Enclosure 12 (SLRA Enclosure 4 Attachment 5)

Structural Integrity Associates Engineering Report No. 1700109.401P, Revision 8 - REDACTED, Evaluation of Environmentally-Assisted Fatigue for Turkey Point Units 3 and 4 for Subsequent License Renewal, October 26, 2018

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Report No. 1700109.401P
Revision 8 REDACTED
Project No. 1700109
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Evaluation of Environmentally-Assisted Fatigue for Turkey Point Units 3 and 4 for Subsequent License Renewal

Prepared for:
Florida Power & Light Company
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Prepared by:
Structural Integrity Associates, Inc.
San Jose, California

Prepared by: David A. Gerber, P.E. Date: 10/26/2018

Reviewed by: Terry J. Herrmann, P.E. Date: 10/26/2018

Approved by: David A. Gerber, P.E. Date: 10/26/2018

Structural Integrity Associates, Inc.®

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Section	Pages	Revision	Date	Comments
1.0 2.0 3.0 4.0 5.0 App. A	1-1 –1-7 2-1 – 2-13 3-1 – 3-17 4-1 – 4-2 5-1 – 5-6 A-1 – A-7	0	11/13/2017	Initial Issue
2.0 3.0 5.0	2-12, 2-13, 3-7 – 3-9 3-11 – 3-14, 3-18 – 3-20, 5-5 – 5-7	1	12/9/2017	Insertion of 60-year CUF values for CRDM components and 80-year CUF, F_{en} and CUF_{en} values for CRDM and RPV components. Annotation of "P" on file names of vendor proprietary documents added. Added new references. Change highlighted text to bold italicized text.
3.0 5.0	3-11 – 3-14 5-5, 5-7	2	12/17/2017	Update of two CRDM CUF values and correction of the Vent Nozzle CUF value in Table 3-3, replacement of Reference [23] in Table 3-3, removal of References [41] and [43] from List of References, addition of three footnotes in Table 3-4, update of CUF_{en} of CRDM Upper Joint and update of revision number of Reference [29].
2.0, 3.0, 5.0	2-15, 3-2, 3-7, 3-8, 3-15, 5-5	3	1/4/2018	Remove GALL Rev. 2 references.
2.0 3.0	2-11, 2-13, 3-11, 3-12	4	3/28/2018	Updated References [32-33] and [35-40]. Changed pressurizer immersion heater CUF values to proprietary.

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5.0	5-5	5	4/2/2018	Updated References [29], [30a] and [30c].
1.0 3.0 5.0 App. A	1-2, 1-4, 1-7 3-1, 3-7, 3-8, 3-10 - 3-14, 3-16 5-1, 5-2, 5-4 - 5-6 A-2, A-4, A-6, A-7	6	10/4/2018	Revised References [5] and [6]. Added References [23] and [27]. Updated F_{en} values to reflect NUREG/CR-6909, Rev. 1 Final from References [35 – 40]. Editorial corrections to faulty hyperlinks.
3.0 5.0	3-11, 3-13, 3-16 5-7	7	10/16/2018	Revised Reference [40] for new revision and revised results for CRDM Lower Joint for bounding result.
3.0	3-11, 3-12	8	10/26/2018	Correct 80-Year CUF_{en} values for Pressurizer Spray Nozzle, Pressurizer Safety and Relief Nozzle , Pressurizer Lower Head, Heater Well, Pressurizer Lower Head/ Perforation and S/G Primary Chamber, Tubesheet and Stub Barrel Complex locations.

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Table of Contents

THIS REPORT CONTAINS VENDOR PROPRIETARY INFORMATION

List of Tables

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1.0 INTRODUCTION

1.1 Background

Florida Power & Light Company (FPL) is developing a Subsequent License Renewal (SLR) application (SLRA) to extend plant operations at Turkey Point Nuclear Plant Units 3 and 4 (PTN) from 60 to 80 years. The PTN SLRA will build off the content from the original PTN license renewal application (LRA) submitted to the U.S. Nuclear Regulatory Commission (NRC) in 2000 [1], and adjust the content as necessary so that it follows the updated NRC guidance for SLR.

In response to FPL's original LRA submittal for PTN, the NRC issued renewed operating licenses on June 6, 2002 to operate an additional 20 years beyond the original 40-year operating licenses [2]. Structural Integrity Associates, Inc. (SI) provided support to FPL in 2000 and 2001 as a part of the LRA development and NRC approval for PTN. SI's support included the preparation of engineering documents, calculations, and evaluations associated with certain time limited aging analyses (TLAAs) for PTN.

This report addresses TLAAs associated with environmentally-assisted fatigue (EAF) for both PTN units. As such, this report documents the EAF TLAAs necessary to support operating license renewals from 60 to 80 years for PTN that satisfy all the requirements specified by the NRC for SLR.

1.2 GALL-SLR Guidance for EAF

Two of the NRC's applicable guidance documents for SLR were published in July 2017, and two others for EAF are still under development. Specifically, this guidance includes the following four documents:

- NUREG-2191, "Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report" [3].

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- NUREG-2192, "Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants" [4].
- Regulatory Guide RG 1.207, Revision 1, "Guidelines for Evaluating the Effects of Light-Water Reactor Water Environments in Fatigue Analyses of Metal Components" [5].
- NUREG/CR-6909, Revision 1, "Effect of LWR Water Environments on the Fatigue Life of Reactor Materials" [6].

This report satisfies all applicable NRC requirements for the EAF TLAAs based on the NRC's final guidance for SLR and EAF, plus any additional knowledge gained from ongoing interactions between the industry and NRC as the documents approach final publication.

The following two subsections provide an overview of the NRC's guidance for EAF, and summarize the PTN EAF TLAAs prepared for FPL's 60-year LRA. Section 1.3 describes the report objectives and identifies the updates needed to assess EAF for PTN for 80 years of operation based on the NRC's SLR guidance.

1.2.1 Summary of the NRC's SLR Guidance for EAF

The NRC's Standard Review Plan for SLR (SRP-SLR) [4] provides guidance to NRC staff reviewers for SLRA content to ensure the quality and uniformity of NRC staff reviews, and to present a well-defined base from which to evaluate applicant programs and activities for SLR. The SRP-SLR is a companion document to the NRC's Generic Aging Lessons Learned Report for SLR (GALL-SLR) [3], which provides guidance for SLR applicants and contains the NRC staff's generic evaluation of plant aging management programs (AMPs) and establishes the technical basis for their adequacy.

Section 4.3, "Metal Fatigue," of the SRP-SLR specifies the areas of review to ensure that the metal component fatigue parameter evaluations are valid for SLR. For EAF, these areas include component fatigue life estimates based on cumulative usage factor (CUF) calculations, and CUF adjusted to account for the effects of the reactor water environment (CUF_{en}). The acceptance criteria for such calculations follow the requirements of Part 54 to Title 10 of the U.S. Code of

Federal Regulations (10 CFR 54) [7]. Specifically, pursuant to 10 CFR 54.21(c)(1)(i) through (iii), an applicant must demonstrate one or more of the following for each analysis:

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.

For components evaluated for CUF_{en}, the acceptance criteria depend on the applicant's choice of 10 CFR 54.21(c)(1)(i), (ii), or (iii). Applicants must also include CUF_{en} calculations for additional component locations if they are more limiting than those previously evaluated for 60 years of operation. Examples of critical components are identified in NUREG/CR-6260 [8]; however, plant-specific component locations in the reactor coolant pressure boundary (RCPB) may be more limiting than those considered in NUREG/CR-6260, and thus must also be considered. Plant-specific justification can be provided to demonstrate that calculations for the NUREG/CR-6260 locations do not need to be included.

Chapter X, "Aging Management Programs That May Be Used to Demonstrate Acceptability of Time-Limited Aging Analyses in Accordance with 10 CFR 54.21(c)(1)(iii)," of the GALL-SLR Report provides an AMP acceptable to the NRC for managing EAF. Specifically, Section X.M1, "Fatigue Monitoring," defines an acceptable basis for managing structures and components (SCs) that are the subject of metal fatigue or cycle-based TLAAs or other analyses that assess fatigue or cyclical loading, in accordance with the 10 CFR 54.21(c)(1), including EAF assessments of CUF_{en}.

In Section X.M1 of the GALL-SLR Report, the NRC staff evaluated an AMP for monitoring and tracking the number of occurrences and the severity of critical cyclic loadings for selected components. The scope of the X.M1 AMP includes those mechanical or structural components with fatigue TLAAs or other analyses that depend on the number of occurrences and severity of transient cycles associated with the plant license renewal period. The X.M1 AMP has two aspects, one that verifies the continued acceptability of existing analyses through cycle counting, and the other that provides periodically updated evaluations of the fatigue analyses to demonstrate that they continue to meet the appropriate limits. In the former, the program assures that the number of occurrences and severity of each transient remains within the limits of the fatigue analyses, which in turn ensure that the analyses remain valid. For the latter, actual plant operating conditions monitored by this program can be used to inform updated evaluations of the fatigue analyses to ensure the analyses continue to meet the design or analysis-specific limit, thus minimizing the likelihood of failures from fatigue-induced cracking of the components caused by cyclic strains in the component's material.

EAF effects on fatigue are evaluated by assessing the specific set of sample critical components for the plant. The X.M1 AMP monitors and tracks the number of occurrences and severity of each of the critical thermal and pressure transients for the selected components to maintain the CUF_{en} below the design limit of 1.0. This program also relies on the GALL-SLR Report AMP XI.M2, "Water Chemistry," to provide monitoring of appropriate environmental parameters for calculating environmental fatigue multiplier (F_{en}) values, which are used to calculate CUF_{en} from CUF.

EAF effects on fatigue for the critical components can be evaluated using the positions described in RG 1.207, Revision 1[1] [5], NUREG/CR-6909, Revision 0 [10] (with "average temperature" used consistent with the clarification that was added to NUREG/CR-6909, Revision 1) [6], or other subsequent NRC-endorsed alternatives.

1 If and when published as RG 1.207, Revision 1 Final.

In some cases, flaw tolerance evaluations are used to establish inspection frequencies for components that, for example, exceed CUF or CUF_{en} fatigue limits. As an example, ASME Code, Section XI, Nonmandatory Appendix L [9] provides guidance on the performance of fatigue flaw tolerance evaluations to determine acceptability for continued service of RCPB components subjected to cyclic loadings. In flaw tolerance evaluations, the predicted size of a postulated fatigue flaw, whose initial size is typically based on the resolution of the inspection method, is a computed parameter that is used to determine the appropriate inspection frequency. The X.M1 AMP monitors and tracks the number of occurrences and severity of critical thermal and pressure transients for the selected components that are used in the fatigue flaw tolerance evaluations to verify that the inspection frequencies remain appropriate.

Component locations within the scope of this program are updated based on operating experience (OE), plant modifications, and inspection findings.

The GALL-SLR Report may be referenced in a SLRA and should be treated in the same manner as an approved topical report. In referencing the GALL-SLR Report, the applicant should indicate that the material referenced is applicable to their plant and should provide the information necessary to support the finding of program acceptability as described and evaluated in the report.

1.2.2 Summary of the EAF TLAAs Performed for PTN for 60 Years

As part of the PTN LRA for 60 years of operation, a report was prepared to describe how the TLAAs on fatigue and the technical issue associated with EAF evaluation in response to Generic Safety Issue (GSI) 190 [11] were addressed for PTN [12]. That work was further enhanced through responses to NRC Requests for Additional Information (RAIs) [13]. Because the work pre-dated the initial revision of the GALL Report [14], guidance for EAF evaluation was still being developed and has continued to evolve based on industry experience. The recommended approach for PTN accomplished two objectives. First, the TLAAs on fatigue design were resolved by confirming that the original transient design limits remain valid for the 60-year operating period. Fatigue monitoring was used to ensure these transient limits would not be exceeded. Second, EAF effects on fatigue life were examined using the most recent data from laboratory simulation of the reactor coolant environment [15,16]. These two evaluations were kept separate, since fatigue design is a TLAA and part of the plant current licensing basis (CLB), while the consideration of reactor water environmental effects on fatigue life, as described in GSI-190, was not a part of the original PTN CLB.

