UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	Docket Nos. 50-247-LR and 50-286-LR
ENTERGY NUCLEAR OPERATIONS, INC.	)	ASLBP No. 07-858-03-LR-BD01
(Indian Point Nuclear Generating Units 2 and 3)	)	

APPLICANT'S MOTION FOR SUMMARY DISPOSITION OF NEW YORK STATE CONTENTIONS 26/26A & RIVERKEEPER TECHNICAL CONTENTIONS 1/1A (METAL FATIGUE OF REACTOR COMPONENTS)

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August 25, 2010

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	Commitment #33 Regarding the Fatigue Monitoring Program" (Aug. 9, 2010)
15	Westinghouse Electric Co., WCAP-17199-P, Revision 0, *Environmental Fatigue Evaluation for Indian Point Unit 2* (June 2010) **(Entergy-Designated Confidential Proprietary Document Subject to Nondisclosure Agreement and ASLB 9/4/2009 Protective Order)**
16	Westinghouse Electric Co., WCAP-17200-P, Revision 0, *Environmental Fatigue Evaluation for Indian Point Unit 3* June 2010) **(Entergy-Designated Confidential Proprietary Document Subject to Nondisclosure Agreement and ASLB 9/4/2009 Protective Order)**

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Pursuant to 10 C.F.R. § 2.1205, Entergy Nuclear Operations, Inc. ("Entergy") seeks summary disposition of consolidated New York State ("NYS") Contentions 26/26A ("NYS-26/26A") and Riverkeeper, Inc. ("Riverkeeper") Technical Contentions 1/1A ("TC-1/1A") (collectively "Consolidated Contention"). As admitted by the Atomic Safety and Licensing Board ("Board"), NYS-26/26A and TC-1/1A allege that Entergy's license renewal application ("LRA")[1] for Indian Point Nuclear Generating Unit 2 and Unit 3 ("IP2" and "IP3"; collectively "Indian Point Energy Center" or "IPEC") does not include an adequate aging management program ("AMP") for metal fatigue because Entergy has not performed the environmentally-assisted fatigue ("EAF") evaluations purportedly required as a condition precedent to license renewal. For the reasons that follow, NYS-26/26A and TC-1/1A should be dismissed as a matter of law.[2]

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1 License Renewal Application (Apr. 23, 2007), *available at* http://www.nrc.gov/reactors/operating/licensing/renewal/applications/indian-point.html (follow links).

2 This Motion is supported by (1) a Statement of Material Facts as to which Entergy asserts there is no genuine dispute (Attach. 1); (2) the Declaration of Nelson F. Azevedo in Support of Entergy's Motion for Summary Disposition of New York State Contention 26/26A and Riverkeeper Technical Contention 1/1A (Metal Fatigue) (Attach. 2); and (3)

I. PRELIMINARY STATEMENT

Since the Board admitted NYS-26/26A and TC-1/1A over two years ago, two recent developments have made summary disposition of the Consolidated Contention appropriate. First, in the *Vermont Yankee* license renewal proceeding, the Commission held unequivocally that EAF evaluations are *not* required as a condition precedent to the renewal of an operating license.[3] Thus, as a legal matter, it is uncontroverted that Entergy's commitment to conduct such EAF evaluations before the period of extended operation ("PEO") is sufficient to meet the applicable requirements in 10 C.F.R. Part 54. Second, Entergy, nonetheless, has completed its EAF evaluations in accordance with applicable NRC guidance, and demonstrated that the refined environmentally-adjusted cumulative usage factors ("CUF_{en}") for the relevant IP2 and IP3 components are below 1.0, the applicable design code limit, when projected to the end of the PEO. By completing these evaluations and fully disclosing the associated methodologies, assumptions, and results, Entergy has both satisfied Commitment 33 and addressed the issues admitted by the Board in NYS-26/26A and TC-1/1A. Accordingly, there is no longer a genuine issue as to any material fact, and the subject contentions should be dismissed in their entirety.[4]

II. STATEMENT OF THE FACTS

A. Overview of Metal Fatigue

Fatigue is the weakening of a metal caused by cyclic mechanical and thermal stresses (*i.e.*, cyclical loading) at a location on a metallic component.[5] All materials have a distinctive number of stress cycles that the material can withstand at a particular applied stress level before

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additional supporting Attachments 3 through 16. All documents referenced in this Motion, the Statement of Material Facts, and the Declaration previously have been disclosed to NYS and Riverkeeper.

3 *See Entergy Vt. Yankee, L.L.C.* (Vermont Yankee Nuclear Power Station), CL-10-17, slip op. at 42-50 (July 8, 2010).

4 Even if the Board disagrees with this conclusion, it still may grant this Motion in part, to clarify and substantially narrow any remaining factual issues that it believes warrant a hearing.

5 Attach. 2, ¶ 4. *See also Vt. Yankee*, CLI-10-17, slip op. at 15 (discussing the definition of metal fatigue).

fatigue failure occurs.[6] The period during which this number of stress cycles occurs is called the material's "fatigue life."[7] The cumulative usage factor ("CUF") represents the fraction of the total allowable fatigue cycles that the component is projected to incur during its operation. [8]

NRC regulations do not specifically define metal fatigue. However, 10 C.F.R. § 50.55a(c)(1) requires that the reactor coolant system ("RCS") pressure boundary, meet the requirements of the American Society of Mechanical Engineers ("ASME") Boiler and Pressure Vessel Code ("ASME Code"), Section III, as endorsed by the NRC.[9] The ASME Code provides a methodology for calculating the CUF for nuclear plant components, and specifies a design limit of 1.0.[10]

ASME Code, Section III fatigue curves, which are used to determine an allowable number of stress cycles at any applied stress, are based on laboratory tests conducted in air at room temperature and a constant strain rate.[11] Concerns over the potential effect of elevated temperature, reactor coolant environments, and different strain rates prompted NRC-sponsored research and studies, including NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components* (Feb. 1995) (Attach. 9).[12] NUREG/CR-6260 evaluated sample locations with high fatigue usage.[13]

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6 Attach. 2, ¶ 5 *Vt. Yankee*, CLI-10-17, slip op. at 15. A "stress cycle" is the time period it takes for a material to go from its minimal stress level to its maximum level and back again to its minimum level. *Id.* n.59.

7 Attach. 2, ¶ 5.

8 *Id.* ¶ 9.

9 *Id.* ¶ 6. *See also Vt. Yankee*, CLI-10-17, slip op. at 16-18.

10 Attach. 2, ¶¶ 7-10. This is an acceptance criterion established by the ASME Code, but exceeding the criterion does not mean the component will fail, given the numerous factors of conservatism in the analytical process. A projected CUF greater than 1.0 does not indicate that fatigue cracking necessarily will occur. Rather, it indicates that there is a potential for cracking to initiate at the affected location. *Id.*, ¶¶ 10-11; LRA at 4.3-22.

11 Attach. 2, ¶ 18.

12 *See* NUREG-1800, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants* at 4.3-2 (Rev. 1, Sept. 2005) ("SRP-LR") (Attach. 4).

13 *Id.*

The NRC Staff found that most locations would have a CUF of less than the ASME Code limit of 1.0 for 40 years.[14] The Staff, however, recommended that license renewal applicants address the effects of the reactor environment on component fatigue life "as aging management programs formulated in support of license renewal."[15] One method acceptable to the Staff for satisfying this recommendation is to assess the impact of the reactor coolant environment (*i.e.*, EAF) on the six critical locations identified in NUREG/CR-6260.[16] These six components may be evaluated by applying environmental correction factors ("F_{en}") to the existing ASME Code fatigue analyses, to obtain CUF_{en} values.[17] NRC-approved formulas for calculating the F_{en}s are contained in NUREG/CR-6583 (carbon and low-alloy steels) and NUREG/CR-5704 (austenitic stainless steels).[18]

B. Aging Management Review of Metal Fatigue

1. *10 C.F.R. Part 54 Regulatory Framework*

Part 54 requires an aging management review ("AMR") of structures and components that are subject to AMR and evaluation of time-limited aging analyses ("TLAA").[19] This review addresses aging management actions identified in § 54.21(a)(3) that *will be* taken and the aging

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[14] *Id. See also* Attach. 2, ¶ 19 (noting that Staff addresses EAF issues in NRC Generic Safety Issue ("GSI") 190 and closed out GSI 190 in December 1999 without the imposing additional requirements on operating reactors for their initial 40-year license term).

[15] Attach. 4, at 4.3-3. *See also* Attach. 2, at ¶ 19.

[16] *See* Attach. 2, ¶ 22; Attach. 4 at 4.3-5; Attach. 9 at 5-62; LRA at 4.3-21. The six critical locations identified in NUREG/CR-6260 are: (1) the reactor vessel shell and lower head, (2) the reactor vessel inlet and outlet nozzles, (3) pressurizer surge line (including hot leg and pressurizer nozzles), (4) RCS piping charging system nozzle, (5) RCS piping safety injection nozzle, and (6) residual heat removal ("RHR") Class 1 piping.

[17] Attach. 2, ¶ 20; Attach. 4 at 4.3-7; LRA at 4.3-21.

[18] Attach. 2, ¶ 21; Attach. 4 at 4.3-5, 4.3-7 (*citing* NUREG/CR-6583, *Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low–Alloy Steels* (Mar. 1998) (Attach. 7); NUREG/CR-5704, *Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels* (Apr. 1999) (Attach. 8)).

[19] 10 C.F.R. § 54.29(a)(1)-(2). Section 54.3(a) defines TLAAs as "those licensee calculations and analyses" that: (1) involve SSCs within the scope of license renewal, as delineated in § 54.4(a); (2) consider the effects of aging; (3) involve time-limited assumptions defined by the current operating term, for example, 40 years; (4) were determined to be relevant by the licensee in making a safety determination; (5) involve conclusions or provide the basis for conclusions related to the capability of the system, structure, and component to perform its intended functions, as delineated in § 54.4(b); and (6) are contained or incorporated by reference in the current licensing basis ("CLB").

management activities identified in § 54.21(c)(1) regarding the evaluation of TLAAs.[20] Section 54.21(c)(1) requires that an applicant include an evaluation of TLAAs demonstrating at least one of the following: (i) the analyses remain valid for the PEO; (ii) the analyses have been projected to the end of the PEO; or (iii) the aging effects *will be* adequately managed for the PEO.[21] Subsection 54.21(c)(1)(iii) can be satisfied through the establishment of an AMP.[22] In contrast, subsections (i) and (ii) require a demonstration that a TLAA is sufficient for the 20-year PEO or has been projected to the end of that period, respectively.[23]

The NRC Staff reviews the LRA and its supporting documents to determine compliance with Part 54, and consistency with two NRC guidance documents: the SRP-LR and the *Generic Aging Lessons Learned Report* ("GALL Report").[24] The SRP-LR describes methods for identifying those SSCs that are subject to aging effects within the scope of license renewal, and provides ten program elements that define an effective AMP.[25] For each of the SSCs identified, an applicant may show that it "*will* adequately manage aging" by demonstrating, prior to issuance of a renewed license, that its AMP is consistent with the GALL Report.[26]

The technical basis behind the SRP-LR is provided in the GALL Report,[27] a document prepared at the Commission's request and cited approvingly by the Commission.[28] The GALL Report identifies generic AMPs that the Staff has found acceptable for meeting the requirements

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20 *See Vt. Yankee*, CLI-10-17, slip op. at 19.

21 Attach. 2, ¶ 13; 10 C.F.R. § 54.21(c)(1)(i)-(iii).

22 *See Vt. Yankee*, CLI-10-17, slip op. at 20.

23 *Id.*

24 2 NUREG-1801, *Generic Aging Lessons Learned Report* (Rev. 1, Sept. 2005) ("GALL Report") (Attach. 5).

25 Attach. 4, at 3.0-2; *id.* app. A at A.1-8.

26 *Vt. Yankee*, CLI-10-17, slip op. at 45.

27 Attach. 4, at 3.0-1.

28 *See Vt. Yankee,* CLI-10-17, slip op. at 45 (*citing AmerGen Energy Co. LLC* (Oyster Creek Nuclear Generating Station), CLI-08-23, 68 NRC 461, 468 (2008)) (stating that "the GALL Report [is] a guidance document that was prepared at our behest and that we have cited with approval").

of Part 54, based on its evaluations of existing programs at operating plants during the initial license period.[29] An applicant's use of an AMP identified in the GALL Report "constitutes *reasonable assurance* that it *will* manage the targeted aging effect during the renewal period."[30]

As stated in Section X.M1 of the GALL Report, an acceptable option for managing metal fatigue of the RCS pressure boundary is to address the effects of the coolant environment on component fatigue life.[31] This can be accomplished, as discussed above, by evaluating the impact of the reactor coolant environment on the six critical component locations identified in NUREG/CR-6260.[32] An applicant also may identify corrective actions to prevent exceeding the ASME Code limit of 1.0 during the PEO, including "repair of the component, replacement of the component, and a more rigorous analysis of the component."[33]

2. *Managing the Aging Effects Caused by Metal Fatigue at IPEC*

In the LRA, Entergy originally chose to address the aging effects of EAF on susceptible components by either projecting the analyses to the end of the PEO, per § 54.21(c)(1)(ii), or demonstrating that aging effects will be adequately managed, per § 54.21(c)(1)(iii).[34] Consistent with the GALL Report, Entergy applied F_{en}s calculated using the formulae in NUREG/CR-6583 and NUREG/CR-5704 to the 60-year-projected CUFs to determine CUF_{en} values, as applicable.[35]

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29 *See, e.g.*, Attach. 5, at X M-1 to X M-2 (sec. X.M1, Metal Fatigue of Reactor Coolant Pressure Boundary).

30 *AmerGen Energy Co., LLC* (Oyster Creek Nuclear Generating Station), CLI-08-23, 68 NRC 461, 468 (2008) (emphasis added).

31 Attach. 5, at X M-1.

32 *See id. See also* Attach. 2, ¶¶ 20-22.

33 Attach. 5, at X M-2.

34 Attach. 2, ¶ 28; LRA at 4.3-22 to 4.2-23.

35 Attach. 2, ¶ 29; LRA at 4.3-21; *id.* tbls. 4.3.13 & 4.3-14 (IP2 and IP3, respectively).

The component locations in LRA Tables 4.3-13 and 4.3-14 without projected CUF_{en} values less than 1.0 include the pressurizer surge line piping for IP2 and IP3; the RCS piping charging system nozzles for IP2 (for which a CUF was available) and for IP3; and the pressurizer surge line nozzles for IP3.[36] To address these locations, Entergy committed to take one of the following actions: (1) refine the fatigue analyses, at least two years before entering the PEO, to determine CUF_{en} values less than 1.0; (2) manage the effects of aging due to fatigue at the affected locations by an inspection program that has been reviewed and approved by the NRC; or (3) repair or replace the affected locations before exceeding CUF of 1.0 (collectively referred to as "Commitment 33").[37]

Subsequently, in response to Staff audit questions, Entergy clarified the relationship between Commitment 33 and the Fatigue Monitoring Program described in Section B.1.12 of Appendix B to the LRA. The Fatigue Monitoring Program is designed to track the number of transients for selected RCS components.[38] Specifically, Entergy amended the LRA to revise Commitment 33 to read as follows:

> At least 2 years prior to entering the period of extended operation, for locations identified in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3), under the Fatigue Monitoring Program, IPEC will implement one or more of the following:
>
> (1) Consistent with the Fatigue Monitoring Program, Detection of Aging Effects, update the fatigue usage calculations using refined fatigue analyses to determine valid CUFs less than 1.0

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36 Attach. 2, ¶ 31; LRA at 4-24 to 4-25. The IP2 pressurizer surge nozzle had a CUFen less than 1.0, whereas the IP3 pressurizer surge nozzle had CUFen greater than 1.0, because the IP3 surge nozzle calculation includes the effects of the insurges/outsurges experienced by these nozzles, while the IP2 analysis did not include these effects. As a result, Entergy committed to re-analyze the pressurizer surge line nozzle for both units to include insurge/outsurge and environmental effects. Attach. 2, ¶¶ 30-31; LRA at 4.3-21 to 4.3-22; 2 NUREG-1930, *Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3, Docket Nos. 50-247 and 50-286, Entergy Nuclear Operations, Inc.*, at 4-45 (Nov. 2009) ("SER"), *available at* ADAMS Accession No. ML093170671.

37 Attach. 2, ¶ 31; LRA at 4.3-22 to 4.3-23.

38 Attach. 2, ¶ 25; LRA app. B at B-44 to B-46.

when accounting for the effects of reactor water environment. This includes applying the appropriate F_{en} factors to valid CUFs determined in accordance with one of the following:

1. For locations in LRA Table 4.3-13(IP2) and LRA Table 4.3-14 (IP3), with existing fatigue analysis valid for the period of extended operation, use the existing CUF.

2. Additional plant-specific locations with valid CUF may be evaluated. In particular, the pressurizer lower shell will be reviewed to ensure the surge nozzle remains the limiting component.

3. Representative CUF values from other plants, adjusted to or enveloping the IPEC plant specific external loads may be used if demonstrated applicable to IPEC.

4. An analysis using an NRC-approved version of the ASME code or NRC-approved alternative (*e.g.*, NRC- approved code case) may be performed to determine a valid CUF.

(2) Consistent with the Fatigue Monitoring Program, Corrective Actions, repair or replace the affected locations before exceeding a CUF of 1.0.[39]

Thus, as revised, Commitment 33 falls within the scope of the Fatigue Monitoring Program and demonstrates that the effects of EAF will be adequately managed for the PEO in accordance with § 54.21(c)(1)(iii).[40] If Entergy does not demonstrate valid CUF_{en} values below 1.0 (Option 1) after conducting refined CUF_{en} analyses, then it must pursue Option 2 of the commitment.[41] Option 2 requires Entergy to repair or replace the affected components before their refined CUF_{en} values exceed 1.0, consistent with the Fatigue Monitoring Program.[42]

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39 NL-08-021, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "License Renewal Application Amendment 2," attach. 2, at 15 (Jan. 22, 2008) ("LRA Amendment 2") (Attach. 10), available at ADAMS Accession No. ML080290659.

40 *See* Attach. 2, ¶ 32.

41 *Id*; NL-08-057, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "Amendment 3 to License Renewal Application (LRA)" attach. 2, at 13 (Mar. 24, 2008) (Attach. 11), *available at* ADAMS Accession No. ML081070255.

42 Attach. 2, ¶ 32.

3. *NRC Staff Findings Concerning the IPEC Fatigue Monitoring Program and the Effects of EAF on Key Components*

As documented in its Safety Evaluation Report ("SER"), the Staff confirmed that the IPEC Fatigue Monitoring Program includes program elements that are acceptable and consistent with the criteria in GALL Report Section X.M1.[43] The Staff further concluded that Entergy has demonstrated that the effects of aging will be adequately managed so that the component intended functions will be maintained consistent with the CLB for the PEO, as required by 10 C.F.R. § 54.21(a)(3).[44] It also reviewed the updated final safety analysis report supplement for this program and concluded that it complies with 10 C.F.R. § 54.21(d).[45]

4. *Recent Completion of the Refined CUF_{en} Analyses for IPEC*

Consistent with Commitment 33, and in view of the Board's admission of the Consolidated Contention, Entergy retained Westinghouse Electric Company LLC ("Westinghouse") in 2008 to prepare refined fatigue analyses to determine CUF_{en}s for the relevant IPEC-specific NUREG/CR-6260 critical component locations.[46] The refined fatigue analyses were completed in late June 2010, and approved by Entergy on July 29, 2010.[47] On

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43 SER at 3-81, 4-43.

44 *Id.* at 3-81, 4-47.

45 *Id.* at 3-81, 4-46 to 4-47. The Staff, in addition to evaluating the IPEC Fatigue Monitoring Program, conducted a review of LRA Section 4.3.3. *Id.* at 4-41. As part of its review, the Staff conducted an onsite audit that concluded, among other things, that IPEC had correctly accounted for the environmental factors used as inputs for calculating F_{en} values. *Id.* at 4-44. The Staff further concluded that Entergy clearly indicated the acceptable component locations (LRA Tables 4.3-13 and 4.3-14) for which it would implement one of the options in Commitment 33. *See id.* at 4-46. Finally, the Staff found that Commitment 33 is consistent with 10 C.F.R. § 54.21(a)(1)(iii). *Id.*

46 Attach. 2, ¶ 34.

47 *Id.* Based on these analyses, which are discussed further below, Westinghouse determined that, for IP2 and IP3, the refined CUF_{en} values for the pressurizer surge line piping, RCS piping charging system nozzle, RCS piping safety injection nozzle, and RHR Class 1 piping are all less than 1.0 for the PEO. Attach. 2, ¶ 39; NL-10-082, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "License Renewal Application – Completion of Commitment #33 Regarding the Fatigue Monitoring Program," attach. 1, at 2-4 (Aug. 9,2010) (Attach. 14) (including revised LRA Tables 4.3-13 and 4.3-14); Westinghouse Electric Co., WCAP-17199-P, *Environmental Fatigue Evaluation for Indian Point Unit 2*, tbls. 5-8 to 5-14 (Rev. 0, June 2010) (proprietary) (Attach. 15) ;Westinghouse Electric Co., WCAP-17200-P, *Environmental Fatigue Evaluation for Indian Point Unit 3*, tbls. 5-8 to 5-14(Rev. 0, June 2010) (proprietary) (Attach. 16).

August 9, 2010, Entergy notified the NRC Staff of the results of the refined EAF analyses; *i.e.*, the refined CUF_{en} values.[48]

III. PROCEDURAL HISTORY

A. The Original Proposed and Amended Contentions

As part of their November 30, 2007, petitions to intervene, NYS and Riverkeeper proffered contentions NYS-26 and TC-1, respectively.[49] Both contentions alleged that, because LRA Tables 4.3-13 and 4.3-14 indicated that the projected CUF_{en} values for certain IPEC components will exceed 1.0 during the PEO, Entergy must demonstrate that the effects of aging on the intended function(s) will be adequately managed for the PEO, as required by 10 C.F.R. § 54.21(c)(1)(iii).[50] Entergy opposed the admission of NYS-26 and TC-1 in their entirety under 10 C.F.R. § 2.309(f)(1).[51] The Staff opposed the admission of both contentions in part.[52]

On the same day that Entergy responded to NYS's and Riverkeeper's Petitions, it filed LRA Amendment 2 to clarify the relationship between Commitment 33 regarding EAF and the

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48 *See* Attach. 14. The Board and Parties also were provided with a copy of NL-10-082 on Aug. 10, 2010.

49 *See* New York State Notice of Intention to Participate and Petition to Intervene 227 (Nov. 30, 2007) ("NYS Petition"); Riverkeeper, Inc.'s Request for Hearing and Petition to Intervene in Indian Point License Renewal Proceeding for the Indian Point Nuclear Power Plant 7 (Nov. 30, 2007) ("Riverkeeper Petition").

50 In TC-1, Riverkeeper also alleged, more broadly, that Entergy must "broaden its TLAA analysis" beyond the scope of the representative components identified in Tables 4.3-13 and 4.3-14 to identify other components whose CUF may be greater than one. Riverkeeper Petition at 7. Relatedly, it further alleged that Entergy's list of components with CUF_{en}s less than 1.0 in LRA Tables 4.3-13 and 4.3-14 is incomplete because Entergy's methods and assumptions for identifying those components are "unrealistic and inadequate." *Id.* Specifically, Riverkeeper argued (incorrectly) that Entergy used an unrealistically low number of 2.45 for an F_{en} (instead of an F_{en} of 17); relied on the "CUF of Record" (40 years) instead of projecting the number of cycles to 60 years; and failed to calculate several NUREG-CR/6260 critical locations because they are designed to ANSI B31.1, despite the availability of "generic CUF values" from NUREG/CR-6260. *Id.* at 14. Finally, it also claimed that Entergy improperly "omit[ted]" consideration of EAF for other components listed in LRA Tables 4.3-3 through 4.3-12. *Id.* at 7-8.

51 Answer of Entergy Nuclear Operations, Inc. Opposing New York State Notice of Intention to Participate and Petition to Intervene 141-49 (Jan. 22, 2008); Answer of Entergy Nuclear Operations, Inc. Opposing Riverkeeper, Inc.'s Request for Hearing and Petition to Intervene 29-43 (Jan. 22, 2008).

52 NRC Staff's Response to Petitions for Leave to Intervene Filed by [the State of New York and Riverkeeper, Inc.] at 77-78 (Jan. 22, 2008) ("NRC Staff Answer") (opposing NYS-26 insofar as it suggested that Entergy will use arbitrary assumptions in performing any refined analyses of the CUFs and contended that Entergy must immediately replace components with CUF_{en} values exceeding 1.0.); *Id.* at 117-18 (opposing TC-1 insofar as it alleged that the lists of components in LRA Tables 4.3-13 and 4.3-14 are incomplete, and that other components need to be considered beyond those listed.)..

Fatigue Monitoring Program described in Section B.1.12 of Appendix B to the LRA. NYS and Riverkeeper filed replies to Entergy's Answer in February 2008, in which they asserted that LRA Amendment 2 did not resolve their concerns.[53] On March 4, 2008, the Staff filed a letter apprising the Board that the LRA omissions alleged in NYS-26 and TC-1 had been cured by LRA Amendment 2, thereby rendering those contentions moot and inadmissible.[54]

Thereafter, on March 5, 2008, and April 7, 2008, Riverkeeper and NYS filed amended contentions TC-1A and NYS-26A, respectively, and argued that LRA Amendment 2 did not cure the deficiencies previously alleged by those parties[55] In short, they contended that LRA Amendment 2 lacks sufficient details concerning the analytical methods that Entergy will use to calculate the refined CUF_{en} values and, by delaying the analyses, fails to meet NRC regulations.[56] NYS argued that "the most prudent way to manage aging for extended operation is to replace those affected components *now*."[57] Both Entergy and the Staff opposed the admission of amended contentions TC-1A and NYS-26A in their entirety, citing Entergy's explicit commitment to manage EAF under the Fatigue Monitoring Program.[58]

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53 New York State Reply in Support of Petition to Intervene 124-30 (Feb. 22, 2008); Riverkeeper, Inc.'s Reply to Entergy's and NRC Staff's Responses to Hearing Request and Petition to Intervene 2-12 (Feb. 15, 2008).

54 *See* Letter from David E. Roth & Kimberly A. Sexton, Counsel for NRC Staff, to Licensing Board 2 (Mar. 4, 2008), *available at* ADAMS Accession No. ML080670286.

55 Riverkeeper, Inc.'s Request for Admission of Amended Contention 6, 2 (Mar. 5, 2008); Petitioner State of New York's Request for Admission of Supplemental Contention No. 26-A, 4 (Metal Fatigue) (Apr. 7, 2008) ("NYS-26A Request").

56 NYS-26A Request at 5.

57 *Id.* at 6.

58 *See* Answer of Entergy Nuclear Operations, Inc. to Riverkeeper's Request for Admission of Amended Contention TC-1 (Concerning Environmentally Assisted Fatigue) (Mar. 31, 2008); Answer of Entergy Nuclear Operations, Inc. Opposing the State of New York's Request for Admission of Supplemental Contention 26-A (Metal Fatigue) (Apr. 21, 2008); NRC Staff's Response to Riverkeeper, Inc.'s Request for Admission of Amended Contention TC-1 ["TC-1A"] (Metal Fatigue) (Apr. 21, 2008); NRC Staff's Response to New York State's Request for Admission of Supplemental Contention 26-A (Metal Fatigue) (Apr. 21, 2008).

B. Scope of Contentions as Admitted by the Board

The Board admitted NYS and Riverkeeper's initial and amended contentions, but limited admission to those aspects "relating to the calculation of the $CUF_{[en]}$s and the adequacy of the resulting AMP for those components with $CUF_{[en]}$s greater than 1.0."[59] Specifically, the Board admitted NYS-26/26A on the following narrow grounds:

> [T]his Board admits NYS-26/26A to the limited extent that it asserts that the LRA is incomplete without the calculations of the CUFs as *threshold* values necessary to assess the need for an AMP, that Entergy's AMP is inadequate *for lack of the final values*, and that the LRA must specify actions to be carried out by the Applicant during extended operations to manage the aging of key reactor components susceptible to metal fatigue.[60]

In this regard, the Board found that Entergy "must" include CUF_{en} calculations as part of its LRA to comply with the TLAA regulations (10 C.F.R. § 54.21(a)(3)), notwithstanding Entergy's stated reliance on an AMP pursuant to § 54.21(c)(1)(iii).[61] The Board, in other words, did not accept Entergy's commitment to perform the refined CUF_{en} analyses within two years of the PEO in accordance with its GALL-consistent Fatigue Monitoring Program.[62]

The Board also found that, even assuming the refined CUF_{en} values were part of an AMP (as opposed to a TLAA), Commitment 33 does not comply with § 54.21(c)(1)(iii) because it does not describe in sufficient detail Entergy's proposed methodologies for recalculating and verifying CUF_{en} values, or provide a summary of the CUF_{en} values for each location.[63] The

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59 *See Entergy Nuclear Operations, Inc.* (Indian Point Nuclear Generating Units 2 & 3), LBP-08-13, 68 NRC 43, 172, 219 (2008).

60 *Id.* at 140 (emphasis added).

61 *Id.* at 137, 140.

62 *Id.* at 138-39

63 *Id.* at 138.

Board cited a similar admissibility ruling by the *Vermont Yankee* Board.[64] As discussed below, however, the specific findings underpinning this Board's admission of NYS-26/26A are legally erroneous in view of the Commission's recent ruling in CLI-10-17, in which it overruled the *Vermont Yankee* Board's interpretation of § 54.21(c)(1)(iii) as it relates to metal fatigue.

The Board also admitted TC-1. In doing so, it incorporated by reference its cited bases for admitting NYS-26/26A, and also identified four ancillary issues raised by TC-1/1A: (1) the extent to which an applicant must expand the scope of its TLAAs to meet GALL Report and NUREG/CR-6260 recommendations; (2) the extent, if any, that refinement of CUF_{en}s is a valid corrective action, and its relationship to the repair or replacement options; (3) the scope of the commitments to monitor, manage, and correct age-related degradation necessary to meet NRC regulations; and (4) the degree of detail and specificity with which the repair or replacement decision criteria must be defined.[65]

IV. APPLICABLE LEGAL STANDARDS[66]

Summary disposition is required when relevant documents and affidavits show that there is no genuine issue as to any material fact, such that the moving party is entitled to judgment as a matter of law.[67] Initially, the burden of proof is on the movant, and the evidence submitted is

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64 *Id.* at 140 (*citing Entergy Nuclear Vt. Yankee, LLC* (Vt. Yankee Nuclear Power Station), LBP-06-20, 64 NRC 131, 186-87 (2006)).

65 *Id.* at 172. The Board consolidated TC-1/1A with NYS-26/26A and directed NYS and Riverkeeper to confer and submit a draft of the Consolidated Contention for the Board's consideration. *Id.* at 172, 219-20. NYS and Riverkeeper submitted the Consolidated Contention on August 21, 2008, and identified NYS as the lead party. *See* Consolidated Contention of Petitioners State of New York (No. 26/26-A) and Riverkeeper, Inc. (TC-1/TC-1A) – Metal Fatigue and Designation of the State of New York as Lead Litigator for this Consolidated Contention (Aug. 21, 2008) ("Consolidated Contention Submittal").

66 Entergy has discussed the legal standards governing summary disposition of an admitted contention in prior pleadings and incorporates its discussion of those standards here. *See* Applicant's Motion for Summary Disposition of New York State's Contention 8 (Electrical Transformers) at 15-17 (Aug. 14, 2009).

67 10 C.F.R. § 2.710(d)(2); *Entergy Nuclear Generation Co.* (Pilgrim Nuclear Power Station), CLI-10-11, slip op. at 12 (Mar. 26, 2010).

construed in favor of the party opposing the motion.[68] If the movant makes a proper showing for summary disposition, then the party opposing the motion must set forth specific facts showing that there is a genuine issue of material fact.[69]

The Commission has emphasized that there is a difference between contentions that allege an "omission" of information and those that challenge substantively and specifically how particular information has been discussed in a license application.[70] Where a contention alleges the omission of particular information or an issue from an application, and the information is later supplied by the applicant, the contention is moot.[71] A motion for summary disposition is an appropriate means for a party to seek dismissal of a moot contention.[72]

V. ARGUMENT

Since the Board admitted NYS-26/26A and TC-1/1A, subsequent events have conclusively resolved the disputed issues, thereby warranting summary disposition of those contentions. First, the Commission held that § 54.21(c)(iii)—the specific regulation relied upon by Entergy in its amended LRA—does *not* require an applicant to calculate refined CUF_{en} values "prior to the issuance of a renewed license."[73] That holding is binding on this Board. Second, Entergy has met Commitment 33 by completing refined EAF analyses for those component locations that, preliminarily, had been projected in the LRA to be less than 1.0. Importantly, the

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68 *See Duke Cogema Stone & Webster* (Savannah River Mixed Oxide Fuel Fabrication Facility), LBP-05-4, 61 NRC 71, 79 (2005).

69 *See Advanced Med. Sys., Inc.* (One Factory Row, Geneva, OH 44041), CLI-93-22, 38 NRC 98, 102 (1993).

70 *See Duke Energy Corp.* (McGuire Nuclear Station, Units 1 & 2; Catawba Nuclear Station, Units 1 & 2), CLI-02-28, 56 NRC 373, 382-83 (2002)).

71 *Id.*

72 *See USEC, Inc.* (Am. Centrifuge Plant), CLI-06-9, 63 NRC 433, 444-45 (2006). *See also Duke Energy Corp.*, CLI-02-28, 56 NRC at 384.

73 *Vt. Yankee,* CLI-10-17, slip op. at 43.

refined CUF_{en} values all are less than 1.0. As a result, NYS's and Riverkeeper's original allegations are now immaterial and moot.

A. In view of the Commission's *Vermont Yankee* Ruling, IPEC Commitment 33 Is Legally Sufficient and Satisfies Applicable Part 54 Requirements

The Commission has made clear that NRC regulations *do not* require an applicant to calculate refined CUF_{en} values as a condition precedent to license renewal.[74] In CLI-10-17, a unanimous Commission reversed the *Vermont Yankee* Board's ruling in LBP-08-25 relative to two metal fatigue contentions that are materially indistinguishable from those admitted in this proceeding. The fundamental issue before the Commission was whether CUF_{en} analyses are TLAAs that must be evaluated as a prerequisite to license renewal, as the *Vermont Yankee* Board had concluded.[75] Specifically, because Entergy had not completed refined CUF_{en} analyses for the Vermont Yankee plant (notwithstanding its formal commitment to do so at least two years before entering the PEO), the *Vermont Yankee* Board held that Entergy had not met the requirements of §§ 54.21(c)(1) and 54.29.[76]

The Commission disagreed. It first noted that the definition of TLAAs is tied directly to TLAAs contained within the plant's CLB.[77] Consequently, an applicant only must evaluate TLAAs and, when doing so, must demonstrate that those analyses are valid for 60 years under § 54.21(c)(1)(i) or, pursuant to subsection (ii), have been projected to 60 years such that no AMP is necessary.[78] Because CUF_{en} analyses are *not* contained within an applicant's CLB, they are *not* TLAAs.[79] The Commission concluded: "None of our regulations requires that a license

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[74] *See id.*, slip op. at 48.

[75] *See id.*, slip op. at 37.

[76] *Entergy Vt. Yankee, LLC.* (Vermont Yankee Nuclear Generating Station) LBP-08-25, 68 NRC 763, 780, 895 (2008).

[77] *Vt. Yankee*, CLI-10-17, slip op. at 41 (discussing § 54.3 definition of TLAAs).

[78] *See id.* at 42.

[79] *Id.* at 41. *See also* Attach. 2, ¶ 20.

renewal applicant calculate CUF_{en}—*that is, adjust the CUF by applying the environmental adjustment factor*—prior to issuance of a renewed license."[80]

When the Commission's recent holding in *Vermont Yankee* is applied to the facts of this case—as it must be—it is clear that (1) Commitment 33 is legally sufficient; (2) there is no genuine issue of material fact; and (3) summary disposition of NYS-26/26A and TC-1/1A is therefore warranted.[81] NYS's and Riverkeeper's contentions are premised on the same arguments and legal conclusions squarely rejected by the Commission in *Vermont Yankee*.[82] Since CUF_{en} values are not contained within IPEC's CLB, refined CUF_{en} analyses are not required to support issuance of the renewed operating licenses.[83] As in *Vermont Yankee*, Entergy's commitment here to perform the refined CUF_{en} analyses, under the Fatigue Monitoring Program and no later than 2 years before entering the PEO, is sufficient under Part 54.

Specifically, the Commission noted that Section 54.21(c)(1)(iii) provides a third option:

> [I]n *Oyster Creek*, we expressly interpreted section 54.21(c)(1) to permit a demonstration *after* the issuance of a renewed license: "an applicant's use of an aging management program identified in the GALL Report constitutes reasonable assurance that it *will* manage the targeted aging effect during the renewal period." We reiterate here that a commitment to implement an AMP that the NRC finds is consistent with the GALL Report constitutes one acceptable method for compliance with 10 C.F.R. § 54.21(c)(1)(iii).[84]

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80 *Id.* at 48.

81 *See, e.g.*, *Pac. Gas & Elec. Co.* (Diablo Canyon Nuclear Power Plant, Units 1 & 2), LBP-86-21, 23 NRC 849, 871-72 (1986) ("In light of the Commission's determination and the regulation, which are binding upon the Board, we find that the contention is not a proper matter for adjudication, and it is therefore rejected.").

82 Indeed, in admitting NYS-26/26A, this Board concluded that "the LRA is incomplete without the calculations of the CUFs as threshold values necessary to assess the need for an AMP." *Indian Point*, LBP-08-13, 68 NRC at 140.

83 *See Vt. Yankee*, CLI-10-17, slip op. at 41.

84 *Id.* at 44 (*citing Oyster Creek*, CLI-08-23, 68 NRC at 468).

Unlike subsection (i) and (ii) TLAAs, a subsection (iii) AMP is not intended to "automatically resolve" the metal-fatigue issue "by use of a single, predictive calculation."[85] Rather, its purpose is to ensure that the ASME Code limit of 1.0 is not exceeded during the PEO.[86] Nonetheless, although not required, refined analyses yielding CUF_{en} values less than 1.0 "would *automatically resolve the metal fatigue issue* in the applicant's favor."[87]

The bottom line is crystal clear: Entergy's commitment to implement a Fatigue Monitoring Program that the Staff has found consistent with the GALL Report establishes Entergy's compliance with § 54.21(c)(1)(iii). For this reason alone, NYS-26/26A and TC-1/1A no longer present a genuine material dispute fit for hearing. Nevertheless, as discussed further below, by proactively completing its EAF analyses and showing that the refined CUF_{en} values for the critical components are all less than 1.0 for the PEO, Entergy has left no doubt that these contentions are now clearly moot and should be "automatically resolved" in its favor.[88]

B. Entergy's Completion of the CUF_{en} Analyses in Accordance With Commitment 33 Further Demonstrates the Lack of Any Material Factual Disputes

At the outset of this proceeding—and long before the Commission's contrary and dispositive ruling in CLI-10-17—the Board found that, without the refined CUF_{en} analyses, the IPEC operating licenses could not be renewed. Consequently, in 2008, Entergy initiated the process of completing its EAF analyses.[89] This process, explained in nearly 200 pages in the attached proprietary Westinghouse reports, is anything but ministerial or simply an attempt to "rework the numbers."[90] Much to the contrary, these evaluations are highly complex, require

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85 *Id.* at 43.

86 *Id.* at 43.

87 *Id.* (emphasis added).

88 *Id.*

89 Attach. 2, ¶ 34.

90 NYS Petition at 232-33.

specialized technical acumen, and involve the use of sophisticated, state-of-the-art computer-based analytics, including finite element analysis.

With the refined CUF_{en} analyses now complete, and the resulting CUF_{en} values all below 1.0, Commitment 33 has been met—and in accordance with what was previously the law of the case (*i.e.*, the Board's ruling that the CUF_{en} values must be completed prior to license renewal). By completing the analyses, the "threshold" issue giving rise to the admission of the Consolidated Contention has been resolved. And so too are any claims that Entergy's AMP is "inadequate for lack of final values."[91] Specifically, the 60-year EAF analyses prepared by Westinghouse demonstrate that the refined CUF_{en}s for all IPEC-specific NUREG/CR-6260 component locations are less than 1.0.[92] Moreover, in accordance with its Fatigue Monitoring Program, Entergy will monitor plant transients to ensure that the numbers of transient cycles experienced by IP2 and IP3 remain within the analyzed numbers of cycles, such that the CUF_{en} values remain less than 1.0 during the PEO.[93]

Entergy's completion of the refined CUF_{en} analyses, while not required by regulation, fully resolves all issues admitted by the Board. For example, NYS and Riverkeeper argue that Commitment 33 lacks sufficient detail regarding the approach to be used in calculating refined CUF_{en} values for the relevant IPEC components. The EAF analyses prepared by Westinghouse, however, refute that allegation and render both NYS-26/26A and TC-1/1A moot.

As documented in WCAP-17199-P (Attach. 15) and WCAP-17200-P (Attach. 16), Westinghouse prepared detailed stress models of the (1) surge line hot leg nozzle, (2) pressurizer surge nozzle, (3) reactor coolant piping charging system nozzle, (4) reactor coolant piping safety

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91 *Indian Point*, LBP-08-13, 68 NRC at 140.

92 Attach. 2, ¶¶ 39, 51.

93 *Id.*, ¶¶54-55.

injection nozzle, and (5) RHR system class 1 piping locations for each unit using standard methods of the ASME Code, Section III.[94] It then developed detailed stress history inputs for all transients relevant to fatigue of the affected component locations enumerated above and used those inputs in the EAF evaluations for those five locations.[95] The ASME Code evaluations were performed for the piping components to remove excess conservatisms in the existing analyses, and because IPEC primary RCS piping was designed to ANSI B31.1, *Power Piping Code*, which did not require a fatigue usage factor calculation.[96]

In determining the refined CUF_{en} values, Westinghouse used NRC-approved F_{en} factors, calculated as described in NUREG/CR-5704, for the stainless steels in the pressurizer surge line, the reactor coolant piping charging and safety injection system nozzles, and the RHR system Class 1 piping.[97] In addition, Westinghouse applied F_{en} factors, calculated as described in NUREG/CR-6583, for the carbon steel associated with the pressurizer surge nozzle dissimilar metal weld.[98] The F_{en} factors were calculated using the detailed inputs described in the applicable NUREG and were applied directly to the ASME Code fatigue results.[99] The CUF_{en} analyses were performed in accordance with Westinghouse's Quality Assurance ("QA") Program, as approved by Entergy, and included design input verification and independent

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94 Attach. 2, ¶¶ 35-38, 45-47. As discussed in LRA Section 4.3.3, the CUF and CUF_{en} values for the three *reactor vessel* locations were not recalculated because the evaluation documented in the LRA used bounding F_{en} values applied to the CUFs for the bottom head to shell region, the reactor vessel inlet nozzle, and the reactor vessel outlet nozzle. LRA at 4.3-21 to 4.3-22. *See also* Attach. 2, ¶¶ 30 & 45; Attach. 14, attach. 1, at 2-4. The original and revised LRA Tables 4.3-13 & 4.3-14 show that the CUF_{en} values for these three locations are all below 1.0 for both units, even without refined EAF analyses, for transients postulated for 60 years of operation. Thus, as the SER concludes, "the analyses performed for these components were projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii)." SER at 4-41.

95 Attach. 2, ¶ 37; Attach. 15, secs. 3, 5; Attach. 16, secs. 3, 5.

96 Attach. 15, at 1-1; Attach. 16, at 1-1. For a more detailed explanation of the purpose and acceptability of "refined" CUF analyses as well as the applicability of ANSI B31.2 to IPEC RCS piping, see paragraphs 17 and 27 of Mr. Azevedo's Declaration (Attach. 2).

97 Attach. 2, ¶¶ 35, 38, 47; Attach. 15, at 5-2 to 5-4 Attach. 16, at 5-2 to 5-4.

98 Attach. 2, ¶¶ 35, 38, 47; Attach. 15, at 5-4 to 5-5; Attach. 16, at 5-4 to 5-5.

99 Attach. 2, ¶ 47; Attach. 15, at 1-2; Attach. 16, at 1-2.

reviews to ensure that valid assumptions, transients, cycles, external loadings, analysis methods, and F_{en} factors were used in the EAF analyses for IP2 and IP3.[100]

In view of the above, no material factual dispute remains relative to the specific issues admitted by the Board in NYS-26/26A. Entergy has provided final refined 60-year CUF_{en} values less than 1.0 for *all* of the NRC-specified locations, including the ANSI B31.1-qualified RCS and RHR component locations for which CUFs of record were not previously available (*i.e.*, the use of generic or default values, as suggested by Riverkeeper, is unnecessary).[101] In addition, the condition of piping and components at the locations of interest will continue to be monitored under IPEC's Inservice Inspection Program and Fatigue Monitoring Program; *i.e.*, AMPs which the NRC Staff has determined to be consistent with the GALL Report.[102] Thus, Entergy has fully addressed the issues raised in NYS-26/26A by (1) completing its EAF analyses; (2) providing "final" CUF_{en} values for the components of interest; and, as discussed further below, by (3) specifying in its Fatigue Monitoring Program the actions to be conducted during the PEO to manage environmentally-assisted metal fatigue.

Similarly, there is no longer any factual basis for the other issues admitted in TC-1/1A. First, there is no need for Entergy to "broaden" its analysis beyond the critical component locations identified in NUREG/CR-6260 and LRA Tables 4.3-13 and 4.3-14.[103] This purported "requirement" is found nowhere in Commitment 33, NRC regulations, or NRC license renewal guidance. In fact, as applied here, the industry guidance (MRP-47) previously quoted by NYS and Riverkeeper on pages 7 to 8 of their Consolidated Contention Submittal explains that "the locations chosen in NUREG/CR-6260 . . . were deemed to be representative of locations with

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100 Attach. 2, ¶ 44.

101 Attach. 2, ¶ 50.

102 Attach. 2, ¶ 53; SER at 3-78 to 3-81, 3-173 to 3-189, 4-43 to 4-45

103 Riverkeeper Petition at 7; Consolidated Contention Submittal at 3, 11.

relatively high usage factors for all plants."[104] MRP-47 further states that: "For cases where acceptable fatigue results are demonstrated for these locations for 60 years of plant operation including environmental effects, *additional evaluations or locations need not be considered.*[105] As discussed above, through plant-specific EAF evaluations, Entergy has shown that the CUF_{en} values for all NUREG/CR-6260 locations are below 1.0 for 60 years. As such, Entergy has not improperly omitted any components listed in its LRA from its EAF analyses.

Second, there is nothing "impermissibly vague" or "unrealistic and inadequate" about the component locations identified in LRA Tables 4.3-13 and 4.3-14, or Entergy's well-documented refined EAF analyses for those component locations.[106] As discussed above, Entergy selected those IPEC component locations that correspond to the locations listed in NUREG/CR-6260. In addition, Entergy applied the formulae for calculating F_{en}s contained in NUREG/CR-6583 and NUREG/CR-5704 for the projected and refined CUF_{en} analyses—as explicitly called for by the GALL Report.[107] Moreover, as documented in the LRA and Westinghouse reports, 60-year (not 40-year) cycle counts were used as inputs for the refined EAF analyses.[108]

Finally, because Entergy has followed Option 1 of Commitment 33 and determined that the CUF_{en}s for the locations identified in NUREG/CR-6260 and LRA Tables 4.3-13 and 4.3-14 are less than 1.0, there is no present need for the corrective actions contemplated in Option 2

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104 EPRI, MRP-47, *Materials Reliability Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application* at 3-4 (Rev. 1, Sept. 2005).

105 *Id* (emphasis added).

106 Consolidated Contention Submittal at 11, 13.

107 *See* Attach. 5, at X M-1. As the *Vermont Yankee* Board found, Entergy's use of NUREG/CR-5704 and -6583 in the determination of the CUFen values is "reasonable and conservative" and produces CUF_{en} values "that are more conservative than those produced by the calculation method espoused by Dr. Hopenfeld (*i.e.*, a hybrid calculation using ASME air curves and NUREG/CR-6909 F_{en} equations)." *Vt. Yankee*, LBP-08-25, 68 NRC at 805-06. Notably, in concluding that Dr. Hopenfeld's own CUF_{en} recalculations were "unsound," that Board rejected his proposed use of a F_{en} factor of 17 (from NUREG/CR-6909) and other ASME default values. *Id.* at 823.

