expected to occur in the piping systems under consideration since these are fabricated from stainless steel which is very ductile at the operating temperatures of these piping systems.

References:

None

Associated SLRA Revisions:

None

Associated Enclosures:

None

NRC RAI Letter Nos. ML18269A209 and ML18269A210 dated October 1, 2018

RAI 4.7.4-7

Issue:

The staff notes that EPRI topical report, "Materials Reliability Program: Assessment of Residual Heat Removal Mixing Tee Thermal Fatigue in PWR Plants (MRP-192, Revision 2, 1024994), August 2012," and "Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines (MRP-146, Revision 1, 1022564), June 2011" are related to thermal fatigue of safety-related piping.

Request:

Discuss whether the thermal fatigue issue in MRP-146 and MRP-192 affects the LBB analysis of Class 1 auxiliary piping. If yes, discuss whether the LBB analysis in SIA Report No. 0901350.401, Revision 3 and SIA Report No. 0901350.304, Revision 2, considered the thermal fatigue discussed in these two MRP reports. If the Class 1 auxiliary piping does experience thermal fatigue, discuss how the auxiliary piping satisfies the screening criteria of SRP 3.6.3 which prohibits LBB be applied to piping experiencing fatigue.

FPL Response:

Thermal fatigue issues identified in MRP-146 were evaluated for all RCS Class 1 auxiliary piping lines at PTN in SIA Calculation PTN-08Q-301, Revision 1 (Reference 1). The evaluation concluded that the surge line and the RHR line and the safety injection line (of which the accumulator lines are part of) screened out and did not warrant any further evaluation. Since the RHR line considered for the LBB analysis does not include mixing tee, MRP-192 is not applicable.

References:

1. Structural Integrity Associates Engineering Calculation PTN-08Q-301, Revision 1, "MRP-146 Assessment of Normally Stagnant Non-Isolable Branch Lines"

Associated SLRA Revisions:

None

Associated Enclosures:

None

NRC RAI Letter Nos. ML18269A209 and ML18269A210 dated October 1, 2018

RAI 4.7.4-8

Issue:

The reactor coolant system primary loop piping in SLRA Section 4.7.3 contains elbows that are made of cast austenitic stainless steel. Section 4 of SIA Report No. 0901350.401, Revision 3, does not mention any pipe components made of cast austenitic stainless steel in the accumulator, RHR and Surge piping.

Request:

Confirm that the accumulator, RHR and Surge piping does not use any components or fittings that are made of cast austenitic stainless steel.

FPL Response:

The accumulator, RHR and surge piping do not use any components or fittings that are made of cast austenitic stainless steel.

References:

None

Associated SLRA Revisions:

None

Associated Enclosures:

None

NRC RAI Letter Nos. ML18269A209 and ML18269A210 dated October 1, 2018

RAI 4.7.4-9

Issue:

Section 5.4 of SIA Report No. 0901350.401, Revision 3, states that "...From the BACs [bounding analysis curves] and load points plotted in Figure 5-7 to Figure 5-12 all the stress points for both Units 3 and 4 are below 10 gpm BACs except for stress point 210M in the Unit 4 Accumulator Line as shown in Figure 5-7. Since the stress point 210M is in the middle of an elbow, removing the conservatism in using the weld material Z factor for pipe/elbow materials, the BAC is plotted using a Z factor of 1.0 as shown in Figure 5-13. Using a Z factor of 1.0, stress point 210M is under the 10 GPM BAC..."

(a) It is not clear what conservatism was included in the original analysis. For example, what is the original Z factor used prior to use 1.0 for the Z factor? Page 5-3 discusses the use of Z factor for welds based on the shield metal arc welding process.

(b) It is not clear why the stress point 210M is located in the middle of an elbow. Figure 4-2 of SIA Report No. 0901350.401, Revision 3, identified only the points associated with the welds at the both ends of the elbow in Loop A of the accumulator line, not in the middle of the elbow.