The SIR-00-089 report [12] examined the PTN fatigue design basis as it applied to the resolution of fatigue design TLAAs and described existing plant programs for managing fatigue crack initiation and growth that were to be used to continue to manage the effects of fatigue during the license renewal term. It also described the results from regulatory and industry studies on the effect of environmental fatigue that were used to address EAF effects in response to GSI-190 for PTN. The results from the PTN plant cycle monitoring program were shown to satisfy license renewal requirements for the fatigue TLAAs, specifically paragraph 10 CFR 54.21(c)(1)(i), where the number of plant transient design cycles for 40 years of operation and transient severity were shown to remain valid for the period of extended operation.

With respect to EAF, the report addressed reactor water environmental effects on the fatigue life of selected fatigue-sensitive reactor coolant system (RCS) components, in accordance with the resolution of GSI-190. The method chosen for this EAF evaluation was based on existing industry evaluations that were pertinent to PTN. Principally, this information was obtained from NUREG/CR-6260, as well as generic industry environmental fatigue studies and additional laboratory data [15,16]. Existing plant programs were described for managing the effects of fatigue and any modifications or enhancements required for the renewal period. EAF effects were shown to be acceptable for all NUREG/CR-6260 components except for the pressurizer surge line. This remaining component was therefore addressed through inspection programs. Based on results of high calculated CUF_{en} values, the surge line was determined to be a candidate for additional inspection considerations during the license renewal period. All welds in the surge line were therefore included as a part of future ISI programs. The surge line inspection approach was submitted to the NRC for review in 2012 [17], and was subsequently approved by the NRC in 2013 [18].

Additional EAF evaluations were undertaken for the reactor pressure vessel (RPV) shell and the RCS components/nozzles and connected systems in 2010 as a part of an Extended Power Uprate (EPU) project for PTN [19]. In addition, the pressurizer spray nozzle was reevaluated for EAF using finite element methods in 2011/2012 [20]. Additional analyses were undertaken by Westinghouse [21, 22] and Areva, NP [21, 23] for EPU. The results of all these evaluations supersede the applicable results from SIR-00-089.
Collectively, these analyses documented that the TLAAs for fatigue and EAF effects were properly evaluated for PTN for 60 years of operation, so the approach was included in FPL's

LRA for PTN. The approach was subsequently approved by the NRC when the PTN renewed operating licenses were issued in 2002. As such, they form the CLB for PTN, and serve as the input for the EAF evaluation for SLR.

1.3 Report Objectives

Based on the summaries provided in Section 1.2, the objectives of this report are as follows:

(i) To perform updated EAF screening for SLR for critical locations using all available CUF values for all PTN Class 1 RCPB components exposed to the water environment for 60 years of operation.

(ii) To develop updated PTN environmental multipliers that adopt the latest F_{en} methods provided in RG 1.207, Revision 1.

(iii) To calculate updated CUF_{en} values for all bounding PTN CUF locations using updated transient projections for 80 years and the updated F_{en} values.

(iv) To demonstrate that the PTN EAF analyses will remain valid for 80 years in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

(v) To serve as a replacement for Attachments 8.1 and 8.2 to FPL Document No. PTN-ENG-LRAM-00-0055 that reflects an updated PTN EAF assessment for SLR.

The relevant plant-specific fatigue background for PTN, which established the CLB for 60 years of operation and serves as the starting point for assessing EAF for SLR, is described in Section 2.0. The plant-specific assessment of EAF effects for PTN for SLR is contained in Section 3.0.

A summary of the key results of this report and the conclusions relevant to SLR are provided in Section 4.0.

2.0 PLANT-SPECIFIC FATIGUE BACKGROUND FOR PTN

The plant-specific fatigue background for PTN was initially compiled and summarized for FPL's 60-year LRA in Section 2.0 of Attachment 8.1 to FPL Document No. PTN-ENG-LRAM-00-0055, Revision 1 (SI Report No. SIR-00-089) [12]. That summary is duplicated in this section, with appropriate updates from the NRC's review and approval of the LRA and EPU, as well as updates made for SLR.

With respect to EAF, as described in Section 1.2.2, there are six sets of previous analyses that form the CLB for PTN and serve as the input for the EAF evaluation for SLR, as follows:

1. Section 3.0 of Attachment 8.1 to FPL Document No. PTN-ENG-LRAM-00-0055, Revision 1 (SI Report No. SIR-00-089) [12], which describes how reactor water environmental effects on the fatigue life of selected fatigue-sensitive RCS components, in accordance with the resolution of GSI-190, were addressed for PTN.
2. Attachment 8.2 to FPL Document No. PTN-ENG-LRAM-00-0055, Revision 1 (SI Report No. SIR-01-042, Rev. 0) [13], which further enhanced the Item 1 report through responses to NRC RAIs[1].
3. Section 4.0 of Attachment 8.1 to FPL Document No. PTN-ENG-LRAM-00-0055, Revision 1 (SI Report No. SIR-00-089), which addressed EAF management of the PTN surge line weld locations through a PTN plant-specific program utilizing the ASME Code Section XI inspection program. The plant-specific surge line inspection program was approved by the NRC in 2013 [18].
4. SI Calculation No. 0900948.302, which performed an additional EAF evaluation for the RPV shell as a part of EPU implementation for PTN in 2010 [19.b]. SI Calculation No. 0900948.301 determined that EAF evaluations for RCS components/nozzles and connected systems were not impacted by EPU [19.a]. Some of the results of the EPU evaluations supersede the applicable results from Items 1 and 2 above.

[1] Throughout this section, reference is made to the RAI responses filed in the NRC's Agencywide Documents Access and Management System (ADAMS) as the source of any data rather than Attachment 8.1 to FPL Document No. PTN-ENG-LRAM-00-0055, Revision 1.

5. SI Calculation Nos. 1100768.301 through 1100768.304, which performed an updated finite element EAF evaluation for the pressurizer spray nozzle in response to license renewal commitments for PTN in 2011/2012 [20]. Those results supersede the applicable results from Items 1 and 2 above.
6. Westinghouse [21, 22] and Areva evaluations [21, 23] for EPU.

PTN's TLAAs on fatigue were resolved for 60 years of operation by the above analyses through a combination of three demonstrations: (i) fatigue monitoring of all relevant plant thermal transients to confirm that the original transient design limits remain valid for the 60-year operating period (i.e., the PTN Cycle Counting Program), (ii) reactor water environmental effects on fatigue life were examined using the most recent data from laboratory simulations of the reactor coolant environment (i.e., the PTN EAF Assessment), and (iii) inspection of the surge line welds under a plant-specific program utilizing the ASME Code Section XI inspection program to verify the absence of fatigue cracking (i.e., the PTN Surge Line Inspection Program). Each of these three demonstrations are summarized in the sections that follow and form the bases for the PTN-specific program described in Section 3.0 for addressing EAF effects for SLR.

2.1 PTN Cycle Counting Program

Section 3.1 of Attachment 8.1 to FPL Document No. PTN-ENG-LRAM-00-0055, Revision 1 (SI Report No. SIR-00-089) [12] summarizes the PTN cycle counting program implemented for 60 years of operation. FPL implemented a cycle counting procedure at PTN to ensure that the design-basis transient counts are not exceeded during 60 years of plant operation [2421]. All significant plant events are captured and recorded via this procedure. The results from this program provide assurance that the structural design bases of the Class 1 plant components (both ASME Section III and ANSI B31.1 piping) are maintained for the 60-year operating period. In addition, a transient evaluation was performed for PTN that extrapolated the actual transient counts established by the FPL cycle counting program to 60 years of plant operation [25]. The results of the extrapolation concluded that the existing primary system design transients are conservative for use as a basis for 60 years of operation.

As stated in Section 4.3.1 of the PTN LRA [1] and discussed in Section 2.2 of this report, the evaluation verified the structural integrity of the RPV, RPV internals, pressurizer, steam generators (SGs), reactor coolant pumps (RCPs), and the pressurizer surge lines will remain valid for the period of extended operation. Similarly, Section 4.3.4 of the PTN LRA [1] stated similar results for the PTN piping designed in accordance with ANSI B31.1, "Power Piping."

The results of the transient evaluation that form the CLB for PTN for 60 years of operation are summarized in Table 2-1 [29]. *[Footnotes for Table 2-2 in [29] are not included in Table 2-1].* These results demonstrated that the design basis (i.e., 40-year) transient definitions were adequate for 60 years of operation, thus satisfying §54.21(c)(l)(i) of the License Renewal Rule [7], as identified in Section 4.3.1 of the PTN LRA [1] and approved by the NRC in the LRA Safety Evaluation Report (SER) [2.b]. The PTN Fatigue Monitoring Program for license renewal is described in Section 16.2.7 of the PTN UFSAR.

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Table 2-1: Projected Number of Transients Compared to Design-Basis Number of Transients – PTN Class 1 Components for 60 Years

Transient Number	Description	PTN Unit 3			PTN Unit 4		
		Projected Number of Cycles for 60 Years	40-Year Design-Basis Number of Cycles	% Used After 60 Years	Projected Number of Cycles for 60 Years	40-Year Design-Basis Number of Cycles	% Used After 60 Years
1	Station Heatup at 100°F/hour	156	200	78.0%	191 (10)	200	95.5%
2	Station Cooldown at 100°F/hour	155	200	77.5%	190 (10)	200	95.0%
3	Pressurizer Cooldown to 400 psia at 200°F/hr	142	200	71.0%	179	200	89.5%
4	Pressurizer Cooldown from 400 psia at 200°F/hr	142	200	71.0%	179	200	89.5%
5	Station Loading at 5% power per minute	2,720	14,500	18.8%	2,320	14,500	16.0%
6	Station Unloading at 5% power per minute	2,140	14,500	14.8%	2,190	14,500	15.1%
7	Step Load Increase of 10% of Full Power	109	2,000	5.5%	112	2,000	5.6%
8	Step Load Decrease of 10% of Full Power	220	2,000	11.0%	123	2,000	6.2%
9	Step Load Decrease of 50% of Full Power	167	200	83.5%	110	200	55.0%
10	Steady State Fluctuations, +/- 100 psi, +/- 6°F	Exempted	Infinite	---	Exempted	Infinite	---
11	Feedwater Cycling at Hot Standby	Exempted	2,000	---	Exempted	2,000	---
12	Boron Concentration Equalization	6,358	36,600	17.4%	6,041	36,600 (7)	16.5%
13	Shipping, Handling, Refueling Events	---	---	---	---	---	---
14	Turbine Roll Test	0	10	0.0%	0	10	0.0%

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Table 2-1: Projected Number of Transients Compared to Design-Basis Number of Transients – PTN Class 1 Components for 60 Years (continued)

Transient Number	Description	PTN Unit 3			PTN Unit 4		
		Projected Number of Cycles for 60 Years	40-Year Design-Basis Number of Cycles	% Used After 60 Years	Projected Number of Cycles for 60 Years	40-Year Design-Basis Number of Cycles	% Used After 60 Years
	Primary Side Hydrostatic Test						
15	a. Hydrostatic Test at 3107 psig Pressure, 100°F Temperature (1)(4)	1 (10))	1	100.0%	1 (10))	1	100.0%
16	b. Hydrostatic Test at 2485 psig Pressure and 400°F Temperature (1)(5)	3 (10)	5	60.0%	3 (10)	5	60.0%
17	Secondary Side Hydrostatic Test to 1356 psig (8)(9)						
	Steam Generator Loop A (pre- and post-1987)	21	35	60.0%	15	35	42.9%
	Steam Generator Loop B (pre- and post-1987)	17	35	48.6%	15	35	42.9%
	Steam Generator Loop C (pre- and post-1987)	17	35	48.6%	13	35	37.1%
18	Primary to Secondary Side Leak Test to 2,250	0	15	0.0%	0	15	0.0%
19	Secondary Leak Test to 1,085 psig (9)						
	Steam Generator Loop A	4	50	8.0%	4	50	8.0%
	Steam Generator Loop B	7	50	14.0%	11	50	22.0%
	Steam Generator Loop C	4	50	8.0%	7	50	14.0%