108 LRA at 4.3-2; Attach. 15, at 2-1 Attach. 16, at 2-1.

(*i.e.*, "repair or replace the affected locations [before exceeding a CUF of 1.0]").[109] Importantly, Entergy will manage the effects of fatigue throughout the PEO by monitoring cycles incurred and ensuring they do not exceed the analyzed numbers of cycles, such that the CUF_{en} analyses remain valid.[110] As required by the Fatigue Monitoring Program, Entergy tracks actual plant transients and evaluates these against the design transients.[111] The plant transient counts will be updated at least once each operating cycle, which is an acceptable frequency since the evaluation during each update determines if the number of design transients could be exceeded prior to the next update.[112]

The Fatigue Monitoring Program requires corrective actions, implemented in accordance with the IPEC Corrective Action Program, *before* the plant exceeds the analyzed number of transient cycles.[113] As discussed in the SER, IPEC procedures contain specific "alert levels" that trigger the initiation of corrective actions under the Fatigue Monitoring Program.[114] Any further reanalysis (*i.e.*, future analysis updates) would be governed by Entergy's QA program, as discussed above.[115] Any repair or replacement of a component, if necessary, would be done in accordance with established plant procedures that are governed by Entergy's ISI and QA programs and meet the applicable repair or replacement requirements of ASME Code Section

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109 Attach. 2, ¶ 52. *See also* SER at 4-46.

110 Attach. 2, ¶ 54; SER at 4-44 to 4-45; NL-08-084, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "Reply to Request for Additional Information Regarding License Renewal Application – Time-Limited Aging Analyses and Boraflex" attach. 1, at 3-4 (May 16, 2008) (Attach. 12).

111 Attach. 2, ¶ 55; Attach. 12, at 4; SER at 4-45

112 Attach. 2, ¶ 55; Attach. 12, at 4; SER at 3-79, 4-44.

113 Attach. 2, ¶ 56; Attach. 12, at 4; SER at 4-44 to 4-45.

114 Attach. 2, ¶ 56; SER at 4-44.

115 Attach. 2, ¶ 57; Attach. 12, at 4.

XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."[116] Consequently, there is no ambiguity concerning IPEC's future monitoring and corrective action programs and activities during the PEO.[117]

VI. CONCLUSION

For the foregoing reasons, the Board should grant summary disposition of NYS-26/26A and TC-1/1A.

CERTIFICATION OF COUNSEL UNDER 10 C.F.R. § 2.323(b)

On August 19, 2010, counsel for Entergy (Mr. Paul Bessette) initiated the consultation process with counsel for NYS (Ms. Janice Dean) and Riverkeeper (Ms. Deborah Brancato) regarding Entergy's planned motion to dismiss the Consolidated Contention. During the course of several phone calls, counsel for NYS and Riverkeeper indicated that their respective experts were reviewing the Westinghouse EAF analyses to determine whether the analyses address the concerns raised in the Consolidated Contention. As such, they requested that the consultation be deferred until August 25th, at which time they would be in a better position to complete the consultation. Counsel for Entergy agreed to a limited extension of the consultation process to August 25, 2010.

To support its expert's review of the Westinghouse EAF analyses, NYS also requested a copy of the user's manual for the proprietary WESTEMS™ computer program used by Westinghouse to perform the EAF analyses. Riverkeeper also requested copies of the proprietary Westinghouse EAF analyses and the WESTEMS™ program manual. Mr. Bessette conveyed Entergy's position that the Consolidated Contention, as admitted, is a *contention of*

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[116] Attach. 2, ¶ 57; Attach. 12 at 4. The purpose of an ASME Code, Section XI evaluation is to show that the component is acceptable at the end of the PEO with projected in-service flaw growth typically predicted based on design thermal and mechanical loading cycles. Attach. 2, ¶ 53.

[117] *Id.* ¶ 57.

omission which alleges that the LRA is incomplete for lack of final revised CUF_{en} values. He further explained that Entergy has cured the alleged omission by completing the final revised EAF analyses, such that the *admitted* contention is now moot. To facilitate possible resolution of this matter, however, Mr. Bessette agreed to request a copy of the WESTEMS™ user's manual from Westinghouse, but emphasized that it is Entergy's position that the acceptability of the revised CUF_{en} values is not the subject of any currently-admitted contention. Entergy requested a copy of the WESTEMS™ user's manual from Westinghouse last week but has yet to receive a copy from Westinghouse.

On August 23, 2010, NYS and Riverkeeper counsel contacted Mr. Bessette, contending that their experts also needed to review any associated "margin of error" analysis prepared by Westinghouse, to assess whether the EAF analyses are sufficiently "complete" to resolve the Consolidated Contention. Entergy objected to this request, noting that even if such an analysis exists, NYS and Riverkeeper are raising an entirely new issue outside the scope of the admitted contention and seeking unauthorized discovery. NYS and Riverkeeper disagreed, contending that evaluation of the "completeness" of the new EAF analyses is essential to determining whether the original contention is now moot.

As agreed, counsel for Entergy, NYS, and Riverkeeper consulted again on August 25, 2010. NRC Staff counsel (Mr. Sherwin Turk) also participated in the call. Unfortunately, the parties remained at an impasse relative to the scope of the admitted contention and the information needed by NYS and Riverkeeper to assess Entergy's claim that the Consolidated Contention is now moot. Accordingly, Mr. Bessette indicated that Entergy would proceed with filing the motion for summary disposition. NYS and Riverkeeper indicated that they do not have sufficient information to take a position at this time on whether the contention is moot. Mr. Turk

stated that, based on the information available to it, the NRC Staff does not oppose the motion and will file a response.

In view of the above, and in accordance with 10 C.F.R. § 2.323(b) and the Board's July, 10, 2010 Scheduling Order, counsel for Entergy certifies that he made a sincere effort to contact the other parties in this proceeding to explain to them the factual and legal issues raised in this Motion, and to resolve those issues, and certifies that these efforts have been unsuccessful. Counsel for Entergy further certifies that this Motion is not interposed for delay or any other improper purpose, that counsel believes in good faith that there is no genuine issue as to any material fact relating to this Motion, and that the moving party is entitled to a decision as a matter of law, as required by 10 C.F.R. §§ 2.1205 and 2.710(d).

Respectfully submitted,

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Dated in Washington, D.C.
this 25th day of August 2010

DB1/65293209

LIST OF SUPPORTING ATTACHMENTS
APPLICANT'S MOTION FOR SUMMARY DISPOSITION OF
NEW YORK STATE CONTENTIONS 26/26A & RIVERKEEPER TECHNICAL CONTENTIONS 1/1A (METAL FATIGUE OF REACTOR COMPONENTS)

Attachment	Description
1	Statement of Material Facts
2	Declaration of Nelson F. Azevedo
3	*Curriculum Vitae* of Nelson F. Azevedo
4	Excerpt from NUREG-1800, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants*, Revision 1 (Sept. 2005)
5	Excerpt from Vol. 2 of NUREG-1801, *Generic Aging Lessons Learned (GALL) Report – Tabulation of Results* (Rev. 1, Sept. 2005)
6	Excerpts from EPRI, MRP-47, *Materials Reliability Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application* (Rev. 1, Sept. 2005)
7	Excerpt from NUREG/CR-6583, *Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels* (Mar. 1998).
8	Excerpt from NUREG/CR-5704, *Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels* (Apr. 1999)
9	Excerpt from NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components* (Feb. 1995)
10	NL-08-021, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "License Renewal Application Amendment 2," (Jan. 22, 2008)
11	Excerpt from NL-08-057, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "Amendment 3 to License Renewal Application (LRA)" (Mar. 24, 2008))
12	NL-08-084, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "Reply to Request for Additional Information Regarding License Renewal Application – Time-Limited Aging Analyses and Boraflex," (May 16, 2008)
13	NL-08-092, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "Amendment 5 to License Renewal Application"

Attachment	Description
	(June 11, 2008)
14	NL-10-082, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "License Renewal Application – Completion of Commitment #33 Regarding the Fatigue Monitoring Program" (Aug. 9, 2010)
15	Westinghouse Electric Co., WCAP-17199-P, Revision 0, *Environmental Fatigue Evaluation for Indian Point Unit 2* (June 2010) **(Entergy-Designated Confidential Proprietary Document Subject to Nondisclosure Agreement and ASLB 9/4/2009 Protective Order)**
16	Westinghouse Electric Co., WCAP-17200-P, Revision 0, *Environmental Fatigue Evaluation for Indian Point Unit 3* June 2010) **(Entergy-Designated Confidential Proprietary Document Subject to Nondisclosure Agreement and ASLB 9/4/2009 Protective Order)**

NYS-26/26A & RIVERKEEPER TC-1/1A: ATTACHMENT 1

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	Docket Nos. 50-247-LR and 50-286-LR
ENTERGY NUCLEAR OPERATIONS, INC.	)	ASLBP No. 07-858-03-LR-BD01
(Indian Point Nuclear Generating Units 2 and 3)	)	August 25, 2010

STATEMENT OF MATERIAL FACTS

Entergy Nuclear Operations, Inc. ("Entergy") submits this statement of undisputed material facts in support of its Motion for Summary Disposition of New York State Contentions 26/26A ("NYS-26/26A") and Riverkeeper Technical Contentions 1/1A ("TC-1/1A").

A. Applicable 10 C.F.R. Part 54 Regulations and License Renewal Guidance

1. 10 C.F.R. § 54.21(a)(3) requires a license renewal applicant to demonstrate that it will adequately manage the effects of aging on structures, systems, and components ("SSCs") subject to aging management review ("AMR"), so that there is "reasonable assurance" that their intended functions will be maintained consistent with the current licensing basis ("CLB") for the period of extended operation ("PEO").

2. 10 C.F.R. § 54.21(d) requires that the final safety analysis report ("FSAR") supplement for the facility contain a summary description of the programs and activities for managing the effects of aging.

3. 10 C.F.R. § 54.21(c)(1) lists the technical information that must be contained in a license renewal application ("LRA") related specifically to time-limited aging analyses ("TLAAs"), which 10 C.F.R. § 54.3(a) defines as those licensee calculations and analyses that: (1) involve SSCs within the scope of license renewal, as delineated in § 54.4(a); (2) consider the effects of aging; (3) involve time-limited assumptions defined by the current operating term, for example, 40 years; (4) were determined to be relevant by the licensee in making a safety determination; (5) involve conclusions or provide the basis for conclusions related to the

capability of the system, structure, and component to perform its intended functions, as delineated in § 54.4(b); and (6) are contained or incorporated by reference in the CLB.

4. For all TLAAs, an applicant must demonstrate that (i) the analyses remain valid for the PEO; (ii) the analyses have been projected to the end of the PEO, or (iii) the aging effects will be adequately managed for the PEO. 10 C.F.R. § 54.21(c)(1)(i)-(iii).

5. NUREG-1800 describes methods for identifying those SSCs that are subject to aging effects within the scope of license renewal, and provides ten program elements that define an effective AMP: (1) Scope of the Program, (2) Preventive Actions, (3) Parameters Monitored or Inspected, (4) Detection of Aging Effects, (5) Monitoring and Trending, (6) Acceptance Criteria, (7) Corrective Actions, (8) Confirmation Process, (9) Administrative Controls, and (10) Operating Experience. NUREG-1800, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants* at 3.0-2, A.1-8 (Rev. 1, Sept. 2005) ("SRP-LR") (Attach. 4).

6. The technical basis behind the SRP-LR is provided in 2 NUREG-1801, *Generic Aging Lessons Learned Report* (Rev. 1, Sept. 2005) ("GALL Report") (Attach. 5), an NRC license renewal guidance document prepared at the request of the Commission and commonly cited by it with approval. *Entergy Vt. Yankee, L.L.C.* (Vermont Yankee Nuclear Power Station), CLI-10-17, slip op. at 44 (*citing AmerGen Energy Co. LLC* (Oyster Creek Nuclear Generating Station), CLI-08-23, 68 NRC 461, 468 (2008)).

7. The GALL Report identifies generic AMPs that the Staff has found acceptable for meeting the requirements of Part 54, based on its evaluations of existing programs at operating plants during the initial license period. An applicant's use of an AMP identified in the GALL Report "constitutes reasonable assurance that it will manage the targeted aging effect during the renewal period." *Oyster Creek*, CLI-08-23, 68 NRC at 468.

B. Metal Fatigue and the Cumulative Usage Factor

8. Fatigue is the weakening of a metal caused by cyclic mechanical and thermal stresses (*i.e.*, cyclical loading) at a location on a metallic component. Metal components experience these stress cycles during "transients" such as plant startup and shutdown that result in significant temperature changes. A "stress cycle" is the time period it takes for a material to

go from its minimum stress level to its maximum level and back again to its minimum level. An excessive number of cycles may result in a significant reduction in the strength of a component. Declaration of Nelson Azevedo in Support of Applicant's Motion for Summary Disposition of Consolidated Contentions NYS-26/26A and Riverkeeper TC-1/1A, ¶ 4 (Attach. 2).

9. All materials have a distinctive number of stress cycles that the material can withstand at a particular applied stress level before fatigue failure occurs. The period during which this number of load cycles occurs is called the material's "fatigue life." Attach. 2, ¶ 5.

10. 10 C.F.R. Part 50 does not specifically mention metal fatigue; however, 10 C.F.R. § 50.55a(c)(1) requires that the reactor coolant system ("RCS") pressure boundary meet the requirements of the American Society of Mechanical Engineers ("ASME") Boiler and Pressure Vessel Code ("ASME Code"), Section III, as endorsed by the NRC. Attach. 2, ¶ 6.

11. The original design specifications for a given safety-related component specify the number of mechanical and thermal cycles that the component must be designed to withstand, and define the safety limits and applicable codes that must be satisfied. For components constituting the primary RCS pressure boundary, the specified requirements for evaluation of cyclic loading and thermal conditions generally are contained in Section III of the ASME Code for Class 1 components. Attach. 2, ¶ 7.

12. For a Class 1 component, different stress cycles from the loadings specified in the governing design specification will produce total stresses of several different magnitudes. The number of times these stresses of different magnitudes occur also varies. The allowable number of cycles for a given alternating stress range is determined from the ASME Code design fatigue curve for the material being evaluated. Attach. 2, ¶ 8.

13. The fatigue usage for that stress cycle is the ratio of the number of analyzed applied stress cycles (n) to the allowable number of stress cycles (N) from the ASME Code design fatigue curve. The cumulative usage factor ("CUF") represents the fraction of the total allowable fatigue cycles that the component is projected to incur during its operation. Attach. 2, ¶ 9.

14. ASME Code Section III requires that the CUF for a Class 1 component not exceed unity or 1.0; *i.e.*, the total number of applied stress cycles is not to exceed the allowable

number of stress cycles. This is an acceptance criterion established by the Code, but exceeding the criterion does not mean the component will fail, given the numerous factors of conservatism in the analytical process. Attach. 2, ¶ 10.

15. CUF values less than 1.0 indicate that the equipment can withstand the fatigue effects of cyclic stress due to operation during the analyzed number of design transients at the analyzed transient severity. A projected CUF greater than 1.0 does not indicate that fatigue cracking necessarily will occur. Rather, it indicates that there is a potential for cracking to initiate at the affected location. This is not necessarily failure of the component. Attach. 2, ¶¶ 10 & 11.

16. CUF values result from conservative fatigue calculations. Applicants may revise fatigue analyses to take advantage of excess conservatisms inherent in the original fatigue analyses (*e.g.*, excess conservatisms in loading definitions, predicted numbers of cycles, transient groupings, etc.) as well as improvements in available analytical tools. Attach. 2, ¶¶ 10, 17; EPRI, MRP-47, *Materials Reliability Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application* at 4-4 (Rev. 1, Sept. 2005) (Attach. 6).

C. Environmentally-Assisted Fatigue and Related NRC Guidance

17. For components (equipment and piping) exposed to reactor coolant water, the fatigue life, or allowable number of stress cycles, may be reduced compared to a component's fatigue life in air. The ASME Code design fatigue curves were developed based on laboratory testing of specimens in air at a constant strain rate, with safety factors incorporated into the curves to account for variables such as surface finish. Laboratory testing of specimens in water under reactor operating conditions indicate that additional environmental factors may need to be applied to the calculated CUF to fully account for reactor coolant environmental conditions. Fatigue analysis accounting for the effects of operating in a reactor coolant environment is called environmentally-assisted fatigue ("EAF") analysis. Attach. 2, ¶ 18.

18. EAF analysis was addressed in NRC Generic Safety Issue ("GSI") 190, "Fatigue Evaluation of Metal Components for 60-year Plant Life." The NRC Staff closed out GSI 190 in December 1999 without imposing additional requirements because it found only negligible calculated increases in core damage frequency in going from 40-year to 60-year license periods. However, the NRC Staff concluded that license renewal applicants should address the effects of

coolant environment on component fatigue life as aging management programs that are formulated for license renewal. Attach. 2, ¶ 19; Attach. 4, at 4.3-2 to 4.3-3.

19. To address EAF, an applicant may apply an environmental correction factor, or "F_{en}", to a CUF, to calculate an environmentally-adjusted CUF, or "CUF_{en}". The F_{en} is defined as the ratio of the fatigue life in air at room temperature to that in water at the service temperature. The fatigue usage derived from current ASME Code fatigue design (air) curves is multiplied by the F_{en} to account for the environmental effects. Because EAF (CUF_{en}) analyses are not contained within an applicant's CLB, they are not TLAAs. Attach. 2, ¶ 20.

20. NRC guidance recommends the use of formulae in NUREG/CR-6583, *Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels* (Mar. 1998) (Attach. 7), to determine F_{en}s for carbon and low alloy steel components, and NUREG/CR-5704, *Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels* (Apr. 1999) (Attach. 8), to determine F_{en}s for stainless steel components. Attach. 2, ¶ 21; Attach. 4, at 4.3-5, 4.3-7; Attach 5, at X M-1.

21. As discussed in the SRP-LR and GALL Report, one method for addressing EAF found acceptable by the NRC Staff is the conduct of CUF_{en} analyses for the six critical locations identified in NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components* (Mar. 1995) (Attach. 9). Those locations include: (1) the reactor vessel shell and lower head, (2) the reactor vessel inlet and outlet nozzles, (3) the pressurizer surge line (including hot leg and pressurizer nozzles), (4) RCS piping charging system nozzle, (5) RCS piping safety injection nozzle, and (6) residual heat removal ("RHR") Class 1 piping. Attach. 2, ¶ 22; Attach. 4, at 4.3-5; Attach. 5, at X M-1; Attach. 9, at 5-62.

22. NUREG/CR-6260 states that the six component locations identified therein "were chosen to give a representative overview of components that had higher CUFs and/or were important from a risk perspective." Attach. 9, at 4-1. Another guidance document, MRP-47, Revision 1, states that the six NUREG/CR-6260 locations "were considered representative enough that the effects of LWR environment on fatigue could be assessed." Attach. 6, at 3-4. MRP-47 further states: "For cases where acceptable fatigue results are demonstrated for these locations for 60 years of plant operation including environmental effects, additional evaluations or locations need not be considered." *Id.*

D. Overview of the Indian Point Energy Center ("IPEC") Fatigue Monitoring Program

23. LRA Chapter 4 summarizes Entergy's evaluation of the effects of metal fatigue and EAF on in-scope piping and components for the PEO. Section 4.3.1 (Class 1 Fatigue) addresses components (and subcomponents) that were designed in accordance with ASME Code, Section III. Broadly speaking, those components include the reactor vessel, reactor vessel internals, pressurizer, steam generators, reactor coolant pumps, and control rod drive mechanisms, Class 1 heat exchangers, and Class 1 piping and components. Section 4.3.2 (Non-Class 1 Fatigue) addresses ANSI Code B31.1 and ASME Code Section III Class 2 and 3 piping systems. Section 4.3.3 (Effects of Reactor Water Environment on Fatigue) addresses EAF with respect to the critical component locations identified in NUREG/CR-6260. Attach. 2, ¶ 23.

24. The appendices to the LRA contain a description of Entergy's Fatigue Monitoring Program. Appendix A presents the information required by 10 C.F.R. § 54.21(d) relating to the AMP for fatigue monitoring that supplements the updated FSAR ("UFSAR") for IPEC. The supplement to the UFSAR, presented in Sections A.2 and A.3 of Appendix A for IP2 and IP3, respectively, contains a summary description of the program and activities for managing the effects of metal fatigue during the PEO. Appendix A states that the Fatigue Monitoring Program will be implemented prior to the PEO. Attach. 2, ¶ 24; LRA app. A at A-1, A-22, A-49, *available at* ADAMS Accession No. ML071210520.

25. Appendix B to the LRA describes those AMPs credited by Entergy to manage the effects of aging. Section B.1.12 describes the IPEC Fatigue Monitoring Program, which is an existing program that is designed to track the number of transients for selected RCS components. LRA app. B at B-1, B-44 to B-46, B-54, *available at* ADAMS Accession No. ML071210523. Section B.1.12 indicates that it is consistent with, but takes one exception to the "detection of aging effects," the program described in the GALL Report Section X.M1. *Id.* at B-44. However, Entergy later removed this exception in an amendment to the LRA. Attach. 2, ¶ 25; NL-08-092, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "Amendment 5 to License Renewal Application" (June 11, 2008) (Attach. 13); 2 NUREG-1930, *Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3, Docket Nos. 50-247 and 50-286, Entergy Nuclear Operations, Inc.* at 3-79 to 3-80 (Aug. 2009) ("SER"), *available at* ADAMS Accession No. ML093170671.

26. As discussed in LRA Section 4.3.1, design basis fatigue evaluations for Class 1 components produce calculated CUFs for components and subcomponents based on selected numbers and magnitudes of design transient cycles. For RCS components, Entergy projected the numbers of cycles accrued to date to the end of the PEO; *i.e.*, the numbers of cycles expected at the end of 60 years of plant operation. Attach. 2, ¶ 26.

27. LRA Tables 4.3-1 and 4.3-2 show the specific transient conditions, the analyzed numbers of cycles, and the projected numbers of cycles for 60 years of operation for IP2 and IP3, respectively. LRA Tables 4.3-3 to 4.3-12 list the CUFs for the various Class 1 components and subcomponents based on the numbers of cycles assumed in the analyses. The numbers of cycles expected at the end of 60 years of plant operation are less than the numbers considered in the analyses for all but a few of the transients listed in LRA Tables 4.3-1 and 4.3-2. Attach. 2, ¶ 26.

28. IPEC RCS piping was designed to ANSI B31.1, *Power Piping Code*, which did not require a detailed fatigue usage factor evaluation. Instead, a stress analysis was performed on the main primary coolant piping, in accordance with the criteria set forth in ANSI B31.1, to ensure that the stress range was within the prescribed limits. Stress range reduction factors were used to account for anticipated transients for RCS piping (a stress range reduction factor of 1.0 is acceptable in the stress analyses for up to 7000 cycles). Therefore, Entergy evaluated the projected thermal cycles for 60 years of plant operation at IP2 and IP3 and determined that the total cycles in 60 years of operation are well below the 7000 cycles allowed by ANSI B31.1 for a stress range reduction factor of 1.0. Attach. 2, ¶ 27; LRA at 4.3-18.

29. In preparing Section 4.3.3 of its April 2007 LRA, Entergy addressed EAF effects on those IPEC-specific components corresponding to the six critical locations identified in NUREG/CR-6260 by either projecting the analyses to the end of the PEO, per § 54.21(c)(1)(ii), or demonstrating that aging effects will be adequately managed, per § 54.21(c)(1)(iii). Attach. 2, ¶ 28; LRA at 4.3-20 to 4.3-25.

30. Consistent with the GALL Report, Entergy applied Fens calculated as described in NUREG/CR-6583 and NUREG/CR-5704 to the 60-year-projected CUFs to determine CUFen values. LRA at 4.3-21, tbls. 4.3.13 (IP2) & 4.3-14 (IP3). (As explained above, CUFs were not available for all RCS piping and RHR Class 1 piping locations because those locations were designed to ANSI B.31.1.) Entergy's Fen calculation methods are documented in a response to a

Staff request for additional information. Attach. 2, ¶ 29; NL-08-084, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "Reply to Request for Additional Information Regarding License Renewal Application – Time-Limited Aging Analyses and Boraflex" attach. 1, at 3-4 (May 16, 2008) (Attach. 12).

31. As indicated in LRA Tables 4.3.13 and 4.3-14, the projected CUFen values for the three NUREG/CR-6260 reactor vessel locations were less than 1.0 for both IP2 and IP3. These three reactor vessel locations included the (1) bottom head to shell, (2) reactor vessel inlet nozzle, and (3) reactor vessel outlet nozzle. In addition, the IP2 (but not IP3) pressurizer surge line nozzle had a CUFen less than 1.0. Accordingly, because these CUFen values fell below 1.0, the EAF analyses for those components were projected to the end of the PEO (in the LRA) in accordance with Section 54.21(c)(1)(ii). Attach. 2, ¶ 30; LRA at 4.3-22.

32. The component locations in LRA Tables 4.3-13 and 4.3-14 without projected CUF_{en} values less than 1.0 include the pressurizer surge line piping for IP2 and IP3; the RCS piping charging system nozzles for IP2 (for which a CUF was available) and for IP3; and the pressurizer surge line nozzles for IP3. LRA tbls. 4.3.13 & 4.3-14. As stated in LRA Section 4.3.3, the IP2 pressurizer surge nozzle had a CUF_{en} less than 1.0, whereas the IP3 pressurizer surge nozzle had CUF_{en} greater than 1.0, because the IP3 surge nozzle calculation includes the effects of the insurges/outsurges experienced by these nozzles, while the IP2 analysis did not include these effects. As a result, Entergy committed to re-analyze the pressurizer surge line nozzle for both units to include insurge/outsurge and environmental effects. Attach. 2, ¶ 31; LRA at 4.3-21 to 4.3-22, tbls. 4.3.13 & 4.3-14.

33. To address these locations, Entergy committed to take one of the following actions: (1) refine the fatigue analyses, at least two years before entering the PEO, to determine valid CUF_{en} values less than 1.0; (2) manage the effects of aging due to fatigue at the affected locations by an inspection program that has been reviewed and approved by the NRC; or (3) repair or replace the affected locations before exceeding CUF of 1.0 (collectively referred to as "Commitment 33"). Attach. 2, ¶ 31; LRA 4.3-22 to 4.3-23.

34. In January 2008, as a part of the LRA Amendment 2, Entergy amended the Fatigue Monitoring Program to provide information on cycle counting and the methodology used to determine stresses and fatigue usage, including environmental effects, in accordance with the

NRC-endorsed ASME Code. Entergy also amended the LRA to place Commitment 33 within the scope of the Fatigue Monitoring Program, by stating that it will use that AMP to manage the effects of reactor water environment on fatigue life, in accordance with § 54.21(c)(1)(iii). Attach. 2, ¶ 32; NL-08-021, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "License Renewal Application Amendment 2" (Jan. 22, 2008) ("LRA Amendment 2") (Attach. 10).

35. Pursuant to revised Commitment 33, if Entergy does not demonstrate valid CUF_{en} values below 1.0 (Option 1) by conducting refined CUF_{en} analyses, then it must implement Option 2 of the commitment. Option 2 requires Entergy to repair or replace the affected locations before their refined CUF_{en}s exceed 1.0, consistent with the Fatigue Monitoring Program. Attach. 2, ¶ 32; Attach. 10, attach. 2, at 15).

36. During the PEO, Entergy will continue to monitor the condition of piping and components at the locations of interest under IPEC's ISI Program, which includes periodic visual, surface, and volumetric examination of Classes 1, 2, and 3 pressure-retaining components, their attachments, and supports. If a flaw is detected during an ISI Program inspection or by other means, then the component may be replaced or repaired, or evaluated for continued service in accordance with the criteria contained in ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components." Acceptance standards for examination evaluations, repair procedures, inservice test requirements, and replacements for ASME Class 1 components are defined in ASME Section XI paragraphs IWB-3000, IWB-4000, IWB-5000 and IWB-7000, respectively. Attach. 2, ¶ 53.

37. Under the Fatigue Monitoring Program, Entergy will manage the effects of fatigue by monitoring cycles incurred and ensuring they do not exceed the analyzed numbers of cycles, such that the CUFen analyses remain valid. Attach. 2, ¶ 54; LRA at 4.3-2 to 4.3-3; SER at 4-44 to 4-45; Attach. 12, attach. 1, at 3-4.

38. As required by the Fatigue Monitoring Program, Entergy tracks actual plant transients and evaluates these against the design transients. The plant transient counts will be updated at least once each operating cycle, which is an acceptable frequency because the evaluation during each update determines if the number of design transients could be exceeded

prior to the next update. Attach. 2, ¶ 55; LRA at 4.3-2 to LRA 4.3-3; Attach. 10, attach. 1, at 6); Attach. 12, attach. 1, at 4).

39. Consistent with GALL Report Section X.M1, the Fatigue Monitoring Program requires that corrective actions be implemented in accordance with the IPEC Corrective Action Program before the plant exceeds the analyzed number of transient cycles. IPEC procedures contain specific "alert levels" that trigger the initiation of corrective actions under the Fatigue Monitoring Program. SER at 4-44. Specifically, IPEC calculates alert levels by adding twice the number of cycles that occurred in the last fuel cycle to the total number of cycles to date. Corrective action is initiated if this alert level exceeds the number of analyzed transients. If the number of cycles is projected to remain at or below the analyzed level for two additional fuel cycles, then no corrective action is required Attach. 2, ¶ 56; Attach. 10, attach. 1, at 5-6; Attach. 12, attach. 1, at 4.

40. 57. Any further reanalysis (*i.e.*, future analysis updates), if necessary, would be governed by applicable QA procedures. Any repair or replacement of a component, if necessary, would be done in accordance with established plant procedures that must comply with Entergy's QA program and meet the applicable repair or replacement requirements of ASME Code Section XI. Attach. 2, ¶ 57; Attach. 10, attach. 1, at 6; Attach. 12, attach. 1, at 4.

E. <u>The NRC Staff's November 2009 Safety Evaluation Report Findings</u>

41. As documented in its final Safety Evaluation Report ("SER"), the NRC Staff, after conducting a detailed technical review of the LRA and performing related site audits, found that IPEC's Fatigue Monitoring Program includes acceptable program elements that are consistent with criteria in GALL Report Section X.M1. 2 SER at 3-81. It further concluded that Entergy has demonstrated that the effects of aging will be adequately managed so that the intended component functions will be maintained consistent with the CLB for the PEO, as required by 10 C.F.R. § 54.21(a)(3). *Id.* In addition to evaluating IPEC's Fatigue Monitoring Program, the Staff reviewed LRA Section 4.3.3 regarding EAF. As part of its review, the NRC Staff confirmed that IPEC had correctly accounted for the environmental factors used as inputs for calculating F_{en} factors. It also found that Commitment 33 is consistent with 10 C.F.R. § 54.21(a)(1)(iii). *Id.* at 4-43 to 4-45.

F. The 2010 Westinghouse Environmental Fatigue Evaluations for IPEC

42. To resolve Commitment 33, Entergy retained Westinghouse Electric Company LLC ("Westinghouse") in 2008, to update the fatigue usage calculations by performing refined fatigue analyses to determine environmentally-adjusted CUFs. Westinghouse completed the refined fatigue analyses in late June 2010. Attach. 2, ¶ 34; Westinghouse Electric Co., WCAP-17199-P, Westinghouse Electric Co., *Environmental Fatigue Evaluation for Indian Point Unit 2* (Rev. 0, June 2010) (proprietary) (Attach. 15); WCAP-17200-P, Rev. 0, *Environmental Fatigue Evaluation for Indian Point Unit 3* (June 2010) (proprietary) (Attach. 16).

43. Entergy approved the Westinghouse EAF analyses on July 29, 2010, and shortly thereafter, notified the Staff of the results of the refined EAF analyses. Attach. 2, ¶ 34; NL-10-082, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "License Renewal Application – Completion of Commitment #33 Regarding the Fatigue Monitoring Program" (Aug. 9, 2010) (Attach. 14).

44. Consistent with the Fatigue Monitoring Program, the refined EAF analyses were performed in accordance with Entergy's 10 C.F.R. Part 50, Appendix B Quality Assurance ("QA") program and included design input verification and independent reviews to ensure that valid assumptions, transients, cycles, external loadings, analysis methods, and F_{en} factors were used in the EAF analyses for IP2 and IP3.

45. As documented in WCAP-17199-P and WCAP-17200-P, Westinghouse performed 60-year EAF evaluations using F_{en} factors derived from NUREG/CR-5704 for stainless steel components and from NUREG/CR-6583 for those components containing carbon and low-alloy steel components. Westinghouse calculated the F_{en} factors, which it then applied to the ASME Code fatigue results. Attach. 2, ¶ 35.

46. The Westinghouse evaluations included the following four major steps: (1) determination of the limiting components for each location; (2) determination of the transients for each limiting component; (3) performance of ASME Code Section III stress and fatigue evaluations for each limiting component; and (4) determination of the CUF_{en}, for each limiting component. As necessary, plant-specific projections of numbers of accrued cycles at the end of the PEO replaced conservative assumptions for numbers of cycles in the analyses. The fatigue

evaluations followed the procedures given in ASME Code Section III, NB-3200. Attach. 2, ¶ 36; Attach. 15, at 2-2 to 2-4, 5-20; Attach. 16, at 2-2 to 2-4, 5-20.

47. Westinghouse developed transients using plant data and/or Westinghouse standard design transients, assuming the total period of operation to be 60 years. For the charging, safety injection, and RHR locations selected, Westinghouse developed stress inputs to the fatigue analyses using detailed finite element analysis models, and ran those models to create unit load transfer functions for each type of mechanical and thermal transient condition loading considered in the analyses. It also developed piping mechanical load functions to produce time history moment loads as a function of system parameters during a transient. The transfer functions and load functions were used in a proprietary Westinghouse computer code to determine detailed stress histories for each applicable transient, considering all applicable mechanical and thermal transient loads during each transient, and to calculate fatigue usage. Attach. 2, ¶ 37; Attachs. 15 & 16.

48. Section 5 of each Westinghouse EAF report details the manner in which Westinghouse evaluated EAF for the relevant IPEC-specific NUREG/CR-6260 component locations. Westinghouse's evaluations were based upon CUF analyses using transients that envelop the 60-year cycle projections. Sections 5.1 and 5.2, in particular, describe Westinghouse F_{en} calculations using the methodologies set forth in NUREG/CR-5704 for stainless steel and NUREG/CR-6583 for carbon steel. Attach. 2, ¶ 38.

49. The 60-year fatigue results for the critical component locations are provided in Tables 5-8 through 5-14 of WCAP-17199-P and WCAP-17200-P. Westinghouse determined that, for IP2 and IP3, the refined CUFen values for pressurizer surge line piping, RCS piping charging system nozzle, RCS piping safety injection nozzle, and RHR Class 1 piping all are below 1.0 when projected to the end of the PEO. Attach. 2, ¶ 39; Attach. 14, attach. 1, at 2-4; Attach. 15, at 6-1; Attach. 16, at 6-1.

Respectfully submitted,

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Dated in Washington, D.C.
this 25th day of August 2010

DB1/65517382.1

NYS-26/26A & RIVERKEEPER TC-1/1A: ATTACHMENT 2

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	Docket Nos. 50-247-LR and 50-286-LR
ENTERGY NUCLEAR OPERATIONS, INC.	)	ASLBP No. 07-858-03-LR-BD01
(Indian Point Nuclear Generating Units 2 and 3)	)	August 20, 2010

DECLARATION OF NELSON F. AZEVEDO IN SUPPORT OF APPLICANT'S MOTION FOR SUMMARY DISPOSITION OF CONTENTIONS NYS-26/26A AND RIVERKEEPER TC-1/1A

Nelson F. Azevedo states as follows under penalties of perjury:

I. PERSONAL BACKGROUND

1. My name is Nelson Azevedo. I am employed by Entergy Nuclear Operations, Inc. ("Entergy") as Supervisor, Code Programs, at Indian Point Energy Center ("IPEC") in Buchanan, New York. I am providing this Declaration in support of the "Applicant's Motion for Summary Disposition of New York State Contentions 26/26A & Riverkeeper Technical Contentions 1/1A (Metal Fatigue of Reactor Components)" in the above-captioned proceeding. In April 2007, Entergy submitted a license renewal application ("LRA") to the U.S. Nuclear Regulatory Commission ("NRC") seeking to renew its operating licenses for Indian Point Nuclear Generating Units 2 and 3 ("IP2" and "IP3," collectively "IPEC") for another 20 years.

2. I have described my professional and educational qualifications in the *curriculum vitae* appended as Attachment 3. Briefly summarized, I hold a B.S. in Mechanical and Materials Engineering from the University of Connecticut and an M.S. in Mechanical Engineering from the Rensselaer Polytechnic Institute ("RPI") in Troy, New York. In

addition, I have received an M.B.A. from RPI. I have over 28 years of professional experience in the nuclear power industry. During that time, I have held engineering and supervisory positions with Northeast Utilities and Entergy. As a Department Manager with Northeast Utilities, I managed five engineering sections responsible for implementing numerous engineering programs at Millstone Station, including the fatigue monitoring program. Currently, I oversee the IPEC engineering section responsible for implementing American Society of Mechanical Engineers ("ASME") Code programs, including the inservice inspection ("ISI"), inservice testing, flow-accelerated corrosion, snubber testing, boric acid corrosion control, non-destructive examination, fatigue monitoring, steam generators, buried piping, alloy 600 cracking, reactor vessel embrittlement, welding, and 10 C.F.R. Part 50, Appendix J containment leakage programs. I also am responsible to ensure compliance with the ASME Code, Section XI requirements for repair and replacement activities at IPEC. I also represent IPEC before industry organizations, including the pressurized water reactor ("PWR") Owners Group Management Committee.

3. From January 2001 through approximately 2007, I was directly responsible for the IP2 Fatigue Monitoring Program and developed the plant procedure used to track the cycles for IP2. I now supervise the IPEC engineering staff responsible for implementing the IP2 and IP3 Fatigue Monitoring Programs. I reviewed draft versions of the Westinghouse environmental fatigue evaluations for IP2 and IP3 discussed below, and directly interfaced with Westinghouse personnel in resolving technical comments on those drafts before their final approval by IPEC. During my career, I have performed pipe stress analysis, finite element analysis of large components, ASME Code Section XI flaw evaluations, and ASME Code Section III, Class 1 fatigue analysis. Accordingly, I have personal knowledge of the

IPEC Fatigue Monitoring Program, including the description of that program in the LRA; relevant NRC requirements and guidance; and applicable industry codes.

II. TECHNICAL BACKGROUND CONCERNING METAL FATIGUE

A. Metal Fatigue and the Cumulative Usage Factor

4. Fatigue is the weakening of a metal caused by cyclic mechanical and thermal stresses (*i.e.*, cyclical loading) at a location on a metallic component. Metal components experience these stress cycles during "transients" such as plant startup and shutdown that result in significant temperature changes. A "stress cycle" is the time period it takes for a material to go from its minimum stress level to its maximum level and back again to its minimum level. An excessive number of cycles may result in a significant reduction in the strength of a component.

5. All materials have a distinctive number of stress cycles that the material can withstand at a particular applied stress level before fatigue failure occurs. The period during which this number of load cycles occurs is called the material's "fatigue life."

6. 10 C.F.R. Part 50 does not specifically mention metal fatigue; however, 10 C.F.R. § 50.55a(c)(1) requires that the reactor coolant system ("RCS") pressure boundary meet the requirements of the ASME Boiler and Pressure Vessel Code ("ASME Code"), Section III, as endorsed by the NRC. For an IPEC-vintage Westinghouse pressurized water reactor, the RCS pressure boundary includes components such as the reactor vessel primary inlet and outlet nozzles and pressurizer surge line nozzles.

7. The original design specifications for a given safety-related component specify the number of mechanical and thermal cycles that the component must be designed to withstand, and define the safety limits and applicable codes that must be satisfied. For components constituting the primary RCS pressure boundary, the specified requirements for

evaluation of cyclic loading and thermal conditions generally are contained in Section III of the ASME Code for Class 1 components.

8. For a Class 1 component, different stress cycles from the loadings specified in the governing design specification will produce total stresses of several different magnitudes. The number of times these stresses of different magnitudes occur also varies. The allowable number of cycles for a given alternating stress range is determined from the ASME Code design fatigue curve for the material being evaluated.

9. The fatigue usage for that stress cycle is the ratio of the number of analyzed applied stress cycles (n) to the allowable number of stress cycles (N) from the ASME Code design fatigue curve. The cumulative usage factor ("CUF") represents the fraction of the total allowable fatigue cycles that the component is projected to incur during its operation.

10. ASME Code Section III requires that the CUF for a Class 1 component not exceed unity or 1.0; *i.e.*, the total number of applied stress cycles is not to exceed the allowable number of stress cycles. This is an acceptance criterion established by the Code, but exceeding the criterion does not mean the component will fail, given the numerous factors of conservatism in the analytical process. CUF values result from conservative fatigue calculations. CUF values less than 1.0 indicate that the equipment can withstand the fatigue effects of cyclic stress due to operation during the analyzed number of design transients at the analyzed transient severity.

11. It is important to understand that a projected CUF greater than 1.0 does not indicate that fatigue cracking necessarily will occur. Rather, it indicates that there is a potential for cracking to initiate at the affected location. This is not necessarily failure of the component.

12. With regard to license renewal, 10 C.F.R. § 54.21(c)(1) lists the technical information that must be contained in an LRA relatedly specifically to time-limited aging analyses ("TLAAs"), which 10 C.F.R. § 54.3(a) defines as those licensee calculations and analyses that: (1) involve structures, systems, and components ("SSCs") within the scope of license renewal, as delineated in Section 54.4(a); (2) consider the effects of aging; (3) involve time-limited assumptions defined by the current operating term, for example, 40 years; (4) were determined to be relevant by the licensee in making a safety determination; (5) involve conclusions or provide the basis for conclusions related to the capability of the SSC to perform its intended functions, as delineated in Section 54.4(b); and (6) are contained or incorporated by reference in the current licensing basis ("CLB").

13. For all TLAAs, an applicant must demonstrate that (i) the analyses remain valid for the PEO; (ii) the analyses have been projected to the end of the PEO, or (iii) the aging effects will be adequately managed for the PEO. 10 C.F.R. § 54.21(c)(1)(i)-(iii).

14. The fatigue analyses of the RCS are treated as TLAAs because they are based on the numbers of cycles estimated during the original design to be adequate for the initial license term (40 years of operation.). A license renewal applicant may update the projected cycles to account for 60 years of plant operation. One possible outcome is that numbers of analyzed cycles at the end of the extended operating period will remain at or below those originally projected for the initial 40-year plant license period, in which case the governing fatigue analyses will remain valid for the PEO, in accordance with 10 C.F.R. § 54.21(c)(1)(i).

15. Another possibility is that more cycles are projected to occur for 60 years of plant operation than were analyzed for the first 40 years. In this case, an applicant must address the increased cycle counts. One possible approach is to revise the fatigue analysis to confirm that

the increased number of cycles still will result in a CUF less than or equal to the allowable design code limit, in accordance with 10 C.F.R. § 54.21(c)(1)(ii).

16. Finally, another acceptable approach is to determine the most limiting number of cycles assumed in the fatigue analyses. This cycle quantity then becomes the allowable limit against which the actual operation is tracked through an appropriate AMP during the PEO in accordance with 10 C.F.R. § 54.21(c)(1)(iii). In this regard, CUFs may provide predictive or tracking/monitoring functions.

17. As noted above, applicants may revise fatigue analyses to take advantage of excess conservatisms inherent in the original fatigue analyses (*e.g.*, excess conservatisms in loading definitions, predicted numbers of cycles, transient groupings, etc.) as well as improvements in available analytical tools. An industry guidance document issued by the Electric Power Research Institute ("EPRI") further explains this important concept:

> Whereas fatigue calculations have varied over the years, their basic content is the same. With the advent of computer technology, the calculations have basically maintained the same content, but computations have become more refined and exhaustive. For example, 30 years ago it was computationally difficult for a stress analyst to evaluate 100 different transients in a fatigue calculation. Therefore, the analyst would have grouped the transients into as few as one transient grouping and performed as few incremental fatigue calculations as possible. With today's computer technology and desire to show more margin, it is relatively easy for the modern-day analyst to evaluate all 100 incremental fatigue calculations for this same problem. Also, older technology would have likely utilized conservative shell interaction hand solutions for computing stress, whereas today finite element techniques are commonly deployed. This improvement in technology would not have changed the basic inputs to the fatigue calculation (*i.e.*, stress), but it would have typically yielded significantly more representative input values.

EPRI, MRP-47, *Materials Reliability Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application* at 4-4 (Rev. 1, Sept. 2005) (Attach. 6).

B. Environmentally-Assisted Metal Fatigue

18. For components (equipment and piping) exposed to reactor coolant water, the fatigue life, or allowable number of stress cycles, may be reduced compared to a component's fatigue life in air. The ASME Code design fatigue curves were developed based on laboratory testing of specimens in air at a constant strain rate, with safety factors incorporated into the curves to account for variables such as surface finish. Laboratory testing of specimens in water under reactor operating conditions indicate that additional environmental factors may need to be applied to the calculated CUF to fully account for reactor coolant environmental conditions. Fatigue analysis accounting for the effects of operating in a reactor coolant environment is called environmentally-assisted fatigue ("EAF") analysis.

19. EAF analysis was addressed in NRC Generic Safety Issue ("GSI") 190, "Fatigue Evaluation of Metal Components for 60-year Plant Life." The NRC Staff closed out GSI 190 in December 1999 without imposing additional requirements because it found only negligible calculated increases in core damage frequency in going from 40-year to 60-year license periods. However, the NRC Staff concluded that license renewal applicants should address the effects of coolant environment on component fatigue life as aging management programs ("AMPs") that are formulated for license renewal.

20. To address EAF, an applicant may apply an environmental correction factor, or "F_{en}", to a CUF, to calculate an environmentally-adjusted CUF, or "CUF_{en}". The F_{en} is defined as the ratio of the fatigue life in air at room temperature to that in water at the service temperature. The fatigue usage derived from current ASME Code fatigue design (air) curves is multiplied by the F_{en} to account for the environmental effects. Because EAF (CUF_{en}) analyses are not contained within an applicant's CLB, they are not TLAAs.

21. NRC guidance recommends the use of formulae in NUREG/CR-6583, *Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels* (Mar. 1998) (Attach. 7) to determine F_{en} values for carbon and low alloy steel components, and NUREG/CR-5704, *Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels* (Apr. 1999) (Attach. 8) to determine F_{en} values for stainless steel components. *See* NUREG-1800, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants*, Revision 1 at 4.3-5, 4.3-7 (Sept. 2005) ("SRP-LR") (Attach. 4); NUREG 1801, *Generic Aging Lessons Learned Report*, Rev. 1, Vol. 2 at X M-1 (Sept. 2005) ("GALL Report") (Attach. 5).

22. As discussed in the SRP-LR and GALL Report, one method for addressing EAF found acceptable by the NRC Staff is the conduct of CUF_{en} analyses for the six critical locations identified in NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components* (Mar. 1995) (Attach. 9). Those locations include: (1) the reactor vessel shell and lower head, (2) the reactor vessel inlet and outlet nozzles, (3) the pressurizer surge line (including hot leg and pressurizer nozzles), (4) RCS piping charging system nozzle, (5) RCS piping safety injection nozzle, and (6) residual heat removal ("RHR") Class 1 piping. Attach. 4 at 4.3-5; Attach. 5 at X M-1; Attach. 5 at 5-62.

III. IPEC'S AGING MANAGEMENT PROGRAM FOR METAL FATIGUE

A. The IPEC Fatigue Monitoring Program as Described in the LRA

23. Chapter 4 of the IPEC LRA summarizes Entergy's evaluation of the effects of metal fatigue and EAF on in-scope piping and components for the PEO. Section 4.3.1 (Class 1 Fatigue) addresses components (and subcomponents) that were designed in accordance with ASME Code, Section III. Broadly speaking, those components include the reactor vessel, reactor vessel internals, pressurizer, steam generators, reactor coolant pumps, and control rod drive mechanisms, Class 1 heat exchangers, and Class 1 piping and components. Section 4.3.2 (Non-Class 1 Fatigue) addresses ANSI Code B31.1 and ASME Code Section III Class 2 and 3 piping systems. Section 4.3.3 (Effects of Reactor Water Environment on Fatigue) addresses EAF with respect to the critical component locations identified in NUREG/CR-6260.

24. The appendices to the LRA contain a description of Entergy's Fatigue Monitoring Program. Appendix A presents the information required by 10 C.F.R. § 54.21(d) relating to the AMP for fatigue monitoring that supplements the updated FSAR ("UFSAR") for IPEC. The supplement to the UFSAR, presented in Sections A.2 and A.3 of Appendix A for IP2 and IP3, respectively, contains a summary description of the program and activities for managing the effects of metal fatigue during the PEO. Appendix A states that the Fatigue Monitoring Program will be implemented prior to the PEO. LRA app. A at A-1, A-22, A-49, *available at* ADAMS Accession No. ML071210520.