Request:

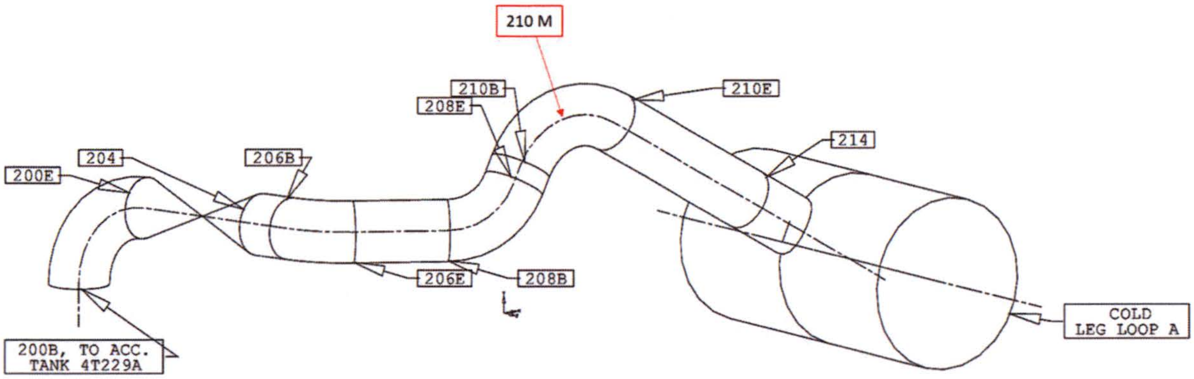
(a) Provide the original Z factor used for the 210M stress point in the Unit 4 accumulator pipe in the limit load analysis. Discuss the calculated leak rate using the original Z factor. Discuss the conservatism in the original analysis in terms of the Z factor.

(b) Clarify the exact location of the stress point 210M as modeled in the pipe stress analysis.

FPL Response:

(a) The original Z factor used for the 210M stress point in the Unit 4 accumulator pipe is 1.444 as provided in Table 4-6 of SIA Report No. 0901350.401, Revision 3. The Z factor is only used in calculating the critical flaw size. The leakage crack size is not affected by the Z factor. The original Z factor of 1.444 is for the weld material which provides conservative results when used for base material location (such as 210M stress point) where Z factor of 1.0 can be used.

(b) The figure below shows the location of the stress point 210M which is the mid-point in the elbow.



References:

None

Associated SLRA Revisions:

None

Associated Enclosures:

None

NRC RAI Letter Nos. ML18269A209 and ML18269A210 dated October 1, 2018

RAI 4.7.4-10

Issue:

Figures 5-9 to 5-12 of SIA Report No. 0901350.401, Revision 3, show BACs [bounding analysis curves] and Load Points for pressurizer surge lines with a single stress point whereas Figures 5-7 and 5-8 show many stress points for the accumulator and RHR lines, respectively. The staff notes that each of Figures 5-9 to 5-12 represents a specific location of the surge line.

Request:

Explain why pressurizer surge lines have only one stress data point in each of Figures 5-9 to 5-12 whereas for the accumulator and RHR piping multiple stress points are indicated in Figures 5-7 and 5-8, respectively.

FPL Response:

For the accumulator and RHR piping, stresses were available for all the nodal points and therefore, all these locations were considered in the BACs. However, for the surge line, only the bounding stress locations, which happened to be at the terminal ends, were available. These terminal end locations were therefore used for the LBB evaluation. This is deemed acceptable because as noted in subparagraphs III(11)(c)(ii) and III(11)(c)(iii) in SRP 3.6.3, for each pipe size in the piping system, the through-wall flaw can be postulated at the location that has the least favorable combination of stress and material properties for base metal, weldments, nozzles, and safe ends. For the surge line, as noted in Table 4-10 of SIA Report No. 0901350.401, Revision 3, node points corresponding to bounding locations in the 12" pipe and the 14" pipe at the nozzle end was selected for both the pressurizer end and the hot leg end. Since the BACs are a function of the pipe diameter and operating conditions, separate figures are need for the 12" pipe (Figures 5-11 and 5-12) and the 14" pipe at the nozzle end (Figures 5-9 and 5-10). For each of the configurations, there are two figures due to two different operating conditions.

References:

None

Associated SLRA Revisions:

None

Associated Enclosures:

None

NRC RAI Letter Nos. ML18269A209 and ML18269A210 dated October 1, 2018

RAI 4.7.4-11

Issue:

Section 5.3 of SIA Report No. 0901350.401, Revision 3, discusses bounding analysis curves which "...represent the maximum allowable membrane (pressure) plus bending stress (as determined from piping analysis for the system) as a function of the applied membrane (pressure) plus bending stress during normal plant operation. The latter condition represents the conditions during which leakage would have to be detected..."