THIS REPORT CONTAINS VENDOR PROPRIETARY INFORMATION

Table 2-1: Projected Number of Transients Compared to Design-Basis Number of Transients – PTN Class 1 Components for 60 Years (concluded)

Transient Number	Description	PTN Unit 3			PTN Unit 4		
		Projected Number of Cycles for 60 Years	40-Year Design-Basis Number of Cycles	% Used After 60 Years	Projected Number of Cycles for 60 Years	40-Year Design-Basis Number of Cycles	% Used After 60 Years
20	Secondary to Primary Side Leak Test to 840 psig (11)						
	Steam Generator Loop A	8	15	53.3%	14	15	93.3%
	Steam Generator Loop B	15	15	100.0%	15	15	100.0%
	Steam Generator Loop C	9	15	60.0%	15	15	100.0%
21	Loss of Load without Immediate Turbine Trip	43	80	53.8%	38	80	47.5%
22	Loss of AC Power	28	40	70.0%	29	40	72.5%
23	Partial Loss of Flow (Reverse Flow)	43	80	53.8%	43	80	53.8%
24	Loss of Secondary Pressure	2	6	33.3%	0	6	0.0%
25	Reactor Trip (1)	291 (10)	400	72.8%	337 (10)	400	84.3%
26	Inadvertent Auxiliary Spray	0	10	0.0%	0	10	0.0%
27	Operating Basis Earthquake (OBE)	0	50	0.0%	0	50	0.0%
28	Loss of Coolant Accident	0	1	0.0%	0	1	0.0%
29	Steam Line Break	0	1	0.0%	0	1	0.0%
30	Safe Shutdown Earthquake (SSE)	0	1	0.0%	0	1	0.0%

2.2 PTN EAF Assessment

Section 3.2 of Attachment 8.1 to FPL Document No. PTN-ENG-LRAM-00-0055, Revision 1 (SI Report No. SIR-00-089) [12] summarizes the PTN evaluation of reactor water environmental effects for 60 years of operation. Attachment 8.2 to FPL Document No. PTN-ENG-LRAM-00-0055, Revision 1 (SI Report No. SIR-01-042, Rev. 0) [13] further enhanced the PTN EAF assessment through responses to NRC RAIs.

The PTN LRA pre-dated GALL requirements, so treatment of EAF effects was still being finalized by the NRC in the wake of the resolution of GSI 190. Based on this, FPL evaluated EAF effects using a combination of methods that included NUREG/CR-6260 [8], NUREG/CR-5704 [16], and NUREG/CR-6583 [15]. The use of NUREG/CR-6260 was directly relevant to PTN because the "Older Vintage Westinghouse Plant" evaluated in Section 5.5 of that document matched PTN in the design codes used, as well as the analytical approach and techniques used. In addition, the evaluated transient cycles matched or bounded PTN.

The PTN EAF assessment for 60 years was performed in five parts:

- First, Section 3.2.1 of Attachment 8.1 to FPL Document No. PTN-ENG-LRAM-00-0055, Revision 1 performed an evaluation of the selected components from NUREG/CR-6260. This included the RPV at the core support guide weld, the RPV inlet and outlet nozzles, the surge line hot leg nozzle safe end, the charging nozzle, the safety injection nozzle, and the RHR line tee.
- Second, Section 3.2.2 of Attachment 8.1 to FPL Document No. PTN-ENG-LRAM-00-0055, Revision 1 performed an evaluation of Class 1 components that FPL originally excluded from the transient evaluation to demonstrate long-term structural acceptability for continued operation beyond 40 years for PTN. This included the pressurizer lower head, pressurizer surge and spray nozzles, pressurizer surge and spray lines, charging lines and associated charging nozzles, and all branch lines subject to NRC Bulletin 88-08 loading conditions.

- Third, some of the above results were enhanced in responses to NRC RAIs.
- Fourth, some of the above results were updated as a part of EPU implementation at PTN in 2010 [21].
- Fifth, some of the above results were updated as a part of fulfilling license renewal commitments for PTN in 2011/2012.

A summary of the EAF assessments that form the CLB for PTN for 60 years of operation is provided in Table 2-2. These results demonstrated that the CUF_{en} values for all evaluated components except the surge line weld locations were within the allowable value of 1.0 for 60 years of plant operation, as identified in Section 4.3.5 of the PTN LRA [1].

THIS REPORT CONTAINS VENDOR PROPRIETARY INFORMATION

Table 2-2: Summary of the PTN EAF Assessment for 60 Years of Operation

Component	60-Year CLB CUF	F_{en}	Material	Latest 60-Year CUF_{en}
RPV Shell at Core Support Pads (1,2)	0.478 (2) / 0.509 (UFSAR) (2)	1.76 (2)	Low Alloy/Carbon (3)	0.8413 (2)
RPV Inlet Nozzle (1,4) (inside surface)	0.073 (4) / 0.066 (UFSAR) (4)	2.45 (4)	Low Alloy/Carbon (4)	0.179 (4)
RPV Outlet Nozzle (1,5) (inside surface)	0.056 (5) / 0.063 (UFSAR) (5)	2.45 (5)	Low Alloy/Carbon (3)	0.137 (5)
RCS Piping Surge Line Hot Leg Nozzle (1,6)	0.944 (6)	4.5 (6)	Stainless steel (3)	4.248 (6)
RCS Piping Safety Injection Nozzle (1,7)	0.046 (7)	7.11 (7)	Stainless steel (3)	0.327 (7)
RCS Piping Charging Nozzle (1,8)	0.030 (8)	10.63 (8)	Stainless steel (3)	0.319 (8)
RCS Piping Residual Heat Removal (RHR) Piping Tee (1,9)	0.022 (9)	9.32 (9)	Stainless steel (3)	0.205 (9)
Pressurizer Surge Nozzle (10)	{ } (10)	Addressed by inspection AMP (see Section 2.3)		
Pressurizer Spray Nozzle (11)	{ } / 0.24667 (11)	4.0 (11)	Not reported in [26]	< 1.0 (11)
Pressurizer Safety and Relief Nozzle (12)	{ } (12)	4.0 (12)	Not reported in [26]	< 1.0 (12)
Pressurizer Lower Head, Heater Well (13)	0.461 / { } (13)	4.2 (13)	Stainless steel (13)	1.94 (13)
Pressurizer Lower Head/Perforation (14)	{ } (14)	4.0 (14)	Not reported in [26]	< 1.0 (14)
Pressurizer Upper Head and Shell (15)	Negligible / { } (15)	4.0 (15)	Not reported in [26]	< 1.0 (15)
Pressurizer Support Skirt/Flange (16)	0.0165 (16)	4.0 (16)	Not reported in [26]	< 1.0 (16)
Pressurizer Manway Pad (17)	{ } (17)	4.0 (17)	Not reported in [26]	< 1.0 (17)
Pressurizer Manway Cover (18)	0.0 (18)	4.0 (18)	Not reported in [26]	< 1.0 (18)
Pressurizer Manway Bolts (18)	0.0 (18)	4.0 (18)	Not reported in [26]	< 1.0 (18)
Pressurizer Welded Manway Diaphragm (19)	{ } (19)	4.0 (19)	Not reported in [26]	< 1.0 (19)
Pressurizer Support Lug (20)	Not Installed (20)	---	---	---
Pressurizer Instrument Nozzle (21)	{ } (21)	4.0 (21)	Not reported in [26]	< 1.0 (21)
Pressurizer Immersion Heater (22)	{ } (22)	4.0 (22)	Not reported in [26]	< 1.0 (22)
Pressurizer Valve Support Bracket (20)	Not Installed (20)	---	---	---

For notes, see next page.

THIS REPORT CONTAINS VENDOR PROPRIETARY INFORMATION

Table 2-2: Summary of the PTN EAF Assessment for 60 Years of Operation (concluded)

Notes for Table 2-2:

1. NUREG/CR-6260 component location, per Table 1 in the response to RAI 4.3.5-1 [26].
2. For the RPV Shell at Core Support Pads location, a revised CUF value of 0.4030 and a CUF_{en} value of 0.9894 was obtained in Table 5 of SI Calculation No. 0900948.302, Revision 1 [19.b]. The value of 0.478 shown was reported for EPU [21, Table 2.2.2.3-1]. The F_{en} was calculated as (0.8413/0.478) = 1.76 for PTN-4. The CUF was reported in the UFSAR as 0.509 [22].
3. As identified in the response to RAI 4.3.5-1 [26].
4. For the RPV Inlet Nozzle, the value reported as U_{PTN} in Table 1 in the response to RAI 4.3.5-1 [26] was updated to the value shown as a part of EPU implementation [21]. The refined CUF_{en} value of 0.372 shown in Table 3 in the response to RAI 4.3.5-1 [26] was also revised, and the value of 0.073 shown was reported for EPU [21, Table 2.2.2.3-1]. The F_{en} was calculated as (0.179/0.073) = 2.45.
5. For the RPV Outlet Nozzle, the value reported as U_{PTN} in Table 1 in the response to RAI 4.3.5-1 [26] was updated to the value shown as a part of EPU implementation [21]. The CUF_{en} value was also revised, and the value of 0.056 shown was reported for EPU [21, Table 2.2.2.3-1]. The F_{en} was calculated as (0.137/0.056) = 2.45.
6. For the RCS Piping Surge Line Hot Leg Nozzle, the CUF value is U_{PTN} in Table 1 in the response to RAI 4.3.5-1 [26]. The CUF_{en} value is U_{6260} in Table 1 in the response to RAI 4.3.5-1 [26]. The F_{en} was calculated as (4.248/0.944) = 4.5.
7. For the RCS Piping Safety Injection Nozzle, the PTN location was evaluated to ANSI B31.1 rules and does not produce a fatigue usage value, so the NUREG/CR-6260 (U_{code}) value was used from Table 1 of the response to RAI 4.3.5-1 [26]. The value was not revised for EPU. The CUF_{en} value is U_{6260} from Table 1 in the response to RAI 4.3.5-1 [26]. The F_{en} was calculated as (0.327/0.046) = 7.11.
8. For the RCS Piping Charging Nozzle, the PTN location was evaluated to ANSI B31.1 rules and does not produce a fatigue usage value, so the NUREG/CR-6260 (U_{code}) value was used from Table 1 of the response to RAI 4.3.5-1 [26]. The value was not revised for EPU. The CUF_{en} value is U_{6260} from Table 1 in the response to RAI 4.3.5-1 [26]. The F_{en} was calculated as (0.319/0.030) = 10.63.
9. For the RCS Piping RHR Piping Tee Nozzle, the PTN location was evaluated to ANSI B31.1 rules and does not produce a fatigue usage value, so the NUREG/CR-6260 (U_{code}) value was used from Table 1 of the response to RAI 4.3.5-1 [26]. The value was not revised for EPU. The CUF_{en} value is U_{6260} from Table 1 in the response to RAI 4.3.5-1 [26]. The F_{en} was calculated as (0.205/0.022.) = 9.32.
10. For the Pressurizer Surge Nozzle, the CUF value of 0.5202 from the table in the response to RAI 4.3.1-4 [26] was updated to { } as a part of EPU implementation [21, 22]. EAF for the PTN surge line components is satisfactorily addressed through a PTN plant-specific program utilizing the ASME Code Section XI inspection program, as discussed in Section 2.3.
11. For the Pressurizer Spray Nozzle, the CUF value of 0.8906 from the table in the response to RAI 4.3.1-4 [26] was updated to 0.24667 in Table 5 of SI Calculation No. 1100768.304, Revision 0 [20.d] in 2012. A value of { } was reported as a part of EPU implementation in 2010 [21, 22]. In the response to RAI 4.3.1-4 [26], a screening F_{en} of 4.0 was used to justify acceptability of all pressurizer locations with a CLB CUF less than 0.25.
12. For the Pressurizer Safety and Relief Nozzle, the CUF value of 0.148 from the table in the response to RAI 4.3.1-4 [26] was updated to { } as a part of EPU implementation [21, 22]. In the response to RAI 4.3.1-4 [26], a screening F_{en} of 4.0 was used to justify acceptability of all pressurizer locations with a CLB CUF less than 0.25.
13. For the Pressurizer Lower Head, Heater Well, the CUF value of 0.461 from the table in the response to RAI 4.3.1-4 [26] was updated to { } as a part of EPU implementation [21, 22]. In

the response to RAI 4.3.1-4 [26], an F_{en} of 4.2 was used to determine a maximum CUF_{en} value of 1.94, which was qualitatively dispositioned based on inherent margins in the calculational process, the low risk significance associated with these penetrations, current visual inspections performed on the penetrations as part of the ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program, and the fact that the surge line is significantly more limiting from a fatigue perspective when considering reactor water environmental effects.