25. Appendix B to the LRA describes those AMPs credited by Entergy to manage the effects of aging. Section B.1.12 describes the IPEC Fatigue Monitoring Program, which is designed to track the number of transients for selected RCS components. LRA app. B at B-1, B-44-B-46, B-54, *available at* ADAMS Accession No. ML071210523. Section B.1.12

indicates that it is consistent with the program described in GALL Report Section X.M1, but takes one exception to the "detection of aging effects" program element. *Id.* Entergy, however, later removed this exception via an amendment to the LRA. NL-08-092, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "Amendment 5 to License Renewal Application" (June 11, 2008) (Attach. 13); 2 NUREG-1930, *Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3, Docket Nos. 50-247 and 50-286, Entergy Nuclear Operations, Inc.* at 3-79 to 3-80 (Aug. 2009) ("SER"), *available at* ADAMS Accession No. ML093170671.

26. As discussed in LRA Section 4.3.1, design basis fatigue evaluations for Class 1 components produce calculated CUFs for components and subcomponents based on selected numbers and magnitudes of design transient cycles. For RCS components, Entergy projected the numbers of cycles accrued to date to the end of the PEO; *i.e.*, the numbers of cycles expected at the end of 60 years of plant operation. LRA Tables 4.3-1 and 4.3-2 show the specific transient conditions, the analyzed numbers of cycles, and the projected numbers of cycles for 60 years of operation for IP2 and IP3, respectively. LRA Tables 4.3-3 to 4.3-12 list the CUFs for the various Class 1 components and subcomponents based on the numbers of cycles assumed in the analyses. The numbers of cycles expected at the end of 60 years of plant operation are less than the numbers considered in the analyses for all but a few of the transients listed in LRA Tables 4.3-1 and 4.3-2.

27. As explained in LRA Section 4.3.1.8, IPEC RCS piping was designed to ANSI B31.1, *Power Piping Code*, which did not require a detailed fatigue usage factor evaluation.[1]

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[1] 10 C.F.R. § 50.55a(c)(1) states that components that are part of the reactor coolant pressure boundary must meet the requirements for Class 1 components in Section III of the ASME Boiler and Pressure Vessel Code,

Footnote continued on next page

Instead, a stress analysis was performed on the main primary coolant piping, in accordance with the criteria set forth in ANSI B31.1, to ensure that the stress range was within the prescribed limits. Stress range reduction factors were used to account for anticipated transients for RCS piping (a stress range reduction factor of 1.0 is acceptable in the stress analyses for up to 7000 cycles). Therefore, Entergy evaluated the projected thermal cycles for 60 years of plant operation at IP2 and IP3 and determined that the total cycles in 60 years of operation are well below the 7000 cycles allowed by ANSI B31.1 for a stress range reduction factor of 1.0.

28. In preparing Section 4.3.3 of its April 2007 LRA, Entergy addressed EAF effects on those IPEC-specific components corresponding to the six critical locations identified in NUREG/CR-6260 by either projecting the analyses to the end of the PEO, per Section 54.21(c)(1)(ii), or demonstrating that aging effects will be adequately managed, per Section 54.21(c)(1)(iii). LRA at 4.3-20 to 4.3-25, *available at* ADAMS Accession No. ML071210517.

29. Consistent with the GALL Report, Entergy applied F_{en}s calculated as described in NUREG/CR-6583 and NUREG/CR-5704 to the 60-year-projected CUFs to determine CUF_{en} values. LRA at 4.3-21; tbls. 4.3.13 (IP2) & 4.3-14 (IP3). (As explained above, CUFs were not available for all RCS piping and RHR Class 1 piping locations because those locations were designed to ANSI B.31.1.). Entergy's F_{en} calculation methods are documented

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Footnote continued from previous page

except for the components described in 10 C.F.R. § 50.55a(c)(2), (c)(3), and (c)(4). As relevant here, Section 50.55a(c)(4) states: "For a nuclear power plant whose construction permit was issued prior to May 14, 1984 the applicable Code Edition and Addenda for a component of the reactor coolant pressure boundary continue to be that Code Edition and Addenda that were required by Commission regulations for such component at the time of issuance of the construction permit." The IP2 and IP3 RCS piping and associated nozzles were designed and fabricated to meet ANSI B31.1 requirements.

in a response to a Staff request for additional information. NL-08-084, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "Reply to Request for Additional Information Regarding License Renewal Application – Time-Limited Aging Analyses and Boraflex" attach. 1, at 3-4 (May 16, 2008) (Attach. 12).

30. As indicated in LRA Tables 4.3.13 and 4.3-14, the projected CUF_{en} values for the three NUREG/CR-6260 reactor vessel locations were less than 1.0 for both IP2 and IP3. These three reactor vessel locations included the (1) bottom head to shell, (2) reactor vessel inlet nozzle, and (3) reactor vessel outlet nozzle. In addition, the IP2 (but not IP3) pressurizer surge line nozzle had a CUF_{en} less than 1.0. Accordingly, because these CUF_{en} values fell below 1.0, the EAF analyses for those components were projected to the end of the PEO (in the LRA) in accordance with Section 54.21(c)(1)(ii) and no further action was necessary. LRA at 4.3-22.

31. The component locations in LRA Tables 4.3-13 and 4.3-14 without projected CUF_{en} values less than 1.0 include the pressurizer surge line piping for IP2 and IP3; the RCS piping charging system nozzles for IP2 (for which a CUF was available) and for IP3; and the pressurizer surge line nozzles for IP3.[2] LRA tbls. 4.3.13 & 4.3-14. To address these locations, Entergy committed to take one of the following actions: (1) refine the fatigue analyses, at least two years before entering the PEO, to determine valid CUF_{en} values less than 1.0; (2) manage the effects of aging due to fatigue at the affected locations by an inspection program that has been reviewed and approved by the NRC; or (3) repair or replace

[2] As explained above and in LRA Section 4.3.3, the IP2 pressurizer surge nozzle had a CUF_{en} less than 1.0, whereas the IP3 pressurizer surge nozzle had CUF_{en} greater than 1.0, because the IP3 surge nozzle calculation includes the effects of the insurges/outsurges experienced by these nozzles, while the IP2 analysis did not include these effects. As a result, Entergy committed to re-analyze the pressurizer surge line nozzle for both units to include insurge/outsurge and environmental effects. LRA at 4.3-21 to 4.3-22.

the affected locations before exceeding CUF of 1.0 (collectively referred to as "Commitment 33"). *Id.* at 4.3-22 to 4.3-23.

32. Subsequently, in January 2008, as a part of LRA Amendment 2, Entergy amended the Fatigue Monitoring Program to provide information on cycle counting and the methodology used to determine stresses and fatigue usage, including environmental effects, in accordance with the NRC-endorsed ASME Code. NL-08-021, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "License Renewal Application Amendment 2," (Jan. 22, 2008) ("LRA Amendment 2") (Attach. 10). Entergy also amended the LRA to place Commitment 33 within the scope of the Fatigue Monitoring Program, by stating that it will use that AMP to manage the effects of reactor water environment on fatigue life, in accordance with § 54.21(c)(1)(iii). *Id.* Pursuant to revised Commitment 33, if Entergy does not demonstrate valid CUF_{en} values below 1.0 (Option 1) by conducting refined CUF_{en} analyses, then it must implement Option 2 of the commitment. Option 2 requires Entergy to repair or replace the affected locations before their refined CUF_{en}s exceed 1.0, consistent with the Fatigue Monitoring Program. *See* Attach. 10, attach. 2, at 15.

B. <u>Summary of the NRC Staff's Safety Evaluation Report Findings</u>

33. As documented in its final Safety Evaluation Report ("SER"), the NRC Staff, after conducting a detailed technical review of the LRA and performing related site audits, found that IPEC's Fatigue Monitoring Program includes acceptable program elements that are consistent with criteria in GALL Report Section X.M1. 2 SER at 3-81. The NRC Staff further concluded that Entergy has demonstrated that the effects of aging will be adequately managed so that the component intended functions will be maintained consistent with the CLB for the PEO, as required by 10 C.F.R. § 54.21(a)(3). *Id.* In addition to evaluating IPEC's Fatigue Monitoring Program, the Staff reviewed LRA Section 4.3.3 regarding EAF.

As part of its review, the NRC Staff confirmed that IPEC had correctly determined the environmental factors used as inputs for calculating F_{en} factors. It also found that Commitment 33 is consistent with 10 C.F.R. § 54.21(a)(1)(iii). *Id.* at 4-43 to 4-45.

C. Overview of the 2010 Westinghouse Environmental Fatigue Evaluations

34. To meet Commitment 33 and address the admitted contentions, Entergy retained Westinghouse Electric Company LLC ("Westinghouse") in 2008 to perform refined fatigue analyses to determine environmentally-adjusted CUFs. Westinghouse completed the refined fatigue analyses in late June 2010. Westinghouse Electric Co., WCAP-17199-P, Rev. 0, *Environmental Fatigue Evaluation for Indian Point Unit 2* (June 2010) (proprietary) (Attach. 15); Westinghouse Electric Co., WCAP-17200-P, Rev. 0, *Environmental Fatigue Evaluation for Indian Point Unit 3* (June 2010) (proprietary) (Attach. 16). Entergy approved the analyses on July 29, 2010, and shortly thereafter, notified the Staff of the results of the refined EAF analyses. NL-10-082, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "License Renewal Application – Completion of Commitment #33 Regarding the Fatigue Monitoring Program" (Aug. 9, 2010) (Attach. 14).

35. As documented in WCAP-17199-P and WCAP-17200-P, and discussed further below, Westinghouse performed 60-year EAF evaluations using F_{en} factors calculated as described in NUREG/CR-5704 for stainless steel components and in NUREG/CR-6583 for those components containing carbon and low-alloy steel components. Westinghouse calculated the F_{en} factors, which it then applied to the ASME Code fatigue results.

36. As documented in WCAP-17199-P and WCAP-17200-P, the Westinghouse evaluations included the following four major steps: (1) determination of the limiting components for each location; (2) determination of the transients for each limiting component; (3) performance of ASME Code Section III stress and fatigue evaluations for

each limiting component; and (4) determination of the CUF_{en}, for each limiting component. Attach. 15, at 2-2 to 2-4; Attach. 16, at 2-2 to 2-4. As necessary, plant-specific projections of numbers of accrued cycles at the end of the PEO replaced conservative assumptions for numbers of cycles in the analyses. Attach. 15, at 2-4; Attach. 16, at 2-4. The fatigue evaluations followed the procedures given in ASME Code Section III, NB-3200. Attach. 15, at 5-20; Attach. 16, at 5-20.

37. Westinghouse developed transients using plant data and/or Westinghouse standard design transients, assuming the total period of operation to be 60 years. For the charging, safety injection, and RHR locations selected, Westinghouse developed stress inputs to the fatigue analyses using detailed finite element analysis models, and ran those models to create unit load transfer functions for each type of mechanical and thermal transient condition loading considered in the analyses. It also developed piping mechanical load functions to produce time history moment loads as a function of system parameters during a transient. The transfer functions and load functions were used in a proprietary Westinghouse computer code to determine detailed stress histories for each applicable transient, considering all applicable mechanical and thermal transient loads during each transient, and to calculate fatigue usage.

38. Section 5 of each Westinghouse EAF report details the manner in which Westinghouse evaluated EAF for the NRC-specified component locations at IP2 and IP3. As noted therein, Westinghouse's evaluations were based upon CUF analyses using transients that envelop the 60-year cycle projections. Sections 5.1 and 5.2, in particular, describe Westinghouse F_{en} calculations using the methodologies set forth in NUREG/CR-5704 for stainless steel and NUREG/CR-6583 for carbon steel.

39. The 60-year fatigue results for the critical component locations are provided in Tables 5-8 through 5-14 of WCAP-17199-P and WCAP-17200-P. Westinghouse determined that, for IP2 and IP3, the refined CUF_{en} values for pressurizer surge line piping, RCS piping charging system nozzle, RCS piping safety injection nozzle, and RHR Class 1 piping all are below 1.0 when projected to the end of the PEO. *See* Attach. 14, attach. 1, at 2-4; Attach. 15, at 6-1; Attach. 16, at 6-1.

IV. ISSUES RAISED IN NYS-26/26A AND RIVERKEEPER TC-1/1A

40. I understand that, as admitted by the Board, NYS-26/26A asserts that: (1) the LRA is incomplete without the calculations of the CUF_{en} values as threshold values necessary to assess the need for an AMP; (2) Entergy's AMP is inadequate for lack of final CUF_{en} values; and (3) the LRA must specify actions to be carried out by Entergy during the PEO to manage the aging of key reactor components susceptible to metal fatigue. *Entergy Nuclear Operations, Inc.* (Indian Point Nuclear Generating Units 2 & 3), LBP-08-13, 68 NRC 43, 116 (2008).

41. I have reviewed NYS-26/26A and NYS's supporting arguments. *See* New York State's Notice of Intention to Participate and Petition to Intervene at 227-233 (Nov. 30, 2007) ("NYS Petition"); Declaration of Dr. Richard T. Lahey, Jr. [in Support of New York State Contention 26], at 8-10 (Nov. 30, 2007); New York State Reply in Support of Petition to Intervene, at 124-130 (Feb. 22, 2008); Petitioner State of New York's Request for Admission of Supplemental Contention No. 26-A (Metal Fatigue) (Apr. 7, 2008); Declaration of Dr. Richard T. Lahey, Jr. in Support of the State of New York's Supplemental Contention 26-A (Apr. 7, 2008). I further understand that NYS alleges that Entergy, by not assessing certain projected CUF_{en} values exceeding 1.0, has not adequately shown that the TLAAs for metal fatigue are valid for the PEO. NYS further argues that Entergy has not provided sufficient

details concerning the analytical methods and assumptions to be used to calculate revised CUF_{en} values.

42. As admitted by the Board, I understand that Riverkeeper TC-1/1A asserts that issues exist relative to: (1) the extent to which an applicant must expand the scope of its TLAAs to meet the recommendations of the GALL Report and NUREG/CR-6260; (2) the extent, if any, that refinement of the CUF_{en} values is a valid corrective action and what relationship it has to the repair and replacement options; (3) the scope of the commitments needed to monitor, manage, and correct age-related degradation to meet NRC regulations; and (4) the degree of detail and specificity with which the repair or replacement decision criteria must be defined. *Indian Point,* LBP-08-13, 68 NRC at 167-68.

43. I have reviewed Riverkeeper TC-1/1A and its supporting arguments. *See* Riverkeeper, Inc.'s Request for Hearing and Petition to Intervene in Indian Point License Renewal Proceedings at 7-15 (Nov. 30, 2007) ("Riverkeeper Petition"); Declaration of Dr. Joram Hopenfeld in Support of Riverkeeper's Contentions TC-1 and TC-1 (Nov. 28, 2007); Riverkeeper, Inc.'s Reply to Entergy's and the NRC Staff's Responses to Hearing Request and Petition to Intervene, at 2-12 (Feb. 15, 2008); Riverkeeper, Inc.'s Request for Admission of Amended Contention (Mar. 5, 2008). Riverkeeper alleges, in principal part, that:

- Entergy's TLAA is "facially non-compliant" with 10 C.F.R. § 54.21(c)(1)(i)-(ii) because four representative components of the RCS have projected CUF_{en} values greater than 1.0. Entergy cannot avoid the "legal requirement" that the LRA is "required to demonstrate" under § 54.21(c)(1)(iii) that CUF_{en} values are less than 1.0. Any refined CUF_{en} analyses cannot be deferred until after a renewed license is granted. Riverkeeper Petition at 7-13.

- Entergy must submit a list of all components with CUF larger than unity, as well as an AMP that includes "clear criteria for determining when a defect in any one of these components is acceptable, when it is acceptable but requires monitoring, and when it is unacceptable and requires repairs." *Id.* at 13.

- Entergy must "broaden its TLAA analysis" beyond the scope of the representative components identified in Tables 4.3-13 and 4.3-14 to identify other components whose CUF may be greater than one. *Id.* at 7.

- Entergy's list of components with CUF_{en}s less than 1.0 in LRA Tables 4.3-13 and 4.3-14 is incomplete because Entergy's methods and assumptions for identifying those components are "unrealistic and inadequate." *Id.* at 7, 14. Entergy used an unrealistically low number of 2.45 for an F_{en} (instead of an F_{en} of 17); relied on the "CUF of Record" (40 years) instead of projecting the number of cycles to 60 years; and failed to calculate several NUREG-CR/6260 limiting locations because they are designed to ANSI B31.1, despite the availability of "generic CUF values" from NUREG/CR-6260. *Id.*

V. RESPONSE TO ISSUES RAISED IN CONTENTIONS NYS-26/26A AND RIVERKEEPER TC-1/1A

44. As a technical matter, NYS's and Riverkeeper's claim that Entergy's approach to addressing EAF is "vague" or "inadequate" is unfounded. The specific methodology and assumptions used by Westinghouse to determine refined CUF_{en} values for the relevant IPEC components are fully documented in WCAP-17199-P and WCAP-17200-P. The refined CUF_{en} analyses were performed in accordance with Westinghouse's Quality Assurance ("QA") Program, as approved by Entergy, and included design input verification and independent reviews to ensure that valid assumptions, transients, cycles, external loadings, analysis methods, and F_{en} factors were used in the EAF analyses for IP2 and IP3.

45. In accordance with Commitment 33, and at Entergy's request, Westinghouse prepared detailed stress models of the (1) surge line hot leg nozzle, (2) pressurizer surge nozzle, (3) RCS piping charging system nozzle, (4) RCS piping safety injection nozzle, and (5) RHR system class 1 piping locations for each unit using standard methods of the ASME Code, Section III.[3] As discussed above and in LRA Section 4.3.3, and reflected in LRA

[3] The pressurizer surge line nozzles were evaluated for environmental fatigue to address the surge line locations identified in NUREG/CR-6260. As explained in the Westinghouse's refined EAF analyses, the IP2 and IP3 surge lines were previously evaluated (in WCAP-12937) for the effects of thermal stratification and plant-specific transients. For IP2 and IP3, the controlling fatigue location was the surge line weld to the

Footnote continued on next page

Tables 4.3.13 and 4.3.14, Entergy previously applied bounding F_{en} values to the CUFs for the bottom head to shell region, the reactor vessel inlet nozzle, and the reactor vessel outlet nozzle and determined that the projected CUF_{en} values for these three locations are all below 1.0 for both units Therefore, refined CUF_{en} analyses were not necessary for these three reactor vessel locations. LRA at 4.3-21, 4.3-24 to 4.3-25; Attach. 14, attach. 1, at 2-4). As the Staff concluded in its SER, "the analyses performed for these components were projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii)." 2 SER at 4-42.

46. Westinghouse developed detailed stress history inputs for all transients relevant to fatigue of the affected component locations enumerated above and used those inputs in the EAF evaluations for those five locations As explained previously, these ASME Code evaluations were performed for the piping components to reduce excess conservatisms in the analyses, or because the CLB qualification for the component was to the ANSI B31.1 Power Piping Code, which did not require a fatigue usage factor (*i.e.*, CUF) calculation at the time of plant design.

47. In determining the refined CUF_{en} values for the component locations listed above, Westinghouse used F_{en} factors calculated using the NRC-approved methodology in NUREG/CR-5704 for the stainless steels in the pressurizer surge line, the reactor coolant piping charging and safety injection system nozzles, and the RHR system Class 1 piping. In addition, Westinghouse applied F_{en} factors calculated using the NRC-approved methodology in NUREG/CR-6583 for the carbon steel associated with the pressurizer surge nozzle

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Footnote continued from previous page

pressurizer surge nozzle in WCAP-12937. The surge line hot leg nozzle was the location evaluated in NUREG/CR-6260. Therefore, the IP2 and IP3 pressurizer surge nozzles and surge line hot leg nozzles were both evaluated to account for EAF effects. Attach. 15, at 2-1, 3-1; Attach. 16, at 2-1, 3-1.

dissimilar metal weld. The F_{en} factors were calculated using the detailed inputs described in the applicable NUREG and were applied directly to the ASME Code fatigue results. Attach. 15, at 1-2; Attach. 16, at 1-2. This approach is consistent with that described in both the SRP-LR and GALL Report.

48. There is no technical basis for Riverkeeper's claim that Entergy must use F_{en} values derived from NUREG/CR-6909, *Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials* (Feb. 2007). As noted above, Entergy's use of the formulae for calculating refined CUF_{en}s in NUREG/CR-6583 and NUREG/CR-5704 is consistent with NRC license renewal guidance. Additionally, use of the F_{en} factors derived from NUREG/CR-6909, as suggested by Riverkeeper, would actually yield less conservative CUF_{en} values, because the ASME Code design air curves for carbon steel and low-alloy steels contained in air in NUREG/CR-6583 and NUREG/CR-5704 are more conservative than the newer air curves in NUREG/CR-6909, as noted on page 81 of the latter document.

49. Riverkeeper's assertion that Entergy improperly relied on the "CUF of Record" (*i.e.*, 40 years) instead of projecting the number of cycles to 60 years is incorrect. As discussed above and clearly stated in LRA Section 4.3, Entergy projected the numbers of cycles accrued to date to determine the numbers of cycles expected at the end of 60 years of operation. LRA at 4.3-2. Furthermore, in its refined EAF analyses, Westinghouse performed 60-year fatigue analyses. Attach. 15 at 2-1, 6-1; Attach. 16 at 2-1, 6-1.

50. In the case of IPEC-specific components designed to ANSI B31.1, Entergy did not improperly refuse to use "generic CUF values." No such requirement exists. In any case, Westinghouse determined plant-specific CUF_{en} values for both IP2 and IP3 ANSI B31.1 piping that, without exception, are less than 1.0.

51. Contrary to Riverkeeper's claim, there is no requirement or need for Entergy to "broaden" its EAF analyses beyond the components identified in LRA Tables 4.3-13 and 4.3-14. As noted above, the refined CUF_{en} values for the critical locations listed in those tables, corresponding to the NUREG/CR-6260 locations, all are less than the ASME code limit of 1.0. Therefore, no further analyses are required for purposes of assessing EAF in the context of an LRA. As NUREG/CR-6260 explains, the six component locations identified therein "were chosen to give a representative overview of components that had higher CUFs and/or were important from a risk perspective." Attach. 9, at 4-1. Similarly, MRP-47, Revision 1 (which NYS and Riverkeeper reference multiple times) states that the six NRC-specified locations "were considered representative enough that the effects of LWR environment on fatigue could be assessed." Attach. 6, at 3-4. It further states: "For cases where acceptable fatigue results are demonstrated for these locations for 60 years of plant operation including environmental effects, additional evaluations or locations need not be considered." *Id.* (emphasis added).

52. Riverkeeper alleges that if the CUF_{en}s exceed 1.0, the second part of Commitment 33, or Option 2 (*i.e.*, repair or replace the affected locations), is too vague to meet the demonstration requirement of Part 54. As an initial matter, because the refined CUF_{en} values are less than 1.0 when projected to the end of the PEO (*i.e.*, 60 years of plant operation), there is no present need for corrective actions. Regardless, the second portion of Commitment 33 is consistent with the GALL Report and 10 C.F.R. § 54.21(c)(1)(iii), as the NRC Staff concluded in its SER. 2 SER at 4-44 to 4-47. GALL Report Section X.M1 states, in relevant part: "Acceptable corrective actions include repair of the component, replacement of the component, and a more rigorous analysis of the component to demonstrate that the

design code limit will not be exceeded during the extended period of operation." Attach. 5 at X M-2.

53. During the PEO, Entergy will continue to monitor the condition of piping and components at the locations of interest under IPEC's ISI Program, which includes periodic visual, surface, and volumetric examination of Classes 1, 2, and 3 pressure-retaining components, their attachments, and supports. If a flaw is detected during an ISI Program inspection or by other means, then the component may be replaced or repaired, or evaluated for continued service in accordance with the criteria contained in ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components." Acceptance standards for examination evaluations, repair procedures, inservice test requirements, and replacements for ASME Class 1 components are defined in ASME Section XI paragraphs IWB-3000, IWB-4000, IWB-5000 and IWB-7000, respectively.

54. Under the Fatigue Monitoring Program, Entergy will manage the effects of fatigue by monitoring cycles incurred and ensuring they do not exceed the analyzed numbers of cycles, such that the CUF_{en} analyses remain valid. LRA at 4.3-2 to 4.3-3; 2 SER at 4-44 to 4-45; NL-08-084, Attach. 12, attach. 1, at 3-4.

55. As required by the Fatigue Monitoring Program, Entergy tracks actual plant transients and evaluates these against the design transients. LRA at 4.3-2 to LRA 4.3-3; SER at 4-45; Attach. 10, attach. 1, at 6); Attach. 12, attach. 1, at 4. The plant transient counts will be updated at least once each operating cycle, which is an acceptable frequency because the evaluation during each update determines if the number of design transients could be exceeded prior to the next update. Attach. 12, attach. 1, at 4); SER at 3-79, 4-44.

56. Consistent with GALL Report Section X.M1, the Fatigue Monitoring Program requires that corrective actions be implemented in accordance with the IPEC Corrective

Action Program before the plant exceeds the analyzed number of transient cycles. Attach. 10, attach. 1, at 5-6; Attach. 12, attach. 1, at 4); SER at 4-44 to 4-45. IPEC procedures contain specific "alert levels" that trigger the initiation of corrective actions under the Fatigue Monitoring Program. SER at 4-44. Specifically, IPEC calculates alert levels by adding twice the number of cycles that occurred in the last fuel cycle to the total number of cycles to date. Corrective action is initiated if this alert level exceeds the number of analyzed transients. If the number of cycles is projected to remain at or below the analyzed level for two additional fuel cycles, then no corrective action is required. *Id.*. ; Attach. 10, attach. 1, at 5-6; Attach. 12, attach. 1, at 4.

57. Any further reanalysis (*i.e.*, future analysis updates), if necessary, would be governed by applicable QA procedures, as discussed above. Attach. 10, attach. 1, at 6; Attach. 12, attach. 1, at 4. Any repair or replacement of a component, if necessary, would be done in accordance with established plant procedures that must comply with Entergy's QA program and meet the applicable repair or replacement requirements of ASME Code Section XI. Attach. 10, attach. 1, at 6; Attach. 12, attach. 1, at 4. SER Sections 3.0.3.2.6, 4.1 and 4.3 discuss the IPEC Fatigue Monitoring Program in detail. 2 SER at 3-78 to 3-81, 4-19 to 4-39, 4-41 to 4-47.

58. Accordingly, Entergy's ISI Program, Fatigue Monitoring Program, and related ASME Code rules provide detailed and sufficient information concerning IPEC's future monitoring and corrective action activities related to environmentally-assisted fatigue during the PEO.

In accordance with 28 U.S.C. § 1746, I declare under penalty of perjury that the foregoing is true and correct.

Executed on AUGUST 20TH, 2010.

Nelson F. Azevedo

NYS-26/26A & RIVERKEEPER TC-1/1A: ATTACHMENT 3

Nelson Azevedo

Entergy, Indian Point Energy Center

Supervisor, Code Programs — January 2001 to present

Responsible for the Engineering Section that implements the American Society of Mechanical Engineers (ASME) Code programs, including Inservice Inspection (ISI), Inservice Testing (IST), Flow Accelerated Corrosion (FAC), Snubber Testing, Boric Acid Corrosion Control, Non-destructive Examination (NDE), Fatigue Monitoring, Steam Generators, Buried Piping, Alloy 600 cracking, Reactor Vessel Embrittlement, Welding and 10CFR50, Appendix J containment leakage program. Also responsible to ensure compliance with the ASME XI Code requirements for all repair and replacement activities at IPEC. Represent IPEC at industry organizations including the PWR Owners Group Management Committee.

Northeast Utilities

Department Manager, Materials Eng. & Code Programs — Jan. 1999 to Jan. 2001

Managed five Sections, with a staff of 38 engineers and other technical support staff responsible for implementation of engineering programs and structural integrity issues, at the Millstone Station. Examples of these programs include the ASME XI Inservice Inspection (ISI), Inservice Testing (IST), Non-Destructive Examinations (NDE), Steam Generator structural integrity (NEI 97-06), Reactor Vessel Embrittlement Management, Flow Accelerated Corrosion (FAC), Paints and Coatings, Fatigue Management and Welding programs. Represented the Seabrook and Millstone Units in the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) Senior Representatives Committee and in other ASME XI and Industry organizations. Responsible for development and management of department budgets, resources and implementation of policies and procedures. Part of the management team that recovered the Millstone Units after they were shutdown for safety and regulatory non-compliance issues.

Supervisor, Structural and Design Engineering — Sept. 1993 to Jan. 1999

Supervised a Section responsible for ensuring compliance with ASME XI, ASME III and ANSI B31.1 Code requirements. Technical areas of responsibility included flaw evaluations, piping stress analyses, ASME III piping and component fatigue evaluations, leak before break (LBB) implementation, finite element analysis, civil structures, FAC program and Reactor Vessel embrittlement. Responsible for structural integrity of large components including Low Pressure turbines, reactor vessels and steam generators as well as implementation of the A-46, seismic verification program for older units. Represented Seabrook, Connecticut Yankee and Millstone Units in several industry organizations including BWR Vessel Internals Project (BWRVIP), ASME Section XI and EPRI.

Engineer/Senior Engineer March 1982 to Sept. 1993

Responsible for resolving a variety of structural and equipment performance issues within the NU, Power Generation Division, which supported the Nuclear, Fossil and Hydro generating stations. Some of the issues included both high pressure and low pressure steam turbine creep and cracking issues, boiler tube failures, pump and valve performance issues and pipe support issues. Also responsible for performing pipe stress analyses, finite element analysis of large components, ASME XI Flaw evaluations, BWR intergranular stress corrosion cracking (IGSCC) evaluations, and reactor vessel structural integrity and embrittlement issues. Performed ASME III, Class 1 fatigue analyses, developed 10CFR50, Appendix G heat up and cooldown curves, designed weld overlays to mitigate IGSCC and performed upper shelf energy analysis for Connecticut Yankee to respond to GL 92-01 issues. Responsible for implementation of several large projects, including steam generator tube sleeving and plugging at Millstone 2 and the Connecticut Yankee FAC program following pipe ruptures. Active participant in the industry resolution of Pressurized Thermal Shock issues (PTS rule) and implementation of the LBB methodology that was published by the NRC in NUREG-1061. A member of the EPRI team that developed the SAFER computer program, which was designed to perform run/repair/retire evaluations of degraded High and Low Pressure turbines.

Education

BS, Mechanical and Materials Engineering
University of Connecticut

MS, Mechanical Engineering
Rensselaer Polytechnic Institute

MBA, General Management
Rensselaer Polytechnic Institute

NYS-26/26A & RIVERKEEPER TC-1/1A: ATTACHMENT 4

NUREG-1800, Rev. 1

Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants

Manuscript Completed: September 2005
Date Published: September 2005

Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

3.0 INTRODUCTION TO STAFF REVIEW OF AGING MANAGEMENT

The NRC project manager (PM) responsible for the safety review of the license renewal application (LRA) is responsible for assigning to appropriate NRC Office of Nuclear Reactor Regulation (NRR) divisions the review or audit of aging management reviews (AMRs) or aging management programs (AMPs) identified in the applicant's LRA. The PM should document to which organization each AMR or AMP is assigned. The assigned AMRs and AMPs should be reviewed per the criteria described in Sections 3.1 through 3.6 of this standard review plan (SRP-LR, NUREG-1800) for review of license renewal applications, as directed by the scope of each of these sections.

The NRC divisions that are usually assigned responsibility for the review of AMRs and AMPs are the Division of Engineering (DE), Division of System Safety Analysis (DSSA), and the Division of Regulatory Improvement Program (DRIP) License Renewal and Environmental Impacts Program (RLEP). Typically, the PM will assign DRIP/RLEP to review the AMRs and AMPs that the LRA identifies as being consistent with the GALL Report or NRC-approved precedents. As common exceptions to this assignment, the PM will assign to DE those AMRs and AMPs that address issues identified as emerging technical issues. Usually, AMRs and AMPs that are not in one of the aforementioned categories are assigned to DE.

Review of the AMPs requires assessment of ten program elements as defined in this SRP-LR. The NRC divisions assigned the AMP should review the ten program elements to verify their technical adequacy. For three of the ten program elements (corrective actions, confirmation process, and administrative controls) the NRC division responsible for quality assurance should verify that the applicant has documented a commitment in the FSAR Supplement to expand the scope of its 10 CFR Part 50, Appendix B program to address the associated program elements for each AMP. If the applicant chooses alternate means of addressing these three program elements (e.g., use of a process other than the applicant's 10 CFR Part 50, Appendix B program), the NRC divisions assigned to review the AMP should request that the Division responsible for quality assurance review the applicant's proposal on a case-by-case basis.

3.0.1 Background on the Types of Reviews

10 CFR 54.21(a)(3) requires that the LRA must demonstrate, for systems, structures, and components (SSCs) identified in the scope of license renewal and subject to an AMR pursuant to 10 CRF 54.21(a)(1), that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis (CLB) for the period of extended operation. This AMR consists of identifying the material, environment, aging effects, and the AMP(s) credited for managing the aging effects.

Sections 3.1 through 3.6 of this SRP-LR describe how the AMRs and AMPs are reviewed. One method that the applicant may use to conduct its AMRs is to satisfy the NUREG-1801 (GALL Report) recommendations. The applicant may choose to use methodology other than that in the GALL Report to demonstrate compliance with 10 CFR 54.21(a)(3).

As stated in the GALL Report:

> The GALL Report is a technical basis document to the SRP-LR, which provides the staff with guidance in reviewing a license renewal application. The GALL Report should be treated in the same manner as an approved topical report that is generically applicable. An applicant may reference the GALL Report in a license renewal application to

demonstrate that the programs at the applicant's facility correspond to those reviewed and approved in the GALL Report and that no further staff review is required, as described in the next paragraph. If the material presented in the GALL Report is applicable to the applicant's facility, the staff should find the applicant's reference to the GALL Report acceptable. In making this determination, the staff should consider whether the applicant has identified specific programs described and evaluated in the GALL Report. The staff, however, should not conduct a re-review of the substance of the matters described in the GALL Report. Rather, the staff should ensure that the applicant verifies that the approvals set forth in the GALL Report for generic programs apply to the applicant's programs. The focus of the staff review should be on augmented programs for license renewal. The staff should also review information that is not addressed in the GALL Report or is otherwise different from that in the GALL Report.

If an applicant takes credit for a program in the GALL Report, it is incumbent on the applicant to ensure that the plant program contains all the elements of the referenced GALL Report program. In addition, the conditions at the plant must be bounded by the conditions for which the GALL Report program was evaluated. The above verifications must be documented on-site in an auditable form. The applicant should include a certification in the license renewal application that the verifications have been completed and are documented on-site in an auditable form.

The GALL Report contains one acceptable way to manage aging effects for license renewal. An applicant may propose alternatives for staff review in its plant-specific license renewal application. Use of the GALL Report is not required, but its use should facilitate both preparation of a license renewal application by an applicant and timely, uniform review by the NRC staff.

In addition, the GALL Report does not address scoping of structures and components for license renewal. Scoping is plant-specific, and the results depend on the plant design and current licensing basis. The inclusion of a certain structure or component in the GALL Report does not mean that this particular structure or component is within the scope of license renewal for all plants. Conversely, the omission of a certain structure or component in the GALL Report does not mean that this particular structure or component is not within the scope of license renewal for any plants.

The GALL Report contains an evaluation of a large number of structures and components that may be in the scope of a typical LRA. The evaluation results documented in the GALL Report indicate that many existing, typical generic aging management programs are adequate to manage aging effects for particular structures or components for license renewal without change. The GALL Report also contains recommendations on specific areas for which generic existing programs should be augmented (require further evaluation) for license renewal and documents the technical basis for each such determination. In addition, the GALL Report identifies certain SSCs that may or may not be subject to particular aging effects, and for which industry groups are developing generic aging management programs or investigating whether aging management is warranted. To the extent the ultimate generic resolution of such an issue will need NRC review and approval for plant-specific implementation, as indicated in a plant-specific FSAR supplement, and reflected in the SER associated with a particular LR application, an amendment pursuant to 10 CFR 50.90 will be necessary.

In this SRP-LR, Subsection 3.X.2 (where X denotes number 1-6) presents the acceptance criteria describing methods to determine whether the applicant has met the requirements of NRC's regulations in 10 CFR 54.21. Subsection 3.X.3 presents the review procedures to be followed. Some rows (line-items) in the AMR tables (in Chapters II through VIII of the GALL Report, Vol. II) establish the need to perform "further evaluations." The acceptance criteria for satisfying these "further evaluations" are found in Subsections 3.X.2.2. The related review procedures are provided in Subsections 3.X.3.2.

In Regulatory Guide 1.188, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses," the NRC has endorsed an acceptable methodology for applicants to structure license renewal applications. Using the guidance described in the aforementioned Regulatory Guide, the applicant documents in the LRA whether its AMR line-item is consistent or not consistent with the GALL Report.

A portion of the AMR includes the assessment of the AMPs in the GALL Report. The applicant may choose to use an AMP that is consistent with the GALL Report AMP, or may choose a plant-specific AMP.

If a GALL Report AMP is selected to manage aging, the applicant may take one or more exceptions to specific GALL Report AMP program elements. However, any deviation or exception to the GALL Report AMP should be described and justified. Exceptions are portions of the GALL Report AMP that the applicant does not intend to implement.

In some cases, an applicant may choose an existing plant program that does not currently meet all the program elements defined in the GALL Report AMP. If this is the situation, the applicant may make a commitment to augment the existing program to satisfy the GALL Report AMP element prior to the period of extended operation. This commitment is an AMP enhancement.

Enhancements are revisions or additions to existing aging management programs that the applicant commits to implement prior to the period of extended operation. Enhancements include, but are not limited to, those activities needed to ensure consistency with the GALL Report recommendations. Enhancements may expand, but not reduce, the scope of an AMP.

An audit and review is conducted at the applicant's facility to evaluate those AMRs or AMPs that the applicant claims to be consistent with the GALL Report. An audit also includes technical assessments of exceptions or enhancements to the GALL Report AMP program elements. Reviews are performed to address those AMRs or AMPs related to emergent issues, stated to be not consistent with the GALL Report, or based on an NRC-approved precedent (e.g., AMRs and AMPs addressed in an NRC SER of a previous LRA). As a result of the criteria established in 10 CFR Part 54, and the guidance provided in SRP-LR, GALL Report, Regulatory Guide 1.188, and the applicant's exceptions and/or enhancements to a GALL Report AMP, the following types of AMRs and AMPs should be audited or reviewed by the NRC staff.

AMRs

- AMR results consistent with the GALL Report
- AMR results for which further evaluation is recommended by the GALL Report
- AMR results not consistent with or not addressed in the GALL Report

AMPs

- Consistent with GALL Report AMPs

- Plant-specific AMPs

FSAR Supplement

- Each LRA AMP will provide an FSAR Supplement which defines changes to the FSAR that will be made as a condition of a renewed license. This FSAR Supplement defines the aging management programs the applicant is crediting to satisfy 10 CFR 54.21(a)(3).
- The FSAR Supplement should also contain a commitment to implement the LRA AMP enhancement prior to the period of extended operation.

3.0.2 Applications with approved Extended Power Uprates

Extended power uprates (EPU) are licensing actions that some licensees have recently requested the NRC staff to approve. This can affect aging management. In a NRC staff letter to the Advisory Committee on Reactor Safeguards, dated October 26, 2004, (ADAMS Accession ML042790085), the NRC Executive Director for Operation states that, "All license renewal applications with an approved EPU will be required to perform an operating experience review and its impact on [aging] management programs for structures, and components before entering the period of extended operation." One way for an applicant with an approved EPU to satisfy this criterion is to document its commitment to perform an operating experience review and its impact on aging management programs for systems, structures, and components (SSCs) before entering the period of extended operation as part of its license renewal application. Such licensee commitments should be documented in the NRC staff's SER written in support of issuing a renewed license. The staff expects to impose a license condition on any renewed license to ensure that the applicant will complete these activities no later than the committed date. EPU impact on SSCs should be part of the license renewal review. If necessary, the PM will assign a responsible group to address EPU.

4.3 METAL FATIGUE ANALYSIS

Review Responsibilities

Primary - Branch responsible for the TLAA issues
Secondary - None

4.3.1 Areas of Review

A metal component subjected to cyclic loading at loads less than the static design load may fail because of fatigue. Metal fatigue of components may have been evaluated based on an assumed number of transients or cycles for the current operating term. The validity of such metal fatigue analysis is reviewed for the period of extended operation.

The metal fatigue analysis review includes, as appropriate, a review of in service flaw growth analyses, reactor vessel underclad cracking analysis, reactor vessel internals fatigue analysis, postulated high energy line break, leak-before-break, RCP flywheel, and metal bellows.

4.3.1.1 Time-Limited Aging Analysis

Metal components may be designed or analyzed based on requirements in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code or the American National Standards Institute (ANSI) guidance. These codes contain explicit metal fatigue or cyclic considerations based on TLAAs.

4.3.1.1.1 ASME Section III, Class 1

ASME Class 1 components, which include core support structures, are analyzed for metal fatigue. ASME Section III (Ref. 1) requires a fatigue analysis for Class 1 components that considers all transient loads based on the anticipated number of transients. A Section III Class 1 fatigue analysis requires the calculation of the "cumulative usage factor" (CUF) based on the fatigue properties of the materials and the expected fatigue service of the component. The ASME Code limits the CUF to a value of less than or equal to one for acceptable fatigue design. The fatigue resistance of these components during the period of extended operation is an area of review.

4.3.1.1.2 ANSI B31.1

ANSI B31.1 (Ref. 2) applies only to piping. It does not call for an explicit fatigue analysis. It specifies allowable stress levels based on the number of anticipated thermal cycles. The specific allowable stress reductions due to thermal cycles are listed in Table 4.3-1. For example, the allowable stress would be reduced by a factor of 1.0, i.e., no reduction, for piping that is not expected to experience more than 7,000 thermal cycles during plant service, but would be reduced to half of the maximum allowable static stress for 100,000 or more thermal cycles. The fatigue resistance of these components during the period of extended operation is an area of review.

4.3.1.1.3 Other Evaluations Based on CUF

The codes also contain metal fatigue analysis criteria based on a CUF calculation [the 1969 edition of ANSI B31.7 (Ref. 3) for Class 1 piping, ASME NC-3200 vessels, ASME NE-3200

Class MC components, and metal bellows designed to ASME NC-3649.4(e)(3), ND-3649.4(e)(3), or NE-3366.2(e)(3)]. For these components, the discussion relating to ASME Section III, Class 1 in Subsection 4.3.1.1.1 of this review plan section applies.

4.3.1.1.4 ASME Section III, Class 2 and 3

ASME Section III, Class 2 and 3 piping cyclic design requirements are similar to the guidance in ANSI B31.1. The discussion relating to B31.1 in Subsection 4.3.1.1.2 of this review plan section applies.

4.3.1.2 Generic Safety Issue

The fatigue design criteria for nuclear power plant components have changed as the industry consensus codes and standards have developed. The fatigue design criteria for a specific component depend on the version of the design code that applied to that component, i.e., the code of record. There is a concern that the effects of the reactor coolant environment on the fatigue life of components were not adequately addressed by the code of record.

The NRC has decided that the adequacy of the code of record relating to metal fatigue is a potential safety issue to be addressed by the current regulatory process for operating reactors (Refs. 4 and 5). The effects of fatigue for the initial 40-year reactor license period were studied and resolved under Generic Safety Issue (GSI)-78, "Monitoring of Fatigue Transient Limits for reactor coolant system," and GSI-166, "Adequacy of Fatigue Life of Metal Components" (Ref. 6). GSI-78 addressed whether fatigue monitoring was necessary at operating plants. As part of the resolution of GSI-166, an assessment was made of the significance of the more recent fatigue test data on the fatigue life of a sample of components in plants where Code fatigue design analysis had been performed. The efforts on fatigue life estimation and ongoing issues under GSI-78 and GSI-166 for 40-year plant life were addressed separately under a staff generic task action plan (Refs. 7 and 8). The staff documented its completion of the fatigue action plan in SECY-95-245 (Ref. 9).

SECY-95-245 was based on a study described in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components" (Ref. 10). In NUREG/CR-6260, sample locations with high fatigue usage were evaluated. Conservatisms in the original fatigue calculations, such as actual cycles versus assumed cycles, were removed, and the fatigue usage was recalculated using a fatigue curve considering the effects of the environment. The staff found that most of the locations would have a CUF of less than the ASME Code limit of 1.0 for 40 years. On the basis of the component assessments, supplemented by a 40-year risk study, the staff concluded that a backfit of the environmental fatigue data to operating plants could not be justified. However, because the staff was less certain that sufficient excessive conservatisms in the original fatigue calculations could be removed to account for an additional 20 years of operation for renewal, the staff recommended in SECY-95-245 that the samples in NUREG/CR-6260 should be evaluated considering environmental effects for license renewal. GSI-190, "Fatigue Evaluation of Metal Components for 60-year Plant Life," was established to address the residual concerns of GSI-78 and GSI-166 regarding the environmental effects on fatigue of pressure boundary components for 60 years of plant operation.

The scope of GSI-190 included design basis fatigue transients. It studied the probability of fatigue failure and its effect on core damage frequency (CDF) of selected metal components for 60-year plant life. The results showed that some components have cumulative probabilities of

crack initiation and through-wall growth that approach one within the 40- to 60-year period. The maximum failure rate (through-wall cracks per year) was in the range of 10^{-2} per year, and those failures were generally associated with high cumulative usage factor locations and components with thinner walls, i.e., pipes more vulnerable to through-wall cracks. In most cases, the leakage from these through-wall cracks is small and not likely to lead to core damage. It was concluded that no generic regulatory action is necessary and that GSI-190 is resolved based on results of probabilistic analyses and sensitivity studies, interactions with the industry (NEI and EPRI), and different approaches available to licensees to manage the effects of aging (Refs. 11 and 12).

However, the calculations supporting resolution of this issue, which included consideration of environmental effects, indicate the potential for an increase in the frequency of pipe leaks as plants continue to operate. Thus, the staff concluded that licensees are to address the effects of coolant environment on component fatigue life as aging management programs are formulated in support of license renewal.

The applicant's consideration of the effects of coolant environment on component fatigue life for license renewal is an area of review.

4.3.1.3 FSAR Supplement

Detailed information on the evaluation of TLAAs is contained in the renewal application. A summary description of the evaluation of TLAAs for the period of extended operation is contained in the applicant's FSAR supplement. The FSAR supplement is an area of review.

4.3.2 Acceptance Criteria

The acceptance criteria for the areas of review described in Subsection 4.3.1 of this review plan section delineate acceptable methods for meeting the requirements of the NRC's regulations in 10 CFR 54.21(c)(1).

4.3.2.1 Time-Limited Aging Analysis

Pursuant to 10 CFR 54.21(c)(1)(i) - (iii), an applicant must demonstrate one of the following:

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

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

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

Specific acceptance criteria for metal fatigue are:

4.3.2.1.1 ASME Section III, Class 1

For components designed or analyzed to ASME Class 1 requirements, the acceptance criteria, depending on the applicant's choice of 10 CFR 54.21(c)(1)(i), (ii), or (iii), are:

4.3.2.1.1.1 10 CFR 54.21(c)(1)(i)

The existing CUF calculations remain valid because the number of assumed transients would not be exceeded during the period of extended operation.

4.3.2.1.1.2 10 CFR 54.21(c)(1)(ii)

The CUF calculations have been reevaluated based on an increased number of assumed transients to bound the period of extended operation. The resulting CUF remains less than or equal to unity for the period of extended operation.

4.3.2.1.1.3 10 CFR 54.21(c)(1)(iii)

In Chapter X of the GALL report (Ref. 13), the staff has evaluated a program for monitoring and tracking the number of critical thermal and pressure transients for the selected reactor coolant system components. The staff has determined that this program is an acceptable aging management program to address metal fatigue of the reactor coolant system components according to 10 CFR 54.21(c)(1)(iii). The GALL report may be referenced in a license renewal application and should be treated in the same manner as an approved topical report. In referencing the GALL report, the applicant should indicate that the material referenced is applicable to the specific plant involved and should provide the information necessary to adopt the finding of program acceptability as described and evaluated in the report. The applicant should also verify that the approvals set forth in the GALL report for the generic program apply to the applicant's program.

4.3.2.1.2 ANSI B31.1

For piping designed or analyzed to B31.1, the acceptance criteria, depending on the applicant's choice of 10 CFR 54.21(c)(1)(i), (ii), or (iii), are:

4.3.2.1.2.1 10 CFR 54.21(c)(1)(i)

The existing fatigue strength reduction factors remain valid because the number of cycles would not be exceeded during the period of extended operation.

4.3.2.1.2.2 10 CFR 54.21(c)(1)(ii)

The fatigue strength reduction factors have been reevaluated based on an increased number of assumed thermal cycles and the stress reduction factors (e.g., Table 4.3-1) given in the applicant's code of record to bound the period of extended operation. The adjusted fatigue strength reduction factors are such that the component design basis remains valid during the period of extended operation.

4.3.2.1.2.3 10 CFR 54.21(c)(1)(iii)

The effects of aging on the intended function(s) will be adequately managed for the period of extended operation. The component could be replaced and the allowable stresses for the replacement will be sufficient as specified by the code during the period of extended operation.