Request:

Explain the objective of the bounding analysis curves. It is not clear how the bounding analysis curves demonstrate (a) the margins on the crack size and leakage detection in SRP 3.6.3 have been satisfied, and (b) the crack stability.

FPL Response:

The bounding analysis curves (BACs) provide the loci of normal operating (NOP) stresses and maximum (NOP+SSE) stresses that must be met to achieve the margins of 10 on leakage detection and 2 on leakage-to-critical crack sizes as required by SRP 3.6.3. Points on or below the BAC curve meet the stability margin for a particular leakage detection capability while points above the BAC curve do not meet the stability margin. Thus, the BACs are simply a pictorial representation demonstrating whether the LBB margins have been met or not. The procedure described in Section 5.3 of the report provides details of how the BACs are determined including the margins.

References:

None

Associated SLRA Revisions:

None

Associated Enclosures:

None

NRC RAI Letter Nos. ML18269A209 and ML18269A210 dated October 1, 2018

RAI 4.7.4-12

Issue:

Page 5-5 of SIA Report No. 0901350.401, Revision 3, states that "...The maximum allowable bending stress is determined from the curve of critical crack size (a) versus applied bending moment such that acritical = 2aleakage..." It is not evident that Section 5 of SIA Report No. 0901350.401, Revision 3, provided the curves of critical crack sizes.

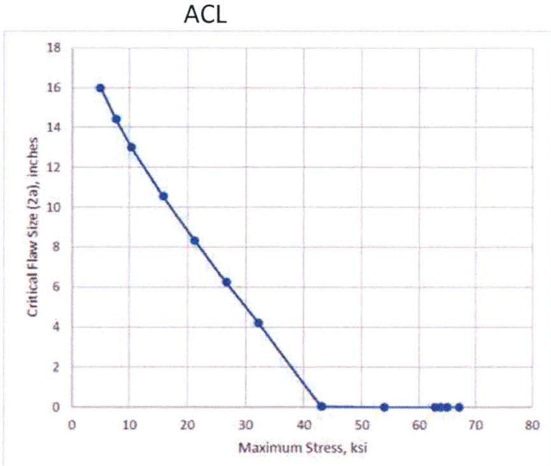
Request:

Provide curves of critical crack sizes for each subject piping.

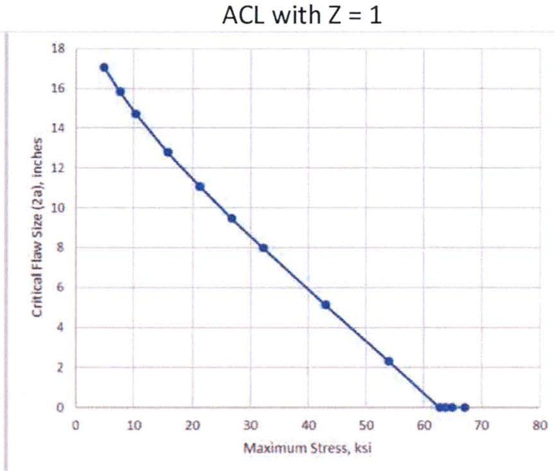
FPL Response:

Curves of critical crack size versus maximum stress are provided below. They were inadvertently left out of the report. They have been incorporated in the revised SIA Report No. 0901350.401, Revision 4.