14. For the Pressurizer Lower Head/Perforation, the CUF value of 0.0165 from the table in the response to RAI 4.3.1-4 [26] was updated to { } as a part of EPU implementation [21, 22]. In the response to RAI 4.3.1-4 [26], a screening F_{en} of 4.0 was used to justify acceptability of all pressurizer locations with a CLB CUF less than 0.25.
15. For the Pressurizer Upper Head and Shell, the CUF value of 0.7737 from the table in the response to RAI 4.3.1-4 [26] was qualitatively revised to a negligible value in the RAI response. That value was updated to { } as a part of EPU implementation [21, 22]. In the response to RAI 4.3.1-4 [26], a screening F_{en} of 4.0 was used to justify acceptability of all pressurizer locations with a CLB CUF less than 0.25.
16. For the Pressurizer Support Skirt/Flange, the CUF value of 0.0165 from the table in the response to RAI 4.3.1-4 [26] was not revised as a part of EPU implementation [21, 22]. In the response to RAI 4.3.1-4 [26], a screening F_{en} of 4.0 was used to justify acceptability of all pressurizer locations with a CLB CUF less than 0.25.
17. For the Pressurizer Manway Pad, the CUF value of 0.0 from the table in the response to RAI 4.3.1-4 [26] was updated to { } as a part of EPU implementation [21, 22]. In the response to RAI 4.3.1-4 [26], a screening F_{en} of 4.0 was used to justify acceptability of all pressurizer locations with a CLB CUF less than 0.25.
18. For the Pressurizer Manway Cover and Bolts, the CUF values of 0.0 from the table in the response to RAI 4.3.1-4 [26] were not updated as a part of EPU implementation [21, 22]. In the response to RAI 4.3.1-4 [26], a screening F_{en} of 4.0 was used to justify acceptability of all pressurizer locations with a CLB CUF less than 0.25.
19. For the Pressurizer Manway Welded Diaphragm, the CUF value of 0.0321 from the table in the response to RAI 4.3.1-4 [26] was updated to { } as a part of EPU implementation [21, 22]. In the response to RAI 4.3.1-4 [26], a screening F_{en} of 4.0 was used to justify acceptability of all pressurizer locations with a CLB CUF less than 0.25.
20. For the Pressurizer Support Lug and Valve Support Bracket, there was no CUF value reported in the table in the response to RAI 4.3.1-4 [26] as these components are not installed at PTN.
21. For the Pressurizer Instrument Nozzle, the CUF value of 0.0627 from the table in the response to RAI 4.3.1-4 [26] was updated to { } as a part of EPU implementation [21, 22]. In the response to RAI 4.3.1-4 [26], a screening F_{en} of 4.0 was used to justify acceptability of all pressurizer locations with a CLB CUF less than 0.25.
22. For the Pressurizer Immersion Heater, the CUF value of { } is from [22]. In the response to RAI 4.3.1-4 [26], a screening F_{en} of 4.0 was used to justify acceptability of all pressurizer locations with a CLB CUF less than 0.25.

2.3 PTN Surge Line Inspection Program

As discussed in Section 3.2.2.2 of Attachment 8.1 to FPL Document No. PTN-ENG-LRAM-00-0055, Revision 1 (SI Report No. SIR-00-089) [12] and indicated by the results shown in Table 2-2, the CUF_{en} results for locations in the PTN surge line (the RCS Piping Surge Line Hot Leg Nozzle and the Pressurizer Surge Nozzle) could not be reduced to below the allowable value of 1.0. Therefore, FPL identified the surge line as a candidate for additional inspection considerations during the license renewal period. Thus, the entire surge line was included as a part of the ASME Code Section XI risk informed inservice inspection (RI-ISI) program, as described in Section 4.0 of Attachment 8.1 to FPL Document No. PTN-ENG-LRAM-00-0055, Revision 1.

In their response to RAI 4.3.5-2 [26], FPL committed to an inspection-based AMP to address fatigue of the PTN pressurizer surge lines during the period of extended operation using an approach like that documented in the ASME Code, Section XI, Nonmandatory Appendix L [9]. Because the NRC had not endorsed the Appendix L approach at the time of PTN's LRA submittal, FPL committed to inspection of all surge line welds on both PTN units during the fourth inservice inspection interval, prior to entering the extended period of operation, and using the inspection results to assess the appropriate approach for addressing EAF of the surge lines using one or more of the following:

1. Further refinement of the fatigue analysis to lower the CUF(s) to below 1.0, or
2. Repair of the affected locations, or
3. Replacement of the affected locations, or
4. Manage the effects of fatigue by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method accepted by the NRC).

If selected, the inspection details for Option 4, such as scope, qualification, method, and frequency, required submittal to the NRC for review and approval prior to entering the period of extended operation.

The surge line inspection program was developed and submitted to the NRC for review in 2012 [17]. This program was based on a flaw tolerance evaluation performed in accordance with ASME Code, Section XI, Appendix L using initial postulated flaw sizes consistent with sizes that are detectable by qualified nondestructive examination (NDE) techniques and fatigue crack growth rates that account for the effects of the reactor coolant environment. The results also determined that there was additional margin between the inspection frequency and the shortest allowable operating period for the most limiting flaw assumed in the evaluation. The analysis concluded that a 10-year inspection frequency was adequate for detecting cracking caused by EAF of the pressurizer surge line welds before there is a loss of intended function.

The surge line inspection program was approved by the NRC in 2013 [18]. Based on its review of the surge line inspection program, the NRC found the program acceptable because it satisfies the ten elements for an acceptable AMP, as described in Section A.1.2.3 of the SRP-SLR [4], and it adequately manages cracking caused by EAF in the pressurizer surge line welds. The NRC staff also found that FPL determined an appropriate approach for addressing EAF of the pressurizer surge lines and thus fulfilled their license renewal commitment.

Based on this, the PTN CLB for assessing EAF of the pressurizer surge line welds is inspection every ten years to verify the absence of fatigue cracking. Therefore, CUF_{en} analyses for surge line components are not necessary for SLR.

3.0 PLANT-SPECIFIC ASSESSMENT OF ENVIRONMENTALLY-ASSISTED FATIGUE EFFECTS FOR SUBSEQUENT LICENSE RENEWAL

The relevant plant-specific fatigue background for 60 years of operation summarized in Section 2.0 forms the CLB for PTN, and serves as the starting point for assessing EAF for SLR. The plant-specific assessment of EAF effects for PTN for SLR is contained in this section. The plant-specific assessment is separated into four parts, with the following objectives:

1. **PTN Cycle Counting Program**. Section 3.1 describes the latest results from the PTN Cycle Counting Program and provides a summary of the updated evaluation of transient counts and severities performed for SLR.
2. **PTN EAF Assessment**. Section 3.2 describes the PTN-specific EAF calculations for all limiting locations for SLR. The assessment incorporates the CUFs for all Class 1 RCPB components that are exposed to a water environment, revises the prior PTN environmental multipliers in Table 2-2 to adopt the latest F_{en} methods provided in RG 1.207, Revision 1, and calculates CUF_{en} values for all relevant PTN components listed in Table 2-2 using the updated transient projections for SLR and the updated F_{en} values.
3. **Plant-Specific Limiting Locations**. Section 3.3 provides the bases for why the plant-specific limiting locations evaluated for EAF in Section 3.2are limiting locations per GALL-SLR Chapter X.M1 requirements for PTN.
4. **Summary of PTN EAF Assessment**. Section 3.4 summarizes the plant-specific EAF assessment results and the relevant changes to the PTN fatigue bases for SLR.

These four parts are addressed in the sections that follow, and collectively, they serve as a replacement to the PTN 60-year EAF assessment that reflects an updated PTN EAF assessment for SLR.

3.1 PTN Cycle Counting Program

The PTN Cycle Counting Program is addressed through FPL Procedure No. 0-ADM-553 [21]. For PTN's first license renewal for 60 years of operation, this program was investigated to project transient counts out to 60 years and evaluate transient severities. That investigation concluded that the original 40-year design basis transient counts and severities remained valid for 60 years of operation. Therefore, as discussed in Section 1.2.2, FPL concluded that the PTN Cycle Counting Program satisfied the license renewal requirements for fatigue TLAAs in 10 CFR 54.21(c)(1)(i).

For SLR, an updated investigation was performed to develop revised transient projections and evaluate transient severities for 80 years of operation [29]. That investigation concluded that the original 40-year design basis transient counts and severities, which are also the CLB for PTN for 60 years of operation, remain valid for SLR. A summary of the transient projections for 80 years of operation from the investigation are shown in **Table 3-1** (Unit 3) and **Table 3-2** (Unit 4), excerpted from [29] *(footnoted References for the tables are for [29])*.

Based on the 80-year evaluation performed for SLR, the same transients and transient counts used for the 60-year EAF assessment remain applicable for 80 years of operation, and were used in the updated EAF assessment for 80 years that is discussed in the following section.

THIS REPORT CONTAINS VENDOR PROPRIETARY INFORMATION

Table 3-1: Summary of Transient Projections for 80 Years of Operation (PTN Unit 3)

PTN-3		Design Number	Through 2016		60 Year Projection [9]	80-Year Projections			Source of Allowables			
Transient Number [3]	Transient		Count through 2016	Percent of Design Number		80 Year Projection	Percent of Design Number	Weighted Projection Method [1]	UFSAR Table 4.1-8	UFSAR Table 4.1-10	0-ADM-553	Minimum
	Normal											
1	Station Heatup at 100°F/hour	200	109	55%	156	164	82%	X	200	200	200	200
2	Station Cooldown at 100°F/hour	200	109	55%	155	164	82%	X	200	200	200	200
3	Pressurizer Cooldown at 200°F/hour [5][17]	200	95	48%	142	148	74%	X	---	200	200	200
5	Station Loading at 5% power per minute	14500	293	2%	2720	533	4%		14500 [11]	---	14500	2200 [16]
6	Station Unloading at 5% power per minute	14500	242	2%	2140	440	3%		14500 [11]	---	14500	2200 [16]
7	Step Load Increase of 10% of Full Power	2000	43	2%	109	79	4%		2000	---	2000	2000
8	Step Load Decrease of 10% of Full Power	2000	90	5%	220	164	8%		2000	---	2000	2000
9	Step Load Decrease of 50% of Full Power	200	68	34%	167	82	41%	X	200	---	200	200
	Steady State Fluctuations [12]	0	Exempted						0			----
	Feedwater Cycling at Hot Standby [15]	2000	Exempted						2000	---	---	2000
	Boron Concentration Equalization [5]	36000	Not Counted						---	---	---	36600
	Test											
14	Turbine Roll Test	10	1 [4][20]	10%	1 [20]	1	10%		---	---	---	----
15	Hydrostatic Test at 3107 psig Pressure, 100°F Temperature [6][19]	1	1	100%	1	1	100%		1	---	5	1
16	Hydrostatic Test at 2485 psig Pressure and 400°F Temperature [7]	5	1	20%	3	2	40%		5	5	---	5
17	Secondary Side Hydrostatic Test to 1356 psig	----	----	----	----	----	---	---	---	---	---	----
	Steam Generator Loop A [10]	10	17 / 9 [14]	90%	21	9	90%		---	10	35 [23]	10
	Steam Generator Loop B [10]	10	13 / 7 [14]	70%	17	7	70%		---	10	35 [23]	10
	Steam Generator Loop C [10]	10	13 / 7 [14]	70%	17	7	70%		---	10	35 [23]	10
	Primary to Secondary Side Leak Test to 2435 psig [7]	150	1	1%	----	2	1%		---	150	150	150
18	Primary to Secondary Side Leak Test to 2250 psig [6]	15	1	7%	0	2	13%		---	15	---	15
19	Secondary Side Leak Test ≥ 1085 psig [2]	----	----	----	----	----	---	---	---	---	---	----
	Steam Generator Loop A [13]	50	9	18%	4	21	42%		---	50	50	50
	Steam Generator Loop B [13]	50	7	14%	7	16	32%		---	50	50	50
	Steam Generator Loop C [13]	50	7	14%	4	16	32%		---	50	50	50
	Secondary to Primary Side Leak Test to 840 psig [6][8]	---	---	---	----	---	---	---	---	---	---	----
	Steam Generator Loop A	15	8	53%	8	8	53%		---	15	---	15
	Steam Generator Loop B	15	15	100%	15	15	100%		---	15	---	15
	Steam Generator Loop C	15	9	60%	9	9	60%		---	15	---	15

Table continued on next page.