Alternative acceptance criteria under 10 CFR 54.21(c)(1)(iii) have yet to be developed. They will be evaluated on a case-by-case basis to ensure that the aging effects will be managed such that the intended functions(s) will be maintained during the period of extended operation.

4.3.2.1.3 Other Evaluations Based on CUF

The acceptance criteria in Subsection 4.3.2.1.1 of this review plan section apply.

4.3.2.1.4 ASME Section III, Class 2 and 3

The acceptance criteria in Subsection 4.3.2.1.2 of this review plan section apply.

4.3.2.2 Generic Safety Issue

The staff recommendation for the closure of GSI-190 is contained in a December 26, 1999 memorandum from Ashok Thadani to William Travers (Ref. 11). The staff recommended that licensees address the effects of the coolant environment on component fatigue life as aging management programs are formulated in support of license renewal. One method acceptable to the staff for satisfying this recommendation is to assess the impact of the reactor coolant environment on a sample of critical components. These critical components should include, as a minimum, those selected in NUREG/CR-6260 (Ref. 10). The sample of critical components can be evaluated by applying environmental correction factors to the existing ASME Code fatigue analyses. Formulas for calculating the environmental life correction factors for carbon and low-alloy steels are contained in NUREG/CR-6583 (Ref. 14) and those for austenitic SSs are contained in NUREG/CR-5704 (Ref. 15).

4.3.2.3 FSAR Supplement

The specific criterion for meeting 10 CFR 54.21(d) is:

> The summary description of the evaluation of TLAAs for the period of extended operation in the FSAR supplement is appropriate such that later changes can be controlled by 10 CFR 50.59. The description should contain information associated with the TLAAs regarding the basis for determining that the applicant has made the demonstration required by 10 CFR 54.21(c)(1).

4.3.3 Review Procedures

For each area of review described in Subsection 4.3.1, the following review procedures should be followed:

4.3.3.1 Time-Limited Aging Analysis

4.3.3.1.1 ASME Section III, Class 1

For components designed or analyzed to ASME Class 1 requirements, the review procedures, depending on the applicant's choice of 10 CFR 54.21(c)(1)(i), (ii), or (iii), are:

4.3.3.1.1.1 10 CFR 54.21(c)(1)(i)

The operating transient experience and a list of the assumed transients used in the existing CUF calculations for the current operating term are reviewed to ensure that the number of assumed transients would not be exceeded during the period of extended operation.

4.3.3.1.1.2 10 CFR 54.21(c)(1)(ii)

The operating transient experience and a list of the increased number of assumed transients projected to the end of the period of extended operation are reviewed to ensure that the transient projection is adequate. The revised CUF calculations based on the projected number of assumed transients are reviewed to ensure that the CUF remains less than or equal to one at the end of the period of extended operation.

The code of record should be used for the reevaluation, or the applicant may update to a later code edition pursuant to 10 CFR 50.55a. In the latter case, the reviewer verifies that the requirements in 10 CFR 50.55a are met.

4.3.3.1.1.3 10 CFR 54.21(c)(1)(iii)

The applicant may reference the GALL report in its license renewal application, as appropriate. The review should verify that the applicant has stated that the report is applicable to its plant with respect to its program that monitors and tracks the number of critical thermal and pressure transients for the selected reactor coolant system components. The reviewer verifies that the applicant has identified the appropriate program as described and evaluated in the GALL report. The reviewer also ensures that the applicant has stated that its program contains the same program elements that the staff evaluated and relied upon in approving the corresponding generic program in the GALL report. No further staff evaluation is necessary.

4.3.3.1.2 ANSI B31.1

For piping designed or analyzed to ANSI B31.1 guidance, the review procedures, depending on the applicant's choice of 10 CFR 54.21(c)(1)(i), (ii), or (iii), are:

4.3.3.1.2.1 10 CFR 54.21(c)(1)(i)

The operating cyclic experience and a list of the assumed thermal cycles used in the existing allowable stress determination are reviewed to ensure that the number of assumed thermal cycles would not be exceeded during the period of extended operation.

4.3.3.1.2.2 10 CFR 54.21(c)(1)(ii)

The operating cyclic experience and a list of the increased number of assumed thermal cycles projected to the end of the period of extended operation are reviewed to ensure that the thermal cycle projection is adequate. The revised allowable stresses based on the projected number of assumed thermal cycles and the stress reduction factors given in the applicant's code of record are reviewed to ensure that they remain sufficient as specified by the code during the period of extended operation. Typical stress reduction factors based on thermal cycles are given in Table 4.3-1.

The code of record should be used for the reevaluation, or the applicant may use the criteria of 10 CFR 50.55a. In the latter case, the reviewer verifies that the requirements in 10 CFR 50.55a are met.

4.3.3.1.2.3 10 CFR 54.21(c)(1)(iii)

The applicant's proposed program to ensure that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation is reviewed. If the applicant proposed a component replacement before it exceeds the assumed thermal cycles, the reviewer verifies that the allowable stresses for the replacement will remain sufficient as specified by the code during the period of extended operation. Other applicant-proposed programs will be reviewed on a case-by-case basis.

4.3.3.1.3 Other Evaluations Based on CUF

The review procedures in Subsection 4.3.3.1.1 of this review plan section apply.

4.3.3.1.4 ASME Section III, Class 2 and 3

The review procedures in Subsection 4.3.3.1.2 of this review plan section apply.

4.3.3.2 Generic Safety Issue

The reviewer verifies that the applicant has addressed the staff recommendation for the closure of GSI-190 contained in a December 26, 1999 memorandum from Ashok Thadani to William Travers (Ref. 11). The reviewer verifies that the applicant has addressed the effects of the coolant environment on component fatigue life as aging management programs are formulated in support of license renewal. If an applicant has chosen to assess the impact of the reactor coolant environment on a sample of critical components, the reviewer verifies the following:

1. The critical components include, as a minimum, those selected in NUREG/CR-6260 (Ref. 10).

2. The sample of critical components has been evaluated by applying environmental correction factors to the existing ASME Code fatigue analyses.

3. Formulas for calculating the environmental life correction factors are those contained in NUREG/CR–6583 (Ref. 14) for carbon and low-alloy steels, and in NUREG/CR-5704 (Ref. 15) for austenitic SSs, or an approved technical equivalent.

4.3.3.3 FSAR Supplement

The reviewer verifies that the applicant has provided information, to be included in the FSAR supplement that includes a summary description of the evaluation of the metal fatigue TLAA. Table 4.3-2 contains examples of acceptable FSAR supplement information for this TLAA. The reviewer verifies that the applicant has provided a FSAR supplement with information equivalent to that in Table 4.3-2.

The staff expects to impose a license condition on any renewed license to require the applicant to update its FSAR to include this FSAR supplement at the next update required pursuant to 10 CFR 50.71(e)(4). As part of the license condition, until the FSAR update is complete, the

applicant may make changes to the programs described in its FSAR supplement without prior NRC approval, provided that the applicant evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59. If the applicant updates the FSAR to include the final FSAR supplement before the license is renewed, no condition will be necessary.

As noted in Table 4.3-2, an applicant need not incorporate the implementation schedule into its FSAR. However, the reviewer should verify that the applicant has identified and committed in the license renewal application to any future aging management activities, including enhancements and commitments to be completed before the period of extended operation. The staff expects to impose a license condition on any renewed license to ensure that the applicant will complete these activities no later than the committed date.

4.3.4 Evaluation Findings

The reviewer determines whether the applicant has provided sufficient information to satisfy the provisions of this section and whether the staff's evaluation supports conclusions of the following type, depending on the applicant's choice of 10 CFR 54.21(c)(1)(i), (ii), or (iii), to be included in the staff's safety evaluation report:

> On the basis of its review, as discussed above, the staff concludes that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1), that, for the metal fatigue TLAA, [choose which is appropriate] (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. The staff also concludes that the FSAR Supplement contains an appropriate summary description of the metal fatigue TLAA evaluation for the period of extended operation as reflected in the license condition.

4.3.5 Implementation

Except in those cases in which the applicant proposes an acceptable alternative method, the method described herein will be used by the staff in its evaluation of conformance with NRC regulations.

4.3.6 References

1. ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Power Plant Components," American Society of Mechanical Engineers.

2. ANSI/ASME B31.1, "Power Piping," American National Standards Institute.

3. ANSI/ASME B31.7-1969, "Nuclear Power Piping," American National Standards Institute.

4. SECY-93-049, "Implementation of 10 CFR Part 54, 'Requirements for Renewal of Operating Licenses for Nuclear Power Plants,'" March 1, 1993.

5. Staff Requirements Memorandum from Samuel J. Chilk, dated June 28, 1993.

6. NUREG-0933, "A Prioritization of Generic Safety Issues," Supplement 20, July 1996.

7. Letter from William T. Russell of NRC to William Rasin of the Nuclear Management and Resources Council, dated July 30, 1993.

8. SECY-94-191, "Fatigue Design of Metal Components," July 26, 1994.

9. SECY-95-245, "Completion of The Fatigue Action Plan," September 25, 1995.

10. NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," March 1995.

11. Letter from Ashok C. Thadani of the Office of Nuclear Regulatory Research to William D. Travers, Executive Director of Operations, dated December 26, 1999.

12. NUREG/CR-6674, "Fatigue Analysis of Components for 60-Year Plant Life," June 2000.

13. NUREG-1801, "Generic Aging Lessons Learned (GALL)," U.S. Nuclear Regulatory Commission, March 2001.

14. NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low–Alloy Steels," March 1998.

15. NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," April 1999.

Table 4.3-1. Stress Range Reduction Factors

Number of Equivalent Full Temperature Cycles	Stress Range Reduction Factor
7,000 and less	1.0
7,000 to 14,000	0.9
14,000 to 22,000	0.8
22,000 to 45,000	0.7
45,000 to 100,000	0.6
100,000 and over	0.5

Table 4.3-2. Example of FSAR Supplement for Metal Fatigue TLAA Evaluation

10 CFR 54.21(c)(1)(iii) Example

TLAA	Description of Evaluation	Implementation Schedule*
Metal fatigue	The aging management program monitors and tracks the number of critical thermal and pressure test transients, and monitors the cycles for the selected reactor coolant system components. The aging management program will address the effects of the coolant environment on component fatigue life by assessing the impact of the reactor coolant environment on a sample of critical components that include, as a minimum, those components selected in NUREG/CR-6260. The sample of critical components can be evaluated by applying environmental correction factors to the existing ASME Code fatigue analyses. Formulas for calculating the environmental life correction factors are contained in NUREG/CR-6583 for carbon and low-alloy steels and in NUREG/CR-5704 for austenitic SSs.	Evaluation should be completed before the period of extended operation

* An applicant need not incorporate the implementation schedule into its FSAR. However, the reviewer should verify that the applicant has identified and committed in the license renewal application to any future aging management activities to be completed before the period of extended operation. The staff expects to impose a license condition on any renewed license to ensure that the applicant will complete these activities no later than the committed date.

Table A.1-1. Elements of an Aging Management Program for License Renewal

Element	Description
1. Scope of program	Scope of program should include the specific structures and components subject to an AMR for license renewal.
2. Preventive actions	Preventive actions should prevent or mitigate aging degradation.
3. Parameters monitored or inspected	Parameters monitored or inspected should be linked to the degradation of the particular structure or component intended function(s).
4. Detection of aging effects	Detection of aging effects should occur before there is a loss of structure or component intended function(s). This includes aspects such as method or technique (i.e., visual, volumetric, surface inspection), frequency, sample size, data collection and timing of new/one-time inspections to ensure timely detection of aging effects.
5. Monitoring and trending	Monitoring and trending should provide predictability of the extent of degradation, and timely corrective or mitigative actions.
6. Acceptance criteria	Acceptance criteria, against which the need for corrective action will be evaluated, should ensure that the structure or component intended function(s) are maintained under all CLB design conditions during the period of extended operation.
7. Corrective actions	Corrective actions, including root cause determination and prevention of recurrence, should be timely.
8. Confirmation process	Confirmation process should ensure that preventive actions are adequate and that appropriate corrective actions have been completed and are effective.
9. Administrative controls	Administrative controls should provide a formal review and approval process.
10. Operating experience	Operating experience of the aging management program, including past corrective actions resulting in program enhancements or additional programs, should provide objective evidence to support the conclusion that the effects of aging will be managed adequately so that the structure and component intended function(s) will be maintained during the period of extended operation.

NYS-26/26A & RIVERKEEPER TC-1/1A: ATTACHMENT 5

NUREG-1801, Vol. 2, Rev. 1

Generic Aging Lessons Learned (GALL) Report

Tabulation of Results

Manuscript Completed: September 2005
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Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

X.M1 METAL FATIGUE OF REACTOR COOLANT PRESSURE BOUNDARY

Program Description

In order not to exceed the design limit on fatigue usage, the aging management program (AMP) monitors and tracks the number of critical thermal and pressure transients for the selected reactor coolant system components.

The AMP addresses the effects of the coolant environment on component fatigue life by assessing the impact of the reactor coolant environment on a sample of critical components for the plant. Examples of critical components are identified in NUREG/CR-6260. The sample of critical components can be evaluated by applying environmental life correction factors to the existing ASME Code fatigue analyses. Formulae for calculating the environmental life correction factors are contained in NUREG/CR-6583 for carbon and low-alloy steels and in NUREG/CR-5704 for austenitic stainless steels.

As evaluated below, this is an acceptable option for managing metal fatigue for the reactor coolant pressure boundary, considering environmental effects. Thus, no further evaluation is recommended for license renewal if the applicant selects this option under 10 CFR 54.21(c)(1)(iii) to evaluate metal fatigue for the reactor coolant pressure boundary.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program includes preventive measures to mitigate fatigue cracking of metal components of the reactor coolant pressure boundary caused by anticipated cyclic strains in the material.

2. ***Preventive Actions:*** Maintaining the fatigue usage factor below the design code limit and considering the effect of the reactor water environment, as described under the program description, will provide adequate margin against fatigue cracking of reactor coolant system components due to anticipated cyclic strains.

3. ***Parameters Monitored/Inspected:*** The program monitors all plant transients that cause cyclic strains, which are significant contributors to the fatigue usage factor. The number of plant transients that cause significant fatigue usage for each critical reactor coolant pressure boundary component is to be monitored. Alternatively, more detailed local monitoring of the plant transient may be used to compute the actual fatigue usage for each transient.

4. ***Detection of Aging Effects:*** The program provides for periodic update of the fatigue usage calculations.

5. ***Monitoring and Trending:*** The program monitors a sample of high fatigue usage locations. This sample is to include the locations identified in NUREG/CR-6260, as minimum, or propose alternatives based on plant configuration.

6. ***Acceptance Criteria:*** The acceptance criteria involves maintaining the fatigue usage below the design code limit considering environmental fatigue effects as described under the program description.

7. ***Corrective Actions:*** The program provides for corrective actions to prevent the usage factor from exceeding the design code limit during the period of extended operation.

Acceptable corrective actions include repair of the component, replacement of the component, and a more rigorous analysis of the component to demonstrate that the design code limit will not be exceeded during the extended period of operation. For programs that monitor a sample of high fatigue usage locations, corrective actions include a review of additional affected reactor coolant pressure boundary locations. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. ***Confirmation Process:*** Site quality assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of Appendix B to 10 CFR Part 50. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process and administrative controls.

9. ***Administrative Controls:*** See Item 8, above.

10. ***Operating Experience:*** The program reviews industry experience regarding fatigue cracking. Applicable experience with fatigue cracking is to be considered in selecting the monitored locations.

References

NUREG/CR-5704, *Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels,* U.S. Nuclear Regulatory Commission, April 1999.

NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components,* U.S. Nuclear Regulatory Commission, March 1995.

NUREG/CR-6583, *Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels,* U.S. Nuclear Regulatory Commission, March 1998.

NYS-26/26A & RIVERKEEPER TC-1/1A: ATTACHMENT 6

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Materials Reliability Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application (MRP-47, Revision 1)

Technical Report

Materials Reliability Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application (MRP-47 Revision 1)

1012017

Final Report, September 2005

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REPORT SUMMARY

For about the last 15 years, the effects of light water reactor environment on fatigue have been the subject of research in both the United States and abroad. Based on a risk study reported in NUREG/CR-6674, the NRC concluded that reactor water environmental effects were not a safety issue for a 60-year operating life, but that some limited assessment of its effect would be required for a license renewal extended operating period beyond 40 years. This guideline offers methods for addressing environmental fatigue in a license renewal submittal.

Background

Many utilities are currently embarking upon efforts to renew their operating licenses. One of the key areas of uncertainty in this process relates to fatigue of pressure boundary components. Although the NRC has determined that fatigue is not a significant contributor to core damage frequency, they believe that the frequency of pipe leakage may increase significantly with operating time and have requested that license renewal applicants perform an assessment to determine the effects of reactor water coolant environment on fatigue, and, where appropriate, manage this effect during the license renewal period. As the license renewal application process progressed starting in 1998, several utilities addressed this request using different approaches. In more recent years, a unified approach has emerged that has obtained regulator approval and allowed utilities to satisfactorily address this issue and obtain a renewed operating license for 60 years of plant operation.

Objectives

- To provide guidance for assessment and management of reactor coolant environmental effects
- To minimize the amount of plant-specific work necessary to comply with NRC requirements for addressing this issue in a license renewal application
- To provide "details of execution" for applying the environmental fatigue approach currently accepted by the NRC in the license renewal application process.

Approach

The project team reviewed previous work by EPRI and utilities related to fatigue environmental effects and license renewal including reports on this subject created by EPRI, NRC, and NRC contractors. Recent license renewal applications, NRC Requests for Additional Information, and the commitments made by the past license renewal applicants provided insight into NRC expectations. After evaluation of all this information, the project team developed alternatives for addressing fatigue environmental effects. This revision provides guidelines based on industry experience, consensus, and insight gained from more than six years of experience with this issue and the license renewal approval process.

Results

The report describes a fatigue environmental effect license renewal approach that can be applied by any license renewal applicant. It provides guidelines for performing environmental fatigue assessments using fatigue environmental factors from currently accepted F_{en} methodology.

EPRI Perspective

Utilities have committed significant resources to license renewal activities related to fatigue. Based on input from applicants to-date, NRC requirements for addressing fatigue environmental effects continued to change for the first few applicants, but more recently have become more unified. These guidelines were developed to provide stability, refined guidance, and assurance of NRC acceptance and include an approach that may be taken to address fatigue environmental effects in a license renewal application. Use of the approach provided in this document should limit the amount of effort necessary by individual license renewal applicants in addressing this requirement and putting activities in place for the extended operating period to manage reactor water environmental effects on fatigue.

Keywords

Fatigue
License Renewal
Reactor Water Environmental Fatigue Effects

ABSTRACT

For about the last 15 years. the effects of light water reactor environment on fatigue have been the subject of research in both the United States and abroad. The conclusions from this research are that the reactor water temperature and chemical composition (particularly oxygen content or ECP) can have a significant effect on the fatigue life of carbon, low alloy, and austenitic stainless steels. The degree of fatigue life reduction is a function of the tensile strain rate during a transient, the specific material, the temperature, and the water chemistry. The effects of other than moderate environment were not considered in the original development of the ASME Code Section III fatigue curves.

This issue has been studied by the Nuclear Regulatory Commission (NRC) for many years. One of the major efforts was a program to evaluate the effects of reactor water environment for both early and late vintage plants designed by all U.S. vendors. The results of that study, published in NUREG/CR-6260, showed that there were a few high usage factor locations in all reactor types, and that the effects of reactor water environment could cause fatigue usage factors to exceed the ASME Code-required fatigue usage limit of 1.0. On the other hand, it was demonstrated that usage factors at many locations could be shown acceptable by refined analysis and/or fatigue monitoring of actual plant transients.

Based on a risk study reported in NUREG/CR-6674, the NRC concluded that reactor water environmental effects were not a safety issue for a 60-year operating life, but that some limited assessment of its effect would be required for a license renewal extended operating period beyond 40 years. Thus, for all license renewal submittals to-date, there have been formal questions raised on the topic of environmental fatigue and, in all cases, utility commitments to address the environmental effects on fatigue in the extended operating period. Many plants have already performed these commitments.

This guideline offers methods for addressing environmental fatigue in a license renewal submittal. It requires that a sampling of the most affected fatigue sensitive locations be identified for evaluation and tracking in the extended operating period. NUREG/CR-6260 locations are considered an appropriate sample for F_{en} evaluation as long as none exceed the acceptance criteria with environmental effects considered. If this occurs, the sampling is to be extended to other locations. For these locations, evaluations similar to those conducted in NUREG/CR-6260 are required. In the extended operating period, fatigue monitoring is used for the sample of locations to show that ASME Code limits are not exceeded. If these limits are exceeded, corrective actions are identified for demonstrating acceptability for continued operation.

Using the guidance provided herein, the amount of effort needed to justify individual license renewal submittals and respond to NRC questions should be minimized, and a more unified, consistent approach should be achieved throughout the industry. More importantly, this revision provides "details of execution" for applying the environmental fatigue approach currently accepted by the NRC in the license renewal application process.

9. ADMINISTRATIVE CONTROLS

   The Thermal Fatigue Licensing Basis Monitoring Guideline actions are implemented by plant work processes.

10. OPERATING EXPERIENCE

   Refer to Sections 1.1 and 2.5.2.3 of Reference [23] for a discussion of how operating experience becomes part of the Thermal Fatigue Licensing Basis Monitoring Guideline implementation.

3.2 Method for Evaluation of Environmental Effects

There are several methods that have been published to assess the effects of reactor water environment on fatigue for each specific location to be considered. In this document, guidance is provided for performing evaluations in accordance with NUREG/CR-6583 [3] for carbon and low alloy steels and NUREG/CR-5704 [4] for austenitic stainless steels, since these are the currently accepted methodologies for evaluating environmental fatigue effects. Other methods that have been published, including those currently being used in Japan, are documented in References [18] and [22].

Figure 3-1 is a flowchart that shows an overview of the assessment approach.

- The first step is to identify the locations to be used in the assessment. This step is discussed in Section 3.2.1
- The second step is to perform an assessment of the effects of environmental fatigue on the locations identified in Step 1. This includes an assessment of the actual expected fatigue usage factor including the influence of environmental effects. Inherent conservatisms in design transients may be removed to arrive at realistic CUFs that include environmental effects. This approach is most applicable to locations where the design transients significantly envelope actual operating conditions in the plant. Further discussion is provided in Section 3.2.2. Specific guidance on performing such evaluation is provided in Section 4.0.
- The bottom of Figure 3-1 indicates that fatigue management occurs after the evaluation from Step 2 is performed for each location. This may be as simple as counting the accumulated cycles and showing that they remain less than or equal to the number of cycles utilized in the assessment performed in Step 2. On the other hand, it may not be possible to show continued acceptance throughout the extended operating period such that additional actions are required. Such options are discussed in Section 3.3. Refer also to Reference [23] for a discussion of cycle counting.

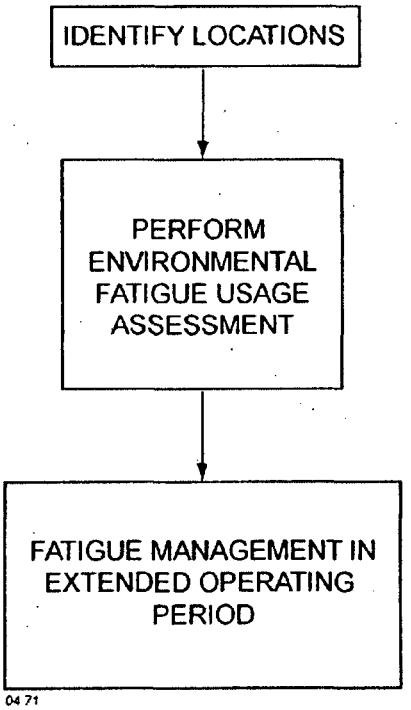

Figure 3-1
Overview of Fatigue Environmental Effects Assessment and Management

3.2.1 Identification of Locations for Assessment of Environmental Effects

A sampling of locations is chosen for the assessment of environmental effects. The purpose of identifying this set of locations is to focus the environmental assessment on just a few components that will serve as leading indicators of fatigue reactor water environmental effects. Figure 3-2 shows an overview of the approach identified for selecting and evaluating locations.

For both PWR and BWR plants, the locations chosen in NUREG/CR-6260 [2] were deemed to be representative of locations with relatively high usage factors for all plants. Although the locations may not have been those with the highest values of fatigue usage reported for the plants evaluated, they were considered representative enough that the effects of LWR environment on fatigue could be assessed.

The locations evaluated in NUREG/CR-6260 [2] for the appropriate vendor/vintage plant should be evaluated on a plant-unique basis. For cases where acceptable fatigue results are demonstrated for these locations for 60 years of plant operation including environmental effects, additional evaluations or locations need not be considered. However, plant-unique evaluations may show that some of the NUREG/CR-6260 [2] locations do not remain within allowable limits for 60 years of plant operation when environmental effects are considered. In this situation, plant specific evaluations should expand the sampling of locations accordingly to include other locations where high usage factors might be a concern.

In original stress reports, usage factors may have been reported in many cases that are unrealistically high, but met the ASME Code requirement for allowable CUF. In these cases, revised analysis may be conducted to derive a more realistic usage factor or to show that the revised usage factor is significantly less than reported.

If necessary, in identifying the set of locations for the expanded environmental assessment, it is important that a diverse set of locations be chosen with respect to component loading (including thermal transients), geometry, materials, and reactor water environment. If high usage factors are presented for a number of locations that are similar in geometry, material, loading conditions, and environment, the location with the highest expected CUF, considering typical environmental fatigue multipliers, should be chosen as the bounding location to use in the environmental fatigue assessment. Similar to the approach taken in NUREG/CR-6260 [2], the final set of locations chosen for expanded environmental assessment should include several different types of locations that are expected to have the highest CUFs and should be those most adversely affected by environmental effects. The basis of location choice should be described in the individual plant license renewal application.

In conclusion, the following steps should be taken to identify the specific locations that are to be considered in the environmental assessment:

- Identify the locations evaluated in NUREG/CR-6260 [2] for the appropriate vintage/vendor plant.
- Perform a plant-unique environmental fatigue assessment for the NUREG/CR-6260 locations.
- If the CUF results for all locations above are less than or equal to the allowable (typically 1.0) for the 60-year operating life, the environmental assessment may be considered complete; additional evaluations or locations need not be considered.
- If the CUF results for any locations above are greater than the allowable for the 60-year operating life, expand the locations evaluated, considering the following:
  - Identify all Class 1 piping systems and major components. For the reactor pressure vessel, there may be multiple locations to consider.
  - For each system or component, identify the highest usage factor locations. By reasons of geometric discontinuities or local transient severity, there will generally be a few locations that have the highest usage factors when considering environmental effects.
  - From the list of locations that results from the above steps, choose a set of locations that are a representative sampling of locations with the highest expected usage factors when considering environmental effects. Considerations for excluding locations can include: (1) identification of excess conservatism in the transient grouping or other aspects of the design fatigue analysis, or (2) locations that have similar loading conditions, geometry, material, and reactor water environment compared to another selected location.

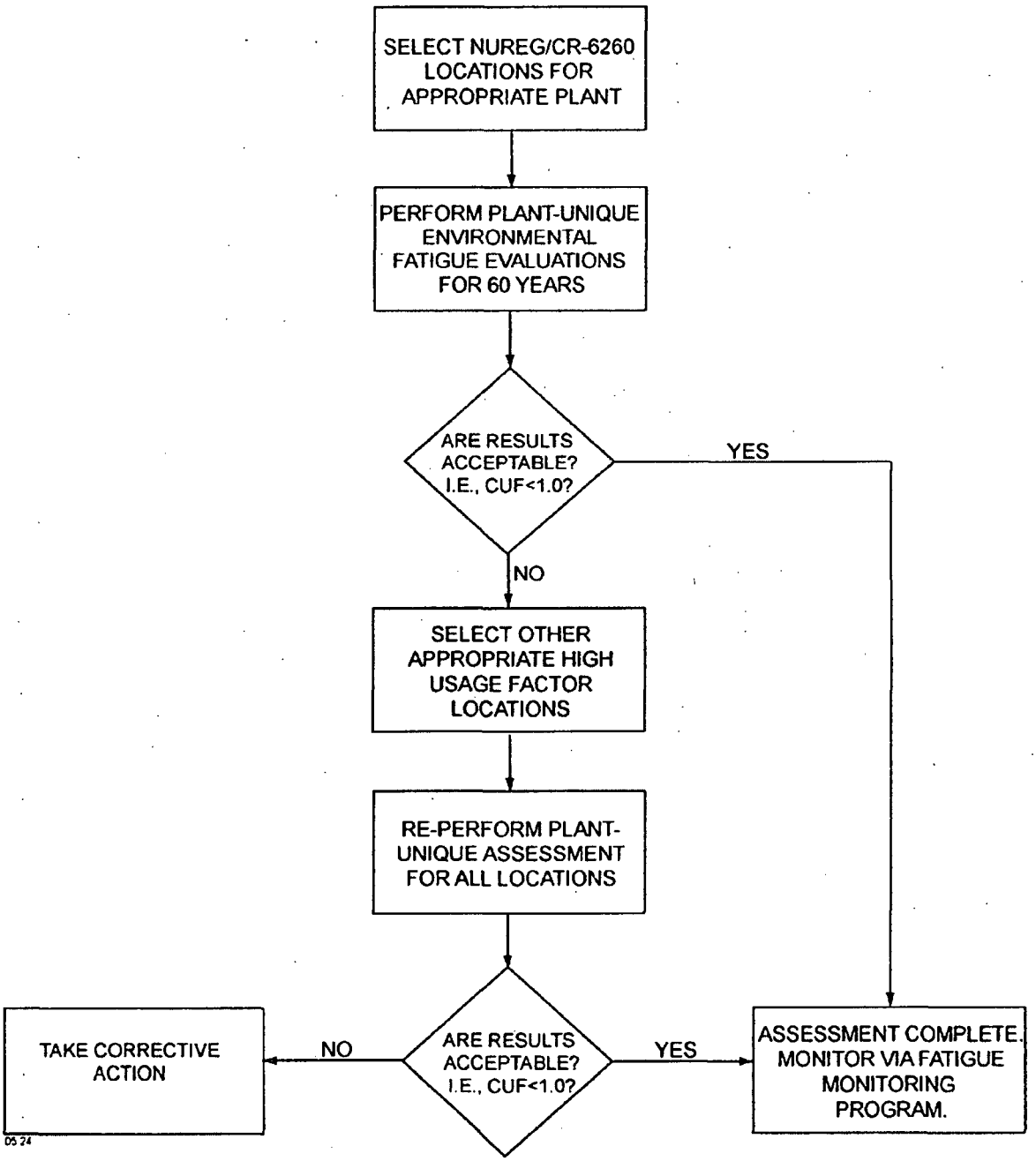

Figure 3-2
Identification of Component Locations and Fatigue Environmental Effects Assessment

3.2.2 Fatigue Assessment Using Environmental Factors

In performing an assessment of environmental fatigue effects, factors to account for environmental effects are incorporated into an updated fatigue evaluation for each selected location using the F_{en} approach documented in NUREG/CR-6583 [3] for carbon and low alloy steels and NUREG/CR-5704 [4] for austenitic stainless steels. Excess conservatism in the loading definitions, number of cycles, and the fatigue analyses may be considered. Figure 3-3 shows the approach for performing the assessment and managing fatigue in the extended operating period.

Determination of Existing Licensing Basis

Existing plant records must be reviewed to determine the cyclic loading specification (transient definition and number of cycles) and stress analysis for the location in question. Review of the analysis may or may not show that excess conservatism exists. Reference [23] provides guidance on reviewing the original design basis, the operating basis, and additions imposed by the regulatory oversight process, to determine the fatigue licensing basis events for which the component is required to be evaluated.

Consideration of Increased Cycles for Extended Period

As a part of the license renewal application process, the applicant must update the projected cycles to account for 60 years of plant operation. The first possible outcome is that the number of expected cycles in the extended operating period will remain at or below those projected for the initial 40-year plant life. In this case, the governing fatigue analyses will not require modification to account for the extended period of operation.

The second possibility is that more cycles are projected to occur for 60 years of plant operation than were postulated for the first 40 years. In this case, an applicant must address the increased cycle counts. One possible solution is to perform a revised fatigue analysis to confirm that the increased number of cycles will still result in a CUF less than or equal to the allowable. A second possibility is to determine the number of cycles at which the CUF would be expected to reach the allowable. This cycle quantity then becomes the allowable against which the actual operation is tracked. Section 3.3 discusses options to be employed if this lower allowable is projected to be exceeded.

Fatigue Assessment

Fatigue assessment includes the determination of CUF considering environmental effects. This may be accomplished conservatively using information from design documentation and bounding F_{en} factors from NUREG/CR-6583 [3] and NUREG/CR-5704 [4], or it may require a more extensive approach (as discussed in Section 4.0).

A revised fatigue analysis may or may not be required. Possible reasons for updating the fatigue analysis could include:

- Excess conservatism in original fatigue analysis with respect to modeling, transient definition, transient grouping and/or use of an early edition of the ASME Code.

- For piping, use of an ASME Code Edition prior to 1979 Summer Addenda, which included the ΔT_1 term in Equation (10) of NB-3650. Use of a later code reduces the need to apply conservative elastic-plastic penalty factors.
- Re-analysis may be needed to determine strain rate time histories possibly not reported in existing component analyses, such that bounding environmental multipliers (i.e., very low or "saturated" strain rates) would not have to be used.

A simplified revised fatigue analysis may be performed using results from the existing fatigue analysis, if sufficient detail is available. Alternatively, a new complete analysis could be conducted to remove additional conservatisms. Such an evaluation would not necessarily need the full pedigree of a certified ASME Code Section III analysis (i.e.. Certified Design Specification, etc.), but it should utilize all of the characteristic methods from Section III for computing CUF. In the environmental fatigue assessment, the environmental fatigue usage may be calculated using the following steps:

- For each load set pair in the fatigue analysis, determine an environmental factor F_{en}. This factor should be developed using the equations in NUREG/CR-6583 [3] or NUREG/CR-5704 [4]. (Section 4.0 provides specific guidance on performing an F_{en} evaluation)
- The environmental partial fatigue usage for each load set pair is then determined by multiplying the original partial usage factor by F_{en}. In no case shall the F_{en} be less than 1.0.
- The usage factor is the sum of the partial usage factors calculated with consideration of environmental effects.

Fatigue Management Approach

As shown in Figure 3-3, the primary fatigue management approaches for the extended operating period consist of tracking either the CUF or number of accumulated cycles.

- For cycle counting, an updated allowable number of cycles may be needed if the fatigue assessment determined the CUF to be larger than allowable. One approach is to derive a reduced number of cycles that would limit the CUF to less than or equal to the allowable value (typically 1.0). On the other hand, if the assessed CUF was shown to be less than or equal to the allowable, the allowable number of cycles may remain as assumed in the evaluation, or increased appropriately. As long as the number of cycles in the extended operating period remains within this allowed number of cycles, no further action is required.
- For CUF tracking, one approach would be to utilize fatigue monitoring that accounts for the actual cyclic operating conditions for each location. This approach would track the CUF due to the actual cycle accumulation, and would take credit for the combined effects of all transients. Environmental factors would have to be factored into the monitoring approach or applied to the CUF results of such monitoring. No further action is required as long as the computed usage factor remains less than or equal to the allowable value.

Prior to such time that the CUF is projected to exceed the allowable value, or the number of actual cycles is projected to exceed the allowable number of cycles, action must be taken such that the allowable limits will not be exceeded. If the cyclic or fatigue limits are expected to be exceeded during the license renewal period, further approaches to fatigue management would be required prior to reaching the limit, as described in Section 3.3. Further details on guidelines for thermal fatigue monitoring and compliance/mitigation options are provided in Reference [23].

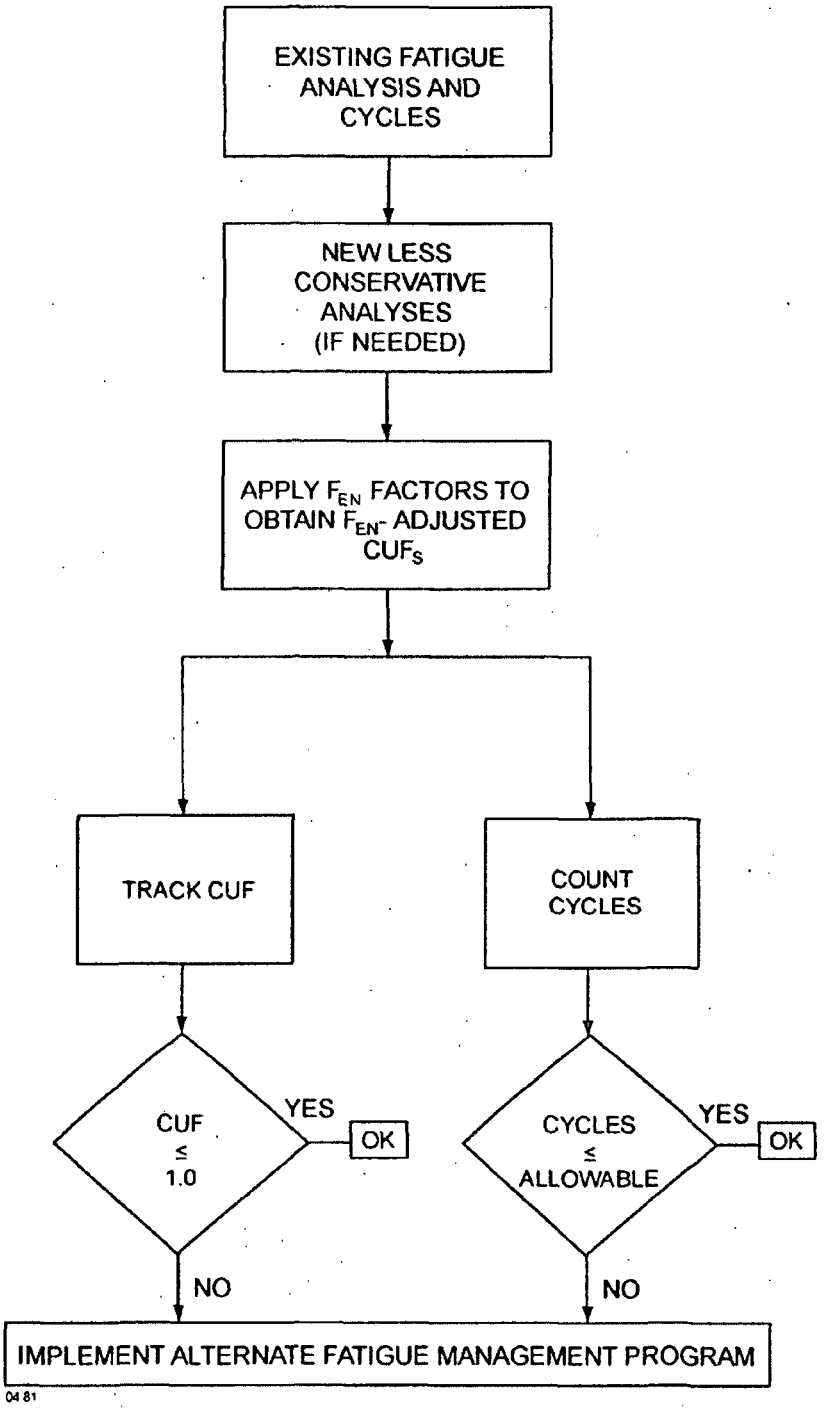

Figure 3-3
Fatigue Management if Environmental Assessment Conducted

A separate section follows for each parameter utilized in the F_{en} expressions, that is transformed sulfur content (S^*), transformed temperature (T^*), transformed dissolved oxygen (O^*), and transformed strain rate ($\dot{\varepsilon}^*$). For the transformed strain rate, temperature, and oxygen parameters, the three approaches are discussed. Transformed sulfur does not vary over the three approaches. A single approach should be utilized for all of the transformed parameters in a single load-pair F_{en} determination, although different approaches may be utilized for different load-pair F_{en}s.

First, the typical content of a fatigue calculation is presented.

4.2.1 Contents of a Typical Fatigue Evaluation

This section provides the content of a typical fatigue calculation. Whereas fatigue calculations have varied over the years, their basic content is the same. With the advent of computer technology, the calculations have basically maintained the same content, but computations have become more refined and exhaustive. For example, 30 years ago it was computationally difficult for a stress analyst to evaluate 100 different transients in a fatigue calculation. Therefore, the analyst would have grouped the transients into as few as one transient grouping and performed as few incremental fatigue calculations as possible. With today's computer technology and desire to show more margin, it is relatively easy for the modern-day analyst to evaluate all 100 incremental fatigue calculations for this same problem. Also, older technology would have likely utilized conservative shell interaction hand solutions for computing stress, whereas today finite element techniques are commonly deployed. This improvement in technology would not have changed the basic inputs to the fatigue calculation (i.e., stress), but it would have typically yielded significantly more representative input values.

The discussion here is limited to the general content of most typical fatigue calculations. Discussions of removing excess conservatisms from the input (stress) values of these calculations are not included, as it is assumed that those techniques are generally well understood by engineers performing these assessments throughout the industry.

Two typical fatigue calculations are shown in Figures 4-1 through 4-4. Figure 4-1 reflects an "old" calculation, i.e., one that is typical from a stress report from a plant designed in the 1960s. Figures 4-2 through 4-4 reflect a "new" calculation, i.e., one that is typical from a 1990s vintage stress report. A description of the content of these two calculations is provided below.

The same basic content is readily apparent in both CUF calculations shown in Figures 4-1 through 4-4. However, it is also apparent that much more detail is present in Figures 4-2 through 4-4 for the "new" calculation compared to Figure 4-1 for the "old" calculation. Therefore, with respect to applying F_{en} methodology to a CUF calculation, the guidance provided in the following sections equally applies to both vintages of calculations. The main difference is in assumptions that need to be made for the F_{en} transformed variables due to a lack of detail backing up the calculations in the stress report. Guidance for these assumptions is described in Sections 4.2.2 through 4.2.5, with appropriate reference to the calculations shown in Figures 4-1 through 4-4.

NYS-26/26A & RIVERKEEPER TC-1/1A: ATTACHMENT 7

NUREG/CR-6583
ANL-97/18

Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels

Argonne National Laboratory

U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, DC 20555-0001

NUREG/CR-6583
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Prepared for
U.S. Nuclear Regulatory Commission

EFFECTS OF LWR COOLANT ENVIRONMENTS ON FATIGUE DESIGN CURVES OF CARBON AND LOW-ALLOY STEELS

by

O. K. Chopra and W. J. Shack

Abstract

The ASME Boiler and Pressure Vessel Code provides rules for the construction of nuclear power plant components. Figures I-9.1 through I-9.6 of Appendix I to Section III of the Code specify fatigue design curves for structural materials. While effects of reactor coolant environments are not explicitly addressed by the design curves, test data indicate that the Code fatigue curves may not always be adequate in coolant environments. This report summarizes work performed by Argonne National Laboratory on fatigue of carbon and low-alloy steels in light water reactor (LWR) environments. The existing fatigue S-N data have been evaluated to establish the effects of various material and loading variables such as steel type, dissolved oxygen level, strain range, strain rate, temperature, orientation, and sulfur content on the fatigue life of these steels. Statistical models have been developed for estimating the fatigue S-N curves as a function of material, loading, and environmental variables. The results have been used to estimate the probability of fatigue cracking of reactor components. The different methods for incorporating the effects of LWR coolant environments on the ASME Code fatigue design curves are presented.

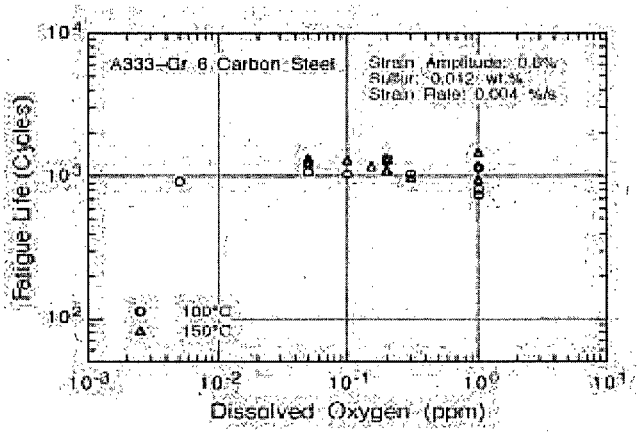

Figure 68.
Fatigue life of A333-Gr 6 carbon steel as a function of dissolved oxygen in water at 100 and 150°C

5 Statistical Model

5.1 Modeling Choices

In attempting to develop a statistical model from incomplete data and where physical processes are only partially understood, care must be taken to avoid overfitting the data. Different functional forms of the predictive equations (e.g., different procedures for transforming the measured variables into data used for fitting equations) were tried for several aspects of the model. Fatigue S-N data are generally expressed by Eq. 1.1, which may be rearranged to express fatigue life N in terms of strain amplitude ε_a as

$$\ln(N) = [\ln B - \ln(\varepsilon_a - A)]/b. \tag{5.1}$$

Additional terms may be added to the model that would improve agreement with the current data set. However, such changes may not hold true in other data sets, and the model would typically be less robust, i.e., it would not predict new data well. In general, complexity in a statistical model is undesirable unless it is consistent with accepted physical processes. Although there are statistical tools that can help manage the tradeoff between robustness and detail in the model, engineering judgment is required. Model features that would be counter to known effects are excluded. Features that are consistent with previous studies use such results as guidance, e.g., defining the threshold or saturation values for an effect, but where there are differences from previous findings, the reasons for the differences are evaluated and an appropriate set of assumptions is incorporated into the model.

5.2 Least-Squares Modeling within a Fixed Structure

The parameters of the model are commonly established through least-squares curve-fitting of the data to either Eq. 1.1 or 5.1. An optimization program sets the parameters so as to minimize the sum of the square of the residual errors, which are the differences between the predicted and actual values of ε_a or ln(N). A predictive model based on least-squares fit on ln(N) is biased for low ε_a; in particular, runoff data cannot be included. The model also leads to probability curves that converge to a single value of threshold strain.

However, the model fails to address the fact that at low ε_a, most of the error in life is due to uncertainty associated with either measurement of stress or strain or variation in threshold strain caused by material variability. On the other hand, a least-squares fit on ε_a does not work well for higher strain amplitudes. The two kinds of models are merely transformations of each other, although the precise values of the coefficients differ.

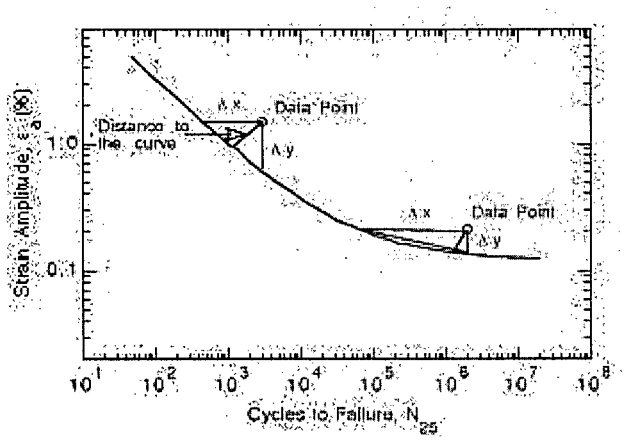

Figure 69.
Schematic of least-squares curve-fitting of data by minimizing sum of squared Cartesian distances from data points to predicted curve

The statistical models[27,28] were developed by combining the two approaches and minimizing the sum of squared Cartesian distances from the data points to the predicted curve (Fig. 69). For low ε_a, this is very close to optimizing the sum of squared errors in predicted ε_a; at high ε_a, it is very close to optimizing the sum of squared errors in predicted life; and at medium ε_a, this model combines both factors. However, because the model includes many nonlinear transformations of variables and because different variables affect different parts of the data, the actual functional form and transformations are partly responsible for minimizing the squares of the errors. The functional forms and transformation are chosen a priori, and no direct computational means exist for establishing them.

To perform the optimization, it was necessary to normalize the x and y axes by assigning relative weights to be used in combining the error in life and strain amplitude because x and y axes are not in comparable units. In this analysis, errors in strain amplitude (%) are weighted 20 times as heavily as errors in ln(N). A value of 20 was selected for two related reasons. First, this factor leads to approximately equal weighting of low- and high-strain-amplitude data in the least-squared error computation of model coefficients. Second, when applied to the model to generate probability curves, it yielded a standard deviation on strain amplitude comparable to that obtained from the best-fit of the high cycle fatigue data to Eq. 1.1. Because there is necessarily judgment applied in the selection of this value, a sensitivity analysis was performed, and it showed that the coefficients of the model do not change much for weight factors between 10 and 25. Distance from the curve was estimated as

$$D = \left\{(x - \hat{x})^2 + \left[k(y - \hat{y})\right]^2\right\}^{1/2}, \tag{5.2}$$

where $\hat{x}$ and $\hat{y}$ represent predicted values, and k = 20.