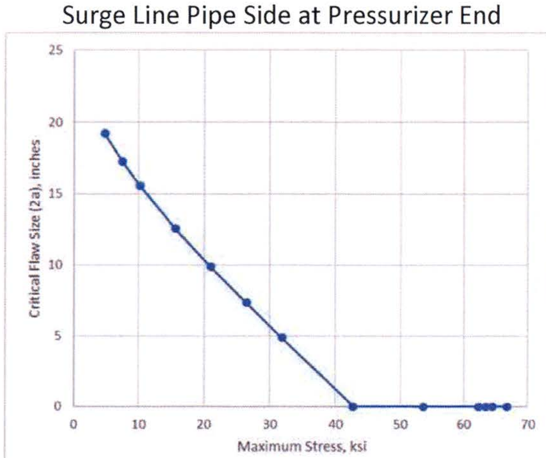
ACL



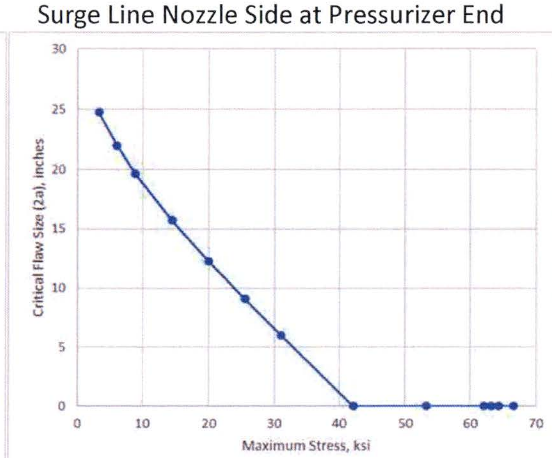
ACL with Z = 1

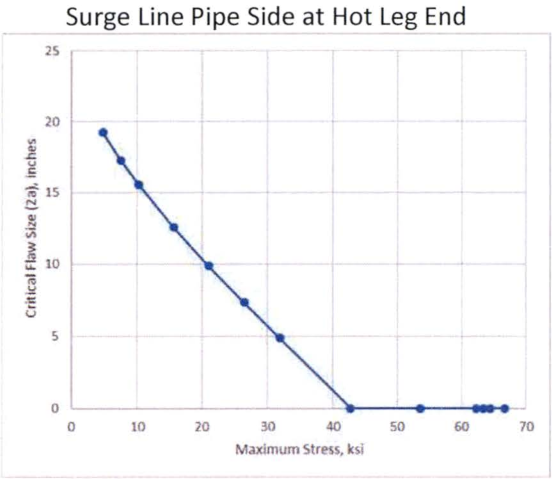


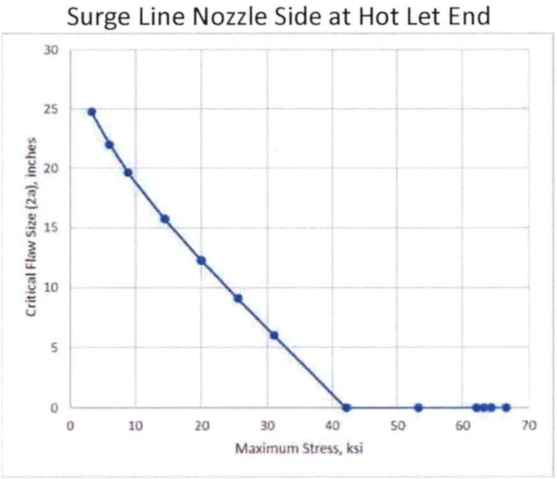
Surge Line Pipe Side at Pressurizer End

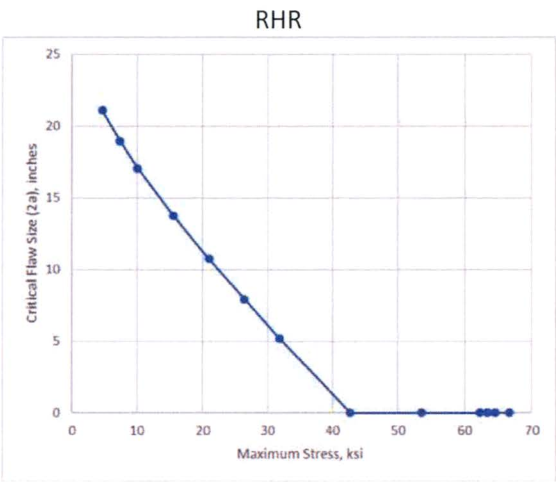


Surge Line Nozzle Side at Pressurizer End









References:

None

Associated SLRA Revisions:

(Refer to) Attachment 19 FPL Response to NRC RAI No. 4.7.4-3, Associated SLRA Revisions that are applicable to NRC RAI No. 4.7.4-12.

Associated Enclosures:

(Refer to) Attachment 19 FPL Response to NRC RAI No. 4.7.4-3, Enclosure 2.

NRC RAI Letter Nos. ML18269A209 and ML18269A210 dated October 1, 2018

RAI 4.7.4-13

Issue:

Section 6.1 of SIA Report No. 0901350.401, Revision 3, states that the transient information from generic Westinghouse nuclear steam supply system documents is used to perform the crack growth evaluation.

Request:

Discuss whether the generic transient information bounds the plant-specific (a) transients that the accumulator, RHR and surge piping will have experienced at the end of 80 years, (b) transients that are specified in the current licensing design basis, and (c) transients that are predicted to the end of the 80 years.