THIS REPORT CONTAINS VENDOR PROPRIETARY INFORMATION

Table 3-1: Summary of Transient Projections for 80 Years of Operation (PTN Unit 3) (concluded)

PTN-3		Design Number	Through 2016		60 Year Projection [9]	80-Year Projections			Source of Allowables			
Transient Number [3]	Transient		Count through 2016	Percent of Design Number		80 Year Projection	Percent of Design Number	Weighted Projection Method [1]	UFSAR Table 4.1-8	UFSAR Table 4.1-10	0-ADM-553	Minimum
	Upset											
21	Loss of Load without Immediate Turbine Trip or Reactor Trip	80	15	19%	43	28	35%		---	80	80	80
22	Loss of Off-Site AC Electrical Power	40	6	15%	28	10	25%	X	---	40	40	40
23	Loss of Flow in One Reactor Coolant Loop	80	14	18%	43	26	33%		---	80	80	80
25	Reactor Trip	400	183	46%	291	272	68%	X	400	400	400	400
26	Inadvertent Auxiliary Spray [18][21]	10	0	0%	0	1	10%		---	10	---	10
27	OBE [22]	50	0	0%	0	10	20%		---	---	---	20
	Loss of Secondary Pressure (Press Loss) [6]	6	1	17%	2	2	33%		---	---	6	6

Footnotes

(1) Weighted projection method used for counted normal and upset transients in which 60-year projections for either unit are over 70% of design numbers in SIR-00-089 [1].
(2) Labelled as "Secondary Leak Test" in 0-ADM-553 [7]. Labelled as "Hydrostatic Pressure Test" in Table 4.1-10 [28].
(3) Transient numbers from Table 3-1 of SIR-00-089 [1].
(4) Not expected to have any additional cycles on RSGs.
(5) Applies to Pressurizer only.
(6) Applies to Steam Generator only. Labelled as "Secondary Leak Test" in 0-ADM-553 [7].
(7) Limited by Reactor Coolant Pump Analysis [16, Attachment 1, pages 44 and 45].
(8) Leak Test Procedure cancelled per [30].
(9) 60-year projections from [5, PTN-LR-00-0127 Table 10.3-1].
(10) Not expected to have any additional cycles on RSGs.
(11) Cycle limits for baffle-former bolts only is being lowered from 14,500 to 2,200 due EPU RCS conditions (Table 4.1-8 of UFSAR [8]).
(12) Not counted, not significant contributor to fatigue usage factor.
(13) 80-year plant life projected cycles computed using 65 years of life for the RSGs.
(14) Values are [(pre-and post- 1987) / (post- 1987)] cycles [5, PTN-LR-00-0127 Table 10.3-1].
(15) Not counted, intermittent slug feeding at hot standby not performed.
(16) Limit of 2,200 cycles established for baffle former bolts only per UFSAR Table 4.1-8 [8].
(17) Represents 200 cycles each of: (1) pressurizer cooldown cycles at ≤ 200°F/hr from nominal pressure and (2) pressurizer cooldown cycles at ≤ 200°F/hr from 400 psia [28].
(18) Spray water temperature differential to 560°F.
(19) Applies to Steam Generator only. Represents pre-operational test [16, Note 3 on Attachment 1, pages 44 and 45].
(20) Adjustment in 60-year projection in [5, PTN-LR-00-0127 Table 10.3-1] - recorded as a value of 0 when 1 was assumed in pre-operational startup.
(21) One cycle is projected for 80 years to remain within the analytical basis if that event occurs.
(22) One cycle of 10 events is projected for 80 years to remain within the analytical basis if that event occurs.
(23) Recommended revision 0-ADM-553 to align with UFSAR Table 4.1-10.

THIS REPORT CONTAINS VENDOR PROPRIETARY INFORMATION

Table 3-2: Summary of Transient Projections for 80 Years of Operation (PTN Unit 4)

PTN-4		Design Number	Through 2016		60 Year Projection [9]	80-Year Projections			Source of Allowables			
Transient Number [3]	Transient		Count through 2016	Percent of Design Number		80 Year Projection	Percent of Design Number	Weighted Projection Method [1]	UFSAR Table 4.1-8	UFSAR Table 4.1-10	0-ADM-553	Minimum
	Normal											
1	Station Heatup at 100°F/hour	200	121	61%	191	181	91%	X	200	200	200	200
2	Station Cooldown at 100°F/hour	200	121	61%	190	181	91%	X	200	200	200	200
3	Pressurizer Cooldown at 200°F/hour [5][17]	200	104	52%	179	158	79%	X	---	200	200	200
5	Station Loading at 5% power per minute	14500	260	2%	2320	484	3%		14500 [11]	---	14500	2200 [16]
6	Station Unloading at 5% power per minute	14500	242	2%	2190	451	3%		14500 [11]	---	14500	2200 [16]
7	Step Load Increase of 10% of Full Power	2000	44	2%	112	82	4%		2000	---	2000	2000
8	Step Load Decrease of 10% of Full Power	2000	57	3%	123	107	5%		2000	---	2000	2000
9	Step Load Decrease of 50% of Full Power	200	42	21%	110	51	26%	X	200	---	200	200
	Steady State Fluctuations [12]	0	Exempted						0			----
	Feedwater Cycling at Hot Standby [15]	2000	Exempted						2000	---	---	2000
	Boron Concentration Equalization [5]	36000	Not Counted						---	---	---	36600
	Test											
14	Turbine Roll Test	10	1 [4][20]	10%	1 [20]	1	10%		---	---	---	----
15	Hydrostatic Test at 3107 psig Pressure, 100°F Temperature [6][19]	1	1	100%	1	1	100%		1	---	5	1
16	Hydrostatic Test at 2485 psig Pressure and 400°F Temperature [7]	5	1	20%	3	2	40%		5	5	---	5
17	Secondary Side Hydrostatic Test to 1356 psig	----	----	----	----	----	---	---	---	---	---	----
	Steam Generator Loop A [10]	10	11 / 6 [14]	60%	15	6	90%		---	10	35 [23]	10
	Steam Generator Loop B [10]	10	11 / 6 [14]	60%	15	6	70%		---	10	35 [23]	10
	Steam Generator Loop C [10]	10	9 / 5 [14]	50%	13	5	70%		---	10	35 [23]	10
	Primary to Secondary Side Leak Test to 2435 psig [7]	150	1	1%	----	2	1%		---	150	150	150
18	Primary to Secondary Side Leak Test to 2250 psig [6]	15	1	7%	0	2	13%		---	15	---	15
19	Secondary Side Leak Test ≥ 1085 psig [2]	----	----	----	----	----	----	---	---	---	---	----
	Steam Generator Loop A [13]	50	6	12%	4	14	28%		---	50	50	50
	Steam Generator Loop B [13]	50	6	12%	11	14	28%		---	50	50	50
	Steam Generator Loop C [13]	50	5	10%	7	12	24%		---	50	50	50
	Secondary to Primary Side Leak Test to 840 psig [6][8]				---	----	----	---	---	---	---	----
	Steam Generator Loop A	15	14	93%	14	14	93%		---	15	---	15
	Steam Generator Loop B	15	15	100%	15	15	100%		---	15	---	15
	Steam Generator Loop C	15	15	100%	15	15	100%		---	15	---	15

Table continued on next page.

THIS REPORT CONTAINS VENDOR PROPRIETARY INFORMATION

Table 3-2: Summary of Transient Projections for 80 Years of Operation (PTN Unit 4) (concluded)

PTN-4		Design Number	Through 2016		60 Year Projection (9)	80-Year Projections			Source of Allowables			
Transient Number (3)	Transient		Count through 2016	Percent of Design Number		80 Year Projection	Percent of Design Number	Weighted Projection Method (1)	UFSAR Table 4.1-8	UFSAR Table 4.1-10	0-ADM-553	Minimum
Upset												
21	Loss of Load without Immediate Turbine Trip or Reactor Trip	80	14	18%	38	27	34%		---	80	80	80
22	Loss of Off-Site AC Electrical Power	40	13	33%	29	19	48%	X	---	40	40	40
23	Loss of Flow in One Reactor Coolant Loop	80	11	14%	43	21	26%		---	80	80	80
25	Reactor Trip	400	187	47%	337	292	73%	X	400	400	400	400
26	Inadvertent Auxiliary Spray (18)(21)	10	0	0%	0	1	10%		---	10	---	10
27	OBE (22)	50	0	0%	0	10	20%		---	---	---	20
	Loss of Secondary Pressure (Press Loss) (6)(21)	6	0	0%	0	1	17%		---	---	6	6

Footnotes
(1) Weighted projection method used for counted normal and upset transients in which 60-year projections for either unit are over 70% of design numbers in SIR-00-089 [1].
(2) Labelled as "Secondary Leak Test" in 0-ADM-553 [7]. Labelled as "Hydrostatic Pressure Test" in Table 4.1-10 [28].
(3) Transient numbers from Table 3-1 of SIR-00-089 [1].
(4) Not expected to have any additional cycles on RSGs.
(5) Applies to Pressurizer only.
(6) Applies to Steam Generator only. Labelled as "Secondary Leak Test" in 0-ADM-553 [7].
(7) Limited by Reactor Coolant Pump Analysis [16, Attachment 1, pages 44 and 45].
(8) Leak Test Procedure cancelled per [30].
(9) 60-year projections from [5, PTN-LR-00-0127 Table 10.3-1].
(10) Not expected to have any additional cycles on RSGs.
(11) Cycle limits for baffle-former bolts only is being lowered from 14,500 to 2,200 due EPU RCS conditions (Table 4.1-8 of UFSAR [8]).
(12) Not counted, not significant contributor to fatigue usage factor.
(13) 80-year plant life projected cycles computed using 66 years of life for the RSGs.
(14) Values are [(pre-and post- 1987) / (post- 1987)] cycles [5, PTN-LR-00-0127 Table 10.3-1].
(15) Not counted, intermittent slug feeding at hot standby not performed.
(16) Limit of 2,200 cycles established for baffle former bolts only per UFSAR Table 4.1-8 [8].
(17) Represents 200 cycles each of: (1) pressurizer cooldown cycles at ≤ 200°F/hr from nominal pressure and (2) pressurizer cooldown cycles at ≤ 200°F/hr from 400 psia [28].
(18) Spray water temperature differential to 560°F.
(19) Applies to Steam Generator only. Represents pre-operational test [16, Note 3 on Attachment 1, pages 44 and 45].
(20) Adjustment in 60-year projection in [5, PTN-LR-00-0127 Table 10.3-2] - recorded as a value of 0 when 1 was assumed in pre-operational startup.
(21) One cycle is projected for 80 years to remain within the analytical basis if that event occurs.
(22) One cycle of 10 events is projected for 80 years to remain within the analytical basis if that event occurs.
(23) Recommended revision 0-ADM-553 to align with UFSAR Table 4.1-10.