5.3 The Model

Based on the existing fatigue S-N data base, statistical models have been developed for estimating the effects of material and loading conditions on the fatigue lives of CSs and LASs.[27,28] The dependence of fatigue life on DO level has been modified because it was determined that in the range of 0.05-0.5 ppm, the effect of DO was more logarithmic than linear.[45,93] In this report, the models have been further optimized with a larger fatigue S-N data base. Because of the conflicting possibilities that with decreasing strain rate, fatigue life may either be unaffected, decrease for some heats, or increase for others, effects of strain rate in air were not explicitly considered in the model. The effects of orientation, i.e., size and distribution of sulfide inclusions, on fatigue life were also excluded because the existing data base does not include information on sulfide distribution and morphology. In air, the fatigue data for CSs are best represented by

$$\ln(N_{25}) = 6.595 - 1.975 \ln(\varepsilon_a - 0.113) - 0.00124\ T \tag{5.3a}$$

and for LASs by

$$\ln(N_{25}) = 6.658 - 1.808 \ln(\varepsilon_a - 0.151) - 0.00124\ T. \tag{5.3b}$$

In LWR environments, the fatigue data for CSs are best represented by

$$\ln(N_{25}) = 6.010 - 1.975 \ln(\varepsilon_a - 0.113) + 0.101\ S^*\ T^*\ O^*\ \dot{\varepsilon}^* \tag{5.4a}$$

and for LASs by

$$\ln(N_{25}) = 5.729 - 1.808 \ln(\varepsilon_a - 0.151) + 0.101\ S^*\ T^*\ O^*\ \dot{\varepsilon}^*, \tag{5.4b}$$

where S^*, T^*, O^*, and $\dot{\varepsilon}^*$ = transformed sulfur content, temperature, DO, and strain rate, respectively, defined as follows:

$$\begin{aligned} S^* &= S && (0 < S \le 0.015 \text{ wt.\%}) \\ S^* &= 0.015 && (S > 0.015 \text{ wt.\%}) \end{aligned} \tag{5.5a}$$

$$\begin{aligned} T^* &= 0 && (T < 150°C) \\ T^* &= T - 150 && (T = 150\text{–}350°C) \end{aligned} \tag{5.5b}$$

$$\begin{aligned} O^* &= 0 && (DO < 0.05 \text{ ppm}) \\ O^* &= \ln(DO/0.04) && (0.05 \text{ ppm} \le DO \le 0.5 \text{ ppm}) \\ O^* &= \ln(12.5) && (DO > 0.5 \text{ ppm}) \end{aligned} \tag{5.5c}$$

$$\begin{aligned} \dot{\varepsilon}^* &= 0 && (\dot{\varepsilon} > 1 \text{ \%/s}) \\ \dot{\varepsilon}^* &= \ln(\dot{\varepsilon}) && (0.001 \le \dot{\varepsilon} \le 1 \text{ \%/s}) \\ \dot{\varepsilon}^* &= \ln(0.001) && (\dot{\varepsilon} < 0.001 \text{ \%/s}) \end{aligned} \tag{5.5d}$$

The functional form and bounding values of the transformed parameters S^*, T^*, O^*, and $\dot{\varepsilon}^*$ were based upon experimental observations and data trends discussed in Section 4.2. Significant features of the model for estimating fatigue life in LWR environments are as follows:

(a) The model assumes that environmental effects on fatigue life occur primarily during the tensile-loading cycle; minor effects during the compressive loading cycle have been excluded. Consequently, the loading and environmental conditions, e.g., temperature, strain rate, and DO, during the tensile-loading cycle are used for estimating fatigue lives.

(b) When any one of the threshold condition is not satisfied, e.g., <0.05 ppm DO in water, the effect of strain rate is not considered in the model, although limited data indicate that heats of steel that are sensitive to strain rate in air also show a decrease in life in water with decreasing strain rate.

(c) The model assumes a linear dependence of S^* on S content in steel and saturation at 0.015 wt.% S.

The model is recommended for predicted fatigue lives of $\leq 10^6$ cycles. For fatigue lives of 10^6 to 10^8 cycles, the results should be used with caution because, in this range, the model is based on very limited data obtained from relatively few heats of material.

The estimated and experimental S-N curves for CS and LAS in air at room temperature and 288°C are shown in Fig. 70. The mean curves used in developing the ASME Code design curve and the average curves of Higuchi and Iida[7] are also included in the figure. The results indicate that the ASME mean curve for carbon steels is not consistent with the experimental data; at strain amplitudes <0.2%, the mean curve predicts significantly lower fatigue lives than those observed experimentally. The estimated curve for low-alloy steels is comparable with the ASME mean curve. For both steels, Eq. 5.3 shows good agreement with the average curves of Higuchi and Iida.

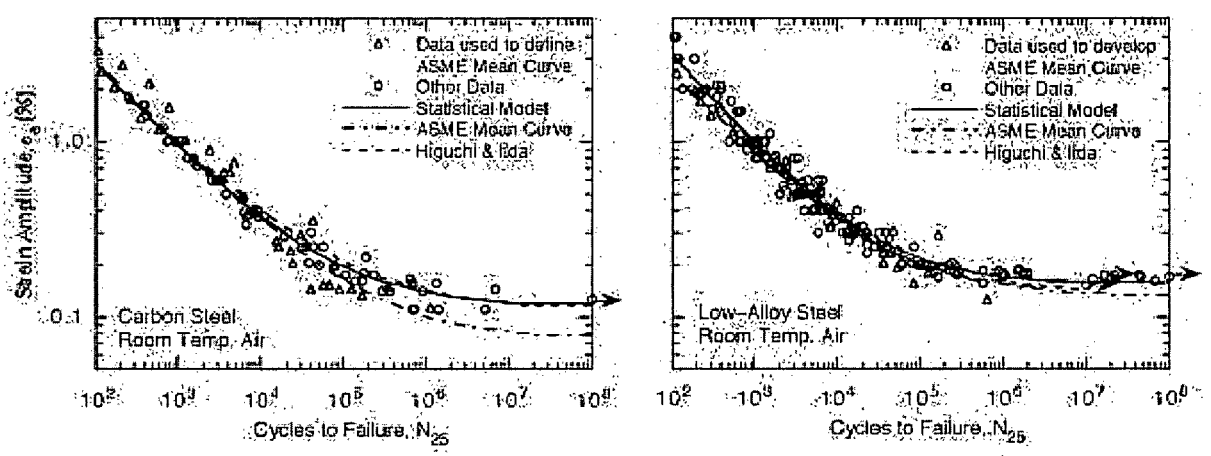

Figure 70. Fatigue S-N behavior for carbon and low-alloy steels estimated from model and determined experimentally in air at room temperature

5.4 Distribution of Fatigue Life

For a given steel type, the average distance of data points from the mean curve does not vary much for different environmental conditions. To develop a distribution on life, we start with the assumption that there are three sources of prediction error: (a) measurement errors

for the applied strain amplitude, (b) variations in the threshold strain amplitude due to material variability, and (c) errors due to uncertainty in test and material conditions or other unexplained variation. Because measurement errors are small at high strain amplitudes, the standard deviation of distance from the mean curve at high strain amplitudes is a good measure of the scatter in fatigue life due to unexplained variations. At low amplitudes where the S-N curve is almost horizontal, the errors (as measured by the distance from the mean curve) are dominated by the variation in strain amplitude. The standard deviation of the error in strain amplitude was taken to be equal to the standard deviation in the predicted fatigue life divided by a factor of 20 consistent with the weighting factor used for optimization. The standard deviation on life was 0.52 for CSs and LASs. These results can be combined with Eq. 5.3 to estimate the distribution in life for smooth test specimens. In air, the xth percentile of the distribution on life $N_{25}[x]$ for CSs is

$$\ln(N_{25}) = 6.595 + 0.52\, F^{-1}[x] - 1.975 \ln(\varepsilon_a - 0.113 + 0.026\, F^{-1}[1-x]) - 0.00124\, T \qquad (5.6a)$$

and for LASs it is

$$\ln(N_{25}) = 6.658 + 0.52\, F^{-1}[x] - 1.808 \ln(\varepsilon_a - 0.151 + 0.026\, F^{-1}[1-x]) - 0.00124\, T\,. \qquad (5.6b)$$

In LWR environments, the xth percentile of the distribution on life $N_{25}[x]$ for CSs is

$$\ln(N_{25}) = 6.010 + 0.52\, F^{-1}[x] - 1.975 \ln(\varepsilon_a - 0.113 + 0.026\, F^{-1}[1-x]) + 0.101\, S^*\, T^*\, O^*\, \dot{\varepsilon}^* \qquad (5.7a)$$

and for LASs it is

$$\ln(N_{25}) = 5.729 + 0.52\, F^{-1}[x] - 1.808 \ln(\varepsilon_a - 0.151 + 0.026\, F^{-1}[1-x]) + 0.101\, S^*\, T^*\, O^*\, \dot{\varepsilon}^*. \qquad (5.7b)$$

The parameters S^*, T^*, O^*, and $\dot{\varepsilon}^*$ are defined in Eqs. 5.5, and $F^{-1}[\cdot]$ denotes the inverse of the standard normal cumulative distribution function. The coefficients of distribution functions $F^{-1}[x]$ and $F^{-1}[1-x]$ represent the standard deviation on life and strain amplitude, respectively. For convenience, values of the inverse of standard normal cumulative distribution function in Eqs. 5.6 and 5.7 are given in Table 3. The standard deviation of 0.026 on strain amplitude obtained from the analysis may be an overly conservative value. A more realistic value for the standard deviation on strain could be obtained by analysis of the fatigue limits of different heats of material. The existing data are inadequate for such an analysis because (a) not enough heats of materials are included in the data base, and (b) there are very few high-cycle fatigue data for accurate estimations of the fatigue limit for specific heats.

The estimated probability curves for the fatigue life of CSs and LASs in an air and LWR environments in Figs. 71-73 show good agreement with experimental data; nearly all of the data are bounded by the 5% probability curve. Relative to the 50% probability curve, the 5% probability curve is a factor of ≈2.5 lower in life at strain amplitudes >0.3% and a factor of 1.4-1.7 lower in strain at <0.2% strain amplitudes. Similarly, the 1% probability curve is a factor of ≈3.7 lower in life and a factor of 1.7-2.2 lower in strain.

Table 3. Inverse of standard cumulative distribution function

Probability	$F^{-1}[x]$	$F^{-1}[1-x]$	Probability	$F^{-1}[x]$	$F^{-1}[1-x]$
0.01	-3.7195	3.7195	3.00	-1.8808	1.8808
0.02	-3.5402	3.5402	5.00	-1.6449	1.6449
0.03	-3.4319	3.4319	7.00	-1.4758	1.4758
0.05	-3.2905	3.2905	10.00	-1.2816	1.2816
0.07	-3.1947	3.1947	20.00	-0.8416	0.8416
0.10	-3.0902	3.0902	30.00	-0.5244	0.5244
0.20	-2.8782	2.8782	50.00	0.0000	0.0000
0.30	-2.7478	2.7478	65.00	0.3853	-0.3853
0.50	-2.5758	2.5758	80.00	0.8416	-0.8416
0.70	-2.4573	2.4573	90.00	1.2816	-1.2816
1.00	-2.3263	2.3263	95.00	1.6449	-1.6449
2.00	-2.0537	2.0537	98.00	2.0537	-2.0537

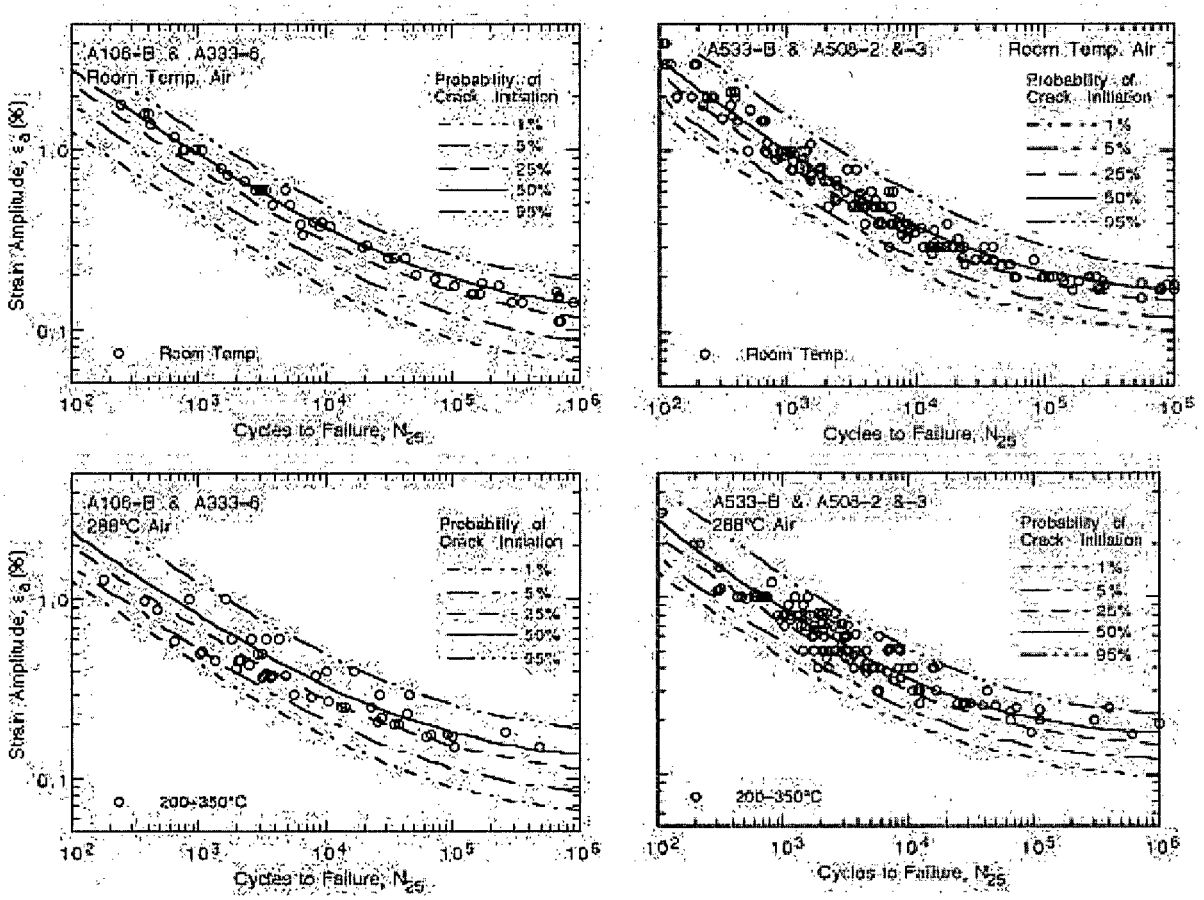

Figure 71. Experimental data and probability of fatigue cracking in carbon and low-alloy steel test specimens in air

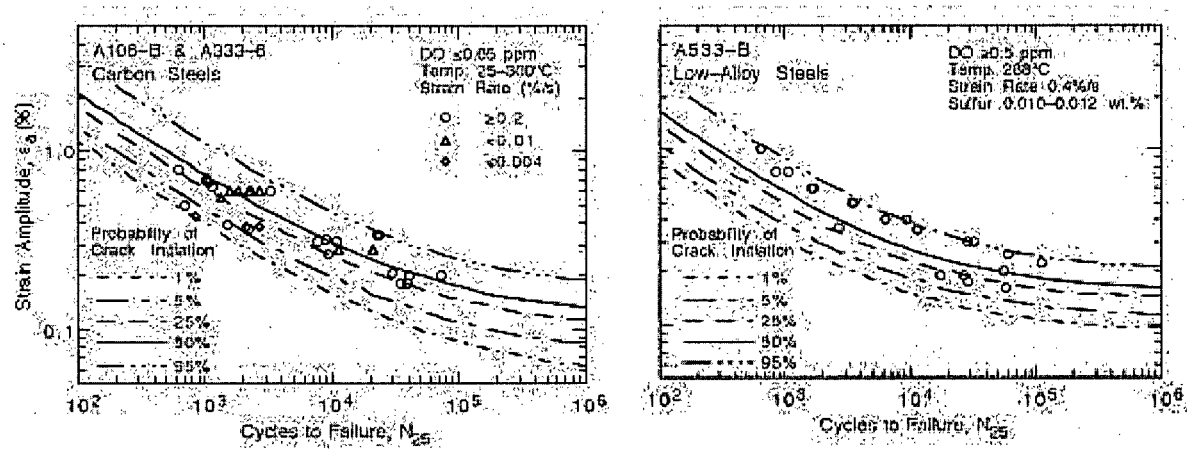

Figure 72. Experimental data and probability of fatigue cracking in carbon and low-alloy steel test specimens in simulated PWR environments

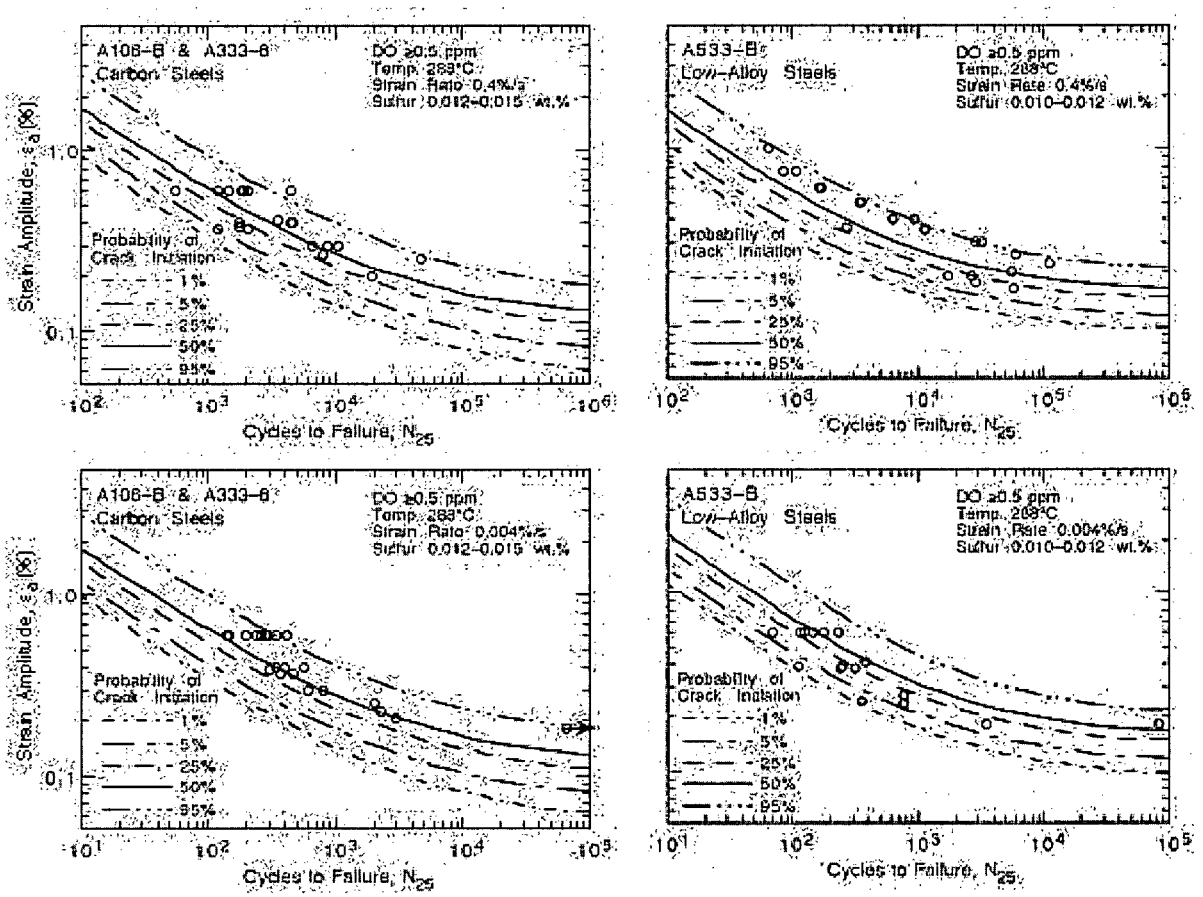

Figure 73. Experimental data and probability of fatigue cracking in carbon and low-alloy steel test specimens in high-dissolved-oxygen water

As with other aspects of this model, the estimates of the probability of cracking should not be extrapolated much beyond the data. The probabilities assume a normal distribution, which is consistent with the data for most of the range. The existing data are not sufficient to determine precise distributions because more data are required to estimate distributions than to estimate the mean curve. However, the assumption of normality is reasonable (and conservative) down to 0.1–1% probability of cracking and it is empirically verified by the number of data points that fall below the respective curves. The probability is not expected to deviate significantly from the normal curve for another order of magnitude (one more standard deviation) even if the probability distribution is not the same. Because estimates of extremely low or high probabilities are sensitive to the choice of distribution, the probability distribution curves should not be extrapolated beyond 0.02% probability.

6 Fatigue Life Correction Factor

An alternative approach for incorporating the effects of reactor coolant environments on fatigue S–N curves has been proposed by the Environmental Fatigue Data (EFD) Committee of the Thermal and Nuclear Power Engineering Society (TENPES) of Japan.* A fatigue life correction factor F_{en} is defined as the ratio of the life in air at room temperature to that in water at the service temperature. The fatigue usage for a specific load pair based on the current Code fatigue design curve is multiplied by the correction factor to account for the environmental effects. Note that the fatigue life correction factor does not account for any differences that might exist between the current ASME mean air curves and the present mean air curves developed from a larger data base. The specific expression for F_{en}, proposed initially by Higuchi and Iida,[7] assumes that life in the environment N_{water} is related to life in air N_{air} at room temperature through a power-law dependence on the strain rate

$$F_{en} = \frac{N_{air}}{N_{water}} = (\dot{\varepsilon})^{-P}, \tag{6.1a}$$

or

$$\ln(F_{en}) = \ln(N_{air}) - \ln(N_{water}) = -P\ln(\dot{\varepsilon}). \tag{6.1b}$$

In air at room temperature, the fatigue life N_{air} of CSs is expressed as

$$\ln(N_{air}) = 6.653 - 2.119\ \ln(\varepsilon_a - 0.108) \tag{6.2a}$$

and for LASs by

$$\ln(N_{air}) = 6.578 - 1.761\ \ln(\varepsilon_a - 0.140), \tag{6.2b}$$

where ε_a is the applied strain amplitude (%). Only the tensile loading cycle is considered to be important for environmental effects on fatigue life. The exponent P is a product of a environmental factor R_p, which depends on temperature T (°C) and DO level (ppm), and a material factor P_c, which depends on the ultimate tensile strength σ_u (MPa) and sulfur content S (wt/.%) of the steel. Thus

$$P = R_p\ P_c, \tag{6.3a}$$

* Presented at the Pressure Vessel Research Council Meeting, April 1996, Orlando, FL.

$$P_c = 0.864 - 0.00092\,\sigma_u + 14.6\,S, \qquad (6.3b)$$

$$R_p = \frac{R_{pT} - 0.2}{2.64}\ln(DO) + 1.75R_{pT} - 0.035, \qquad 0.2 \le R_p \le R_{pT} \qquad (6.3c)$$

and $R_{pT} = 0.198\exp(0.00557T)$. (6.3d)

The fatigue lives of carbon and low-alloy steels measured experimentally and those estimated from the statistical and EFD models are shown in Figs. 74–78. Although the EFD correlations for exponent P were based entirely on data for carbon steels, Eqs. 6.3a–6.3d were also used for estimating the fatigue lives of LASs. Also, σ_u in Eq. 6.3b was assumed to be 520 and 650 MPa, respectively, for CSs and LASs. The significant differences between the two models are as follows:

(a) The EFD correlations have been developed from data for CSs alone.

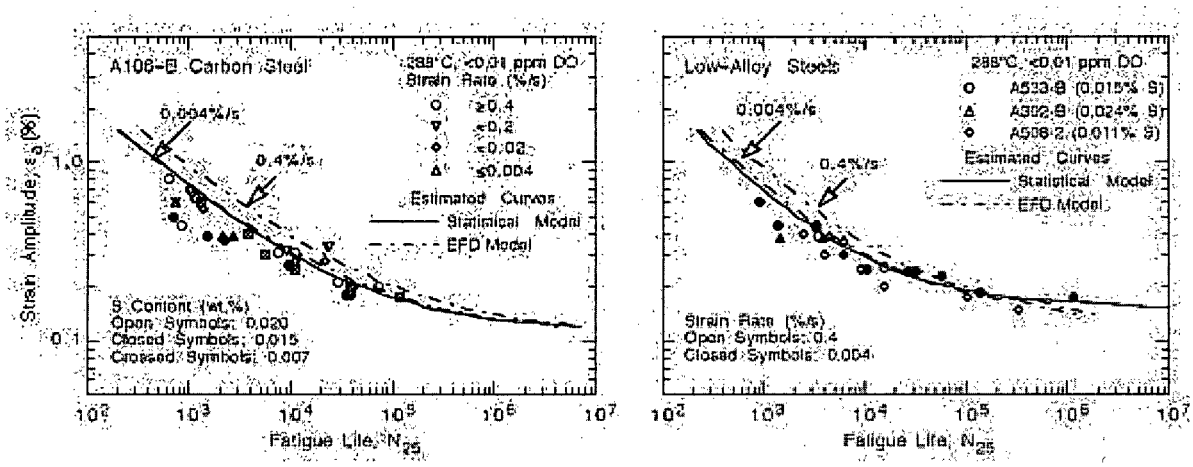

Figure 74. Experimental fatigue lives and those estimated from statistical and EFD models for carbon and low-alloy steels in simulated PWR water

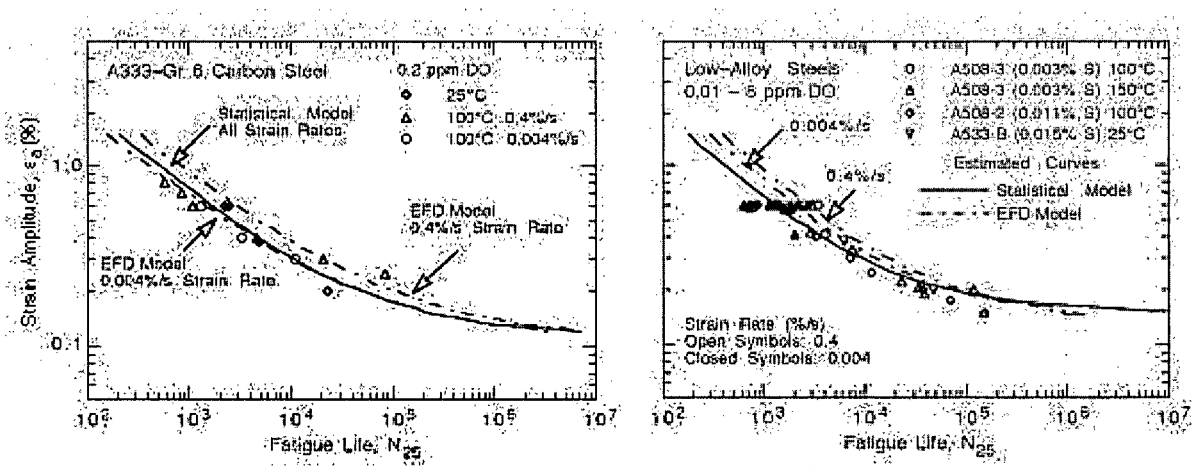

Figure 75. Experimental fatigue lives and those estimated from statistical and EFD models for carbon and low-alloy steels in water at temperatures below 150°C

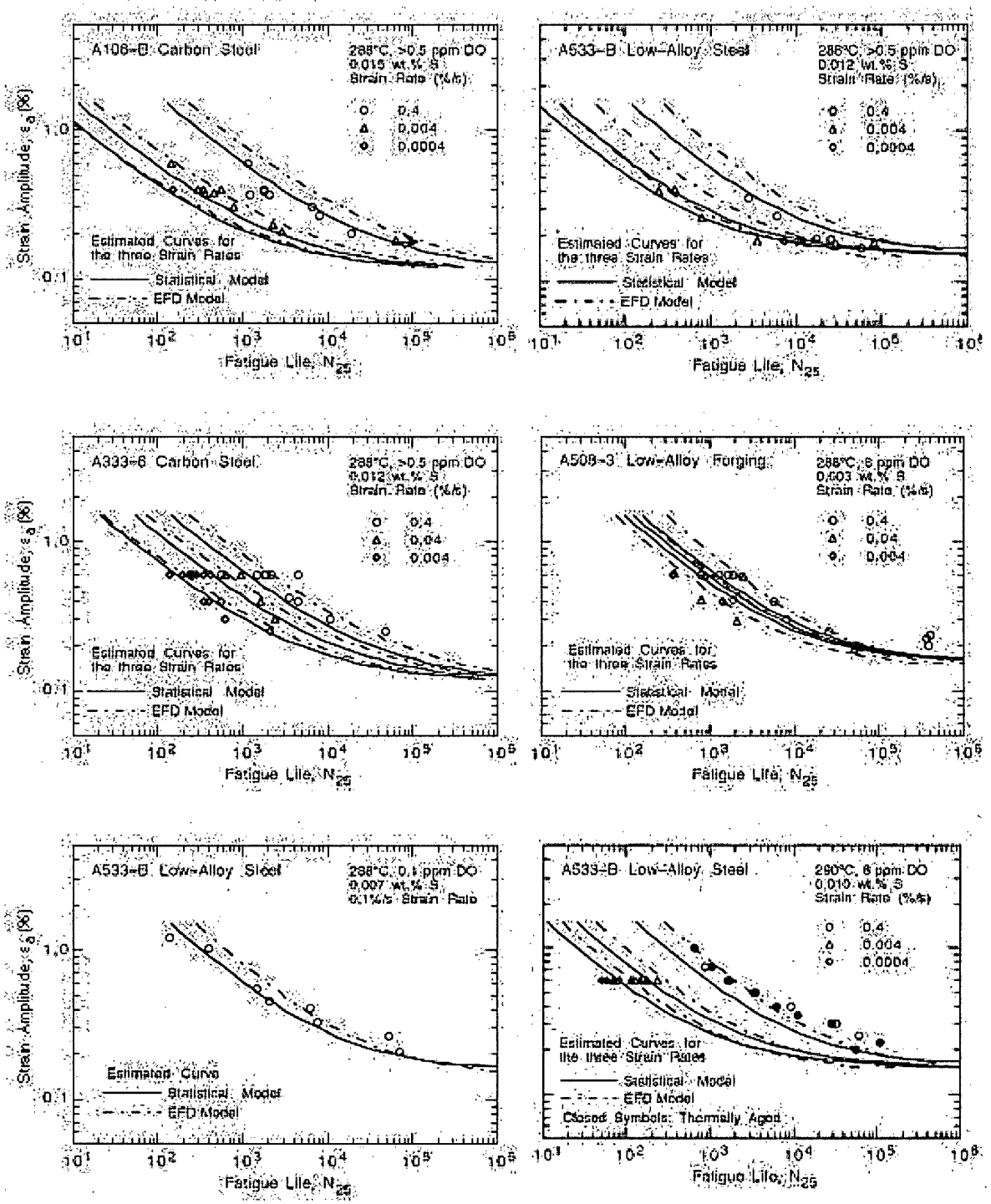

Figure 76. Experimental fatigue lives and those estimated from statistical and EFD models for carbon and low-alloy steels in high-dissolved-oxygen water

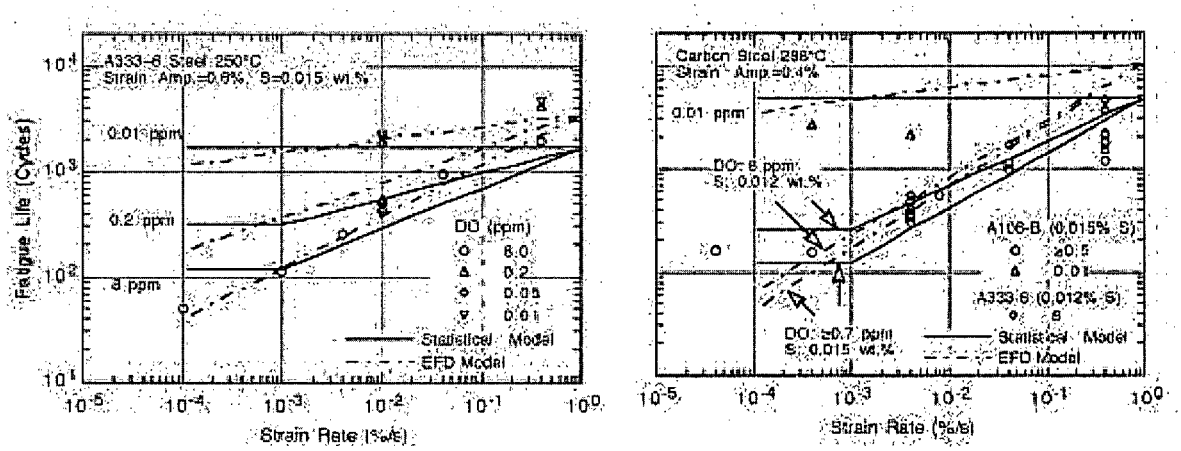

Figure 77. Dependence on strain rate of fatigue life of carbon steels observed experimentally and that estimated from statistical and EFD models

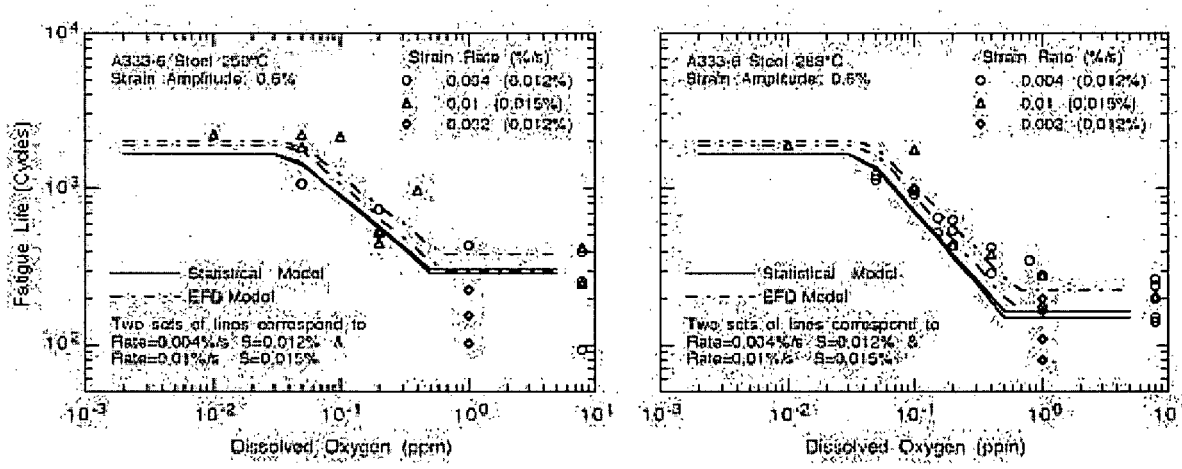

Figure 78. Dependence on dissolved oxygen of fatigue life of carbon steels observed experimentally and that estimated from statistical and EFD models

(b) The statistical model assumes that the effects of strain rate on fatigue life saturate below 0.001%/s, Fig. 77. Such a saturation is not considered in the EFD model.

(c) A threshold temperature of 150°C below which environmental effects on fatigue life are modest is incorporated in the statistical model but not in the EFD model.

(d) The EFD model includes the effect of tensile strength on fatigue life of CSs in LWR environments.

Another estimate of the fatigue life correction factor F_{en} can also be obtained from the statistical model. Since

$$\ln(F_{en}) = \ln(N_{air}) - \ln(N_{water}), \tag{6.4}$$

from Eqs. 5.3a and 5.4a, the fatigue life correction factor for CSs is given by

$$\ln(F_{en}) = 0.585 - 0.00124T - 0.101S^*T^*O^*\dot{\varepsilon}^* \tag{6.5a}$$

and from Eqs. 5.3b and 5.4b, the fatigue life correction factor for LASs is given by

$$\ln(F_{en}) = 0.929 - 0.00124T - 0.101S^*T^*O^*\dot{\varepsilon}^*, \tag{6.5b}$$

where the threshold and saturation values for S^*, T^*, O^*, and $\dot{\varepsilon}^*$ are defined in Eqs. 5.5. A value of 25°C is used for T in Eqs. 6.5a and 6.5b if the fatigue life correction factor is defined relative to RT air. Otherwise, both T and T^* represent the service temperature. A fatigue life correction factor F_{en} based on the statistical model has been proposed as part of a nonmandatory Appendix to ASME Section IX fatigue evaluations.[94,95]

7 Fatigue S–N Curves for Components

The current ASME Section III Code design fatigue curves were based on experimental data on small polished test specimens. The best-fit or mean curve to the experimental data used to develop the Code design curve, expressed in terms of stress amplitude S_a (MPa) and fatigue cycles N, for carbon steels is given by

$$S_a = 59{,}736/\sqrt{N} + 149.24 \tag{7.1a}$$

and for low-alloy steels by

$$S_a = 49{,}222/\sqrt{N} + 265.45. \tag{7.1b}$$

The stress amplitude S_a is the product of strain amplitude ε_a and elastic modulus E; the room temperature value of 206.8 GPa (30,000 ksi) for the elastic modulus for carbon and low-alloy steels was used in converting the experimental strain-versus-life data to stress-versus-life curves. To obtain design fatigue curves the best-fit curves (Eqs. 7.1a and 7.1b) were first adjusted for the effect of mean stress based on the modified Goodman relation

$$S'_a = S_a\left(\frac{\sigma_u - \sigma_y}{\sigma_u - S_a}\right) \qquad \text{for } S_a < \sigma_y, \tag{7.2a}$$

and

$$S'_a = S_a \qquad \text{for } S_a > \sigma_y, \tag{7.2b}$$

where S'_a is the adjusted value of stress amplitude, and σ_y and σ_u are yield and ultimate strengths of the material, respectively. The Goodman relation assumes the maximum possible mean stress and typically gives a conservative adjustment for mean stress at least when environmental effects are not significant. The design fatigue curves were then obtained by lowering the adjusted best-fit curve by a factor of 2 on stress or 20 on cycles, whichever was more conservative, at each point on the curve. The factor of 20 on cycles was intended to account for the uncertainties in fatigue life associated with material and loading conditions, and the factor of 2 on strain was intended to account for uncertainties in threshold strain caused by material variability. This procedure is illustrated for CSs and LASs in Fig. 79.

NYS-26/26A & RIVERKEEPER TC-1/1A: ATTACHMENT 8

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Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels

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EFFECTS OF LWR COOLANT ENVIRONMENTS ON FATIGUE DESIGN CURVES OF AUSTENITIC STAINLESS STEELS

by

O. K. Chopra

Abstract

The ASME Boiler and Pressure Vessel Code provides rules for the construction of nuclear power plant components. Figures I-9.1 through I-9.6 of Appendix I to Section III of the Code specify fatigue design curves for structural materials. While effects of reactor coolant environments are not explicitly addressed by the design curves, test data indicate that the Code fatigue curves may not always be adequate in coolant environments. This report summarizes work performed by Argonne National Laboratory on fatigue of austenitic stainless steels in light water reactor (LWR) environments. The existing fatigue S-N data have been evaluated to establish the effects of various material and loading variables such as steel type, dissolved oxygen level, strain range, strain rate, and temperature on the fatigue lives of these steels. Statistical models are presented for estimating the fatigue S-N curves as a function of material, loading, and environmental variables. Design fatigue curves have been developed for austenitic stainless steel components in LWR environments. The extent of conservatism in the design fatigue curves and an alternative method for incorporating the effects of LWR coolant environments into the ASME Code fatigue evaluations are discussed.

hydrogen-induced cracking. Fatigue striations should not be observed if enhancement of crack growth is caused by the slip oxidation/dissolution process.

5 Statistical Model

The fatigue S-N curves are generally expressed in terms of the Langer equation,[6] which may be used to represent either strain amplitude in terms of life or life in terms of strain amplitude. The parameters of the equation are commonly established through least-squares curve-fitting of the data to minimize the sum of the square of the residual errors for either fatigue life or strain amplitude. A predictive model based on least-squares fit on life is biased for low strain amplitude. The model leads to probability curves that converge to a single value of strain, and fails to address the fact that at low strain values, most of the error in life is due to uncertainty associated with either measurement of strain or variation in fatigue limit caused by material variability. On the other hand, a least-squares fit on strain does not work well for higher strain amplitudes. Statistical models have been developed at ANL[33,34] by combining the two approaches and minimizing the sum of the squared Cartesian distances from the data point to the predicted curve; the models were later updated with a larger fatigue S-N data base.[31] The functional forms and transformation for the different variables were based on experimental observations and data trends.

In air, the model assumes that fatigue life is independent of temperature and that strain rate effects occur at temperatures >250°C. It is also assumed that the effect of strain rate on life depends on temperature. One data set, obtained on Type 316 SS in room-temperature air, was excluded from the analysis. The tests in this data set were conducted in load-control mode at stress levels in the range of 190-230 MPa. The strain amplitudes were calculated only as elastic strains, i.e., strain amplitudes of 0.1-0.12% (the data are shown as circles in Fig. 5, with fatigue lives of 4×10^5 to 3×10^7). Based on cyclic stress vs. strain correlations for Type 316 SS (Eqs. 4a-4f), actual strain amplitudes for these tests should be 0.23-0.32%. In air, the fatigue life N of Types 304 and 316 SS is expressed as

$$\ln(N) = 6.703 - 2.030 \ln(\varepsilon_a - 0.126) + T^* \dot{\varepsilon}^* \quad (5a)$$

and that of Type 316NG, as

$$\ln(N) = 7.422 - 1.671 \ln(\varepsilon_a - 0.126) + T^* \dot{\varepsilon}^*, \quad (5b)$$

where ε_a is the strain amplitude (%) and T^* and $\dot{\varepsilon}^*$ are transformed temperature and strain rate, respectively, defined as follows:

$$T^* = 0 \quad (T < 250°C)$$
$$T^* = [(T - 250)/525]^{0.84} \quad (250 \le T < 400°C) \quad (6a)$$

$$\dot{\varepsilon}^* = 0 \quad (\dot{\varepsilon} > 0.4\%/s)$$
$$\dot{\varepsilon}^* = \ln(\dot{\varepsilon}/0.4) \quad (0.0004 \le \dot{\varepsilon} \le 0.4\%/s)$$
$$\dot{\varepsilon}^* = \ln(0.0004/0.4) \quad (\dot{\varepsilon} < 0.0004\%/s). \quad (6b)$$

In LWR environments, the fatigue lives of austenitic SSs depends on strain rate, DO level, and temperature; the decrease in life is greater at low-DO levels and high temperatures. However, existing data are inadequate to establish the functional form for the dependence of fatigue life on DO level or temperature. Separate correlations have been developed for low- and high-DO levels (< or ≥ 0.05 ppm), and low and high temperatures (< or ≥ 200°C). Also, a threshold strain rate of

0.4%/s and saturation rate of 0.0004%/s is assumed in the model. Furthermore, for convenience in incorporating environmental effects into fatigue evaluations, the slope of the S-N curve in LWR environments was assumed to be the same as that in air although the best-fit of the experimental data in water yielded a slope for the S-N curve that differed from the slope of the curve that was obtained in air. In LWR environments, the fatigue life N of Types 304 and 316 SS is expressed as

$$\ln(N) = 5.768 - 2.030 \ln(\varepsilon_a - 0.126) + T^* \dot{\varepsilon}^* O^* \tag{7a}$$

and that of Type 316NG, as

$$\ln(N) = 6.913 - 1.671 \ln(\varepsilon_a - 0.126) + T^* \dot{\varepsilon}^* O^*, \tag{7b}$$

where the constants for transformed temperature, strain rate, and DO are defined as follows:

$$\begin{aligned} T^* &= 0 && (T < 200°C) \\ T^* &= 1 && (T \geq 200°C) \end{aligned} \tag{8a}$$

$$\begin{aligned} \dot{\varepsilon}^* &= 0 && (\dot{\varepsilon} > 0.4\%/s) \\ \dot{\varepsilon}^* &= \ln(\dot{\varepsilon}/0.4) && (0.0004 \leq \dot{\varepsilon} \leq 0.4\%/s) \\ \dot{\varepsilon}^* &= \ln(0.0004/0.4) && (\dot{\varepsilon} < 0.0004\%/s) \end{aligned} \tag{8b}$$

$$\begin{aligned} O^* &= 0.260 && (DO < 0.05 \text{ ppm}) \\ O^* &= 0.172 && (DO \geq 0.05 \text{ ppm}). \end{aligned} \tag{8c}$$

The model is recommended for predicted fatigue lives $\leq 10^6$ cycles. Recent test results indicate that for high-DO environments, conductivity of water is important for environmental effects on fatigue life of austenitic SSs. Therefore, the above correlations may be conservative for high-DO, i.e., ≥0.05 ppm DO, environments. The experimental values of fatigue life in air and water and those predicted from Eqs. 5-8 are plotted in Fig. 24. The estimated fatigue S-N curves for Types 304, 316, and 316NG SSs in air and LWR environments are shown in Figs. 5 and 25, respectively. The predicted fatigue lives show good agreement with the experimental data. Note that the ASME mean curve is not consistent with the existing fatigue S-N data (Fig. 5). Also, although the best-fit of the S-N data in LWR environments (Fig. 25) yields a steeper slope, the slope of the S-N curve in water was assumed to be the same as in air.