FPL Response:

Based on SIA Report No. 1700109.402P, Rev. 4, Section 3.0, Tables 3-1 and 3-2, the generic transient information bounds the plant-specific transients that are predicted to the end of the 80 years.

References:

None

Associated SLRA Revisions:

None

Associated Enclosures:

None

NRC RAI Letter Nos. ML18269A209 and ML18269A210 dated October 1, 2018

RAI 4.7.4-14

Issue:

Section 6.3 of SIA Report No. 0901350.401, Revision 3, discusses the derivation of the allowable flaw size. SRP 3.6.3 specifies a leakage flaw size and critical flaw size, not an allowable flaw size.

Request:

Discuss how the allowable flaw size is used in the LBB analysis and the relationship among the allowable flaw size, leakage flaw size and critical flaw size.

FPL Response:

The allowable flaw size was not used in the mechanistic LBB analysis as outlined in SRP 3.6.3. Rather, it was used in the fatigue crack growth analysis as an acceptance criterion for part-through wall flaws (Section 6.4.3) to which the final crack growth is compared. The allowable flaw size is represented in terms of the allowable end-of-evaluation period flaw depth-to-thickness ratio per ASME Code Section XI, Appendix C. The leakage flaw size and critical flaw size are used in the fatigue crack growth analysis of through-wall flaws (Section 6.4.4).

References:

None

Associated SLRA Revisions:

None

Associated Enclosures:

None

NRC RAI Letter Nos. ML18269A209 and ML18269A210 dated October 1, 2018

RAI 4.7.4-15

Issue:

(a) Section 6.4.4 of SIA Report No. 0901350.401, Revision 3, provides the leakage flaw size for the subject piping. However, the report does not provide specific calculated critical crack size for the subject piping. The report does state that the critical crack size is twice of the calculated leakage flaw size. However, without showing the actual calculated critical crack size based on material fracture toughness, it is not evident that there is a margin of 2 on the crack size as specified by SRP 3.6.3.

(b) The second paragraph on Page 6-7 of SIA Report No. 0901350.401, Revision 3, compares the calculated crack growth to the circumference of the accumulator pipe to demonstrate the crack stability. However, in a typical LBB evaluation, the crack growth is added to the leakage crack size to obtain the final leakage crack size at the end of 80 years. The final leakage crack size is compared to the critical crack size. The final leakage crack size should not exceed the half of the critical crack size in order to satisfy the margin of 2 as specified in SRP 3.6.3.

(c) The second paragraph on Page 6-7 of SIA Report No. 0901350.401, Revision 3, states that "...For the Accumulator Line, the maximum $\sigma m + \sigma b$ is 19.02 ksi (including internal pressure), and the bounding leakage flaw size is 2.53 inches with bending stress= 0 for 5GPM (Figure 5-1)..." The staff has questions on the use of 5 gpm because the leakage flaw size should provide a leak rate of 10 gpm in order to satisfy the margin of 10 with respect to the RCS detection system capability of 1 gpm.

Request:

(a) Provide the calculated critical crack size based on material fracture toughness for the accumulator, RHR and surge lines.

(b) Provide numbers to show that the leakage flaw size plus the crack growth (i.e., the final leakage crack size at the end of 80 years) still maintain a margin of 2 with respect to the critical crack size for each of the subject piping at the end of 80 years.

(c) stress = 0 for 5 GPM..." Explain why bending stress at a leak rate of 5 gpm is used and not 10 gpm.

FPL Response:

(a) As detailed in the response to the Request in RAI 4.7.4-11, in using the BAC approach, the critical crack size is not specifically calculated for each nodal point in the accumulator, RHR and surge line piping. Rather, a generic relationship between the critical crack size and the maximum stress was developed as part of the BAC approach as shown in the response to RAI 4.7.4-12. In developing the BACs, the critical crack size is assumed to be equal to twice the leakage flaw size to meet the stability margin of 2 stipulated in SRP 3.6.3. Since all points are below the BACs, it implies that a margin of at least 2 between the critical and leakage crack sizes is achieved for all nodal points considered in the analysis. The critical crack size was calculated using the modified limit load approach in SRP 3.6.3 since all piping materials are stainless steel and, therefore, the fracture toughness was not needed.