THIS REPORT CONTAINS VENDOR PROPRIETARY INFORMATION

3.2 PTN EAF Assessment

For SLR, PTN-specific EAF calculations were performed for the limiting PTN components. All components with CUF values were considered in the EAF assessment so that the plant-specific limiting locations were addressed by the assessment to be consistent with GALL-SLR guidance. The assessment serves as an update and replacement to the EAF assessment performed for 60 years that is summarized in Table 2-2.

The PTN EAF assessment for SLR was performed using the applicable guidance from the SRP-SLR [4]. Section 4.3.2.1.2, "Components Evaluated for CUF_{en}," of the SRP-SLR for components evaluated for CUF_{en} states the following:

> Applicants should also include CUF_{en} calculations for additional component locations if they are considered to be more limiting than those previously evaluated. This sample set includes the locations identified in NUREG/CR–6260 and additional plant-specific component locations in the reactor coolant pressure boundary if they may be more limiting than those considered in NUREG/CR–6260. Plant-specific justification can be provided to demonstrate that calculations for the NUREG/CR–6260 locations do not need to be included. Environmental effects on fatigue for these critical components can be evaluated using the positions described in Regulatory Guide (RG) 1.207, Revision 1[1]; NUREG/CR–6909, Revision 0 (with "average temperature" used consistent with the clarification that was added to NUREG/CR–6909, Revision 1); or other subsequent NRC-endorsed alternatives.
>
> 1 If and when published as RG 1.207, Revision 1 Final.

For PTN, the older-vintage Westinghouse PWR plant is applicable; in fact, the "Older Vintage Westinghouse Plant" evaluated in Section 5.5 of NUREG/CR-6260 is directly relevant to PTN because the design codes, analytical approach and techniques used for that plant matches those used for PTN. In addition, the evaluated transient cycles matched or bounded PTN. Therefore, the evaluation from NUREG/CR-6260 is directly applicable to PTN.

Based on the foregoing discussion, the updated SLR EAF assessment for PTN was performed as follows:

- The plant-specific NUREG/CR-6260 locations were reevaluated for SLR.
- To ensure that any locations that may be more limiting than the NUREG/CR- 6260 locations were addressed, the reactor coolant pressure boundary (RCPB) components with existing ASME Code fatigue analyses CUFs presented in Tables 2-2 and 3-3 were evaluated for EAF for SLR.
- Revised plant-specific EAF multipliers applicable for SLR were calculated based on the latest F_{en} methods using the guidance in NUREG/CR–6909, Revision 1 [6].

The SLR EAF assessment results from Appendix A are summarized in Table 3-4. These results are discussed in Section 3.4.

Table 3-3: Other Class 1 RCPB Components with a CUF Value (1)

Component	Location	60-Year CLB CUF
RPV	Head Flange	0.083
	Vessel Flange	0.531
	Closure Studs	0.81
	CRDM Housing -- J-weld	0.730
	CRDM Housing -- Bi-metallic Weld[1]	0.620
	Vent Nozzle	0.49
	Shell-to-Shell Juncture	0.034
	Bottom Head-to-Shell Juncture	0.023
	Bottom Mounted Instrumentation Nozzles	0.002
	Core Support Pads	0.020
Control Rod Drive Mechanisms (CRDMs)	Latch Housing	{ }
	Rod Travel Housing	{ }
	Cap	{ }
	Lower Joint	{ }
	Middle Joint	{ }
	Upper Joint	{ }
Steam Generators (S/Gs) (Primary Side)	Divider Plate	{ }
	Primary Chamber, Tubesheet and Stub Barrel Complex	{ }
	Tube-to-Tubesheet Weld	{ }
	Tubes	{ }
Steam Generators (S/Gs) (Secondary Side)	Upper Shell Drain	{ }
	Feedwater Nozzle	{ }
	Secondary Manway Bolts	{ }
	Upper Shell Remnants	{ }
	Secondary Hand-Hole & Inspection Port	{ }
	Steam Outlet Nozzle Flow Limiters	{ }
Reactor Coolant Pumps (RCPs)	Main Flange Studs	0.29
	Main Flange	0.025
	Casing	0.001
Reactor Internals	Upper Support Plate	{ }
	Deep Beam	{ }
	Upper Core Plate	{ }
	Upper Core Plate Alignment Pins	{ }
	Upper Support Columns	{ }
	Lower Support Plate	{ }
	Lower Support Plate to Core Barrel Weld	{ }
	Lower Core Plate	{ }
	Lower Support Columns	{ }
	Core Barrel Flange	{ }
	Core Barrel Outlet Nozzle	{ }
	Radial Keys and Clevis Insert Assembly	{ }

Footnote for Table 3-3:

1. From Table 2-2 of Reference [29].

Table 3-4: Summary of the PTN EAF Assessment for SLR (80 Years of Operation)

Component (1)	80-Year CUF (2)	F_{en} (3)	Material (4)	80-Year CUF_{en} (5)
RPV Shell at Core Support Pads	0.509 (24) / { } (8)	6.28 / { } (8)	Low alloy steel	3.197 / 0.910 (27)
RPV Inlet Nozzle (inside surface)	0.066 (24)	6.28	Carbon steel with stainless steel clad and a stainless steel weld butter safe end	0.414
RPV Outlet Nozzle (inside surface)	0.063 (24)	6.28	Carbon steel with stainless steel clad and a stainless steel weld butter safe end	0.396
Head Flange	0.083 (25)	6.28	Low Alloy Steel (23)	0.521
Vessel Flange	0.531 (25) / { } (20)	6.28 / { } (20)	Low alloy steel (23)	3.333/ 0.373 (26)
CRDM Housing - J-Groove Weld (14)	0.73 (25) / { } (15)	3.75 (21) / { } (15)	Inconel	2.738/ 0.299 (15)
CRDM Housing - Bi-Metallic (Nozzle-to-Adapter) Weld	0.62 (25) / { } (18)	3.75 / { } (18)	Inconel	2.323 / 0.646 (18)
Vent Nozzle (14)	0.49 (25) / { } (16)	3.75 (21) / { } (16)	Inconel	1.84 / 0.230 (16)
Shell-to-Shell Juncture	0.034 (25)	12.81	Carbon steel with stainless steel clad	0.436
Bottom Head-to-Shell Juncture	0.023 (25)	12.81	Carbon steel with stainless steel clad	0.295
Bottom Mounted Instrumentation Nozzles	0.002 (25)	12.81	Stainless steel	0.026
Core Support Pads	0.020 (25)	3.75	Inconel	0.075
RCS Piping Safety Injection Nozzle	0.046 (24)	12.81	Stainless steel	0.589
RCS Piping Charging Nozzle	0.030 (24)	12.81	Stainless steel	0.384
RCS Piping Residual Heat Removal (RHR) Piping	0.022 (24)	12.81	Stainless steel	0.282

Table continued on next page.

THIS REPORT CONTAINS VENDOR PROPRIETARY INFORMATION

Table 3-4: Summary of the PTN EAF Assessment for SLR (80 Years of Operation) (continued)

Component (1)	80-Year CUF (2)	F_{en} (3)	Material (4)	80-Year CUF_{en} (5)
Pressurizer Spray Nozzle	{ } (24) / 0.0721 (11)	12.81 / 6.80 (11)	Carbon steel with stainless steel clad, stainless steel thermal sleeve, and a stainless steel safe end	{ } / 0.4904 (7)(11)
Pressurizer Safety and Relief Nozzle	{ } (24)	12.81	Carbon steel with stainless steel clad and a stainless steel safe end	{ }
Pressurizer Lower Head, Heater Well	{ } (24) / { } (12)	12.81 / { } (12)	Carbon steel with stainless steel clad and J-groove weld cover filet	{ } / 0.093 (12)
Pressurizer Lower Head/ Perforation	{ } (24)	12.81	Carbon steel with stainless steel clad	{ }
Pressurizer Upper Head and Shell	{ } (24) / { } (8)	6.28 / { } (8)	Low alloy steel	{ } / 0.974 (27)
Pressurizer Support Skirt/Flange	0.0165 (24)	6.28	Carbon steel	0.104
Pressurizer Manway Pad	{ } (24)	6.28	Carbon steel	{ }
Pressurizer Manway Cover	0.0 (24)	6.28	Carbon steel	0.000
Pressurizer Manway Bolts	0.0 (24)	6.28	Low Alloy Steel	0.000
Pressurizer Welded Manway Diaphragm	{ } (24)	3.75	Nickel Alloy	{ }
Pressurizer Instrument Nozzle	{ } (24)	12.81	Stainless steel	{ }
Pressurizer Immersion Heater	{ } (24)	12.81	Stainless steel	{ }
CRDM Latch Housing	{ } (25) / { } (19)	12.81/ { } (19)	Stainless steel	{ } / 0.250 (19)
CRDM Rod Travel Housing	{ } (25)	12.81	Stainless steel	{ }
CRDM Cap	{ } (25)	12.81	Stainless steel	{ }
CRDM Lower Joint	{ } (25) / { } (17)	12.81/ { } (17)	Stainless Steel	{ } / { } (17)
CRDM Middle Joint	{ } (25)	12.81	Stainless Steel	{ }
CRDM Upper Joint	{ } (25)	12.81	Stainless Steel	{ }

Table continued on next page.

THIS REPORT CONTAINS VENDOR PROPRIETARY INFORMATION

Table 3-4: Summary of the PTN EAF Assessment for SLR (80 Years of Operation) (concluded)

Component (1)	80-Year CUF (2)	F_{en} (3)	Material (4)	80-Year CUF_{en} (5)
S/G Divider Plate	{ } (25) / { } (9)	3.75 / { } (9)	Inconel	{ } / 0.881 (28)
S/G Primary Chamber, Tubesheet and Stub Barrel Complex	{ } (25)	12.81	Carbon steel with stainless steel clad	{ }
S/G Tube-to-Tubesheet Weld (13)	{ } (25)	3.75	Inconel	{ } (13)
S/G Tubes	{ } (25) / { } (9)	3.75 / { } (9)	Inconel	{ } / 0.903 (28)
RCP Main Flange	0.025 (25)	12.81	Unknown -- use maximum	0.320
RCP Casing	0.001 (25)	12.81	Unknown -- use maximum	0.013

THIS REPORT CONTAINS VENDOR PROPRIETARY INFORMATION

Footnotes for Table 3-4:

1. Components are those components in Table 2-2.
2. "CUF (for 80 Years)" from Table 2-2. Based on the transient counts and severities remaining valid for SLR (see Section 3.1), the 80-Year CUF value remains equal to the 60-Year CUF value unless noted otherwise.
3. The maximum F_{en} value for the material.
4. The component material corresponding to the maximum F_{en} value obtained from [28].
5. The allowable value for CUF_{en} is 1.0.
6. NOT USED
7. The value of CUF_{en} shown results from evaluation of one (1) Inadvertent Auxiliary Spray (IAS) transient. Evaluation of four (4) IAS transients results in a CUF_{en} = 0.9036.
8. From [32].
9. From [33].
10. NOT USED
11. CUF and CUF_{en} values from [30.c]. Calculation contains vendor proprietary references.
12. CUF and CUF_{en} values from [31.c]. Calculation contains vendor proprietary references.
13. S/G Tube-to-Tubesheet Weld will be included in the S/G inspection program. Accordingly, the S/G Tube-to-Tubesheet Weld is not included in the SLRA.
14. The location with the highest CUF_{en} that is wetted by reactor coolant will be identified and values provided in lieu of the locations currently listed.
15. { } [38].
16. { } [37].
17. From [40]. Bounding analysis.
18. From [35].
19. From [36].
20. From [39].
21. Although the location is not wetted, a F_{en}. value of 3.75 was used.
22. Not Available (NA).
23. Head Flange is SA-508 Class 2 material and Vessel Flange is SA-508 Class 3 material [34]
24. From Table 2-2.
25. From Table 3-3.
26. From [39].
27. From [23].
28. From [27]