Upon completion of the modeling phase, the residual errors (i.e., the Cartesian distance from the prediction curve) should not show significant patterns, such as heteroskedasticity (changing variance), or a nonzero slope. The residual errors for each variable, grouped by steel type and environment (air or water), are plotted in Figs. 26-30. Most data subsets and plots

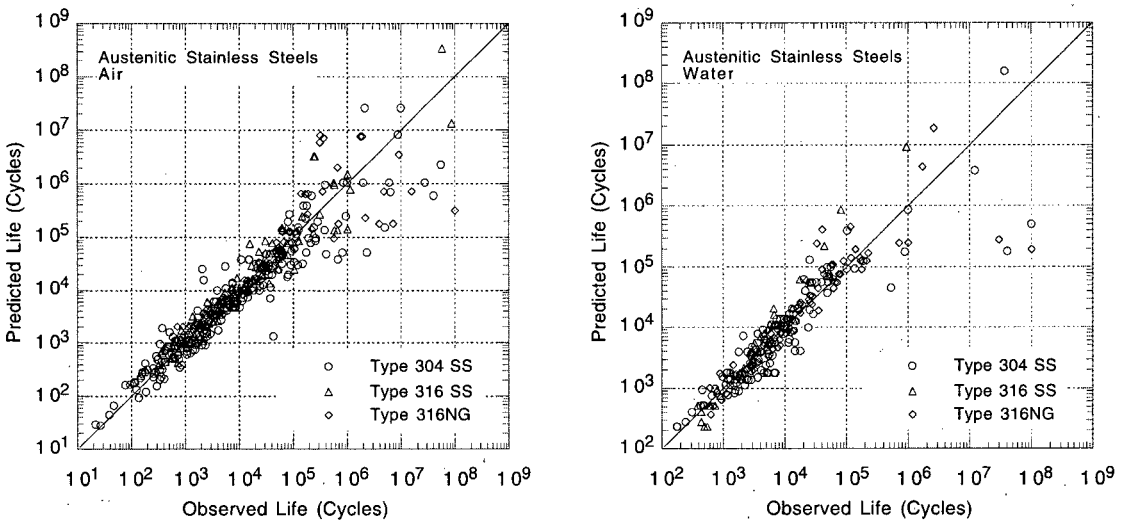


Figure 24. Experimental and predicted values of fatigue lives of austenitic SSs in air and water environments

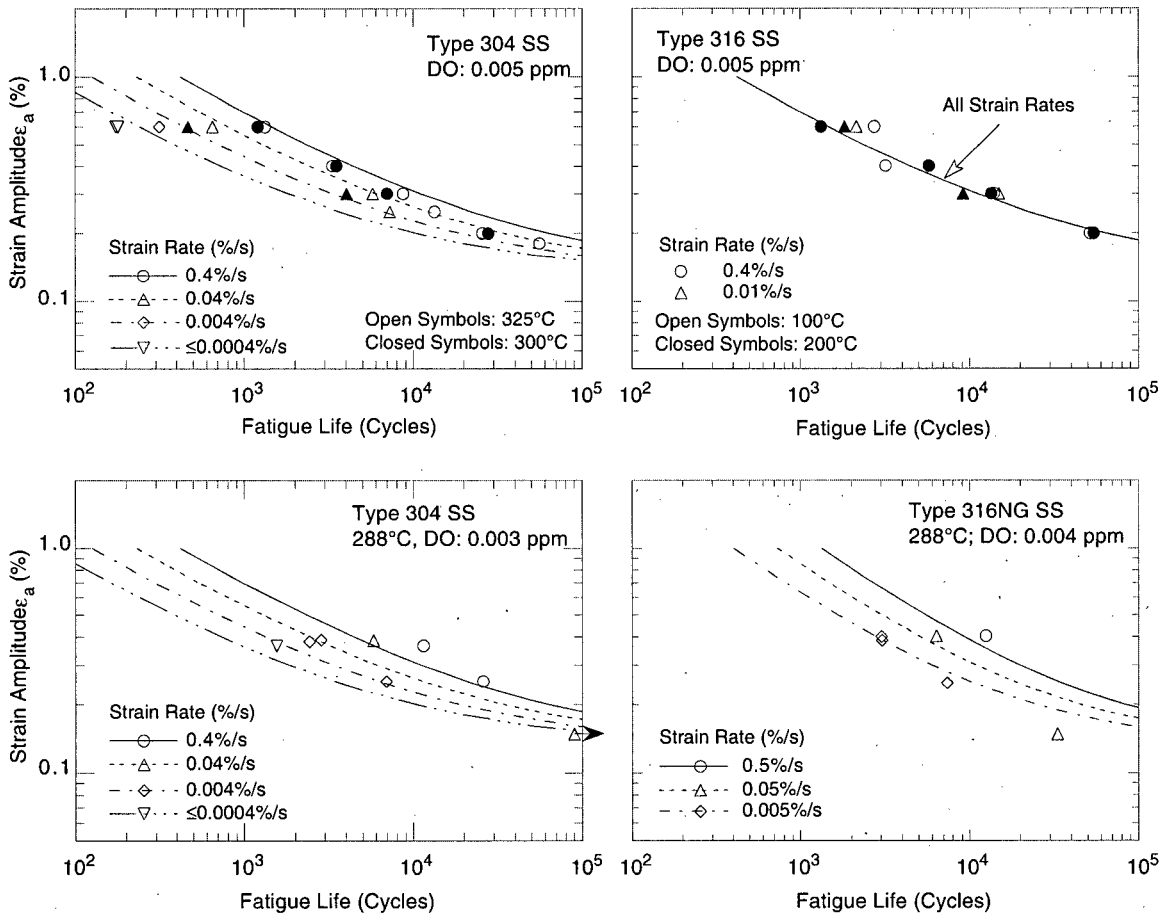


Figure 25. Experimental fatigue lives and those estimated from statistical models for austenitic SSs in water environments

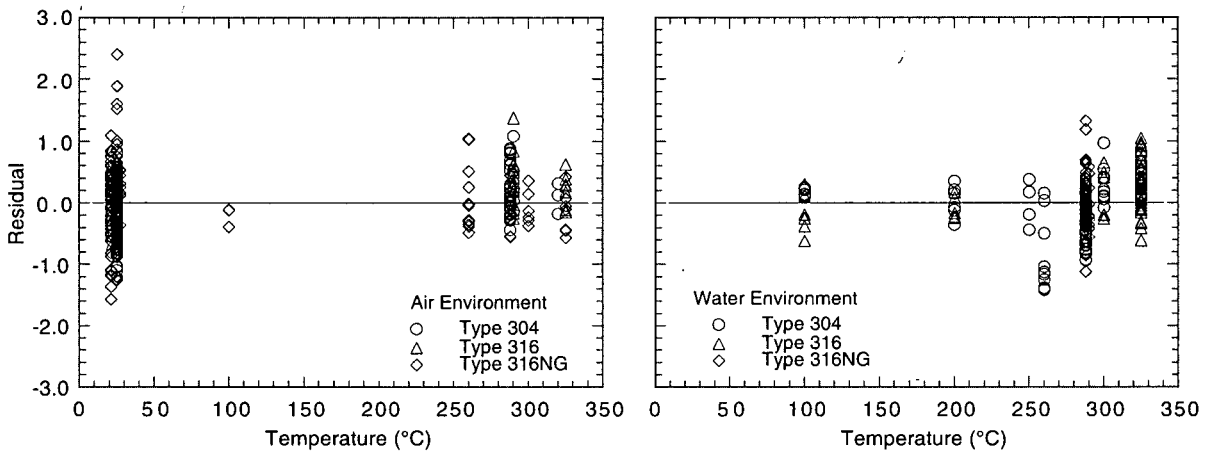


Figure 26. Residual error for austenitic SSs as a function of test temperature

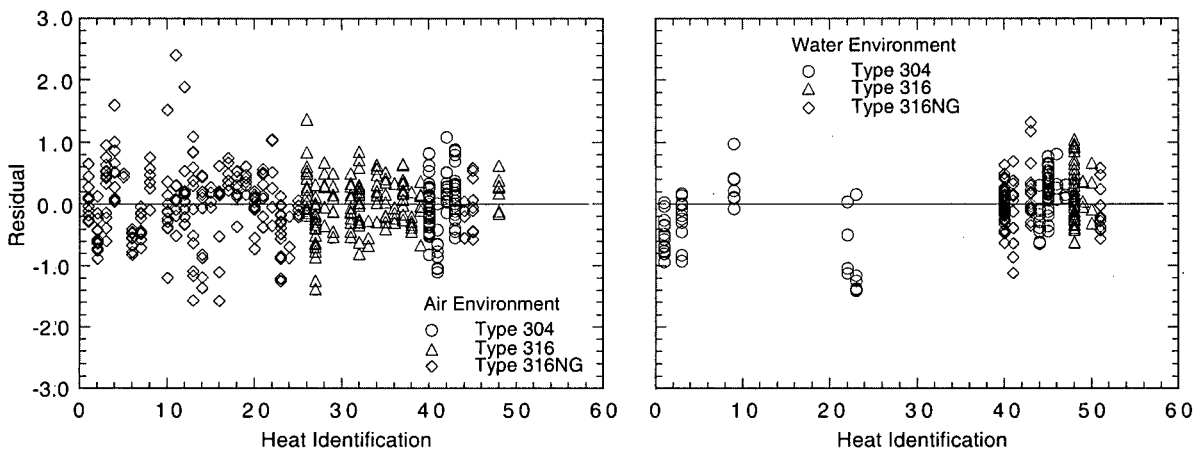


Figure 27. Residual error for austenitic SSs as a function of material heat

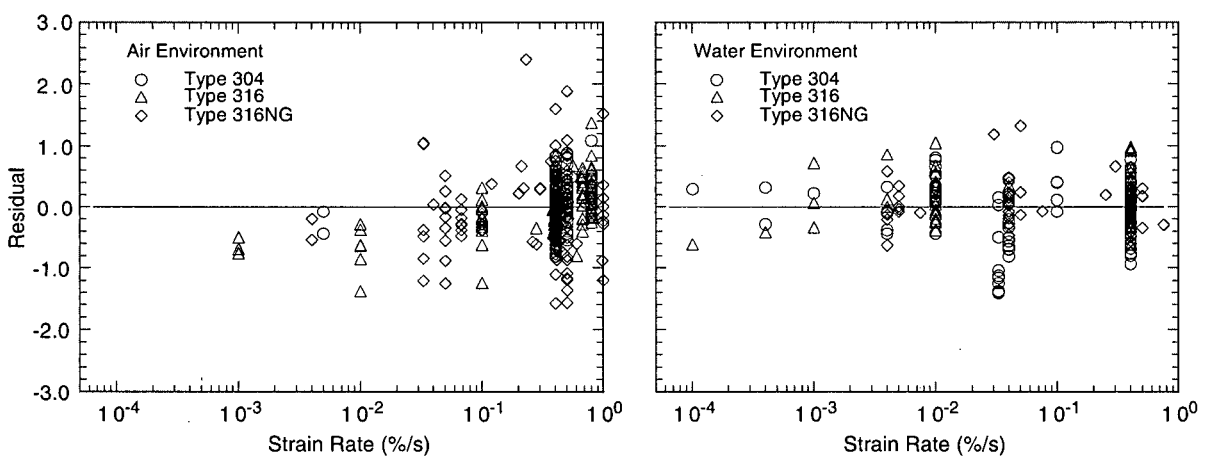


Figure 28. Residual error for austenitic SSs as a function of loading strain rate

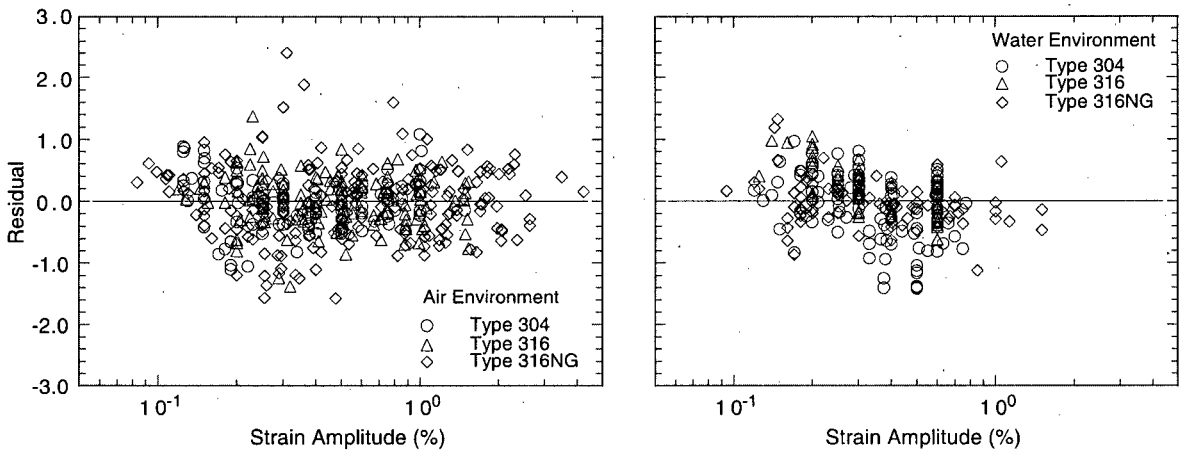


Figure 29. Residual error for austenitic SSs as a function of applied strain amplitude

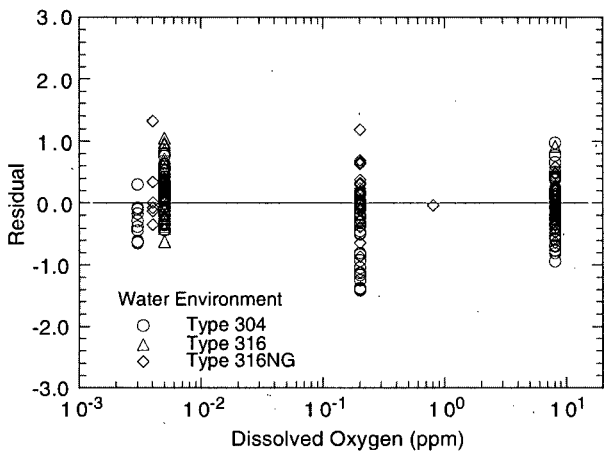


Figure 30.
Residual error for austenitic SSs as a function of dissolved oxygen in water

do not show patterns. In general, high variance tends to be associated with longer lives and lower strain amplitudes. Furthermore, biases seem to be traceable to heat-to-heat variation.

6 Design Fatigue Curves

The design fatigue curves in the current ASME Section III Code were based on experimental data on small polished test specimens. The curves were obtained by adjusting the best-fit curve for the effect of mean stress and then lowering the adjusted curve by a factor of 2 on stress or 20 on life, whichever was more conservative, at each point of the curve. The best-fit curve to the experimental data,[51] expressed in terms of strain amplitude ε_a (%) and fatigue cycles N, for austenitic SSs is given by

$$\ln[N] = 6.954 - 2.0 \ln(\varepsilon_a - 0.167). \quad (9)$$

The mean curve, expressed in terms of stress amplitude S_a (MPa), which is the product of ε_a and elastic modulus E, is given by

$$S_a = 58020/(N)^{1/2} + 299.92. \quad (10)$$

The strain-vs.-life data were converted to stress-vs.-life curves by using the room-temperature value of 195.1 GPa (28300 ksi) for the elastic modulus. The best-fit curves were adjusted for the effect of mean stress by using the modified Goodman relationship[46]

$$S'_a = S_a\left(\frac{\sigma_u - \sigma_y}{\sigma_u - S_a}\right) \qquad \text{for } S_a < \sigma_y, \tag{10a}$$

and $$S'_a = S_a \qquad \text{for } S_a > \sigma_y, \tag{10b}$$

where S'_a is the adjusted value of stress amplitude, and σ_y and σ_u are yield and ultimate strengths of the material, respectively. The Goodman relationship assumes the maximum possible mean stress and typically gives a conservative adjustment for mean stress, at least when environmental effects are not significant. The design fatigue curves were then obtained by lowering the adjusted best-fit curve by a factor of 2 on stress or 20 on cycles, whichever was more conservative, to account for differences and uncertainties in fatigue life associated with material and loading conditions.

The same procedure has been used to develop design fatigue curves for LWR environments. However, because of the differences between the ASME mean curve and the best-fit curve to existing fatigue data (Fig. 5), the margin on strain for the current ASME Code design fatigue curve is closer to 1.5 than 2. Therefore, to be consistent with the current Code design curve, a factor of 1.5 rather than 2 was used in developing the design fatigue curves from the updated statistical models in air and LWR environments.

The design fatigue curves based on the statistical model for Types 304 and 316 SS in air and low- and high-DO water are shown in Figs. 31-33. A similar set of curves can be obtained for Type 316NG SS. Because the fatigue life of Type 316NG is superior to that of Types 304 or 316 SS, Figs. 31-33 may be used conservatively for Type 316NG SS. Also, as mentioned earlier, recent test results indicate that the conductivity of water is important for environmental effects on fatigue life of austenitic SSs in high-DO environments. Therefore, the design fatigue curves for Type 304 and 316 SS in water with ≥0.05 ppm DO (Fig. 33) may be conservative.

Although, in air at low stress levels, the differences between the current ASME Code design curve and the design curve obtained from the updated statistical model at temperatures <250°C have been reduced or eliminated by reducing the margin on stress from 2 to 1.5, significant differences still exist between the two curves. For example, at stress amplitudes >300 MPa, estimates of life from the updated design curve are a factor of ≈2 lower than those from the ASME Code curve. Therefore, the actual margins on stress and life for the current ASME Code design fatigue curve are 1.5 and 10, respectively, instead of 2 and 20.

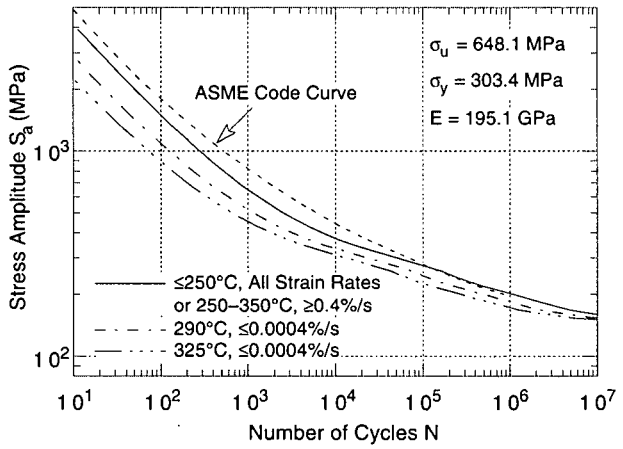


Figure 31.
ASME and statistical-model design fatigue curves for Types 304 and 316 SS in air

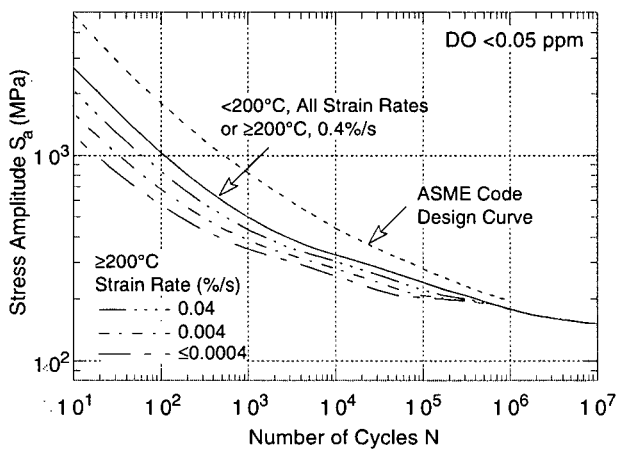


Figure 32.
ASME and statistical-model design fatigue curves for Types 304 and 316 SS in water with <0.05 ppm DO

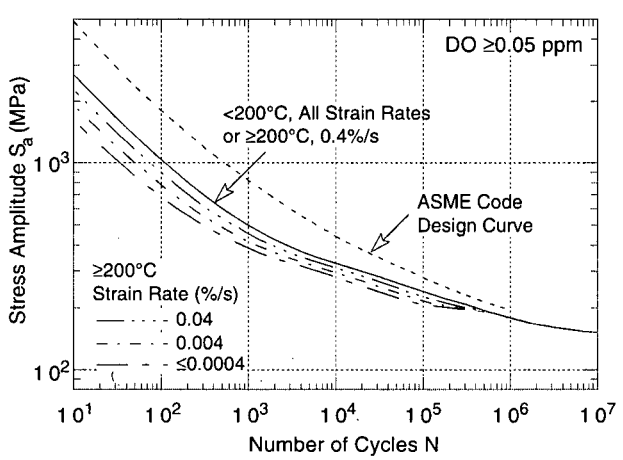


Figure 33.
ASME and statistical-model design fatigue curves for Types 304 and 316 SS in water with ≥0.05 ppm DO

As discussed above, the existing fatigue data indicate a threshold strain range of ≈0.32%, below which environmental effects on the fatigue life of austenitic SSs either do not occur or are insignificant. This value must be adjusted for the effects of mean stress and uncertainties due to material and loading variability. Threshold strain amplitudes are decreased by ≈10% to account for mean stress effects and by a factor of 1.5 to account for uncertainties in fatigue life associated with material and loading variability. Thus, a threshold strain amplitude of 0.097% (stress amplitude of 189 MPa) was selected, below which environmental effects on life are modest and are represented by the design curve for temperatures <200°C (shown by the solid line in Figs. 31 and 32).

These curves can be used to perform ASME Code fatigue evaluations of components that are in service in LWR environments. For each set of load pairs, a partial usage factor is obtained from the appropriate design fatigue curve. Information about the service conditions, such as temperature, strain rate, and DO level, are required for the evaluations. The procedure for obtaining these parameters depends on whether the elapsed-time-vs.-temperature information for the transient is available. The maximum values of temperature and DO level and the slowest strain rate during the transient may be used for a conservative estimate of life. Note that the design curves in LWR environments not only account for environmental effects on life but also include the difference between the current Code design curve and the updated design curve in air, i.e., the difference between the solid and dashed curves in Fig. 31.

7 Fatigue Life Correction Factor

The effects of reactor coolant environments on fatigue life have also been expressed in terms of a fatigue life correction factor F_{en}, which is the ratio of the life in air at room temperature to that in water at the service temperature.[11,52,53] To incorporate environmental effects into the ASME Code fatigue evaluation, a fatigue usage for a specific load pair, based on the current Code fatigue design curve, is multiplied by the correction factor. A fatigue life correction factor F_{en} can also be obtained from the statistical model, where

$$\ln(F_{en}) = \ln(N_{air}) - \ln(N_{water}). \tag{12}$$

From Eqs. 5a and 7a, the fatigue life correction factor relative to room-temperature air for Types 304 and 316 SSs is given by

$$F_{en} = \exp(0.935 - T^* \dot{\varepsilon}^* O^*), \tag{13}$$

where the threshold and saturation values for T^*, $\dot{\varepsilon}^*$, and O^* are defined in Eqs. 8a-8c. At temperatures ≥200°C and strain rates ≤0.0004%/s, Eq. 13 yields an F_{en} of ≈15 in low-DO PWR water (<0.05 ppm DO) and ≈8 in high-DO water (≥0.05 ppm DO). At temperatures <200°C, F_{en} is ≈2.5 in both low- and high-DO water at all strain rates.

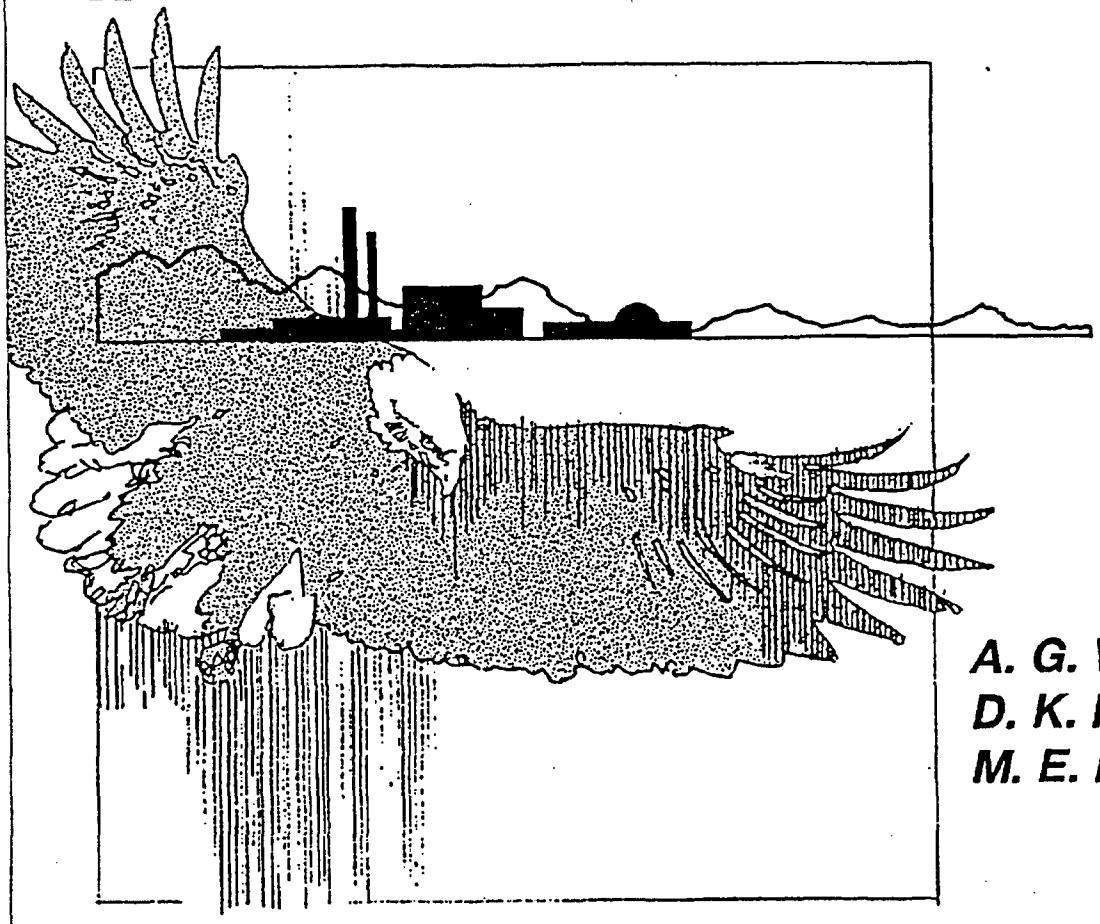
8 Conservatism in Design Fatigue Curves

The overall conservatism in ASME Code fatigue evaluations has also been demonstrated in fatigue tests on piping welds and components.[54] In air, the margins on the number of cycles to failure for austenitic SS elbows and tees were 40-310 and 104-510, respectively. The margins for girth butt welds were significantly lower at 6-77. In these tests, fatigue life was expressed as the number of cycles for the crack to penetrate through the wall, which ranged in thickness from 6 to 18 mm (0.237 to 0.719 in). The fatigue design curves represent the number of cycles that are necessary to form a 3-mm-deep crack. Consequently, depending on wall thickness, the actual margins to failure may be lower by a factor of >2.

Deardorff and Smith[55] have discussed the types and extent of conservatisms present in the ASME Section III fatigue evaluations and the effects of LWR environments on fatigue margins. The sources of conservatism include design transients considerably more severe than those experienced in service, grouping of transients, and simplified elastic-plastic analysis. Environmental effects on two components, the BWR feedwater nozzle/safe end and PWR steam generator feedwater nozzle/safe end, both constructed from LAS and known to be affected by severe thermal transients,

NYS-26/26A & RIVERKEEPER TC-1/1A: ATTACHMENT 9

IDAHO NATIONAL ENGINEERING LABORATORY



NUREG/CR-6260
INEL-95/0045

February 1995

A. G. Ware
D. K. Morton
M. E. Nitzel

Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components

Idaho National Engineering Laboratory

Lockheed Idaho Technologies Company

Work performed under
DOE Contract
No. DE-AC07-94ID13223

NUREG/CR-6260
INEL-95/0045
Distribution Category: R5

Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components

A. G. Ware
D. K. Morton
M. E. Nitzel

Manuscript Completed February 1995

Idaho National Engineering Laboratory
Lockheed Idaho Technologies Company
Idaho Falls, Idaho 83415

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Division of Engineering
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
Under DOE Idaho Operations Office
Contract DE-AC07-94ID13223
FIN J2081

ABSTRACT

Recent test data indicate that the effects of the light water reactor (LWR) environment could significantly reduce the fatigue resistance of materials used in the reactor coolant pressure boundary components of operating nuclear power plants. Argonne National Laboratory has developed interim fatigue curves based on test data simulating LWR conditions, and published them in NUREG/CR-5999. In order to assess the significance of these interim fatigue curves, fatigue evaluations of a sample of the components in the reactor coolant pressure boundary of LWRs were performed. The sample consists of components from facilities designed by each of the four U.S. nuclear steam supply system vendors. For each facility, six locations were studied, including two locations on the reactor pressure vessel. In addition, there are older vintage plants where components of the reactor coolant pressure boundary were designed to codes that did not require an explicit fatigue analysis of the components. In order to assess the fatigue resistance of the older vintage plants, an evaluation was also conducted on selected components of three of these plants. This report discusses the insights gained from the application of the interim fatigue curves to components of seven operating nuclear power plants.

4. APPROACH

4.1 Selection of Components for Analysis

The components chosen for the evaluation of the five PWR plants [B&W, Combustion Engineering (one older vintage and one newer vintage), and Westinghouse (one older vintage and one newer vintage)] are as follows:

1. Reactor vessel shell and lower head.
2. Reactor vessel inlet and outlet nozzles.
3. Pressurizer surge line (including hot leg and pressurizer nozzles).
4. Reactor coolant piping charging system nozzle.
5. Reactor coolant piping safety injection nozzle.
6. Residual heat removal (RHR) system Class 1 piping.

The terminology used above is for Westinghouse plants. The first three components are the same for Combustion Engineering and B&W plants, but the latter three components for the three PWR nuclear steam supply system (NSSS) vendors are different either simply in name or in the routing of the piping. For cases where there is no direct one-for-one correspondence, the location that most nearly corresponded to the Westinghouse component was chosen. These locations are described in Section 5.

The components chosen for the evaluation of the two BWR plants [General Electric (one older vintage and one newer vintage)] are as follows:

1. Reactor vessel shell and lower head.
2. Reactor vessel feedwater nozzle.
3. Reactor recirculation piping (including inlet and outlet nozzles).
4. Core spray line reactor vessel nozzle and associated Class 1 piping.
5. RHR Class 1 piping.
6. Feedwater line Class 1 piping.

For both PWR and BWR plants, these components are not necessarily the locations with the highest design CUFs in the plant, but were chosen to give a representative overview of components that had higher CUFs and/or were important from a risk perspective. For example, the reactor vessel shell and lower head was chosen for its risk importance.

4.2 Application of NUREG/CR-5999 Fatigue Curves

NUREG/CR-5999 includes one fatigue curve for stainless steel, but several curves for carbon/low-alloy steels which are based on the sulfur content of the steel and the oxygen level in the coolant. For the five PWR plants, the curves for high-sulfur steel and a low-oxygen environment (typical for PWRs) were used. For the two BWR plants, the curves for high-sulfur steel and a high-oxygen environment were used. The high-oxygen (greater than 100 ppm) environment considered in the selected curves is consistent with the water chemistry in BWRs without hydrogen water chemistry. Neither of the two BWR plants evaluated have used hydrogen water chemistry.

4.2.1 Interior and Exterior Surfaces. The highest CUFs for components in the seven plants evaluated in this fatigue assessment study generally occur on the interior surfaces which experience the full effects of thermal shocks from fluid temperature changes. In a few cases the highest CUF was found to occur on the exterior surface (because of stress concentration effects), and in other cases no differentiation between interior and exterior surfaces was made in the licensee's calculations. Since it is expected that the interior

5.5 Older Vintage Westinghouse Plant

A comparison of the design CUFs from the licensee's design basis calculations and CUFs using the NUREG/CR-5999 interim fatigue curves was carried out for the locations of highest design CUF for the six components listed below:

1. Reactor vessel shell and lower head
2. Reactor vessel inlet and outlet nozzles
3. Pressurizer surge line (including hot leg and pressurizer nozzles)
4. Reactor coolant piping charging system nozzle (representative design basis fatigue calculation performed by INEL)
5. Reactor coolant piping safety injection nozzle (representative design basis fatigue calculation performed by INEL)
6. Residual Heat Removal system Class 1 piping (representative design basis fatigue calculation performed by INEL).

As of late 1993, the plant has been operated approximately 20 of the 40 years currently approved in its operating license. Table 5-83 shows the design basis cycles for transients that are important from a fatigue standpoint for the six components that were evaluated. The numbers of transients to date have been extrapolated to 40 years by multiplying by 40/20.

The results of a generic Westinghouse plant study of thermal stratification in surge lines was included in the licensee's fatigue analysis of the surge line. There were no plant specific data to remove conservatism assumptions for this particular plant.

5.5.1 Reactor Vessel Shell and Lower Head. The highest CUF on the lower shell and head is 0.290 for the inside surface of the lower head near the shell-to-head transition, where core support guides are welded to the interior of the shell. The SA-302 Grade B head is protected from the coolant by a layer of stainless steel and Alloy 600 cladding. No fatigue analysis is performed for the cladding.

5.5.1.1 NUREG/CR-5999 CUF Based on Licensee's Design Calculation Stresses. The licensee's CUF calculations used the ASME Code, Section III, 1965 edition, through Summer 1966 addenda.

The effect of the NUREG/CR-5999 interim fatigue curve is shown in Table 5-84. As previously discussed, the results shown in Table 5-84 assume that the coolant is in contact with the low-alloy steel base metal underneath the cladding. The S_{alt} values were adjusted for the effect of the modulus of elasticity by multiplying by 30/27, the ratio of the modulus of elasticity on the fatigue curve in the current edition of the Code to the value at 500°F for SA-302 Grade B low-alloy steel. The 1965 Code edition did not require an adjustment for the effect of the modulus of elasticity.

Table 5-83. Number of selected design basis cycles compared to anticipated number of cycles over 40-year license life.

Transient	Design basis cycles	Anticipated cycles for 40 years
Heatup/cooldown	200	172
Reactor trip	400	426
Hydrotest	5	2
5% power change	14500	512
10% power change (up/down)	2000/7000	42/86
50% power change	200	136

NYS-26/26A & RIVERKEEPER TC-1/1A: ATTACHMENT 10

Entergy Nuclear Northeast
Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
Buchanan, NY 10511-0249
Tel (914) 788-2055

Fred Dacimo
Vice President
License Renewal

January 22, 2008

Re: Indian Point Units 2 & 3
Docket Nos. 50-247 & 50-286
NL-08-021

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

SUBJECT: Entergy Nuclear Operations Inc.
Indian Point Nuclear Generating Unit Nos. 2 & 3
Docket Nos. 50-247 and 50-286
License Renewal Application Amendment 2

REFERENCES:

1. Entergy Letter dated April 23, 2007, F. R. Dacimo to Document Control Desk, "License Renewal Application" (NL-07-039)
2. Entergy Letter dated April 23, 2007, F. R. Dacimo to Document Control Desk, "License Renewal Application Boundary Drawings" (NL-07-040)
3. Entergy Letter dated April 23, 2007, F. R. Dacimo to Document Control Desk, "License Renewal Application Environmental Report References" (NL-07-041)

Dear Sir or Madam:

In the referenced letters, Entergy Nuclear Operations, Inc. (Entergy) applied for renewal of the Indian Point Energy Center operating licenses for Unit 2 and 3.

Based on discussions during license renewal audits, clarification to the LRA is provided in Attachment 1. This information clarifies the relationship between Commitment 33 regarding environmentally assisted fatigue and the Fatigue Monitoring Program described in LRA Section B.1.12. The Fatigue Monitoring Program includes the actions identified in Commitment 33 to address the evaluation of the effects of environmentally assisted fatigue in accordance with 10 CFR 54.21(c)(1)(iii). The revised Regulatory Commitment List is provided in Attachment 2.

If you have any questions, or require additional information, please contact Mr. Robert Walpole at 914-734-6710.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 1-22-08.

Sincerely,

Patricia W Conroy for
Fred R. Dacimo per telecon
Vice President
License Renewal

Attachments:

1. Fatigue Monitoring Program Clarification
2. Regulatory Commitment List, Revision 3

cc: Mr. Samuel J. Collins, Regional Administrator, NRC Region I
Mr. Kenneth Chang, NRC Branch Chief, Engineering Review Branch I
Mr. Bo M. Pham, NRC Environmental Project Manager
Mr. John Boska, NRR Senior Project Manager
Mr. Paul Eddy, New York State Department of Public Service
NRC Resident Inspector's Office
Mr. Paul D. Tonko, President, New York State Energy, Research, & Development Authority

ATTACHMENT 1 TO NL-08-021

Fatigue Monitoring Program Clarification

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 & 3
DOCKET NOS. 50-247 AND 50-286

License Renewal Application
Amendment 2

Fatigue Monitoring Program Clarification

LRA and commitment list revisions are provided below. (underline - added, strikethrough – deleted)

LRA Table 4.1-1, List of IP2 TLAA and Resolution, line item titled "Effects of reactor water environment on fatigue life", is revised as follows.

Effects of reactor water environment on fatigue life	~~Analyses remain valid 10 CFR 54.21(c)(1)(i)~~ ~~OR~~ Aging effect managed 10 CFR 54.21(c)(1)(iii)	4.3.3

LRA Table 4.1-2, List of IP3 TLAA and Resolution, line item titled "Effects of reactor water environment on fatigue life", is revised as follows.

Effects of reactor water environment on fatigue life	~~Analyses remain valid 10 CFR 54.21(c)(1)(i)~~ ~~OR~~ Aging effect managed 10 CFR 54.21(c)(1)(iii)	4.3.3

LRA Section 4.3.3, paragraph 10 is revised as follows.

At least 2 years prior to entering the period of extended operation, for the locations identified in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3)~~NUREG/CR-6260 for Westinghouse PWRs of the IPEC vintage~~, under the Fatigue Monitoring Program IPEC will implement one or more of the following.

(1) Consistent with the Fatigue Monitoring Program, Detection of Aging Effects, update the fatigue usage calculations using ~~R~~refined ~~the~~ fatigue analyses to determine valid CUFs less than 1.0 when accounting for the effects of reactor water environment. This includes applying the appropriate F_{en} factors to valid CUFs determined in accordance with one of the following.

For locations in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3)~~including NUREG/CR-6260 locations~~, with existing fatigue analysis valid for the period of

extended operation, use the existing CUF ~~to determine the environmentally adjusted CUF~~.

~~More limiting IPEC~~ Additional plant-specific locations with a valid CUF may be ~~added in addition to the NUREG/CR-6260 locations~~evaluated. In particular, the pressurizer lower shell will be reviewed to ensure the surge nozzle remains the limiting component.

Representative CUF values from other plants, adjusted to or enveloping the IPEC plant-specific external loads may be used if demonstrated applicable to IPEC.

An analysis using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case) may be performed to determine a valid CUF.

~~(2) Manage the effects of aging due to fatigue at the affected locations 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 acceptable to the NRC).~~

(~~3~~2) Consistent with the Fatigue Monitoring Program, Corrective Actions, ~~R~~repair or replace the affected locations before exceeding a CUF of 1.0.

~~Should IPEC select the option to manage the aging effects due to environmental-assisted fatigue during the period of extended operation, details of the aging management program such as scope, qualification, method, and frequency will be submitted to the NRC at least 2 years prior to the period of extended operation.~~

~~Depending on the option chosen, which may vary by component, this TLAA will be projected through the period of extended operation per 10CFR54.21(c)(1)(ii) or t~~The effects of environmentally assisted fatigue will be managed per 10CFR54.21(c)(1)(iii).

LRA Section A.2.1.11, Fatigue Monitoring Program, is revised as follows.

The Fatigue Monitoring Program is an existing program that tracks the number of critical thermal and pressure transients for selected reactor coolant system components. The program ensures the validity of analyses that explicitly analyzed a specified number of fatigue transients by assuring that the actual effective number of transients does not exceed the analyzed number of transients. The program provides for update of the fatigue usage calculations to maintain a CUF of < 1.0 for the period of extended operation. For the locations identified in Section A.2.2.2.3, updated calculations will account for the effects of the reactor water environment. These calculation updates are governed by Entergy's 10 CFR 50 Appendix B Quality Assurance (QA) program and include design input verification and independent reviews ensuring that valid assumptions, transients, cycles, external loadings, analysis methods, and environmental fatigue life correction factors will be used in

the fatigue analyses. The program requires corrective actions including repair or replacement of affected components before fatigue usage calculations determine the CUF exceeds 1.0. Specific corrective actions are implemented in accordance with the IPEC corrective action program. Repair or replacement of the affected component(s), if necessary, will be in accordance with established plant procedures governing repair and replacement activities. These established procedures are governed by Entergy's 10 CFR 50 Appendix B QA program and meet the applicable repair or replacement requirements of the ASME Code Section XI.

LRA Section A.2.2.2.3, Environmental Effects on Fatigue, is revised as follows.

The effects of reactor water environment on fatigue were evaluated for license renewal. Projected cumulative usage factors (CUFs) were calculated for the limiting locations ~~identified in~~ based on NUREG/CR-6260. The identified IP2 locations are those listed in the license renewal application, Table 4.3-13. ~~For the locations with CUFs less than 1.0, the TLAA has been projected through the period of extended operation per 10 CFR 54.21(c)(1)(ii).~~ Several locations may exceed a CUF of 1.0 with consideration of environmental effects during the period of extended operation. The Fatigue Monitoring Program requires that ~~A~~at least two years prior to entering the period of extended operation, the site will implement one or more of the following:

(1) Consistent with the Fatigue Monitoring Program, Detection of Aging Effects, update the fatigue usage calculations using ~~R~~refined ~~the~~fatigue analyses to determine valid CUFs less than 1.0 when accounting for the effects of reactor water environment. This includes applying the appropriate F_{en} factors to valid CUFs determined in accordance with one of the following.

For locations~~, including NUREG/CR-6260 locations,~~ with existing fatigue analysis valid for the period of extended operation, use the existing CUF ~~to determine the environmentally adjusted CUF~~.

~~In addition to the NUREG/CR-6260 locations, more limiting~~Additional plant-specific locations with a valid CUF may be evaluated. In particular, the pressurizer lower shell will be reviewed to ensure the surge nozzle remains the limiting component.

Representative CUF values from other plants, adjusted to or enveloping the plant-specific external loads may be used if demonstrated applicable.

An analysis using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case) may be performed to determine a valid CUF.

~~(2) Manage the effects of aging due to fatigue at the affected locations 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 acceptable to the NRC).~~

(~~3~~2) Consistent with the Fatigue Monitoring Program, Corrective Actions, ~~R~~repair or replace the affected locations before exceeding a CUF of 1.0.

~~Should IPEC select the option to manage the aging effects due to environmental-assisted fatigue during the period of extended operation, details of the aging management program such as scope, qualification, method, and frequency will be submitted to the NRC at least 2 years prior to the period of extended operation.~~

LRA Section A.3.1.11, Fatigue Monitoring Program, is revised as follows.

The Fatigue Monitoring Program is an existing program that tracks the number of critical thermal and pressure transients for selected reactor coolant system components. The program ensures the validity of analyses that explicitly analyzed a specified number of fatigue transients by assuring that the actual effective number of transients does not exceed the analyzed number of transients. The program provides for update of the fatigue usage calculations to maintain a CUF of < 1.0 for the period of extended operation. For the locations identified in Section A.3.2.2.3, updated calculations will account for the effects of the reactor water environment. These calculation updates are governed by Entergy's 10 CFR 50 Appendix B Quality Assurance (QA) program and include design input verification and independent reviews ensuring that valid assumptions, transients, cycles, external loadings, analysis methods, and environmental fatigue life correction factors will be used in the fatigue analyses. The program requires corrective actions including repair or replacement of affected components before fatigue usage calculations determine the CUF exceeds 1.0. Specific corrective actions are implemented in accordance with the IPEC corrective action program. Repair or replacement of the affected component(s), if necessary, will be in accordance with established plant procedures governing repair and replacement activities. These established procedures are governed by Entergy's 10 CFR 50 Appendix B QA program and meet the applicable repair or replacement requirements of the ASME Code Section XI.

LRA Section A.3.2.2.3, Environmental Effects on Fatigue, is revised as follows.

The effects of reactor water environment on fatigue were evaluated for license renewal. Projected cumulative usage factors (CUFs) were calculated for the limiting locations ~~identified in~~ based on NUREG/CR-6260. The identified IP3 locations are those listed in the license renewal application, Table 4.3-14. ~~For the locations with CUFs less than 1.0, the TLAA has been projected through the period of extended operation per 10 CFR 54.21(c)(1)(ii).~~ Several locations may exceed a CUF of 1.0 with consideration of environmental effects during the period of extended operation. The Fatigue Monitoring Program requires that ~~A~~at least two years prior to entering the period of extended operation, ~~for the locations identified in NUREG/CR-6260 for Westinghouse PWRs of this vintage,~~ the site will implement one or more of the following:

(1) Consistent with the Fatigue Monitoring Program, Detection of Aging Effects, update the fatigue usage calculation using ~~R~~refined ~~the~~ fatigue analyses to determine valid CUFs less than 1.0 when accounting for the effects of reactor water environment. This includes applying the appropriate F_{en} factors to valid CUFs determined in accordance with one of the following.

For locations~~, including NUREG/CR-6260 locations,~~ with existing fatigue analysis valid for the period of extended operation, use the existing CUF ~~to determine the environmentally adjusted CUF~~.

~~In addition to the NUREG/CR-6260 locations,more limiting~~Additional plant-specific locations with a valid CUF may be evaluated. In particular, the pressurizer lower shell will be *reviewed to ensure the surge nozzle remains the limiting component.*

Representative CUF values from other plants, adjusted to or enveloping the plant-specific external loads may be used if demonstrated applicable.

An analysis using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case) may be performed to determine a valid CUF.

~~(2) Manage the effects of aging due to fatigue at the affected locations 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 acceptable to the NRC).~~

(~~3~~2) Consistent with the Fatigue Monitoring Program, Corrective Actions, ~~R~~repair or replace the affected locations before exceeding a CUF of 1.0.

~~Should IPEC select the option to manage the aging effects due to environmental-assisted fatigue during the period of extended operation, details of the aging management program such as scope, qualification, method, and frequency will be submitted to the NRC at least 2 years prior to the period of extended operation.~~

LRA Section B.1.12, Fatigue Monitoring, Program Description, is revised as follows.

The Fatigue Monitoring Program is an existing program that tracks the number of critical thermal and pressure transients for selected reactor coolant system components. The program ensures the validity of analyses that explicitly analyzed a specified number of fatigue transients by assuring that the actual effective number of transients does not exceed the analyzed number of transients.

The program provides for update of the fatigue usage calculations to maintain a CUF of < 1.0 for the period of extended operation. For the locations identified in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3), updated calculations will account for the effects of the reactor water environment. These calculation updates are governed by Entergy's 10 CFR 50 Appendix B Quality Assurance (QA) program and include design input verification and

independent reviews ensuring that valid assumptions, transients, cycles, external loadings, analysis methods, and environmental fatigue life correction factors will be used in the fatigue analyses.

The analysis methods for determination of stresses and fatigue usage will be in accordance with an NRC endorsed Edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III Rules for Construction of Nuclear Power Plant Components Division 1 Subsection NB, Class 1 Components, Sub articles NB-3200 or NB-3600 as applicable to the specific component. IPEC will utilize design transients from IPEC Design Specifications to bound all operational transients. The numbers of cycles used for evaluation will be based on the design number of cycles and actual IPEC cycle counts projected out to the end of the license renewal period (60 years).

Environmental effects on fatigue usage will be assessed using methodology consistent with the Generic Aging Lessons Learned Report, NUREG-1801, Rev. 1, (GALL) that states; "The sample of critical components can be evaluated by applying environmental life correction factors to the existing ASME Code fatigue analyses. Formulae for calculating the environmental life correction factors are contained in NUREG/CR-6583 for carbon and low-alloy steels and in NUREG/CR-5704 for austenitic stainless steels."

The Fatigue Monitoring Program tracks actual plant transients and evaluates these against the design transients. Cycle counts show no limits are expected to be approached for the current license term. The Fatigue Monitoring Program will ensure that the numbers of transient cycles experienced by the plant remain within the analyzed numbers of cycles and hence, the component CUFs remain below the values calculated in the design basis fatigue evaluations. If ongoing monitoring indicates the potential for a condition outside that analyzed above, IPEC may perform further reanalysis of the identified configuration using established configuration management processes as described above.

The program requires corrective actions including repair or replacement of affected components before fatigue usage calculations determine the CUF exceeds 1.0. Specific corrective actions are implemented in accordance with the IPEC corrective action program. Repair or replacement of the affected component(s), if necessary, will be in accordance with established plant procedures governing repair and replacement activities. These established procedures are governed by Entergy's 10 CFR 50 Appendix B QA program and meet the applicable repair or replacement requirements

ATTACHMENT 2 TO NL-08-021

Regulatory Commitment List, Revision 3

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 & 3
DOCKET NOS. 50-247 AND 50-286

List of Regulatory Commitments

Rev. 3

The following table identifies those actions committed to by Entergy in this document.

Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
1	Enhance the Aboveground Steel Tanks Program for IP2 and IP3 to perform thickness measurements of the bottom surfaces of the condensate storage tanks, city water tank, and fire water tanks once during the first ten years of the period of extended operation. Enhance the Aboveground Steel Tanks Program for IP2 and IP3 to require trending of thickness measurements when material loss is detected.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.1 A.3.1.1 B.1.1
2	Enhance the Bolting Integrity Program for IP2 and IP3 to clarify that actual yield strength is used in selecting materials for low susceptibility to SCC and clarify the prohibition on use of lubricants containing MoS_2 for bolting. The Bolting Integrity Program manages loss of preload and loss of material for all external bolting.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.2 A.3.1.2 B.1.2 Audit Items 201, 241, 270
3	Implement the Buried Piping and Tanks Inspection Program for IP2 and IP3 as described in LRA Section B.1.6. This new program will be implemented consistent with the corresponding program described in NUREG-1801 Section XI.M34, Buried Piping and Tanks Inspection.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.5 A.3.1.5 B.1.6 Audit Item 173

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
4	Enhance the Diesel Fuel Monitoring Program to include cleaning and inspection of the IP2 GT-1 gas turbine fuel oil storage tanks, IP2 and IP3 EDG fuel oil day tanks, IP2 SBO/Appendix R diesel generator fuel oil day tank, and IP3 Appendix R fuel oil storage tank and day tank once every ten years. Enhance the Diesel Fuel Monitoring Program to include quarterly sampling and analysis of the IP2 SBO/Appendix R diesel generator fuel oil day tank, IP2 security diesel fuel oil day tank, and IP3 Appendix R fuel oil storage tank. Particulates, water and sediment checks will be performed on the samples. Filterable solids acceptance criterion will be less than or equal to 10mg/l. Water and sediment acceptance criterion will be less than or equal to 0.05%. Enhance the Diesel Fuel Monitoring Program to include thickness measurement of the bottom surface of the following tanks once every ten years. IP2: EDG fuel oil storage tanks, EDG fuel oil day tanks, SBO/Appendix R diesel generator fuel oil day tank, GT-1 gas turbine fuel oil storage tanks, and diesel fire pump fuel oil storage tank; IP3: EDG fuel oil day tanks, Appendix R fuel oil storage tank, and diesel fire pump fuel oil storage tank. Enhance the Diesel Fuel Monitoring Program to change the analysis for water and particulates to a quarterly frequency for the following tanks. IP2: GT-1 gas turbine fuel oil storage tanks and diesel fire pump fuel oil storage tank; IP3: Appendix R fuel oil day tank and diesel fire pump fuel oil storage tank. Enhance the Diesel Fuel Monitoring Program to specify acceptance criteria for thickness measurements of the fuel oil storage tanks within the scope of the program. Enhance the Diesel Fuel Monitoring Program to direct samples be taken near the tank bottom and include direction to remove water when detected. Enhance the Diesel Fuel Monitoring Program to direct the addition of chemicals including biocide when the presence of biological activity is confirmed.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.8 A.3.1.8 B.1.9 Audit items 128, 129, 132

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
5	Enhance the External Surfaces Monitoring Program for IP2 and IP3 to include periodic inspections of systems in scope and subject to aging management review for license renewal in accordance with 10 CFR 54.4(a)(1) and (a)(3). Inspections shall include areas surrounding the subject systems to identify hazards to those systems. Inspections of nearby systems that could impact the subject systems will include SSCs that are in scope and subject to aging management review for license renewal in accordance with 10 CFR 54.4(a)(2).	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.10 A.3.1.10 B.1.11
6	Enhance the Fatigue Monitoring Program for IP2 to monitor steady state cycles and feedwater cycles or perform an evaluation to determine monitoring is not required. Review the number of allowed events and resolve discrepancies between reference documents and monitoring procedures. Enhance the Fatigue Monitoring Program for IP3 to include all the transients identified. Assure all fatigue analysis transients are included with the lowest limiting numbers. Update the number of design transients accumulated to date.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.11 A.3.1.11 B.1.12, Audit Item 164

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
7	Enhance the Fire Protection Program to inspect external surfaces of the IP3 RCP oil collection systems for loss of material each refueling cycle. Enhance the Fire Protection Program to explicitly state that the IP2 and IP3 diesel fire pump engine sub-systems (including the fuel supply line) shall be observed while the pump is running. Acceptance criteria will be revised to verify that the diesel engine does not exhibit signs of degradation while running; such as fuel oil, lube oil, coolant, or exhaust gas leakage. Enhance the Fire Protection Program to specify that the IP2 and IP3 diesel fire pump engine carbon steel exhaust components are inspected for evidence of corrosion and cracking at least once each operating cycle. Enhance the Fire Protection Program for IP3 to visually inspect the cable spreading room, 480V switchgear room, and EDG room CO_2 fire suppression system for signs of degradation, such as corrosion and mechanical damage at least once every six months.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.12 A.3.1.12 B.1.13

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
8	Enhance the Fire Water Program to include inspection of IP2 and IP3 hose reels for evidence of corrosion. Acceptance criteria will be revised to verify no unacceptable signs of degradation. Enhance the Fire Water Program to replace all or test a sample of IP2 and IP3 sprinkler heads required for 10 CFR 50.48 using guidance of NFPA 25 (2002 edition), Section 5.3.1.1.1 before the end of the 50-year sprinkler head service life and at 10-year intervals thereafter during the extended period of operation to ensure that signs of degradation, such as corrosion, are detected in a timely manner. Enhance the Fire Water Program to perform wall thickness evaluations of IP2 and IP3 fire protection piping on system components using non-intrusive techniques (e.g., volumetric testing) to identify evidence of loss of material due to corrosion. These inspections will be performed before the end of the current operating term and at intervals thereafter during the period of extended operation. Results of the initial evaluations will be used to determine the appropriate inspection interval to ensure aging effects are identified prior to loss of intended function. Enhance the Fire Water Program to inspect the internal surface of foam based fire suppression tanks. Acceptance criteria will be enhanced to verify no significant corrosion.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153 NL-08-014	A.2.1.13 A.3.1.13 B.1.14 Audit Items 105, 106

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
9	Enhance the Flux Thimble Tube Inspection Program for IP2 and IP3 to implement comparisons to wear rates identified in WCAP-12866. Include provisions to compare data to the previous performances and perform evaluations regarding change to test frequency and scope. Enhance the Flux Thimble Tube Inspection Program for IP2 and IP3 to specify the acceptance criteria as outlined in WCAP-12866 or other plant-specific values based on evaluation of previous test results. Enhance the Flux Thimble Tube Inspection Program for IP2 and IP3 to direct evaluation and performance of corrective actions based on tubes that exceed or are projected to exceed the acceptance criteria. Also stipulate that flux thimble tubes that cannot be inspected over the tube length and cannot be shown by analysis to be satisfactory for continued service, must be removed from service to ensure the integrity of the reactor coolant system pressure boundary.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.15 A.3.1.15 B.1.16

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
10	Enhance the Heat Exchanger Monitoring Program for IP2 and IP3 to include the following heat exchangers in the scope of the program. • Safety injection pump lube oil heat exchangers • RHR heat exchangers • RHR pump seal coolers • Non-regenerative heat exchangers • Charging pump seal water heat exchangers • Charging pump fluid drive coolers • Charging pump crankcase oil coolers • Spent fuel pit heat exchangers • Secondary system steam generator sample coolers • Waste gas compressor heat exchangers • SBO/Appendix R diesel jacket water heat exchanger (IP2 only) Enhance the Heat Exchanger Monitoring Program for IP2 and IP3 to perform visual inspection on heat exchangers where non-destructive examination, such as eddy current inspection, is not possible due to heat exchanger design limitations. Enhance the Heat Exchanger Monitoring Program for IP2 and IP3 to include consideration of material-environment combinations when determining sample population of heat exchangers. Enhance the Heat Exchanger Monitoring Program for IP2 and IP3 to establish minimum tube wall thickness for the new heat exchangers identified in the scope of the program. Establish acceptance criteria for heat exchangers visually inspected to include no unacceptable signs of degradation.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.16 A.3.1.16 B.1.17, Audit Item 52

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
11	Enhance the ISI Program for IP2 and IP3 to provide periodic visual inspections to confirm the absence of aging effects for lubrite sliding supports used in the steam generator and reactor coolant pump support systems.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.17 A.3.1.17 B.1.18 Audit item 59
12	Enhance the Masonry Wall Program for IP2 and IP3 to specify that the IP1 intake structure is included in the program.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.18 A.3.1.18 B.1.19
13	Enhance the Metal-Enclosed Bus Inspection Program to add IP2 480V bus associated with substation A to the scope of bus inspected. Enhance the Metal-Enclosed Bus Inspection Program for IP2 and IP3 to visually inspect the external surface of MEB enclosure assemblies for loss of material at least once every 10 years. The first inspection will occur prior to the period of extended operation and the acceptance criterion will be no significant loss of material. Enhance the Metal-Enclosed Bus Inspection Program for IP2 and IP3 to inspect bolted connections at least once every five years if performed visually or at least once every ten years using quantitative measurements such as thermography or contact resistance measurements. The first inspection will occur prior to the period of extended operation. The plant will process a change to applicable site procedure to remove the reference to "re-torquing" connections for phase bus maintenance and bolted connection maintenance.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.19 A.3.1.19 B.1.20 Audit Item 124 Audit Item 133
14	Implement the Non-EQ Bolted Cable Connections Program for IP2 and IP3 as described in LRA Section B.1.22.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.21 A.3.1.21 B.1.22

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
15	Implement the Non-EQ Inaccessible Medium-Voltage Cable Program for IP2 and IP3 as described in LRA Section B.1.23. This new program will be implemented consistent with the corresponding program described in NUREG-1801 Section XI.E3, Inaccessible Medium-Voltage Cables Not Subject To 10 CFR 50.49 Environmental Qualification Requirements.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.22 A.3.1.22 B.1.23 Audit item 173
16	Implement the Non-EQ Instrumentation Circuits Test Review Program for IP2 and IP3 as described in LRA Section B.1.24. This new program will be implemented consistent with the corresponding program described in NUREG-1801 Section XI.E2, Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.23 A.3.1.23 B.1.24 Audit item 173
17	Implement the Non-EQ Insulated Cables and Connections Program for IP2 and IP3 as described in LRA Section B.1.25. This new program will be implemented consistent with the corresponding program described in NUREG-1801 Section XI.E1, Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.24 A.3.1.24 B.1.25 Audit item 173

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
18	Enhance the Oil Analysis Program for IP2 to sample and analyze lubricating oil used in the SBO/Appendix R diesel generator consistent with oil analysis for other site diesel generators. Enhance the Oil Analysis Program for IP2 and IP3 to sample and analyze generator seal oil and turbine hydraulic control oil. Enhance the Oil Analysis Program for IP2 and IP3 to formalize preliminary oil screening for water and particulates and laboratory analyses including defined acceptance criteria for all components included in the scope of this program. The program will specify corrective actions in the event acceptance criteria are not met. Enhance the Oil Analysis Program for IP2 and IP3 to formalize trending of preliminary oil screening results as well as data provided from independent laboratories.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.25 A.3.1.25 B.1.26
19	Implement the One-Time Inspection Program for IP2 and IP3 as described in LRA Section B.1.27. This new program will be implemented consistent with the corresponding program described in NUREG-1801, Section XI.M32, One-Time Inspection.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.26 A.3.1.26 B.1.27 Audit item 173
20	Implement the One-Time Inspection – Small Bore Piping Program for IP2 and IP3 as described in LRA Section B.1.28. This new program will be implemented consistent with the corresponding program described in NUREG-1801, Section XI.M35, One-Time Inspection of ASME Code Class I Small-Bore Piping.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.27 A.3.1.27 B.1.28 Audit item 173
21	Enhance the Periodic Surveillance and Preventive Maintenance Program for IP2 and IP3 as necessary to assure that the effects of aging will be managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis through the period of extended operation.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.28 A.3.1.28 B.1.29

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
22	Enhance the Reactor Vessel Surveillance Program for IP2 and IP3 revising the specimen capsule withdrawal schedules to draw and test a standby capsule to cover the peak reactor vessel fluence expected through the end of the period of extended operation. Enhance the Reactor Vessel Surveillance Program for IP2 and IP3 to require that tested and untested specimens from all capsules pulled from the reactor vessel are maintained in storage.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.31 A.3.1.31 B.1.32
23	Implement the Selective Leaching Program for IP2 and IP3 as described in LRA Section B.1.33. This new program will be implemented consistent with the corresponding program described in NUREG-1801, Section XI.M33 Selective Leaching of Materials.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.32 A.3.1.32 B.1.33 Audit item 173
24	Enhance the Steam Generator Integrity Program for IP2 and IP3 to require that the results of the condition monitoring assessment are compared to the operational assessment performed for the prior operating cycle with differences evaluated.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.34 A.3.1.34 B.1.35
25	Enhance the Structures Monitoring Program to explicitly specify that the following structures are included in the program. • Appendix R diesel generator foundation (IP3) • Appendix R diesel generator fuel oil tank vault (IP3) • Appendix R diesel generator switchgear and enclosure (IP3) • city water storage tank foundation • condensate storage tanks foundation (IP3) • containment access facility and annex (IP3) • discharge canal (IP2/3) • emergency lighting poles and foundations (IP2/3) • fire pumphouse (IP2) • fire protection pumphouse (IP3) • fire water storage tank foundations (IP2/3) • gas turbine 1 fuel storage tank foundation • maintenance and outage building-elevated passageway (IP2) • new station security building (IP2)	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.35 A.3.1.35 B.1.36 Audit item 86 Audit item 88 Audit Item 87

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
	• nuclear service building (IP1) • primary water storage tank foundation (IP3) • refueling water storage tank foundation (IP3) • security access and office building (IP3) • service water pipe chase (IP2/3) • service water valve pit (IP3) • superheater stack • transformer/switchyard support structures (IP2) • waste holdup tank pits (IP2/3) Enhance the Structures Monitoring Program for IP2 and IP3 to clarify that in addition to structural steel and concrete, the following commodities (including their anchorages) are inspected for each structure as applicable. • cable trays and supports • concrete portion of reactor vessel supports • conduits and supports • cranes, rails and girders • equipment pads and foundations • fire proofing (pyrocrete) • HVAC duct supports • jib cranes • manholes and duct banks • manways, hatches and hatch covers • monorails • new fuel storage racks • sumps, sump screens, strainers and flow barriers Enhance the Structures Monitoring Program for IP2 and IP3 to inspect inaccessible concrete areas that are exposed by excavation for any reason. IP2 and IP3 will also inspect inaccessible concrete areas in environments where observed conditions in accessible areas exposed to the same environment indicate that significant concrete degradation is occurring. Enhance the Structures Monitoring Program for IP2 and IP3 to perform inspections of elastomers (seals, gaskets, seismic joint filler, and roof elastomers) to identify cracking and change in material properties and for inspection of aluminum vents and louvers to identify loss of material.			

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
	Enhance the Structures Monitoring Program for IP2 and IP3 to perform an engineering evaluation of groundwater samples to assess aggressiveness of groundwater to concrete on a periodic basis (at least once every five years). IPEC will obtain samples from at least 5 wells that are representative of the ground water surrounding below-grade site structures. Samples will be monitored for sulfates, pH and chlorides. Enhance the Structures Monitoring Program for IP2 and IP3 to perform inspection of normally submerged concrete portions of the intake structures at least once every 5 years.			
26	Implement the Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program for IP2 and IP3 as described in LRA Section B.1.37. This new program will be implemented consistent with the corresponding program described in NUREG-1801, Section XI.M12, Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.36 A.3.1.36 B.1.37 Audit item 173
27	Implement the Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program for IP2 and IP3 as described in LRA Section B.1.38. This new program will be implemented consistent with the corresponding program described in NUREG-1801 Section XI.M13, Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steel (CASS) Program.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.37 A.3.1.37 B.1.38 Audit item 173
28	Enhance the Water Chemistry Control – Closed Cooling Water Program to maintain water chemistry of the IP2 SBO/Appendix R diesel generator cooling system per EPRI guidelines. Enhance the Water Chemistry Control – Closed Cooling Water Program to maintain the IP2 and IP3 security generator cooling water system pH within limits specified by EPRI guidelines.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.39 A.3.1.39 B.1.40

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
29	Enhance the Water Chemistry Control – Primary and Secondary Program for IP2 to test sulfates monthly in the RWST with a limit of <150 ppb.	IP2: September 28, 2013	NL-07-039	A.2.1.40 B.1.41
30	For aging management of the reactor vessel internals, IPEC will (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	IP2: September 28, 2011 IP3: December 12, 2013	NL-07-039	A.2.1.41 A.3.1.41
31	Additional P-T curves will be submitted as required per 10 CFR 50, Appendix G prior to the period of extended operation as part of the Reactor Vessel Surveillance Program.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.2.1.2 A.3.2.1.2 4.2.3
32	As required by 10 CFR 50.61(b)(4), IP3 will submit a plant-specific safety analysis for plate B2803-3 to the NRC three years prior to reaching the RT_{PTS} screening criterion. Alternatively, the site may choose to implement the revised PTS (10 CFR 50.61) rule when approved, which would permit use of Regulatory Guide 1.99, Revision 3.	IP3: December 12, 2015	NL-07-039	A.3.2.1.4 4.2.5

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
33	At least 2 years prior to entering the period of extended operation, for the locations identified in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3), under the Fatigue Monitoring Program, IP2 and IP3 will implement one or more of the following: (1) Consistent with the Fatigue Monitoring Program, Detection of Aging Effects, update the fatigue usage calculations using ~~R~~refined ~~the~~ fatigue analyses to determine valid CUFs less than 1.0 when accounting for the effects of reactor water environment. This includes applying the appropriate Fen factors to valid CUFs determined in accordance with one of the following: 1. For locations in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3) ~~including NUREG/CR-6260 locations~~, with existing fatigue analysis valid for the period of extended operation, use the existing CUF. ~~to determine the environmentally adjusted CUF.~~ 2. ~~In addition to the NUREG/CR-6260 locations, more limiting~~ Additional plant-specific locations with a valid CUF may be evaluated. In particular, the pressurizer lower shell will be reviewed to ensure the surge nozzle remains the limiting component. 3. Representative CUF values from other plants, adjusted to or enveloping the IPEC plant specific external loads may be used if demonstrated applicable to IPEC. 4. An analysis using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case) may be performed to determine a valid CUF. ~~(2) Manage the effects of aging due to fatigue at the affected locations 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 acceptable to the NRC).~~ ~~(3)~~ (2) Consistent with the Fatigue Monitoring Program, Corrective Actions, ~~R~~repair or replace the affected locations before exceeding a CUF of 1.0. ~~Should IPEC select the option to manage the aging effects due to environmental assisted fatigue during the period of extended operation, details of the aging management program such as scope, qualification, method, and frequency will be submitted to the NRC at least 2 years prior to the period of extended operation.~~	IP2: September 28, 2011 IP3: December 12, 2013	NL-07-039 NL-07-153 NL-08-021	A.2.2.2.3 A.3.2.2.3 4.3.3 Audit item 146

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
34	IP2 SBO / Appendix R diesel generator will be installed and operational by April 30, 2008. This committed change to the facility meets the requirements of 10 CFR 50.59(c)(1) and, therefore, a license amendment pursuant to 10 CFR 50.90 is not required.	April 30, 2008	NL-07-078	2.1.1.3.5

NYS-26/26A & RIVERKEEPER TC-1/1A: ATTACHMENT 11

Entergy

Entergy Nuclear Northeast
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Fred Dacimo
Vice President
License Renewal

March 24, 2008

Re: Indian Point Units 2 & 3
Docket Nos. 50-247 & 50-286
NL-08-057

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

SUBJECT: Entergy Nuclear Operations Inc.
Indian Point Nuclear Generating Unit Nos. 2 & 3
Docket Nos. 50-247 and 50-286
Amendment 3 to License Renewal Application (LRA)

REFERENCES:

1. Entergy Letter dated April 23, 2007, F. R. Dacimo to Document Control Desk, "License Renewal Application" (NL-07-039)
2. Entergy Letter dated April 23, 2007, F. R. Dacimo to Document Control Desk, "License Renewal Application Boundary Drawings (NL-07-040)
3. Entergy Letter dated April 23, 2007, F. R. Dacimo to Document Control Desk, "License Renewal Application Environmental Report References (NL-07-041)
4. Entergy Letter dated October 11, 2007, F. R, Dacimo to Document Control Desk, "License Renewal Application (LRA)" (NL-07-124)
5. Entergy Letter November 14, 2007, F. R, Dacimo to Document Control Desk, "Supplement to License Renewal Application (LRA) Environmental Report References" (NL-07-133)

Dear Sir or Madam:

In the referenced letters, Entergy Nuclear Operations, Inc. applied for renewal of the Indian Point Energy Center operating license.

This letter contains Amendment 3 of the License Renewal Application (LRA), which consists of five attachments. Attachment 1 consists of an amendment to the LRA to address Regional Inspection items. Attachment 2 consists of an amendment to address Audit Time Limited Aging

A128
NRR

Analyses (TLAA) and other LRA amendment items. Attachment 3 consists of a revision to the list of regulatory commitments associated with the LRA. Attachment 4 provides the responses to the questions raised by the NRC team during the TLAA portion of the LRA. Attachment 5 provides the responses to the questions raised by the NRC team during the Aging Management Programs (AMP) portion of the LRA.

If you have any questions, or require additional information, please contact Mr. Robert Walpole at 914-734-6710.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 3/24/08.

Sincerely,

Fred R. Dacimo
Vice President
License Renewal

Attachments:

1. Regional Inspection LRA Amendment
2. Audit TLAA and other LRA Amendment
3. IPEC LRA List of Regulatory Commitments, Revision 4
4. TLAA Audit Database Report
5. AMP Audit Database Report

cc: Mr. Samuel J. Collins, Regional Administrator, NRC Region I
Mr. Sherwin E. Turk, NRC Office of General Counsel, Special Counsel
Mr. Kenneth Chang, NRC Branch Chief, Engineering Review Branch I
Mr. Bo M. Pham, NRC Environmental Project Manager
Mr. John Boska, NRR Senior Project Manager
Mr. Paul Eddy, New York State Department of Public Service
NRC Resident Inspector's Office
Mr. Paul D. Tonko, President, New York State Energy, Research, & Development Authority

ATTACHMENT 3 TO NL-08-057

IPEC LRA List of Regulatory Commitments, Revision 4

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 & 3
DOCKET NOS. 50-247 AND 50-286

33	At least 2 years prior to entering the period of extended operation, for the locations identified in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3), under the Fatigue Monitoring Program, IP2 and IP3 will implement one or more of the following: (1) Consistent with the Fatigue Monitoring Program, Detection of Aging Effects, update the fatigue usage calculations using refined fatigue analyses to determine valid CUFs less than 1.0 when accounting for the effects of reactor water environment. This includes applying the appropriate Fen factors to valid CUFs determined in accordance with one of the following: 1. For locations in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3), with existing fatigue analysis valid for the period of extended operation, use the existing CUF. 2. Additional plant-specific locations with a valid CUF may be evaluated. In particular, the pressurizer lower shell will be reviewed to ensure the surge nozzle remains the limiting component. 3. Representative CUF values from other plants, adjusted to or enveloping the IPEC plant specific external loads may be used if demonstrated applicable to IPEC. 4. An analysis using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case) may be performed to determine a valid CUF. (2) Consistent with the Fatigue Monitoring Program, Corrective Actions, repair or replace the affected locations before exceeding a CUF of 1.0.	IP2: September 28, 2011 IP3: December 12, 2013	NL-07-039 NL-07-153 NL-08-021	A.2.2.2.3 A.3.2.2.3 4.3.3 Audit item 146
34	IP2 SBO / Appendix R diesel generator will be installed and operational by April 30, 2008. This committed change to the facility meets the requirements of 10 CFR 50.59(c)(1) and, therefore, a license amendment pursuant to 10 CFR 50.90 is not required.	April 30, 2008	NL-07-078	2.1.1.3.5

NYS-26/26A & RIVERKEEPER TC-1/1A: ATTACHMENT 12

Entergy

Entergy Nuclear Northeast
Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
Buchanan, NY 10511-0249
Tel (914) 788-2055

Fred R. Dacimo
Vice President
License Renewal

May 16, 2008

Re: Indian Point Units 2 & 3
Docket Nos. 50-247 & 50-286

NL-08-084

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

SUBJECT: **Reply to Request for Additional Information Regarding License Renewal Application – Time-Limited Aging Analyses and Boraflex**

Reference: NRC letter dated April 18, 2008; "Request for Additional Information for the Review of the Indian Point Nuclear Generating Unit Nos. 2 and 3, License Renewal Application – Time-Limited Aging Analyses and Boraflex"

Dear Sir or Madam:

Entergy Nuclear Operations, Inc is providing, in Attachment I, the additional information requested in the referenced letter pertaining to NRC review of the License Renewal Application for Indian Point 2 and Indian Point 3. The additional information provided in this transmittal addresses staff questions for Time-Limited Aging Analyses and Boraflex.

There are no new commitments identified in this submittal. If you have any questions or require additional information, please contact Mr. R. Walpole, Manager, Licensing at (914) 734-6710.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 5-16-08.

Sincerely,

Patrick W. Conroy for
Fred R. Dacimo
Vice President
License Renewal

A128
NRR

Attachment:

1. Reply to NRC Request for Additional Information Regarding License Renewal Application – Time-Limited Aging Analyses and Boraflex

cc: Mr. Bo M. Pham, NRC Environmental Project Manager
Ms. Kimberly Green, NRC Safety Project Manager
Mr. John P. Boska, NRC NRR Senior Project Manager
Mr. Samuel J. Collins, Regional Administrator, NRC Region I
Mr. Sherwin E. Turk, NRC Office of General Counsel, Special Counsel
IPEC NRC Senior Resident Inspectors Office
Mr. Paul D. Tonko, President, NYSERDA
Mr. Paul Eddy, New York State Dept. of Public Service

ATTACHMENT I TO NL-08-084

REPLY TO NRC REQUEST FOR ADDITIONAL INFORMATION

REGARDING

LICENSE RENEWAL APPLICATION

Time-Limited Aging Analyses and Boraflex

ENTERGY NUCLEAR OPERATIONS, INC
INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 and 3
DOCKETS 50-247 and 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3
LICENSE RENEWAL APPLICATION (LRA)
REQUESTS FOR ADDITIONAL INFORMATION (RAI)
REGARDING TIME-LIMITED AGING ANALYSES AND BORAFLEX

Time-Limited Aging Analyses

RAI 4.3.1.8-1

License renewal application (LRA) Section 4.3.1 states "[c]urrent design basis fatigue evaluations calculate cumulative usage factors (CUFs) for components or sub-components based on design transient cycles." For CUF values listed in LRA Tables 4.3-13 and 4.3-14, please describe the details of how various environmental effects are factored into the calculation of the CUF using F_{en} values.

Response to RAI 4.3.1.8-1

For CUF values listed in LRA Tables 4.3-13 and 4.3-14, the F_{en} values were determined as described below.

NUREG-1801 calls for using formulas provided in NUREG/CR-5704 for austenitic stainless steel and NUREG/CR-6583 for carbon steel and low-alloy steel to calculate environmentally assisted fatigue correction factors (F_{en}). For IPEC, none of the locations identified in Tables 4.3-13 and 4.3-14 (NUREG/CR-6260 locations) are made of carbon steel, so calculation of F_{en} for carbon steel was not required.

The environmentally assisted fatigue correction factor (F_{en}) for **low alloy steel** was calculated as follows.

F_{en} =	$\exp(0.929-0.00124T-0.101\, S^*T^*O^*\varepsilon^*)$		based on NUREG/CR-6583, Eq. 6.5b
T =	25°C	Reference temperature for original fatigue curves	
S^* =	S	(0 < S (Sulfur) ≤ 0.015 wt.%)	
S^* =	0.015	(S ≥ 0.015 wt.%)	NUREG/CR-6583, Eq. 5.5a
T^* =	0	(T (Temperature) < 150°C)	
T^* =	T-150	(T = 150-350°C)	NUREG/CR-6583, Eq. 5.5b
O^* =	0	(DO (Dissolved Oxygen) < 0.05ppm)	
O^* =	ln(DO/0.04)	(0.05 ppm ≤ DO ≤ 0.5 ppm)	
O^* =	ln(12.5) = 2.53	(DO > 0.5 ppm)	NUREG/CR-6583, Eq. 5.5c
$\dot{\varepsilon}^*$ =	0	($\dot{\varepsilon}$ (strain rate) > 1%/s)	
$\dot{\varepsilon}^*$ =	ln($\dot{\varepsilon}$)	(0.001 ≤ $\dot{\varepsilon}$ ≤ 1%/s)	
$\dot{\varepsilon}^*$ =	ln(0.001)	($\dot{\varepsilon}$ < 0.001%/s)	NUREG/CR-6583, Eq. 5.5d

There are four low alloy steel subcomponents for each unit in Tables 4.3-13 and 4.3-14 for the NUREG-6260 locations at IPEC. The F_{en} was calculated for each location on each unit as shown below.

IP2

$T_{(reference\ temperature\ °C)}$ = 25 Reference temperature for original fatigue curves

$O^{*}_{(RCS)}$ = 0.0 RCS dissolved oxygen is ≤ 50 ppb

Since $O^{*}_{(RCS)}$ equals 0.0, S*, T*, and $\dot{\varepsilon}$ terms are eliminated.

F_{en} = exp(0.929-(0.00124)(T))

$F_{en\ (bottom\ head\ to\ shell)}$ = exp(0.929-(0.00124)(25)) = 2.45
$F_{en\ (inlet\ nozzles)}$ = exp(0.929-(0.00124)(25)) = 2.45
$F_{en\ (outlet\ nozzles)}$ = exp(0.929-(0.00124)(25)) = 2.45
$F_{en\ (surge\ line\ nozzles)}$ = exp(0.929-(0.00124)(25)) = 2.45

IP3

$T_{(reference\ temperature\ °C)}$ = 25 Reference temperature for original fatigue curves

Since $O^{*}_{(RCS)}$ equals 0.0, S*, T*, and $\dot{\varepsilon}$ terms are eliminated.

F_{en} = exp(0.929-(0.00124)(T))
$F_{en\ (bottom\ head\ to\ shell)}$ = exp(0.929-(0.00124)(25)) = 2.45
$F_{en\ (inlet\ nozzles)}$ = exp(0.929-(0.00124)(25)) = 2.45
$F_{en\ (outlet\ nozzles)}$ = exp(0.929-(0.00124)(25)) = 2.45
$F_{en\ (surge\ line\ nozzles)}$ = exp(0.929-(0.00124)(25)) = 2.45

The environmentally assisted fatigue correction factor (F_{en}) for **austenitic stainless steel** was calculated as follows.

F_{en} = exp(0.935-T′O′ε′)		NUREG/CR-5704, Eq. 13
T′ = 0	(T < 200ºC)	
T′ = 1	(T ≥ 200ºC)	NUREG/CR-5704, Eq. 8a
O′ = 0.260	(DO < 0.05 ppm)	
O′ = 0.172	(DO ≥ 0.05 ppm)	NUREG/CR-5704, Eq. 8b
$\dot{\varepsilon}'$ = 0	($\dot{\varepsilon}$ > 0.4%/s)	
$\dot{\varepsilon}'$ = ln($\dot{\varepsilon}$ /0.4)	(0.0004 ≤ $\dot{\varepsilon}$ ≤ 0.4%/s)	
$\dot{\varepsilon}'$ = ln(0.0004/0.4)	($\dot{\varepsilon}$ < 0.0004%/s)	NUREG/CR-5704, Eq. 8c

There are four stainless steel subcomponents for each unit in Tables 4.3-13 and 4.3-14 for the NUREG-6260 locations at IPEC. The F_{en} will be calculated for each location on each unit as shown below.

IP2

$T'_{(Surge\ line)}$	= 1.0	(T ≥ 200ºC)
$T'_{(Charging\ nozzle)}$	= 1.0	(T ≥ 200ºC)
$T'_{(SI\ nozzle)}$	= 1.0	(T ≥ 200ºC)
$T'_{(RHR\ piping)}$	= 1.0	(T ≥ 200ºC)

$O'_{(All)}$	= 0.260	RCS dissolved oxygen is ≤ 50 ppb
$\dot{\varepsilon}'_{(all)}$	= -6.91	Assume bounding strain rate
$F_{en\ (Surge\ line)}$	= exp(0.935-(1.0)(0.260)(-6.91)) = 15.35	
$F_{en\ (Charging\ nozzle)}$	= exp(0.935-(1.0)(0.260)(-6.91)) = 15.35	
$F_{en\ (SI\ nozzle)}$	= exp(0.935-(1.0)(0.260)(-6.91)) = 15.35	
$F_{en\ (RHR\ piping)}$	= exp(0.935-(1.0)(0.260)(-6.91)) = 15.35	

IP3

$T'_{(Surge\ line)}$	= 1.0	(T ≥ 200ºC)
$T'_{(Charging\ nozzle)}$	= 1.0	(T ≥ 200ºC)
$T'_{(SI\ nozzle)}$	= 1.0	(T ≥ 200ºC)
$T'_{(RHR\ piping)}$	= 1.0	(T ≥ 200ºC)
$O'_{(All)}$	= 0.260	RCS dissolved oxygen is ≤ 50 ppb
$\dot{\varepsilon}'_{(all)}$	= -6.91	Assume bounding strain rate
$F_{en\ (Surge\ line)}$	= exp(0.935-(1.0)(0.260)(-6.91)) = 15.35	
$F_{en\ (Charging\ nozzle)}$	= exp(0.935-(1.0)(0.260)(-6.91)) = 15.35	
$F_{en\ (SI\ nozzle)}$	= exp(0.935-(1.0)(0.260)(-6.91)) = 15.35	
$F_{en\ (RHR\ piping)}$	= exp(0.935-(1.0)(0.260)(-6.91)) = 15.35	

RAI 4.3.1.8-2

Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants (SRP-LR) Section 4.3.2.1.1.3 provides the basis for the staff acceptance of an aging management program to address environmental fatigue. It states, "[t]he staff has evaluated a program for monitoring and tracking the number of critical thermal and pressure transients for the selected reactor coolant system components. The staff has determined that this program is an acceptable aging management program to address metal fatigue of the reactor coolant system components according to 10 CFR 54.21(c)(1)(iii)." The staff is unable to determine if the Fatigue Monitoring Program of Indian Point 2 and Indian Point 3 contain sufficient details to satisfy this criterion. Please provide adequate details of the Fatigue Monitoring Program such that the staff can make a determination based on the criterion set forth in SRP-LR Section 4.3.2.1.1.3. Also, please explain in detail the corrective actions and the frequency that such actions will be taken so the acceptance criteria will not be exceeded in the period of extended operation.

Response to RAI 4.3.1.8-2

The IPEC Fatigue Monitoring Program was compared to the program described in NUREG-1801 (GALL), Section X.M1, Metal Fatigue of Reactor Coolant Pressure Boundary. The program description in the GALL report is directly applicable to the IPEC units. As indicated in LRA Section B.1.12, during the period of extended operation the IPEC program will be consistent with the GALL program with one exception. The exception to GALL is that rather than performing periodic updates of CUF calculations, IPEC periodically assesses the number of transient cycles compared to calculation assumptions and updates the CUF calculations, if

necessary. Based on this comparison including evaluation of the exception, the approvals set forth in the GALL report apply to the IPEC Fatigue Monitoring Program.

The program description was modified per letter NL-08-21, Indian Point Nuclear Generating Units Nos. 2 & 3 License Renewal Application Amendment 2, dated January 22, 2008. This letter commits to complete CUF calculations for all areas identified in NUREG-6260 (LRA Table 4.3-13 for IP2 and LRA Table 4.3-14 for IP3), incorporating the effect of the reactor coolant environment, for IP2 and IP3. Once these CUF calculations are complete (at least 2 years prior to the period of extended operation), IPEC will ensure that the cycles analyzed in the new or updated calculations are included in the Fatigue Monitoring Program. IPEC will continue to manage the effects of fatigue throughout the period of extended operation by monitoring cycles incurred and assuring they do not exceed the analyzed numbers of cycles.

As required by IPEC technical specifications, the Fatigue Monitoring Program tracks actual plant transients and evaluates these against the design transients. The plant transient counts are updated at least once each operating cycle. This frequency is acceptable since the evaluation during each update determines if the number of design transients could be exceeded prior to the next update. The Fatigue Monitoring Program ensures that the numbers of transient cycles experienced by the plant remain within the analyzed numbers of cycles and hence, the component CUF calculations remain valid.

The program requires corrective actions before exceeding the analyzed number of transient cycles. The corrective actions are implemented in accordance with the IPEC corrective action program. IPEC may perform further reanalysis if cycle counts approach analyzed numbers. These calculation updates will be governed by Entergy's 10 CFR 50 Appendix B Quality Assurance (QA) program and include design input verification and independent reviews ensuring that valid assumptions, transients, cycles, external loadings, analysis methods, and environmental fatigue life correction factors will be used in the fatigue analyses. Repair or replacement of the affected component(s), if necessary, will be done prior to exceeding the allowable CUF in accordance with established plant procedures governing repair and replacement activities. These established procedures are governed by Entergy's 10 CFR 50 Appendix B QA program and meet the applicable repair or replacement requirements of the ASME Code Section XI.

RAI 3.0.3.2.3-1

Indian Point 2 Updated Final Safety Analysis Report, Revision 20, dated 2006, Section 14.2.1 on page 55 of 218, states in part that:

> "Northeast Technology Corporation report NET-173-01 and NET-173-02 are based on conservative projections of amount of boraflex absorber panel degradation assumed in each sub-region. These projections are valid through the end of the year 2006."

Please confirm that the Boraflex neutron absorber panels in the Indian Point Unit 2 spent fuel pool have been re-evaluated for service through the end of the current licensing period. Also, please discuss the plans for updating the Boraflex analysis during the period of extended operation.

Response to RAI 3.0.3.2.3-1

Boron-10 areal density gage for evaluating racks (BADGER) testing was performed in February 2000, July 2003, and again in July 2006. Using the latest test data and RACKLIFE code projections, the Boraflex neutron absorber panels in the Indian Point Unit 2 spent fuel pool will meet the Technical Specification requirements through the end of the current licensing period. The next BADGER test will be performed prior to the end of calendar year 2009. As required by the Boraflex Monitoring Program (LRA Section B.1.3), periodic BADGER testing and RACKLIFE code projections will continue through the period of extended operation to confirm acceptable Boraflex condition.

The referenced section of the Indian Point 2 Updated Final Safety Analysis Report will be updated in the next revision.

NYS-26/26A & RIVERKEEPER TC-1/1A: ATTACHMENT 13

Entergy

Entergy Nuclear Northeast
Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
Buchanan, NY 10511-0249
Tel (914) 788-2055

Fred Dacimo
Vice President
License Renewal

June 11, 2008

Re: Indian Point Units 2 & 3
Docket Nos. 50-247 & 50-286
NL-08-092

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

SUBJECT: Entergy Nuclear Operations Inc.
Indian Point Nuclear Generating Unit Nos. 2 & 3
Docket Nos. 50-247 and 50-286
Amendment 5 to License Renewal Application (LRA)

REFERENCES:

1. Entergy Letter dated April 23, 2007, F. R. Dacimo to Document Control Desk, "License Renewal Application" (NL-07-039)
2. Entergy Letter dated April 23, 2007, F. R. Dacimo to Document Control Desk, "License Renewal Application Boundary Drawings (NL-07-040)
3. Entergy Letter dated April 23, 2007, F. R. Dacimo to Document Control Desk, "License Renewal Application Environmental Report References (NL-07-041)
4. Entergy Letter dated October 11, 2007, F. R, Dacimo to Document Control Desk, "License Renewal Application (LRA)" (NL-07-124)
5. Entergy Letter November 14, 2007, F. R, Dacimo to Document Control Desk, "Supplement to License Renewal Application (LRA) Environmental Report References" (NL-07-133)

Dear Sir or Madam:

In the referenced letters, Entergy Nuclear Operations, Inc. applied for renewal of the Indian Point Energy Center operating license.

This letter contains Amendment 5 of the License Renewal Application (LRA) which consists of three attachments. Attachment 1 consists of the annual update amendment to the LRA.

A128
NRR

Attachment 2 consists of an amendment for (a)(2) clarification. Attachment 3 consists of an amendment for reactor vessel clarification.

If you have any questions, or require additional information, please contact Mr. Robert Walpole at 914-734-6710.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 6/11/08.

Sincerely,

Fred R. Dacimo
Vice President
License Renewal

Attachments:

1. Annual Update Amendment
2. (A)(2) Clarification Amendment
3. Reactor Vessel Clarification Amendment

cc: Mr. Samuel J. Collins, Regional Administrator, NRC Region I
Mr. Sherwin E. Turk, NRC Office of General Counsel, Special Counsel
Mr. Kenneth Chang, NRC Branch Chief, Engineering Review Branch I
Mr. Bo M. Pham, NRC Environmental Project Manager
Mr. John Boska, NRR Senior Project Manager
Mr. Paul Eddy, New York State Department of Public Service
NRC Resident Inspector's Office
Mr. Paul D. Tonko, President, New York State Energy, Research, & Development Authority

ATTACHMENT 1 TO NL-08-092

Annual Update Amendment

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 & 3
DOCKET NOS. 50-247 AND 50-286

LRA Section A.2.1.17, Inservice Inspection – Inservice Inspection (ISI) Program, third paragraph, is revised as follows.

> On ~~July~~ March 1, 2007~~1994~~, ~~the plant~~ IP2 entered the ~~third~~ fourth ISI interval. The ASME code edition and addenda used for the ~~third~~ fourth interval is the ~~1989~~2001 Edition with ~~no~~2003 addenda.

LRA Section B.1.8, Containment Inservice Inspection, Program Description, is revised as follows.

> The Containment Inservice Inspection (CII) Program is an existing program encompassing ASME Section XI Subsection IWE and IWL requirements as modified by 10 CFR 50.55a. The IP2 program uses the ASME Boiler and Pressure Vessel Code, Section XI, ~~1992~~ 2001 Edition, ~~1992~~ 2003Addenda. The IP3 program uses the ASME Boiler and Pressure Vessel Code, Section XI, 1998 Edition, no Addenda. Every 10 years, each unit's program is updated to the latest ASME Section XI code edition and addenda approved by the Nuclear Regulatory Commission in 10 CFR 50.55a.

LRA Section B.1.18, Inservice Inspection, Program Description, seventh paragraph, is revised as follows.

> On ~~July~~ March 1, 2007~~1994~~, IP2 entered the ~~third~~ fourth ISI interval and on July 21, 2000, IP3 entered the third ISI interval. The ASME code edition and addenda used for the IP2 fourth interval is the 2001 Edition with 2003 addenda. The ASME code edition and addenda used for the IP3 third interval ~~for both units~~ is the 1989 Edition with no addenda.

LRA Section B.1.18, Inservice Inspection, Element 4, eighth paragraph, is revised as follows.

> For ~~both~~ IP2 ~~and IP3~~, Article IWF of ASME Section XI, ~~1989 Edition~~ 2001 Edition and 2003 Addenda, does not contain any specific exemption criteria for component supports. For IP3, ~~C~~components exempt from examination are in accordance with the criteria contained in Code Case N-491-2, Alternate Rules for Examination of Class 1, 2, 3 and MC Component Supports of Light-Water Cooled Power Plants, Section XI, Division 1, IWF-1230.

Additional LRA Clarification

LRA Section 4.3.3, Effects of Reactor Water Environment on Fatigue Life, is revised to delete the tenth paragraph as follows.

> ~~For those locations with CUFs less than 1.0, the TLAA has been projected through the period of extended operation per 10CFR54.21(c)(1)(ii).~~

LRA Section A.2.1.20, Nickel Alloy Inspection Program, last paragraph, is revised as follows. (Refer to RAI 3.0.3.3.5-2 response in letter NL-08-051 dated March 12, 2008)

~~The site will continue to implement commitments associated with (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff accepted industry guidelines.~~

The site commits to comply with future applicable NRC Orders. In addition, IPEC commits to implement applicable (1) Bulletins and Generic Letters associated with nickel alloys and (2) staff-accepted industry guidelines associated with nickel alloys.

LRA Section A.3.1.20, Nickel Alloy Inspection Program, last paragraph, is revised as follows. (Refer to RAI 3.0.3.3.5-2 response in letter NL-08-051 dated March 12, 2008)

~~The site will continue to implement commitments associated with (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff accepted industry guidelines.~~

The site commits to comply with future applicable NRC Orders. In addition, IPEC commits to implement applicable (1) Bulletins and Generic Letters associated with nickel alloys and (2) staff accepted industry guidelines associated with nickel alloys.

LRA Section B.1.21, Nickel Alloy Inspection, Program Description, last paragraph, is revised as follows. (Refer to RAI 3.0.3.3.5-2 response in letter NL-08-051 dated March 12, 2008)

~~IPEC will continue to implement commitments associated with (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff accepted industry guidelines.~~

IPEC commits to comply with future applicable NRC Orders. In addition, IPEC commits to implement applicable (1) Bulletins and Generic Letters associated with nickel alloys and (2) staff accepted industry guidelines associated with nickel alloys.

LRA Section B.1.12, Fatigue Monitoring, Exceptions to NUREG-1801, is revised as follows.

The Fatigue Monitoring Program is consistent with the program described in NUREG-1801, Section X.M1, Metal Fatigue of Reactor Coolant Pressure Boundary~~, with the following exception~~.

~~Attributes Affected~~	~~Exceptions~~
~~4. Detection of Aging Effects~~	~~NUREG-1801 specifies periodic updates of fatigue usage calculations. The IPEC program updates fatigue usage calculations when the number of actual cycles approach the analyzed number of cycles.~~ [1]

~~Exception Notes~~

~~[1] Updates of fatigue usage calculations are not necessary unless the number of accumulated fatigue cycles approaches the number of analyzed design cycles. The IPEC program provides for periodic assessment of the number of accumulated cycles. If any transient approaches its number of analyzed cycles, corrective action is taken which may include update of the fatigue usage calculation.~~

NYS-26/26A & RIVERKEEPER TC-1/1A: ATTACHMENT 14

Entergy

Entergy Nuclear Northeast
Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
Buchanan, N.Y. 10511-0249
Tel (914) 788-2055

Fred Dacimo
Vice President
License Renewal

NL-10-082

August 9, 2010

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

SUBJECT: License Renewal Application – Completion of Commitment #33 Regarding the Fatigue Monitoring Program
Indian Point Nuclear Generating Unit Nos. 2 and 3
Docket Nos. 50-247 and 50-286
License Nos. DPR-26 and DPR-64

REFERENCE

1. Entergy Letter dated April 23, 2007, F. R. Dacimo to Document Control Desk, "License Renewal Application" (NL-07-039)
2. Entergy Letter dated April 23, 2007, F. R. Dacimo to Document Control Desk, "License Renewal Application Boundary Drawings (NL-07-040)
3. Entergy Letter dated April 23, 2007, F. R. Dacimo to Document Control Desk, "License Renewal Application-Environmental Report References (NL-07-041)
4. Entergy Letter dated October 11, 2007, F. R, Dacimo to Document Control Desk, "License Renewal Application (LRA)" (NL-07-124)
5. Entergy Letter November 14, 2007, F. R, Dacimo to Document Control Desk, "Supplement to License Renewal Application (LRA) Environmental Report References" (NL-07-133)

Dear Sir or Madam:

In the referenced letters, Entergy Nuclear Operations, Inc. applied for renewal of the Indian Point Energy Center operating license. This letter contains information supporting the completion of commitment 33 to the License Renewal Application regarding the Fatigue Monitoring Program.

There are no new commitments identified in this submittal. If you have any questions, or require additional information, please contact Mr. Robert Walpole at 914-734-6710.

Sincerely,

FD/mb

Attachments:
1. Environmental Fatigue Evaluations
2. List of Regulatory Commitments

cc: Mr. S. J. Collins, Regional Administrator, NRC Region I
Mr. J. Boska, Senior Project Manager, NRC, NRR, DORL
Mr. Sherwin E. Turk, NRC Office of General Counsel, Special Counsel
Ms. Kimberly Green, NRC Safety Project Manager
NRC Resident Inspectors Office, Indian Point
Mr. Paul Eddy, NYS Dept. of Public Service
Mr. Francis J. Murray, Jr., President and CEO, NYSERDA

ATTACHMENT 1 TO NL-10-082

Environmental Fatigue Evaluations

ENTERGY NUCLEAR OPERATIONS, INC
INDIAN POINT NUCLEAR GENERATING UNITS 2 AND 3
DOCKET NOS. 50-247 & 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3
LICENSE RENEWAL APPLICATION
ENVIRONMENTAL FATIGUE EVALUATIONS

Environmental Fatigue Evaluation for Indian Point Unit 2 and Unit 3

Entergy has applied for renewed operating licenses for Indian Point Nuclear Generating Unit 2 and Unit 3 (IP2 and IP3). In the license renewal application, Entergy committed to address environmentally assisted fatigue. Entergy's commitment, amended by letter, NL-08-021, dated January 22, 2008, reads as follows.

> At least 2 years prior to entering the period of extended operation, for the locations identified in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3), under the Fatigue Monitoring Program, IP2 and IP3 will implement one or more of the following:
>
> (1) Consistent with the Fatigue Monitoring Program, Detection of Aging Effects, update the fatigue usage calculations using refined fatigue analyses to determine valid CUFs less than 1.0 when accounting for the effects of reactor water environment. This includes applying the appropriate Fen factors to valid CUFs determined in accordance with one of the following:
>
> 1. For locations in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3), with existing fatigue analysis valid for the period of extended operation, use the existing CUF.
>
> 2. Additional plant-specific locations with a valid CUF may be evaluated. In particular, the pressurizer lower shell will be reviewed to ensure the surge nozzle remains the limiting component.
>
> 3. Representative CUF values from other plants, adjusted to or enveloping the IPEC plant specific external loads may be used if demonstrated applicable to IPEC.
>
> 4. An analysis using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case) may be performed to determine a valid CUF.
>
> (2) Consistent with the Fatigue Monitoring Program, Corrective Actions, repair or replace the affected locations before exceeding a CUF of 1.0.

Entergy has updated the fatigue usage calculations using refined fatigue analyses to determine CUFs when accounting for the effects of reactor water environment. This includes applying the appropriate Fen factors to valid CUFs for the locations shown in LRA Table 4.3-13 and LRA Table 4.3-14.
The tables from the LRA are repeated as follows with the updated CUF values inserted.