(b) The margin of 2 between the leakage crack size and the critical crack size has already been demonstrated in the BAC approach and is not required to be demonstrated in the crack growth analysis.

The objective of the fatigue crack growth analysis is to show that the growth of an initial part through-wall flaw (which is equivalent to the ASME Section XI acceptance standard flaw) will be below the ASME Section XI allowable flaw size and will be detected by the plant in-service inspection program as part of defense-in-depth for the LBB analysis.

Though not required by the SRP 3.6.3, a through-wall crack growth analysis is also performed to show that there is adequate time for the plant to take the necessary action before the crack reaches the critical through-wall crack size. A very conservative analysis was performed, where the initial through-wall flaw size was set at the 'maximum' leakage flaw size considering all the nodal points (even though from the part-wall crack growth analysis, the initial ASME Section XI acceptance standard flaw would not become a through-wall flaw after 80 years). The critical flaw size was also set as the 'minimum' critical flaw size considering all node points. Because of the conservative assumptions made in the original analysis in SIA Report No. 0901350.401, Revision 3, a more realistic crack growth evaluation was performed in response to the present RAI. The results of the analysis presented in the below table demonstrate that in all cases the final crack size does not reach the critical crack size despite the conservative nature of the evaluation. SIA Report 0901350.401 Rev. 3 has been updated with the revised crack growth evaluation.

Components	Half Critical Crack Size[1] (in.)	Half Initial Leakage Crack Size[2] (in.)	Recalculated Half Final Crack Size (in.)
Accumulator Line	4.61	2.53	2.760
RHR Line	6.35	3.12	3.334
Surge Line	4.06	3.30	3.558

Notes

1. Half critical crack size (a_C) is the minimum considering all nodes.
2. Half leakage crack size (a_L) is the maximum considering all nodes (for 5.0 gpm).

(c) Several studies performed by the industry such as that reported in MRP-140 has shown that plants are capable of detecting leakages far smaller than 1.0 gpm which is typically used in LBB analysis. In fact, plants have indicated that they are capable of detecting leakages as low as 0.1 gpm. So, whereas in the mechanistic LBB analysis per SRP 3.6.3, the traditional detectable leakage of 1.0 gpm (10.0 gpm with a factor of 10) was used for conservatism. For the through-wall crack growth analysis, a more realistic upper bound leakage detection limit of 0.5 gpm (5.0 gpm with a factor of 10) was used in determining the initial crack size. As explained in Item (b) above, conservative assumption was made in the through-wall crack growth analysis – i.e., minimum critical crack size and maximum leakage crack size of all node points. In order to not compound the conservatisms, a more realistic initial leakage flaw size for 5 gpm was used in the fatigue crack growth analysis.

References:

Materials Reliability Program: Leak-Before-Break Evaluation for PWR Alloy 82/182 Welds (MRP-140), 2005

Associated SLRA Revisions:

(Refer to) Attachment 19 FPL Response to NRC RAI No. 4.7.4-3, Associated SLRA Revisions that are applicable to NRC RAI No. 4.7.4-15.

Associated Enclosures:

(Refer to) Attachment 19 FPL Response to NRC RAI No. 4.7.4-3, Enclosure 2.

NRC RAI Letter Nos. ML18269A209 and ML18269A210 dated October 1, 2018

RAI 4.7.4-16

Issue:

Reference 51 in Section 8 of SIA Report No. 0901350.401, Revision 3, is titled: SI Report No. 1700109.402, (under Preparation) "Evaluation of Fatigue of ASME Section III, Class 1 Components for Turkey Point Units 3 and 4 for Subsequent License Renewal".

Request:

(a) Justify why data from an incomplete report are valid to be used to perform the fatigue crack growth calculations in a formal submittal to the NRC.

(b) Discuss whether Reference 51 has been published. (b)(1) If yes, discuss whether the data in the incomplete Reference 51 that were used to perform the fatigue crack growth calculations in SIA Report No. 0901350.401, Revision 3 are still valid. (b)(2) If yes, discuss whether No. 0901350.401, Revision 3, needs to be revised to indicate the publication of Reference 51.

(c) If Reference 51 has not been published, discuss when it will be published.

FPL Response:

(a) The incomplete reference was inadvertently left in the report. The current revision of SIA Report 1700109.402 is Revision 4 which has been referenced in the revised SIA Report No. 0901350.401, Revision 4.