3.3 Plant-Specific Limiting Locations

Consistent with the SRP-SLR requirements discussed in Section 3.2, Chapter X.M1 of the GALL-SLR Report [3] states, in part, the following:

> *CUF_{en} is CUF adjusted to account for the effects of the reactor water environment on component fatigue life. For a plant, the effects of reactor water environment on fatigue are evaluated by assessing a set of sample critical components for the plant. Examples of critical components are identified in NUREG/CR-6260; however, plant-specific component locations in the reactor coolant pressure boundary may be more limiting than those considered in NUREG/CR-6260, and thus should also be considered.*

The locations evaluated for EAF in Section 3.2 for PTN represent the limiting locations from a CUF perspective per GALL-SLR Chapter X.M1 requirements for PTN based on the following:

- In the PTN 60-year LRA, the plant-specific NUREG/CR-6260 locations for the older-vintage Westinghouse plant were evaluated. The older-vintage Westinghouse plant evaluated in NUREG/CR-6260 is directly relevant to PTN because the design codes, analytical approach and techniques used match those used at PTN. In addition, the evaluated transient cycles matched or bounded PTN. In all the CUF calculations documented in Section 5.5, "Older Vintage Westinghouse Plant," of NUREG/CR-6260, EAF evaluation was consistently performed for "the locations of highest design CUF." Therefore, use of maximum plant-specific F_{en} values, coupled with these highest CUF values, ensures bounding CUF_{en} values for PTN.
- Based on the comprehensive review of the PTN fatigue bases that was recently performed for SLR [29], all Class 1 locations evaluated for CUF were evaluated in the EAF assessment in Appendix A except for those locations that are not part of the RCPB or those locations that are not exposed to a water environment such that EAF effects do not apply, as follows:
  - All surge line locations were excluded from CUF_{en} calculations because they were evaluated for EAF via inspection management, as discussed in Section 2.3. This

includes the RCS Piping Surge Line Hot Leg Nozzle and the Pressurizer Surge Nozzle.

- The following locations are not exposed to the environment so EAF assessment is not required: The RPV Closure Studs and the RCP Main Flange Studs.
- The following locations are not part of the RCPB: All S/G Secondary Side locations and all Reactor Internals locations. Only RCPB locations require EAF assessment per Section 4.3.2.1.2, "Components Evaluated for CUF_{en}" of the SRP-SLR [4].

- For each CUF value evaluated for EAF, bounding F_{en} multipliers were used for the materials present in the component location. In cases where the materials were not known, F_{en} values for all three F_{en} material groupings (carbon and low alloy steels, stainless steels, and nickel alloys) were determined and the maximum multiplier was used.

Therefore, the SLR EAF assessment for PTN satisfies GALL-SLR Chapter X.M1 guidance for components selection as the PTN EAF assessment considers the NUREG/CR-6260 locations as well as all other plant-specific limiting locations.

3.4 Summary of PTN EAF Assessment

The following summarizes SLR EAF assessment for PTN:

- Based on an 80-year transient evaluation performed for SLR, the same transients, transient severity and transient counts used for the 60-year EAF assessment are bounding for 80 years of operation. Therefore, the 60-year CLB CUF values remain unchanged and were used in the updated EAF assessment for 80 years. These CUF values use design transients counts for all locations and reflect power uprate.
- The SLR EAF assessment for PTN considers the NUREG/CR-6260 locations, as well as all other plant-specific limiting locations with a CUF calculation that are exposed to a water environment and are part of the RCPB. Therefore, it satisfies GALL Chapter X.M1 and related SRP-SLR guidance for component selection.

THIS REPORT CONTAINS VENDOR PROPRIETARY INFORMATION

- The PTN EAF assessment uses plant-specific F_{en} values, coupled with the highest CUF values, to determine bounding CUF_{en} values.
- Revised plant-specific EAF multipliers applicable for SLR were calculated that generally make use of the latest F_{en} methods provided in RG 1.207, Revision 1 [5]. As noted in Appendix A, the PTN CUF values still reflect the use of the fatigue curves from the applicable Section III used in each component's CUF calculation because detailed fatigue tables for each PTN component locations are not available. Therefore, the guidance of Section C.1.1 of RG 1.207, Revision 1 [5] (for carbon and low alloy steel), Section C.2.1 (for SS), and Section C.3.1 (for Ni-Cr-Fe alloys) could not be fully implemented in the calculations for all locations. This shortcoming should be addressed as part of the resolution of high CUF_{en} values discussed below.
- The CUF_{en} values calculated using the ASME Code fatigue curves of record for each location are above the allowable of 1.0 for 13 locations: (1) RPV Shell at Core Support Pads, (2) RPV Vessel Flange, (3) RPV CRDM Housing – J-Groove Weld, (4) RPV CRDM Housing – Bi-Metallic Weld, (5) RPV Vent Nozzle, (6) Pressurizer Spray Nozzle, (7) Pressurizer Lower Head Heater Well, (8) Pressurizer Upper Head and Shell, (9) CRDM Latch Housing, (10) CRDM Lower Joint, (11) S/G Divider Plate, (12) S/G Tube-to-Tubesheet Weld and (13) S/G Tubes. The S/G Tube-to-Tubesheet Weld is no longer subject to further EAF assessment consistent with the SRP-SLR [4] and is excluded from this list because it is no longer part of the reactor coolant pressure boundary. This location is being managed with a permanently approved H* alternate repair criteria for both the hot- and cold-leg sides of the steam generator. For the 12 remaining locations that must be managed for EAF effects, FPL should satisfactorily assess EAF using one or more of the following options:
    - Further refinement of the EAF analysis to lower the CUF_{en} values to below 1.0, or
    - repair of the affected locations, or
    - replacement of the affected locations, or
    - management of EAF effects using an inspection program that has been reviewed and approved by the NRC (the approach intended for the S/G Tube-to-Tubesheet Weld).

To fully satisfy GALL-SLR AMP requirements, FPL should complete one of these options at least two years prior to the period of SLR operation. If the last option is selected, FPL should provide the NRC with the inspection details of the AMP requiring staff approval prior to the period of SLR operation.

- The components shown in Table 3-3 with 80-year CUF_{en} values less than 1.0 are acceptable for 80 years of operation.
- Some of the EAF-analyzed components required refined analysis techniques and use 80-year projected cycles for selected transients to achieve a CUF_{en} value below 1.0. Table 3-5 provides which 80-year projected transients and their minimum values which have been used in the EAF analyses.
- The PTN EAF assessment should be coupled with the PTN Cycle Counting Program to verify the continued acceptability of all EAF analyses through cycle counting and periodically updated evaluations, if necessary, to demonstrate that they continue to meet the appropriate limits throughout the SLR period.

The PTN-specific EAF assessment and the above recommendations to use updated fatigue curves and resolve unacceptable CUF_{en} values serve as an acceptable AMP that satisfies GALL-SLR Report, Chapter X.M1 guidance to manage SCs that are the subject of fatigue TLAAs in accordance with the requirements in 10 CFR 54.21(c)(1)(iii).

THIS REPORT CONTAINS VENDOR PROPRIETARY INFORMATION

Table 3-5: Minimum Cycles Used in EAF Analyses

Component	Design Number	Pressurizer Spray Nozzle	Pressurizer Lower Head	RSG Divider Plate	RSG Tubes	Pzr Upper Head and Shell	RV Core Support Blocks	CRDM Bi-Metallic (Nozzle-to-Adapter) Weld	CRDM Latch Housing	RVCH CRDM Nozzle and J-Groove Weld	Vessel Flange	Minimum Cycles	CRDM Lower Joint (from RVCH installation ot 80 years for bounding Unit 4)
Station Heatup at 100°F/hour	200	181	181	181	181			181	181	181	181	181	92
Station Cooldown at 100°F/hour	200	181	181	181	181			181		181	181	181	
Station Loading at 5% power per minute	14500	484		533	533	533			533	533	533	484	
Station Unloading at 5% power per minute	14500	450		533	533	451				451	440	440	
Step Load Increase of 10% of Full Power	2000	82		164	164							82	82
Step Load Decrease of 10% of Full Power	2000	106		164	164							106	
Step Load Decrease of 50% of Full Power	200	51								82	82	51	82
Loss of Load without Immediate Turbine Trip or	80	27								28	28	27	13
Loss of Flow in One Reactor Coolant Loop	80	21								26	26	21	
Inadvertent Auxiliary Spray	10	2 [(1)]										0	
Hydrostatic Test at 2485 psig Pressure and 400°F	5	3					2					2	
Primary to Secondary Side Leak Test to 2435 psi	150	4										4	
Loss of AC Power	40									19	19	19	8
Reactor Trip	400							311		292	292	292	
OBE	50							10	10			10	
Rod Trips [(2)]	2600								2000			2000	144

Footnotes for Table 3-5:

(1) One (1) occurrence of the IAS transient is specified in Tables 3-1 and 3-2. Evaluation of the pressurizer spray nozzle shows that evaluation of two (2) IAS transients results in an acceptable $CUF_{en} \leq 1.0$ value [30.c].

(2) Transient not counted.

4.0 SUMMARY AND CONCLUSIONS

This report addresses TLAAs associated with EAF for both PTN units. As such, this report documents the EAF TLAAs necessary to support operating license renewals from 60 to 80 years for both PTN units that satisfy all the requirements specified by the NRC for SLR.

Section 2.0 summarizes PTN's TLAAs on fatigue that were resolved for 60 years of operation by analyses that used a combination of three demonstrations: (i) fatigue monitoring of all relevant plant thermal transients to confirm that the original transient design limits remain valid for the 60-year operating period (i.e., the PTN Cycle Counting Program), (ii) reactor water environmental effects on fatigue life using the most recent data from laboratory simulation of the reactor coolant environment (i.e., the PTN EAF Assessment), and (iii) inspection of the surge line welds under the ASME Code Section XI inspection program to verify the absence of fatigue cracking (i.e., the PTN Surge Line Inspection Program). Collectively, these three demonstrations form the EAF CLB for PTN and represent the inputs to the PTN-specific program for addressing EAF effects for SLR.

Section 3.0 contains the plant-specific assessment of EAF effects for PTN for SLR. The plant-specific assessment is separated into four parts, the PTN Cycle Counting Program, the PTN EAF Assessment, discussion of the plant-specific limiting locations evaluated in the PTN EAF Assessment, and a summary of the PTN EAF Assessment and how it satisfies NRC requirements for SLR. Collectively, these four parts serve as a replacement to the PTN 60-year EAF assessment that reflects an updated assessment applicable to SLR. The PTN EAF assessment should be coupled with the PTN Cycle Counting Program to verify the continued acceptability of all EAF analyses through cycle counting and periodically updated evaluations, if necessary, to demonstrate that they continue to remain valid and meet the appropriate limits throughout the SLR period. The PTN-specific EAF assessment and the associated recommended options for resolving unacceptable CUF_{en} values serve as an acceptable AMP that satisfies GALL-SLR Report, Chapter X.M1 guidance to manage SCs that are the subject of EAF TLAAs in accordance with the requirements in 10 CFR 54.21(c)(1)(iii).

Based on the evaluation and results presented in this report, the TLAAs for EAF are adequately evaluated for PTN, and the potential effects of the reactor water environment have been properly evaluated, as required by the GALL-SLR [3]. Therefore, the proposed approach described in this report is recommended for inclusion in FPL's SLRA for PTN to address EAF.

THIS REPORT CONTAINS VENDOR PROPRIETARY INFORMATION

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THIS REPORT CONTAINS VENDOR PROPRIETARY INFORMATION

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APPENDIX A

F_{EN} CALCULATIONS

THIS REPORT CONTAINS VENDOR PROPRIETARY INFORMATION

This appendix contains the details of the environmentally adjusted cumulative usage factor (CUF_{en}) calculations for Turkey Point Nuclear Plant, Units 3 and 4 (PTN) for Subsequent License Renewal (SLR) operation out to 80 years.

Chapter X.M1 of the GALL-SLR Report [3] states, in part, the following:

> *CUF_{en} is CUF adjusted to account for the effects of the reactor water environment on component fatigue life. For a plant, the effects of reactor water environment on fatigue are evaluated by assessing a set of sample critical components for the plant. Examples of critical components are identified in NUREG/CR-6260; however, plant-specific component locations in the reactor coolant pressure boundary may be more limiting than those considered in NUREG/CR-6260, and thus should also be considered. Environmental effects on fatigue for these critical components may be evaluated using the guidance in Regulatory Guide (RG) 1.207, Revision 1[1]; alternatively, the bases in NUREG/CR-6909, Revision 0 (with "average temperature" used consistent with the clarification that was added to NUREG/CR-6909, Revision 1); or other subsequent U.S. Nuclear Regulatory Commission (NRC)-endorsed alternatives.*
>
> 1 If and when published as RG 1.207, Revision 1 Final.

Consistent with this guidance, environmental adjustment factor (F_{en}) calculations are performed in this appendix for PTN for SLR using the methods documented in RG 1.207, Revision 1 [5].

Section C of RG 1.207, Revision 1 refers to the equations in Appendix A of NUREG/CR-6909, Revision 1 for calculating F_{en} values. Therefore, the F_{en} equations from Appendix A of NUREG/CR-6909, Revision 1 [6] are summarized here and used in updated PTN CUF_{en} calculations.

<u>Carbon and Low Alloy Steels</u>

The nominal environmental fatigue adjustment factor for both carbon and low-alloy steels, $F_{en\text{-}CS\text{-}LAS}$, is expressed as:

$$F_{en\text{-}CS\text{-}LAS} = \exp\left((0.003 - 0.031\ \dot{\varepsilon}^*)\ S^*\ T^*\ O^*\right) \qquad \text{(Eqn. A-1)}$$

where S*, T*, O*, and $\dot{\varepsilon}^*$ are transformed sulfur (S) content, transformed material temperature (T), transformed dissolved oxygen (DO) level, and transformed strain rate ($\dot{\varepsilon}$), respectively, defined as:

$$S^* = 2.0 + 98\ S \qquad (S \leq 0.015\text{ wt. \%})$$
$$S^* = 3.47 \qquad (S > 0.015\text{ wt. \%}) \qquad \text{(Eqn. A-2)}$$

$$T^* = 0.395 \qquad (T < 150\ °C)$$
$$T^* = (T - 75)/190 \qquad (150\ °C \leq T \leq 325\ °C) \qquad \text{(Eqn. A-3)}$$

$$O^* = 1.49 \qquad (DO < 0.04\text{ ppm})$$
$$O^* = \ln(DO/0.009) \qquad (0.04\text{ ppm} \leq DO \leq 0.5\text{ ppm})$$
$$O^* = 4.02 \qquad (DO > 0.5\text{ ppm}) \qquad \text{(Eqn. A-4)}$$

$$\dot{\varepsilon}^* = 0 \qquad (\dot{\varepsilon} > 2.2\%/s)$$
$$\dot{\varepsilon}^* = \ln(\dot{\varepsilon}/2.2) \qquad (0.0004\%/s \leq \dot{\varepsilon} \leq 2.2\%/s)$$
$$\dot{\varepsilon}^* = \ln(0.0004/2.2) \qquad (\dot{\varepsilon} < 0.0004\%/s) \qquad \text{(Eqn. A-5)}$$

A threshold value of 0.07% for strain amplitude (one-half the strain range for the cycle, ε_a) is defined, below which environmental effects on the fatigue lives of these steels do not occur. Thus, $F_{en\text{-}CS\text{-}LAS}$ is equal to 1.0 when ε_a is less than or equal to 0.07%.

Section C.1.1 of RG 1.207, Revision 1 [5] states that the CUF for carbon and low alloy steel components should be computed using the design fatigue curves provided in Figures A.1 and A.2 and Table A.1 in Appendix A to NUREG/CR-6909, Revision 1, or, alternatively, using the fatigue design curve for carbon and low-alloy steel in Appendix I to Section III of the 2013 Edition of the ASME Code.

Stainless Steels

The nominal environmental fatigue adjustment factor for wrought and cast austenitic stainless steels (SSs), $F_{en\text{-}SS}$, is expressed as:

$$F_{en\text{-}SS} = \exp(-T^* \, O^* \, \dot{\varepsilon}^*) \qquad \text{(Eqn. A-6)}$$

where T^*, O^*, and $\dot{\varepsilon}^*$ are transformed material temperature (T), transformed dissolved oxygen (DO) level, and transformed strain rate ($\dot{\varepsilon}$), respectively, defined as:

$$\begin{aligned} T^* &= 0 && (T \leq 100\ °C) \\ T^* &= (T - 100)/250 && (100\ °C \leq T \leq 325\ °C) && \text{(Eqn. A-7)} \end{aligned}$$

$$\begin{aligned} \dot{\varepsilon}^* &= 0 && (\dot{\varepsilon} > 7\%/s) \\ \dot{\varepsilon}^* &= \ln(\dot{\varepsilon}/7) && (0.0004\%/s \leq \dot{\varepsilon} \leq 7\%/s) \\ \dot{\varepsilon}^* &= \ln(0.0004/7) && (\dot{\varepsilon} < 0.0004\%/s) && \text{(Eqn. A-8)} \end{aligned}$$

For all wrought and cast SSs and heat treatments, SS weld metals, and sensitized high-carbon wrought and cast SSs:

$O^* = 0.29$ (for any DO level)

For all wrought SSs except sensitized high-carbon SSs:

$O^* = 0.14$ ($DO \geq 0.1$ ppm) (Eqn. A-9)

A threshold value of 0.10% for ε_a is defined, below which environmental effects on the fatigue lives of these steels do not occur. Thus, $F_{en\text{-}SS}$ is equal to 1.0 when ε_a is less than or equal to 0.10%.

Section C.2.1 of RG 1.207, Revision 1 [5] states that the CUF for SS components should be computed using the design fatigue curves provided in Figure A.3 and Table A.2 in Appendix A to NUREG/CR-6909, Revision 1.

Nickel Alloys

The nominal environmental fatigue adjustment factor for Ni-Cr-Fe steels, $F_{en\text{-}Ni}$, is expressed as:

$$F_{en\text{-}Ni} = \exp(-T^*\, O^*\, \dot{\varepsilon}^*) \qquad \text{(Eqn. A-10)}$$

where T^*, O^*, and $\dot{\varepsilon}^*$ are transformed material temperature (T), transformed dissolved oxygen (DO) level, and transformed strain rate ($\dot{\varepsilon}$), respectively, defined as:

$$T^* = 0 \qquad (T \leq 50\ °C)$$
$$T^* = (T - 50)/275 \qquad (50\ °C \leq T \leq 325\ °C) \qquad \text{(Eqn. A-11)}$$

$$\dot{\varepsilon}^* = 0 \qquad (\dot{\varepsilon} > 5.0\%/s)$$
$$\dot{\varepsilon}^* = \ln(\dot{\varepsilon}/5.0) \qquad (0.0004\%/s \leq \dot{\varepsilon} \leq 5.0\%/s)$$
$$\dot{\varepsilon}^* = \ln(0.0004/5.0) \qquad (\dot{\varepsilon} < 0.0004\%/s) \qquad \text{(Eqn. A-12)}$$

$$O^* = 0.06 \qquad (DO \geq 0.1\ ppm)$$
$$O^* = 0.14 \qquad (DO < 0.1\ ppm) \qquad \text{(Eqn. A-13)}$$

A threshold value of 0.10% for ε_a is defined, below which environmental effects on the fatigue lives of these steels do not occur. Thus, $F_{en\text{-}Ni}$ is equal to 1.0 when ε_a is less than or equal to 0.10%.

Section C.3.1 of RG 1.207, Revision 1 [5] states that the CUF for Ni-Cr-Fe alloy components should be computed using the design fatigue curves provided in Figure A.3 and Table A.2 in Appendix A to NUREG/CR-6909, Revision 1, or, alternatively, the fatigue design curve for Ni-Cr-Fe alloys in Section III of the 2013 Edition of the ASME Code may be used.

CUF_{en}

The environmentally adjusted cumulative usage factor for 80 years is computed as:

$$CUF_{en\text{-}80} = F_{en}\, CUF_{80} \qquad \text{(Eqn. A-14)}$$

where:

$CUF_{en\text{-}80}$ = environmentally adjusted cumulative usage factor for 80 years of operation

THIS REPORT CONTAINS VENDOR PROPRIETARY INFORMATION

F_{en} = environmental adjustment factor for the specific component material using Equation A-1 for carbon and low alloy steels, Equation A-6 for SSs, or Equation A-10 for Ni-Cr-Fe alloys.

CUF_{80} = cumulative usage factor for 80 years of operation

PTN-Specific $CUF_{en\text{-}80}$ Calculations

The following is considered with respect to the PTN-specific $CUF_{en\text{-}80}$ calculations:

- In the absence of detailed fatigue tables for each PTN component locations, the following inputs were used for all calculations:
  - **Dissolved Oxygen, DO**. A DO value that bounds normal operating conditions and represents the controlled value of 5 parts per billion (ppb), or 0.005 parts per million (ppm), from the PTN Water Chemistry Program (FPL Procedure No. 0-ADM-651) [28, excerpted] is considered in the calculations to yield bounding F_{en} values. The threshold DO level of 0.040 ppm is used in the carbon and low alloy steel F_{en} calculations, the threshold DO level of 0.1 ppm is used in the Ni-Cr-Fe alloy F_{en} calculations, and the DO level does not affect the SS F_{en} calculations for pressurized water reactors (PWRs). Therefore, the PTN controlled value for the DO level is well removed from any levels that would affect the F_{en} calculations. In addition, PTN's FPL's procedure requires that a Condition Report be initiated to take remedial actions if this level is exceeded.
  - **Temperature, T**. A maximum temperature value of 617 °F (325 °C) was used to yield bounding F_{en} values, which represents the maximum RCS temperature that all components may be exposed to for plant operations.
  - **Sulfur Content, S**. In the absence of Certified Material Test Reports (CMTRs) for all carbon and low alloy steel components, the F_{en} upper bound S threshold of 0.015 wt. % for Equation A-2 is used to yield bounding F_{en} values.
  - **Strain Rate, $\dot{\varepsilon}$**. In the absence of detailed transient load pair information, the bounding $\dot{\varepsilon}$ of 0.0004 %/sec is used in Equations A-5, A-8, and A-12 to yield bounding F_{en} values.

THIS REPORT CONTAINS VENDOR PROPRIETARY INFORMATION

- F_{en} calculations are performed for each of the materials present in each component, and the maximum F_{en} for all materials in each component is used to compute bounding, component-specific $CUF_{en\text{-}80}$ values. The F_{en} value is computed from the CUF_{en} and CUF values where they are available. In cases where the materials are not known, F_{en} values for all three F_{en} material groupings (carbon and low alloy steels, stainless steels, and nickel alloys) are determined and the maximum multiplier is used. The maximum F_{en} values used are:
    - Stainless Steel – 12.81
    - Ni-Cr-Fe –3.75
    - Carbon / Low Alloy Steel – 6.28
- The PTN CUF values still reflect the use of the fatigue curves from the applicable Section III Code of record for each location because detailed fatigue tables for each PTN component locations are not available. Therefore, the guidance of Section C.1.1 of RG 1.207, Revision 1 [5] (for carbon and low alloy steel), Section C.2.1 (for SS), and Section C.3.1 (for Ni-Cr-Fe alloys) could not be fully implemented in the calculations.