(b) Reference 51 (SIA Report No. 1700109.402) has been published and the data in the incomplete Reference 51 that were used to perform the fatigue crack growth calculations in SIA Report No. 0901350.401, Revision 3 are still valid. The current revision of SIA Report 1700109.402P is Revision 4, which will be referenced when the LBB report is updated.

(c) Reference 51 has been published.

References:

None

Associated SLRA Revisions:

(Refer to) Attachment 19 FPL Response to NRC RAI No. 4.7.4-3, Associated SLRA Revisions that are applicable to NRC RAI No. 4.7.4-16.

Associated Enclosures:

(Refer to) Attachment 19 FPL Response to NRC RAI No. 4.7.4-3, Enclosure 2.

NRC RAI Letter Nos. ML18269A209 and ML18269A210 dated October 1, 2018

RAI 4.7.4-17

Issue:

The following issues are related to the fatigue crack growth evaluations in SIA Report No. 0901350.304, Revision 2.

(a) Section 1, Page 4, of the fatigue crack growth evaluation discusses correction for the errors documented in Corrective Action Report (CAR) 17-012.

(b) Section 2 of the fatigue crack growth evaluation states that the ASME Code Section XI fatigue crack growth law for austenitic stainless steels (for 60 years) and the ASME Code Case N-809 fatigue crack growth law (for 80 years) were used to perform the crack growth analysis.

Request:

(a) Discuss the errors in CAR 17-012 that caused the update of the fatigue crack growth calculations for the subject piping. Discuss whether the error affects calculations related to other nuclear power plants.

(b) Discuss why two separate documents (methods) were used to calculate the fatigue crack growth.

FPL Response:

(a) SIA Calculation No. 0901350.304, Revision 2 provides details of the crack growth evaluation regarding the LBB analysis for the accumulator, RHR and surge line lines at PTN. Page 15 of SIA Calculation No. 0901350.304, Revision 2 states "... conservative aspect ratio (a/l) of 0.1." Typically, in fracture mechanics evaluations, 'l' refers to the total crack length. The solution in pc-CRACK uses the crack half-length, which is typically denoted using the variable, c (i.e., l = 2c). So, for an a/l value of 0.1, the corresponding value of a/c is 0.2. However, in the original calculations, an a/c value of 0.05 was used in error. Since this is a user input error, it does not affect calculations related to other nuclear power plants.

(b) The error in CAR 17-012 affected the original fatigue crack growth calculations for 60 years which used the fatigue crack growth law described in Section 3.4.1 of SIA Calculation No. 0901350.304, Revision 2. Hence, for consistency and comparison purposes, the 60 years fatigue crack growth was performed using the original fatigue crack growth law. However, for 80 years, recently developed fatigue crack growth law described in Section 3.4.2 of SIA Calculation No. 0901350.304, Revision 2 was utilized.

References:

1. Structural Integrity Associates Engineering Calculation No. 0901350.304, Revision 3, "Fatigue Crack Growth Evaluation"

Associated SLRA Revisions:

None

Associated Enclosures:

None

NRC RAI Letter Nos. ML18269A209 and ML18269A210 dated October 1, 2018

RAI 4.7.4-18

Issue:

Section 3.2.5 of SIA Report, 0901350.304, Rev. 2, states that 51 cycles of safe shutdown earthquake (SSE) loading (one SSE cycle assumed, along with 50 cycles of operating basis earthquake (OBE)) were used in the fatigue crack growth calculations. The staff notes that the 51 cycles of SSE and OBE loads in calculating the fatigue crack growth for the subject piping are based on generic values, not plant-specific values.

Request:

Demonstrate that the 51 cycles of OBE plus SSE, with associated earthquake loads used in the fatigue crack growth calculations bound the plant-specific transient cycles and earthquake loadings specified in the current licensing basis at Turkey Point Units 3 and 4.

FPL Response:

Per SIA Report No. 1700109.402P, Rev. 4, Section 3.0, the design number of cycles of OBE, 50 cycles, bounds the actual number of cycles for 80-year projections. For the fatigue crack growth calculations, 51 cycles of SSE loading (i.e. stress for SSE event) was used as a conservative input to represent one SSE cycle and 50 cycles of OBE.

References:

1. Structural Integrity Associates Engineering Calculation No. 0901350.304, Revision 3, "Fatigue Crack Growth Evaluation"

Associated SLRA Revisions:

None

Associated Enclosures:

None