The results of the refined analyses are shown in the tables as <u>underlined values.</u>

Table 4.3-13
IP2 Cumulative Usage Factors for NUREG/CR-6260 Limiting Locations

NUREG-6260 Generic Location		IP2 Plant-Specific Location	Material Type	CUF of Record	Per NUREG/CR-6583 or NUREG/CR-5704	
					F_{en}	Environmentally Adjusted CUF
1	Vessel shell and lower head	Bottom head to shell	LAS	0.004	2.45	0.01
2	Vessel inlet and outlet nozzles	Reactor vessel inlet nozzle	LAS	0.05	2.45	0.12
2	Vessel inlet and outlet nozzles	Reactor vessel outlet nozzle	LAS	0.281	2.45	0.69
3	Pressurizer surge line nozzles	Pressurizer surge line nozzle	LAS	~~0.264~~ 0.109	~~2.45~~ 1.74	~~0.646~~ 0.188
3	Pressurizer surge line piping	Surge line piping to safe end weld	SS	~~0.6~~ 0.062	~~15.35~~ 13.26	~~9.21~~ 0.822
4	RCS piping charging system nozzle	Charging system nozzle	SS	~~0.99~~ 0.0323	~~15.35~~ 8.7	~~15.20~~ 0.2809
5	RCS piping safety injection nozzle	NA	SS	~~NA~~[2] 0.1083	~~15.35~~ 7.8975	~~NA~~[2] 0.8553
6	RHR Class 1 piping	NA	SS	~~NA~~[2] 0.0721	~~15.35~~ 13.08	~~NA~~[2] 0.9434

Table 4.3-14
IP3 Cumulative Usage Factors for NUREG/CR-6260 Limiting Locations

	NUREG-6260 Location	IP3 Plant-Specific Location	Material Type	CUF of Record	Per NUREG/CR-6583 or NUREG/CR-5704	
					F_{en}	Environmentally Adjusted CUF
1	Vessel shell and lower head	Bottom head to shell	LAS	0.02	2.45	0.05
2	Vessel inlet and outlet nozzles	Reactor vessel inlet nozzle	LAS	0.049	2.45	0.12
2	Vessel inlet and outlet nozzles	Reactor vessel outlet nozzle	LAS	0.259	2.45	0.64
3	Pressurizer surge line nozzles	Pressurizer surge line nozzle	LAS	~~0.9612~~ 0.0903	~~2.45~~ 1.74	~~2.35~~ 0.157
3	Pressurizer surge line piping	Surge line piping to safe end weld	SS	~~0.6~~ 0.0411	~~15.35~~ 14.45	~~9.21~~ 0.594
4	RCS piping charging system nozzle	NA	SS	~~NA~~[2] 0.1812	~~15.35~~ 3.98	~~NA~~[2] 0.722
5	RCS piping safety injection nozzle	NA	SS	~~NA~~[2] 0.1709	~~15.35~~ 5.0117	~~NA~~[2] 0.8565
6	RHR Class 1 piping	NA	SS	~~NA~~[2] 0.1279	~~15.35~~ 7.79	~~NA~~[2] 0.9961

As described in LRA Section 4.3.3 and shown in Tables 4.3-13 and 4.3-14, the CUFs for the reactor vessel locations were not changed. The evaluation documented in the LRA used bounding Fens applied to the CUFs of record for the bottom head to shell region, the reactor vessel inlet nozzle and the reactor vessel outlet nozzle. The tables show the Cumulative Usage Factors are all below 1.0 for these three locations for both units.

The stress and fatigue evaluations for the remaining piping components listed in Table 4.3-13 and 4.3-14 were performed using standard methods of the ASME Code, Section III. Detailed stress models of the surge line hot leg nozzle, pressurizer surge nozzle, reactor coolant piping charging system nozzle, reactor coolant piping safety injection nozzle, and the RHR system class 1 piping locations were prepared. The analyst developed detailed stress history inputs for all transients considered in the evaluation, which were subsequently used to provide detailed inputs used in the EAF evaluations. The ASME Code evaluations were performed for the piping components to reduce conservatism in the analyses or because the current licensing basis (CLB) qualification was to the ANSI B31.1 Power Piping Code, which did not require a fatigue usage factor calculation. The evaluations were limited to the stress qualifications related to the fatigue requirements of the ASME Code. The primary stress qualifications documented in the CLB remain applicable for the components evaluated.

The evaluation of EAF was accomplished through the application of Fen factors, as described in NUREG/CR-5704 for the stainless steels in the pressurizer surge line, the reactor coolant piping charging and safety injection system nozzles, and the RHR system Class 1 piping. In addition, Fen factors for the pressurizer surge nozzle dissimilar metal weld also considered the carbon steel factors in NUREG/CR-6583. The Fen factors are calculated based on detailed inputs described in the applicable NUREG and are directly applied to the ASME Code fatigue results.

The refined fatigue analyses of the Indian Point Unit 2 and Unit 3 piping locations corresponding to the locations identified in NUREG/CR-6260 for older vintage Westinghouse plants demonstrate that cumulative fatigue usage factors including consideration of reactor water environmental effects are below a value of 1.0 for transients postulated for 60 years of operation.

The results of this evaluation resolve commitment number 33 in the NRC Safety Evaluation Report on license renewal for Indian Point Nuclear Generating Unit 2 and Unit 3.

ATTACHMENT 2 TO NL-10-082

List of Regulatory Commitments

ENTERGY NUCLEAR OPERATIONS, INC
INDIAN POINT NUCLEAR GENERATING UNITS 2 AND 3
DOCKET NOS. 50-247 & 50-286

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
5	Enhance the External Surfaces Monitoring Program for IP2 and IP3 to include periodic inspections of systems in scope and subject to aging management review for license renewal in accordance with 10 CFR 54.4(a)(1) and (a)(3). Inspections shall include areas surrounding the subject systems to identify hazards to those systems. Inspections of nearby systems that could impact the subject systems will include SSCs that are in scope and subject to aging management review for license renewal in accordance with 10 CFR 54.4(a)(2).	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.10 A.3.1.10 B.1.11
6	Enhance the Fatigue Monitoring Program for IP2 to monitor steady state cycles and feedwater cycles or perform an evaluation to determine monitoring is not required. Review the number of allowed events and resolve discrepancies between reference documents and monitoring procedures. Enhance the Fatigue Monitoring Program for IP3 to include all the transients identified. Assure all fatigue analysis transients are included with the lowest limiting numbers. Update the number of design transients accumulated to date.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.11 A.3.1.11 B.1.12, Audit Item 164
7	Enhance the Fire Protection Program to inspect external surfaces of the IP3 RCP oil collection systems for loss of material each refueling cycle. Enhance the Fire Protection Program to explicitly state that the IP2 and IP3 diesel fire pump engine sub-systems (including the fuel supply line) shall be observed while the pump is running. Acceptance criteria will be revised to verify that the diesel engine does not exhibit signs of degradation while running; such as fuel oil, lube oil, coolant, or exhaust gas leakage. Enhance the Fire Protection Program to specify that the IP2 and IP3 diesel fire pump engine carbon steel exhaust components are inspected for evidence of corrosion and cracking at least once each operating cycle. Enhance the Fire Protection Program for IP3 to visually inspect the cable spreading room, 480V switchgear room, and EDG room CO_2 fire suppression system for signs of degradation, such as corrosion and mechanical damage at least once every six months.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.12 A.3.1.12 B.1.13

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
33	At least 2 years prior to entering the period of extended operation, for the locations identified in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3), under the Fatigue Monitoring Program, IP2 and IP3 will implement one or more of the following: (1) Consistent with the Fatigue Monitoring Program, Detection of Aging Effects, update the fatigue usage calculations using refined fatigue analyses to determine valid CUFs less than 1.0 when accounting for the effects of reactor water environment. This includes applying the appropriate Fen factors to valid CUFs determined in accordance with one of the following: 1. For locations in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3), with existing fatigue analysis valid for the period of extended operation, use the existing CUF. 2. Additional plant-specific locations with a valid CUF may be evaluated. In particular, the pressurizer lower shell will be reviewed to ensure the surge nozzle remains the limiting component. 3. Representative CUF values from other plants, adjusted to or enveloping the IPEC plant specific external loads may be used if demonstrated applicable to IPEC. 4. An analysis using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case) may be performed to determine a valid CUF. (2) Consistent with the Fatigue Monitoring Program, Corrective Actions, repair or replace the affected locations before exceeding a CUF of 1.0.	IP2: September 28, 2011 IP3: December 12, 2013 <u>Complete</u>	NL-07-039 NL-07-153 NL-08-021 <u>NL-10-082</u>	A.2.2.2.3 A.3.2.2.3 4.3.3 Audit item 146
34	IP2 SBO / Appendix R diesel generator will be installed and operational by April 30, 2008. This committed change to the facility meets the requirements of 10 CFR 50.59(c)(1) and, therefore, a license amendment pursuant to 10 CFR 50.90 is not required.	April 30, 2008 Complete	NL-07-078 NL-08-074	2.1.1.3.5

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	Docket Nos. 50-247-LR and 50-286-LR
	)	
ENTERGY NUCLEAR OPERATIONS, INC.	)	
	)	
(Indian Point Nuclear Generating Units 2 and 3)	)	
	)	August 25, 2010

CERTIFICATE OF SERVICE

I hereby certify that copies of the "Applicant's Motion for Summary Disposition of New York State Contentions 26/26A and Riverkeeper Technical Contentions 1/1A (Metal Fatigue of Reactor Components" and the Public Version of the Supporting Attachments (i.e., excluding Entergy-Designated Proprietary Attachments 15 and 16), dated August 25, 2010, were served this 25th day of August, 2010 upon (1) all persons listed below by e-mail, and (2) by first-class mail to those persons who are not receiving the Non-Public Confidential Proprietary Version of the Supporting Attachments, as designated below with a double asterisk (**).

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	Docket Nos. 50-247-LR and 50-286-LR
ENTERGY NUCLEAR OPERATIONS, INC.	)	ASLBP No. 07-858-03-LR-BD01
(Indian Point Nuclear Generating Units 2 and 3)	)	

SUPPORTING ATTACHMENTS TO APPLICANT'S MOTION FOR SUMMARY DISPOSITION OF NEW YORK STATE CONTENTIONS 26/26A & RIVERKEEPER TECHNICAL CONTENTIONS 1/1A (METAL FATIGUE OF REACTOR COMPONENTS)

PUBLIC VERSION EXCLUDING ENTERGY-DESIGNATED CONFIDENTIAL PROPRIETARY ATTACHMENTS 15 AND 16

Filed on August 25, 2010

LIST OF SUPPORTING ATTACHMENTS
APPLICANT'S MOTION FOR SUMMARY DISPOSITION OF
NEW YORK STATE CONTENTIONS 26/26A & RIVERKEEPER TECHNICAL CONTENTIONS 1/1A (METAL FATIGUE OF REACTOR COMPONENTS)

Attachment	Description
1	Statement of Material Facts
2	Declaration of Nelson F. Azevedo
3	*Curriculum Vitae* of Nelson F. Azevedo
4	Excerpt from NUREG-1800, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants*, Revision 1 (Sept. 2005)
5	Excerpt from Vol. 2 of NUREG-1801, *Generic Aging Lessons Learned (GALL) Report – Tabulation of Results* (Rev. 1, Sept. 2005)
6	Excerpts from EPRI, MRP-47, *Materials Reliability Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application* (Rev. 1, Sept. 2005)
7	Excerpt from NUREG/CR-6583, *Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels* (Mar. 1998).
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11	Excerpt from NL-08-057, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "Amendment 3 to License Renewal Application (LRA)" (Mar. 24, 2008))
12	NL-08-084, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "Reply to Request for Additional Information Regarding License Renewal Application – Time-Limited Aging Analyses and Boraflex," (May 16, 2008)
13	NL-08-092, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "Amendment 5 to License Renewal Application"

Attachment	Description
	(June 11, 2008)
14	NL-10-082, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "License Renewal Application – Completion of Commitment #33 Regarding the Fatigue Monitoring Program" (Aug. 9, 2010)
15	**EXCLUDED:** Westinghouse Electric Co., WCAP-17199-P, Revision 0, *Environmental Fatigue Evaluation for Indian Point Unit 2* (June 2010) **(Entergy-Designated Confidential Proprietary Document Subject to Nondisclosure Agreement and ASLB 9/4/2009 Protective Order)**
16	**EXCLUDED:** Westinghouse Electric Co., WCAP-17200-P, Revision 0, *Environmental Fatigue Evaluation for Indian Point Unit 3* June 2010) **(Entergy-Designated Confidential Proprietary Document Subject to Nondisclosure Agreement and ASLB 9/4/2009 Protective Order)**

UNITED STATES OF AMERICA
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NYS-26/26A & RIVERKEEPER TC-1/1A: ATTACHMENT 1

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	Docket Nos. 50-247-LR and 50-286-LR
ENTERGY NUCLEAR OPERATIONS, INC.	)	ASLBP No. 07-858-03-LR-BD01
(Indian Point Nuclear Generating Units 2 and 3)	)	August 25, 2010

STATEMENT OF MATERIAL FACTS

Entergy Nuclear Operations, Inc. ("Entergy") submits this statement of undisputed material facts in support of its Motion for Summary Disposition of New York State Contentions 26/26A ("NYS-26/26A") and Riverkeeper Technical Contentions 1/1A ("TC-1/1A").

A. Applicable 10 C.F.R. Part 54 Regulations and License Renewal Guidance

1. 10 C.F.R. § 54.21(a)(3) requires a license renewal applicant to demonstrate that it will adequately manage the effects of aging on structures, systems, and components ("SSCs") subject to aging management review ("AMR"), so that there is "reasonable assurance" that their intended functions will be maintained consistent with the current licensing basis ("CLB") for the period of extended operation ("PEO").

2. 10 C.F.R. § 54.21(d) requires that the final safety analysis report ("FSAR") supplement for the facility contain a summary description of the programs and activities for managing the effects of aging.

3. 10 C.F.R. § 54.21(c)(1) lists the technical information that must be contained in a license renewal application ("LRA") related specifically to time-limited aging analyses ("TLAAs"), which 10 C.F.R. § 54.3(a) defines as those licensee calculations and analyses that: (1) involve SSCs within the scope of license renewal, as delineated in § 54.4(a); (2) consider the effects of aging; (3) involve time-limited assumptions defined by the current operating term, for example, 40 years; (4) were determined to be relevant by the licensee in making a safety determination; (5) involve conclusions or provide the basis for conclusions related to the

capability of the system, structure, and component to perform its intended functions, as delineated in § 54.4(b); and (6) are contained or incorporated by reference in the CLB.

4. For all TLAAs, an applicant must demonstrate that (i) the analyses remain valid for the PEO; (ii) the analyses have been projected to the end of the PEO, or (iii) the aging effects will be adequately managed for the PEO. 10 C.F.R. § 54.21(c)(1)(i)-(iii).

5. NUREG-1800 describes methods for identifying those SSCs that are subject to aging effects within the scope of license renewal, and provides ten program elements that define an effective AMP: (1) Scope of the Program, (2) Preventive Actions, (3) Parameters Monitored or Inspected, (4) Detection of Aging Effects, (5) Monitoring and Trending, (6) Acceptance Criteria, (7) Corrective Actions, (8) Confirmation Process, (9) Administrative Controls, and (10) Operating Experience. NUREG-1800, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants* at 3.0-2, A.1-8 (Rev. 1, Sept. 2005) ("SRP-LR") (Attach. 4).

6. The technical basis behind the SRP-LR is provided in 2 NUREG-1801, *Generic Aging Lessons Learned Report* (Rev. 1, Sept. 2005) ("GALL Report") (Attach. 5), an NRC license renewal guidance document prepared at the request of the Commission and commonly cited by it with approval. *Entergy Vt. Yankee, L.L.C.* (Vermont Yankee Nuclear Power Station), CLI-10-17, slip op. at 44 (*citing AmerGen Energy Co. LLC* (Oyster Creek Nuclear Generating Station), CLI-08-23, 68 NRC 461, 468 (2008)).

7. The GALL Report identifies generic AMPs that the Staff has found acceptable for meeting the requirements of Part 54, based on its evaluations of existing programs at operating plants during the initial license period. An applicant's use of an AMP identified in the GALL Report "constitutes reasonable assurance that it will manage the targeted aging effect during the renewal period." *Oyster Creek*, CLI-08-23, 68 NRC at 468.

B. <u>Metal Fatigue and the Cumulative Usage Factor</u>

8. Fatigue is the weakening of a metal caused by cyclic mechanical and thermal stresses (*i.e.*, cyclical loading) at a location on a metallic component. Metal components experience these stress cycles during "transients" such as plant startup and shutdown that result in significant temperature changes. A "stress cycle" is the time period it takes for a material to

go from its minimum stress level to its maximum level and back again to its minimum level. An excessive number of cycles may result in a significant reduction in the strength of a component. Declaration of Nelson Azevedo in Support of Applicant's Motion for Summary Disposition of Consolidated Contentions NYS-26/26A and Riverkeeper TC-1/1A, ¶ 4 (Attach. 2).

9. All materials have a distinctive number of stress cycles that the material can withstand at a particular applied stress level before fatigue failure occurs. The period during which this number of load cycles occurs is called the material's "fatigue life." Attach. 2, ¶ 5.

10. 10 C.F.R. Part 50 does not specifically mention metal fatigue; however, 10 C.F.R. § 50.55a(c)(1) requires that the reactor coolant system ("RCS") pressure boundary meet the requirements of the American Society of Mechanical Engineers ("ASME") Boiler and Pressure Vessel Code ("ASME Code"), Section III, as endorsed by the NRC. Attach. 2, ¶ 6.

11. The original design specifications for a given safety-related component specify the number of mechanical and thermal cycles that the component must be designed to withstand, and define the safety limits and applicable codes that must be satisfied. For components constituting the primary RCS pressure boundary, the specified requirements for evaluation of cyclic loading and thermal conditions generally are contained in Section III of the ASME Code for Class 1 components. Attach. 2, ¶ 7.

12. For a Class 1 component, different stress cycles from the loadings specified in the governing design specification will produce total stresses of several different magnitudes. The number of times these stresses of different magnitudes occur also varies. The allowable number of cycles for a given alternating stress range is determined from the ASME Code design fatigue curve for the material being evaluated. Attach. 2, ¶ 8.

13. The fatigue usage for that stress cycle is the ratio of the number of analyzed applied stress cycles (n) to the allowable number of stress cycles (N) from the ASME Code design fatigue curve. The cumulative usage factor ("CUF") represents the fraction of the total allowable fatigue cycles that the component is projected to incur during its operation. Attach. 2, ¶ 9.

14. ASME Code Section III requires that the CUF for a Class 1 component not exceed unity or 1.0; *i.e.*, the total number of applied stress cycles is not to exceed the allowable

number of stress cycles. This is an acceptance criterion established by the Code, but exceeding the criterion does not mean the component will fail, given the numerous factors of conservatism in the analytical process. Attach. 2, ¶ 10.

15. CUF values less than 1.0 indicate that the equipment can withstand the fatigue effects of cyclic stress due to operation during the analyzed number of design transients at the analyzed transient severity. A projected CUF greater than 1.0 does not indicate that fatigue cracking necessarily will occur. Rather, it indicates that there is a potential for cracking to initiate at the affected location. This is not necessarily failure of the component. Attach. 2, ¶¶ 10 & 11.

16. CUF values result from conservative fatigue calculations. Applicants may revise fatigue analyses to take advantage of excess conservatisms inherent in the original fatigue analyses (*e.g.*, excess conservatisms in loading definitions, predicted numbers of cycles, transient groupings, etc.) as well as improvements in available analytical tools. Attach. 2, ¶¶ 10, 17; EPRI, MRP-47, *Materials Reliability Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application* at 4-4 (Rev. 1, Sept. 2005) (Attach. 6).

C. Environmentally-Assisted Fatigue and Related NRC Guidance

17. For components (equipment and piping) exposed to reactor coolant water, the fatigue life, or allowable number of stress cycles, may be reduced compared to a component's fatigue life in air. The ASME Code design fatigue curves were developed based on laboratory testing of specimens in air at a constant strain rate, with safety factors incorporated into the curves to account for variables such as surface finish. Laboratory testing of specimens in water under reactor operating conditions indicate that additional environmental factors may need to be applied to the calculated CUF to fully account for reactor coolant environmental conditions. Fatigue analysis accounting for the effects of operating in a reactor coolant environment is called environmentally-assisted fatigue ("EAF") analysis. Attach. 2, ¶ 18.

18. EAF analysis was addressed in NRC Generic Safety Issue ("GSI") 190, "Fatigue Evaluation of Metal Components for 60-year Plant Life." The NRC Staff closed out GSI 190 in December 1999 without imposing additional requirements because it found only negligible calculated increases in core damage frequency in going from 40-year to 60-year license periods. However, the NRC Staff concluded that license renewal applicants should address the effects of

coolant environment on component fatigue life as aging management programs that are formulated for license renewal. Attach. 2, ¶ 19; Attach. 4, at 4.3-2 to 4.3-3.

19. To address EAF, an applicant may apply an environmental correction factor, or "F_{en}", to a CUF, to calculate an environmentally-adjusted CUF, or "CUF_{en}". The F_{en} is defined as the ratio of the fatigue life in air at room temperature to that in water at the service temperature. The fatigue usage derived from current ASME Code fatigue design (air) curves is multiplied by the F_{en} to account for the environmental effects. Because EAF (CUF_{en}) analyses are not contained within an applicant's CLB, they are not TLAAs. Attach. 2, ¶ 20.

20. NRC guidance recommends the use of formulae in NUREG/CR-6583, *Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels* (Mar. 1998) (Attach. 7), to determine F_{en}s for carbon and low alloy steel components, and NUREG/CR-5704, *Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels* (Apr. 1999) (Attach. 8), to determine F_{en}s for stainless steel components. Attach. 2, ¶ 21; Attach. 4, at 4.3-5, 4.3-7; Attach 5, at X M-1.

21. As discussed in the SRP-LR and GALL Report, one method for addressing EAF found acceptable by the NRC Staff is the conduct of CUF_{en} analyses for the six critical locations identified in NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components* (Mar. 1995) (Attach. 9). Those locations include: (1) the reactor vessel shell and lower head, (2) the reactor vessel inlet and outlet nozzles, (3) the pressurizer surge line (including hot leg and pressurizer nozzles), (4) RCS piping charging system nozzle, (5) RCS piping safety injection nozzle, and (6) residual heat removal ("RHR") Class 1 piping. Attach. 2, ¶ 22; Attach. 4, at 4.3-5; Attach. 5, at X M-1; Attach. 9, at 5-62.

22. NUREG/CR-6260 states that the six component locations identified therein "were chosen to give a representative overview of components that had higher CUFs and/or were important from a risk perspective." Attach. 9, at 4-1. Another guidance document, MRP-47, Revision 1, states that the six NUREG/CR-6260 locations "were considered representative enough that the effects of LWR environment on fatigue could be assessed." Attach. 6, at 3-4. MRP-47 further states: "For cases where acceptable fatigue results are demonstrated for these locations for 60 years of plant operation including environmental effects, additional evaluations or locations need not be considered." *Id.*

D. Overview of the Indian Point Energy Center ("IPEC") Fatigue Monitoring Program

23. LRA Chapter 4 summarizes Entergy's evaluation of the effects of metal fatigue and EAF on in-scope piping and components for the PEO. Section 4.3.1 (Class 1 Fatigue) addresses components (and subcomponents) that were designed in accordance with ASME Code, Section III. Broadly speaking, those components include the reactor vessel, reactor vessel internals, pressurizer, steam generators, reactor coolant pumps, and control rod drive mechanisms, Class 1 heat exchangers, and Class 1 piping and components. Section 4.3.2 (Non-Class 1 Fatigue) addresses ANSI Code B31.1 and ASME Code Section III Class 2 and 3 piping systems. Section 4.3.3 (Effects of Reactor Water Environment on Fatigue) addresses EAF with respect to the critical component locations identified in NUREG/CR-6260. Attach. 2, ¶ 23.

24. The appendices to the LRA contain a description of Entergy's Fatigue Monitoring Program. Appendix A presents the information required by 10 C.F.R. § 54.21(d) relating to the AMP for fatigue monitoring that supplements the updated FSAR ("UFSAR") for IPEC. The supplement to the UFSAR, presented in Sections A.2 and A.3 of Appendix A for IP2 and IP3, respectively, contains a summary description of the program and activities for managing the effects of metal fatigue during the PEO. Appendix A states that the Fatigue Monitoring Program will be implemented prior to the PEO. Attach. 2, ¶ 24; LRA app. A at A-1, A-22, A-49, *available at* ADAMS Accession No. ML071210520.

25. Appendix B to the LRA describes those AMPs credited by Entergy to manage the effects of aging. Section B.1.12 describes the IPEC Fatigue Monitoring Program, which is an existing program that is designed to track the number of transients for selected RCS components. LRA app. B at B-1, B-44 to B-46, B-54, *available at* ADAMS Accession No. ML071210523. Section B.1.12 indicates that it is consistent with, but takes one exception to the "detection of aging effects," the program described in the GALL Report Section X.M1. *Id.* at B-44. However, Entergy later removed this exception in an amendment to the LRA. Attach. 2, ¶ 25; NL-08-092, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "Amendment 5 to License Renewal Application" (June 11, 2008) (Attach. 13); 2 NUREG-1930, *Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3, Docket Nos. 50-247 and 50-286, Entergy Nuclear Operations, Inc.* at 3-79 to 3-80 (Aug. 2009) ("SER"), *available at* ADAMS Accession No. ML093170671.

26. As discussed in LRA Section 4.3.1, design basis fatigue evaluations for Class 1 components produce calculated CUFs for components and subcomponents based on selected numbers and magnitudes of design transient cycles. For RCS components, Entergy projected the numbers of cycles accrued to date to the end of the PEO; *i.e.*, the numbers of cycles expected at the end of 60 years of plant operation. Attach. 2, ¶ 26.

27. LRA Tables 4.3-1 and 4.3-2 show the specific transient conditions, the analyzed numbers of cycles, and the projected numbers of cycles for 60 years of operation for IP2 and IP3, respectively. LRA Tables 4.3-3 to 4.3-12 list the CUFs for the various Class 1 components and subcomponents based on the numbers of cycles assumed in the analyses. The numbers of cycles expected at the end of 60 years of plant operation are less than the numbers considered in the analyses for all but a few of the transients listed in LRA Tables 4.3-1 and 4.3-2. Attach. 2, ¶ 26.

28. IPEC RCS piping was designed to ANSI B31.1, *Power Piping Code*, which did not require a detailed fatigue usage factor evaluation. Instead, a stress analysis was performed on the main primary coolant piping, in accordance with the criteria set forth in ANSI B31.1, to ensure that the stress range was within the prescribed limits. Stress range reduction factors were used to account for anticipated transients for RCS piping (a stress range reduction factor of 1.0 is acceptable in the stress analyses for up to 7000 cycles). Therefore, Entergy evaluated the projected thermal cycles for 60 years of plant operation at IP2 and IP3 and determined that the total cycles in 60 years of operation are well below the 7000 cycles allowed by ANSI B31.1 for a stress range reduction factor of 1.0. Attach. 2, ¶ 27; LRA at 4.3-18.

29. In preparing Section 4.3.3 of its April 2007 LRA, Entergy addressed EAF effects on those IPEC-specific components corresponding to the six critical locations identified in NUREG/CR-6260 by either projecting the analyses to the end of the PEO, per § 54.21(c)(1)(ii), or demonstrating that aging effects will be adequately managed, per § 54.21(c)(1)(iii). Attach. 2, ¶ 28; LRA at 4.3-20 to 4.3-25.

30. Consistent with the GALL Report, Entergy applied Fens calculated as described in NUREG/CR-6583 and NUREG/CR-5704 to the 60-year-projected CUFs to determine CUFen values. LRA at 4.3-21, tbls. 4.3.13 (IP2) & 4.3-14 (IP3). (As explained above, CUFs were not available for all RCS piping and RHR Class 1 piping locations because those locations were designed to ANSI B.31.1.) Entergy's Fen calculation methods are documented in a response to a

Staff request for additional information. Attach. 2, ¶ 29; NL-08-084, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "Reply to Request for Additional Information Regarding License Renewal Application – Time-Limited Aging Analyses and Boraflex" attach. 1, at 3-4 (May 16, 2008) (Attach. 12).

31. As indicated in LRA Tables 4.3.13 and 4.3-14, the projected CUFen values for the three NUREG/CR-6260 reactor vessel locations were less than 1.0 for both IP2 and IP3. These three reactor vessel locations included the (1) bottom head to shell, (2) reactor vessel inlet nozzle, and (3) reactor vessel outlet nozzle. In addition, the IP2 (but not IP3) pressurizer surge line nozzle had a CUFen less than 1.0. Accordingly, because these CUFen values fell below 1.0, the EAF analyses for those components were projected to the end of the PEO (in the LRA) in accordance with Section 54.21(c)(1)(ii). Attach. 2, ¶ 30; LRA at 4.3-22.

32. The component locations in LRA Tables 4.3-13 and 4.3-14 without projected CUF_{en} values less than 1.0 include the pressurizer surge line piping for IP2 and IP3; the RCS piping charging system nozzles for IP2 (for which a CUF was available) and for IP3; and the pressurizer surge line nozzles for IP3. LRA tbls. 4.3.13 & 4.3-14. As stated in LRA Section 4.3.3, the IP2 pressurizer surge nozzle had a CUF_{en} less than 1.0, whereas the IP3 pressurizer surge nozzle had CUF_{en} greater than 1.0, because the IP3 surge nozzle calculation includes the effects of the insurges/outsurges experienced by these nozzles, while the IP2 analysis did not include these effects. As a result, Entergy committed to re-analyze the pressurizer surge line nozzle for both units to include insurge/outsurge and environmental effects. Attach. 2, ¶ 31; LRA at 4.3-21 to 4.3-22, tbls. 4.3.13 & 4.3-14.

33. To address these locations, Entergy committed to take one of the following actions: (1) refine the fatigue analyses, at least two years before entering the PEO, to determine valid CUF_{en} values less than 1.0; (2) manage the effects of aging due to fatigue at the affected locations by an inspection program that has been reviewed and approved by the NRC; or (3) repair or replace the affected locations before exceeding CUF of 1.0 (collectively referred to as "Commitment 33"). Attach. 2, ¶ 31; LRA 4.3-22 to 4.3-23.

34. In January 2008, as a part of the LRA Amendment 2, Entergy amended the Fatigue Monitoring Program to provide information on cycle counting and the methodology used to determine stresses and fatigue usage, including environmental effects, in accordance with the

NRC-endorsed ASME Code. Entergy also amended the LRA to place Commitment 33 within the scope of the Fatigue Monitoring Program, by stating that it will use that AMP to manage the effects of reactor water environment on fatigue life, in accordance with § 54.21(c)(1)(iii). Attach. 2, ¶ 32; NL-08-021, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "License Renewal Application Amendment 2" (Jan. 22, 2008) ("LRA Amendment 2") (Attach. 10).

35. Pursuant to revised Commitment 33, if Entergy does not demonstrate valid CUF_{en} values below 1.0 (Option 1) by conducting refined CUF_{en} analyses, then it must implement Option 2 of the commitment. Option 2 requires Entergy to repair or replace the affected locations before their refined CUF_{en}s exceed 1.0, consistent with the Fatigue Monitoring Program. Attach. 2, ¶ 32; Attach. 10, attach. 2, at 15).

36. During the PEO, Entergy will continue to monitor the condition of piping and components at the locations of interest under IPEC's ISI Program, which includes periodic visual, surface, and volumetric examination of Classes 1, 2, and 3 pressure-retaining components, their attachments, and supports. If a flaw is detected during an ISI Program inspection or by other means, then the component may be replaced or repaired, or evaluated for continued service in accordance with the criteria contained in ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components." Acceptance standards for examination evaluations, repair procedures, inservice test requirements, and replacements for ASME Class 1 components are defined in ASME Section XI paragraphs IWB-3000, IWB-4000, IWB-5000 and IWB-7000, respectively. Attach. 2, ¶ 53.

37. Under the Fatigue Monitoring Program, Entergy will manage the effects of fatigue by monitoring cycles incurred and ensuring they do not exceed the analyzed numbers of cycles, such that the CUFen analyses remain valid. Attach. 2, ¶ 54; LRA at 4.3-2 to 4.3-3; SER at 4-44 to 4-45; Attach. 12, attach. 1, at 3-4.

38. As required by the Fatigue Monitoring Program, Entergy tracks actual plant transients and evaluates these against the design transients. The plant transient counts will be updated at least once each operating cycle, which is an acceptable frequency because the evaluation during each update determines if the number of design transients could be exceeded

prior to the next update. Attach. 2, ¶ 55; LRA at 4.3-2 to LRA 4.3-3; Attach. 10, attach. 1, at 6); Attach. 12, attach. 1, at 4).

39. Consistent with GALL Report Section X.M1, the Fatigue Monitoring Program requires that corrective actions be implemented in accordance with the IPEC Corrective Action Program before the plant exceeds the analyzed number of transient cycles. IPEC procedures contain specific "alert levels" that trigger the initiation of corrective actions under the Fatigue Monitoring Program. SER at 4-44. Specifically, IPEC calculates alert levels by adding twice the number of cycles that occurred in the last fuel cycle to the total number of cycles to date. Corrective action is initiated if this alert level exceeds the number of analyzed transients. If the number of cycles is projected to remain at or below the analyzed level for two additional fuel cycles, then no corrective action is required Attach. 2, ¶ 56; Attach. 10, attach. 1, at 5-6; Attach. 12, attach. 1, at 4.

40. 57. Any further reanalysis (*i.e.*, future analysis updates), if necessary, would be governed by applicable QA procedures. Any repair or replacement of a component, if necessary, would be done in accordance with established plant procedures that must comply with Entergy's QA program and meet the applicable repair or replacement requirements of ASME Code Section XI. Attach. 2, ¶ 57; Attach. 10, attach. 1, at 6; Attach. 12, attach. 1, at 4.

E. <u>The NRC Staff's November 2009 Safety Evaluation Report Findings</u>

41. As documented in its final Safety Evaluation Report ("SER"), the NRC Staff, after conducting a detailed technical review of the LRA and performing related site audits, found that IPEC's Fatigue Monitoring Program includes acceptable program elements that are consistent with criteria in GALL Report Section X.M1. 2 SER at 3-81. It further concluded that Entergy has demonstrated that the effects of aging will be adequately managed so that the intended component functions will be maintained consistent with the CLB for the PEO, as required by 10 C.F.R. § 54.21(a)(3). *Id.* In addition to evaluating IPEC's Fatigue Monitoring Program, the Staff reviewed LRA Section 4.3.3 regarding EAF. As part of its review, the NRC Staff confirmed that IPEC had correctly accounted for the environmental factors used as inputs for calculating F_{en} factors. It also found that Commitment 33 is consistent with 10 C.F.R. § 54.21(a)(1)(iii). *Id.* at 4-43 to 4-45.

F. The 2010 Westinghouse Environmental Fatigue Evaluations for IPEC

42. To resolve Commitment 33, Entergy retained Westinghouse Electric Company LLC ("Westinghouse") in 2008, to update the fatigue usage calculations by performing refined fatigue analyses to determine environmentally-adjusted CUFs. Westinghouse completed the refined fatigue analyses in late June 2010. Attach. 2, ¶ 34; Westinghouse Electric Co., WCAP-17199-P, Westinghouse Electric Co., *Environmental Fatigue Evaluation for Indian Point Unit 2* (Rev. 0, June 2010) (proprietary) (Attach. 15); WCAP-17200-P, Rev. 0, *Environmental Fatigue Evaluation for Indian Point Unit 3* (June 2010) (proprietary) (Attach. 16).

43. Entergy approved the Westinghouse EAF analyses on July 29, 2010, and shortly thereafter, notified the Staff of the results of the refined EAF analyses. Attach. 2, ¶ 34; NL-10-082, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "License Renewal Application – Completion of Commitment #33 Regarding the Fatigue Monitoring Program" (Aug. 9, 2010) (Attach. 14).

44. Consistent with the Fatigue Monitoring Program, the refined EAF analyses were performed in accordance with Entergy's 10 C.F.R. Part 50, Appendix B Quality Assurance ("QA") program and included design input verification and independent reviews to ensure that valid assumptions, transients, cycles, external loadings, analysis methods, and F_{en} factors were used in the EAF analyses for IP2 and IP3.

45. As documented in WCAP-17199-P and WCAP-17200-P, Westinghouse performed 60-year EAF evaluations using F_{en} factors derived from NUREG/CR-5704 for stainless steel components and from NUREG/CR-6583 for those components containing carbon and low-alloy steel components. Westinghouse calculated the F_{en} factors, which it then applied to the ASME Code fatigue results. Attach. 2, ¶ 35.

46. The Westinghouse evaluations included the following four major steps: (1) determination of the limiting components for each location; (2) determination of the transients for each limiting component; (3) performance of ASME Code Section III stress and fatigue evaluations for each limiting component; and (4) determination of the CUF_{en}, for each limiting component. As necessary, plant-specific projections of numbers of accrued cycles at the end of the PEO replaced conservative assumptions for numbers of cycles in the analyses. The fatigue

evaluations followed the procedures given in ASME Code Section III, NB-3200. Attach. 2, ¶ 36; Attach. 15, at 2-2 to 2-4, 5-20; Attach. 16, at 2-2 to 2-4, 5-20.

47. Westinghouse developed transients using plant data and/or Westinghouse standard design transients, assuming the total period of operation to be 60 years. For the charging, safety injection, and RHR locations selected, Westinghouse developed stress inputs to the fatigue analyses using detailed finite element analysis models, and ran those models to create unit load transfer functions for each type of mechanical and thermal transient condition loading considered in the analyses. It also developed piping mechanical load functions to produce time history moment loads as a function of system parameters during a transient. The transfer functions and load functions were used in a proprietary Westinghouse computer code to determine detailed stress histories for each applicable transient, considering all applicable mechanical and thermal transient loads during each transient, and to calculate fatigue usage. Attach. 2, ¶ 37; Attachs. 15 & 16.

48. Section 5 of each Westinghouse EAF report details the manner in which Westinghouse evaluated EAF for the relevant IPEC-specific NUREG/CR-6260 component locations. Westinghouse's evaluations were based upon CUF analyses using transients that envelop the 60-year cycle projections. Sections 5.1 and 5.2, in particular, describe Westinghouse F_{en} calculations using the methodologies set forth in NUREG/CR-5704 for stainless steel and NUREG/CR-6583 for carbon steel. Attach. 2, ¶ 38.

49. The 60-year fatigue results for the critical component locations are provided in Tables 5-8 through 5-14 of WCAP-17199-P and WCAP-17200-P. Westinghouse determined that, for IP2 and IP3, the refined CUFen values for pressurizer surge line piping, RCS piping charging system nozzle, RCS piping safety injection nozzle, and RHR Class 1 piping all are below 1.0 when projected to the end of the PEO. Attach. 2, ¶ 39; Attach. 14, attach. 1, at 2-4; Attach. 15, at 6-1; Attach. 16, at 6-1.

Respectfully submitted,

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Dated in Washington, D.C.
this 25th day of August 2010

DB1/65517382.1

NYS-26/26A & RIVERKEEPER TC-1/1A: ATTACHMENT 2

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	Docket Nos. 50-247-LR and 50-286-LR
ENTERGY NUCLEAR OPERATIONS, INC.	)	ASLBP No. 07-858-03-LR-BD01
(Indian Point Nuclear Generating Units 2 and 3)	)	August 20, 2010

DECLARATION OF NELSON F. AZEVEDO IN SUPPORT OF APPLICANT'S MOTION FOR SUMMARY DISPOSITION OF CONTENTIONS NYS-26/26A AND RIVERKEEPER TC-1/1A

Nelson F. Azevedo states as follows under penalties of perjury:

I. PERSONAL BACKGROUND

1. My name is Nelson Azevedo. I am employed by Entergy Nuclear Operations, Inc. ("Entergy") as Supervisor, Code Programs, at Indian Point Energy Center ("IPEC") in Buchanan, New York. I am providing this Declaration in support of the "Applicant's Motion for Summary Disposition of New York State Contentions 26/26A & Riverkeeper Technical Contentions 1/1A (Metal Fatigue of Reactor Components)" in the above-captioned proceeding. In April 2007, Entergy submitted a license renewal application ("LRA") to the U.S. Nuclear Regulatory Commission ("NRC") seeking to renew its operating licenses for Indian Point Nuclear Generating Units 2 and 3 ("IP2" and "IP3," collectively "IPEC") for another 20 years.

2. I have described my professional and educational qualifications in the *curriculum vitae* appended as Attachment 3. Briefly summarized, I hold a B.S. in Mechanical and Materials Engineering from the University of Connecticut and an M.S. in Mechanical Engineering from the Rensselaer Polytechnic Institute ("RPI") in Troy, New York. In

addition, I have received an M.B.A. from RPI. I have over 28 years of professional experience in the nuclear power industry. During that time, I have held engineering and supervisory positions with Northeast Utilities and Entergy. As a Department Manager with Northeast Utilities, I managed five engineering sections responsible for implementing numerous engineering programs at Millstone Station, including the fatigue monitoring program. Currently, I oversee the IPEC engineering section responsible for implementing American Society of Mechanical Engineers ("ASME") Code programs, including the inservice inspection ("ISI"), inservice testing, flow-accelerated corrosion, snubber testing, boric acid corrosion control, non-destructive examination, fatigue monitoring, steam generators, buried piping, alloy 600 cracking, reactor vessel embrittlement, welding, and 10 C.F.R. Part 50, Appendix J containment leakage programs. I also am responsible to ensure compliance with the ASME Code, Section XI requirements for repair and replacement activities at IPEC. I also represent IPEC before industry organizations, including the pressurized water reactor ("PWR") Owners Group Management Committee.

3. From January 2001 through approximately 2007, I was directly responsible for the IP2 Fatigue Monitoring Program and developed the plant procedure used to track the cycles for IP2. I now supervise the IPEC engineering staff responsible for implementing the IP2 and IP3 Fatigue Monitoring Programs. I reviewed draft versions of the Westinghouse environmental fatigue evaluations for IP2 and IP3 discussed below, and directly interfaced with Westinghouse personnel in resolving technical comments on those drafts before their final approval by IPEC. During my career, I have performed pipe stress analysis, finite element analysis of large components, ASME Code Section XI flaw evaluations, and ASME Code Section III, Class 1 fatigue analysis. Accordingly, I have personal knowledge of the

IPEC Fatigue Monitoring Program, including the description of that program in the LRA; relevant NRC requirements and guidance; and applicable industry codes.

II. TECHNICAL BACKGROUND CONCERNING METAL FATIGUE

A. Metal Fatigue and the Cumulative Usage Factor

4. Fatigue is the weakening of a metal caused by cyclic mechanical and thermal stresses (*i.e.*, cyclical loading) at a location on a metallic component. Metal components experience these stress cycles during "transients" such as plant startup and shutdown that result in significant temperature changes. A "stress cycle" is the time period it takes for a material to go from its minimum stress level to its maximum level and back again to its minimum level. An excessive number of cycles may result in a significant reduction in the strength of a component.

5. All materials have a distinctive number of stress cycles that the material can withstand at a particular applied stress level before fatigue failure occurs. The period during which this number of load cycles occurs is called the material's "fatigue life."

6. 10 C.F.R. Part 50 does not specifically mention metal fatigue; however, 10 C.F.R. § 50.55a(c)(1) requires that the reactor coolant system ("RCS") pressure boundary meet the requirements of the ASME Boiler and Pressure Vessel Code ("ASME Code"), Section III, as endorsed by the NRC. For an IPEC-vintage Westinghouse pressurized water reactor, the RCS pressure boundary includes components such as the reactor vessel primary inlet and outlet nozzles and pressurizer surge line nozzles.

7. The original design specifications for a given safety-related component specify the number of mechanical and thermal cycles that the component must be designed to withstand, and define the safety limits and applicable codes that must be satisfied. For components constituting the primary RCS pressure boundary, the specified requirements for

evaluation of cyclic loading and thermal conditions generally are contained in Section III of the ASME Code for Class 1 components.

8. For a Class 1 component, different stress cycles from the loadings specified in the governing design specification will produce total stresses of several different magnitudes. The number of times these stresses of different magnitudes occur also varies. The allowable number of cycles for a given alternating stress range is determined from the ASME Code design fatigue curve for the material being evaluated.

9. The fatigue usage for that stress cycle is the ratio of the number of analyzed applied stress cycles (n) to the allowable number of stress cycles (N) from the ASME Code design fatigue curve. The cumulative usage factor ("CUF") represents the fraction of the total allowable fatigue cycles that the component is projected to incur during its operation.

10. ASME Code Section III requires that the CUF for a Class 1 component not exceed unity or 1.0; *i.e.*, the total number of applied stress cycles is not to exceed the allowable number of stress cycles. This is an acceptance criterion established by the Code, but exceeding the criterion does not mean the component will fail, given the numerous factors of conservatism in the analytical process. CUF values result from conservative fatigue calculations. CUF values less than 1.0 indicate that the equipment can withstand the fatigue effects of cyclic stress due to operation during the analyzed number of design transients at the analyzed transient severity.

11. It is important to understand that a projected CUF greater than 1.0 does not indicate that fatigue cracking necessarily will occur. Rather, it indicates that there is a potential for cracking to initiate at the affected location. This is not necessarily failure of the component.

12. With regard to license renewal, 10 C.F.R. § 54.21(c)(1) lists the technical information that must be contained in an LRA relatedly specifically to time-limited aging analyses ("TLAAs"), which 10 C.F.R. § 54.3(a) defines as those licensee calculations and analyses that: (1) involve structures, systems, and components ("SSCs") within the scope of license renewal, as delineated in Section 54.4(a); (2) consider the effects of aging; (3) involve time-limited assumptions defined by the current operating term, for example, 40 years; (4) were determined to be relevant by the licensee in making a safety determination; (5) involve conclusions or provide the basis for conclusions related to the capability of the SSC to perform its intended functions, as delineated in Section 54.4(b); and (6) are contained or incorporated by reference in the current licensing basis ("CLB").

13. For all TLAAs, an applicant must demonstrate that (i) the analyses remain valid for the PEO; (ii) the analyses have been projected to the end of the PEO, or (iii) the aging effects will be adequately managed for the PEO. 10 C.F.R. § 54.21(c)(1)(i)-(iii).

14. The fatigue analyses of the RCS are treated as TLAAs because they are based on the numbers of cycles estimated during the original design to be adequate for the initial license term (40 years of operation.). A license renewal applicant may update the projected cycles to account for 60 years of plant operation. One possible outcome is that numbers of analyzed cycles at the end of the extended operating period will remain at or below those originally projected for the initial 40-year plant license period, in which case the governing fatigue analyses will remain valid for the PEO, in accordance with 10 C.F.R. § 54.21(c)(1)(i).

15. Another possibility is that more cycles are projected to occur for 60 years of plant operation than were analyzed for the first 40 years. In this case, an applicant must address the increased cycle counts. One possible approach is to revise the fatigue analysis to confirm that

the increased number of cycles still will result in a CUF less than or equal to the allowable design code limit, in accordance with 10 C.F.R. § 54.21(c)(1)(ii).

16. Finally, another acceptable approach is to determine the most limiting number of cycles assumed in the fatigue analyses. This cycle quantity then becomes the allowable limit against which the actual operation is tracked through an appropriate AMP during the PEO in accordance with 10 C.F.R. § 54.21(c)(1)(iii). In this regard, CUFs may provide predictive or tracking/monitoring functions.

17. As noted above, applicants may revise fatigue analyses to take advantage of excess conservatisms inherent in the original fatigue analyses (*e.g.*, excess conservatisms in loading definitions, predicted numbers of cycles, transient groupings, etc.) as well as improvements in available analytical tools. An industry guidance document issued by the Electric Power Research Institute ("EPRI") further explains this important concept:

> Whereas fatigue calculations have varied over the years, their basic content is the same. With the advent of computer technology, the calculations have basically maintained the same content, but computations have become more refined and exhaustive. For example, 30 years ago it was computationally difficult for a stress analyst to evaluate 100 different transients in a fatigue calculation. Therefore, the analyst would have grouped the transients into as few as one transient grouping and performed as few incremental fatigue calculations as possible. With today's computer technology and desire to show more margin, it is relatively easy for the modern-day analyst to evaluate all 100 incremental fatigue calculations for this same problem. Also, older technology would have likely utilized conservative shell interaction hand solutions for computing stress, whereas today finite element techniques are commonly deployed. This improvement in technology would not have changed the basic inputs to the fatigue calculation (*i.e.*, stress), but it would have typically yielded significantly more representative input values.

EPRI, MRP-47, *Materials Reliability Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application* at 4-4 (Rev. 1, Sept. 2005) (Attach. 6).

B. Environmentally-Assisted Metal Fatigue

18. For components (equipment and piping) exposed to reactor coolant water, the fatigue life, or allowable number of stress cycles, may be reduced compared to a component's fatigue life in air. The ASME Code design fatigue curves were developed based on laboratory testing of specimens in air at a constant strain rate, with safety factors incorporated into the curves to account for variables such as surface finish. Laboratory testing of specimens in water under reactor operating conditions indicate that additional environmental factors may need to be applied to the calculated CUF to fully account for reactor coolant environmental conditions. Fatigue analysis accounting for the effects of operating in a reactor coolant environment is called environmentally-assisted fatigue ("EAF") analysis.

19. EAF analysis was addressed in NRC Generic Safety Issue ("GSI") 190, "Fatigue Evaluation of Metal Components for 60-year Plant Life." The NRC Staff closed out GSI 190 in December 1999 without imposing additional requirements because it found only negligible calculated increases in core damage frequency in going from 40-year to 60-year license periods. However, the NRC Staff concluded that license renewal applicants should address the effects of coolant environment on component fatigue life as aging management programs ("AMPs") that are formulated for license renewal.

20. To address EAF, an applicant may apply an environmental correction factor, or "F_{en}", to a CUF, to calculate an environmentally-adjusted CUF, or "CUF_{en}". The F_{en} is defined as the ratio of the fatigue life in air at room temperature to that in water at the service temperature. The fatigue usage derived from current ASME Code fatigue design (air) curves is multiplied by the F_{en} to account for the environmental effects. Because EAF (CUF_{en}) analyses are not contained within an applicant's CLB, they are not TLAAs.

21. NRC guidance recommends the use of formulae in NUREG/CR-6583, *Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels* (Mar. 1998) (Attach. 7) to determine F_{en} values for carbon and low alloy steel components, and NUREG/CR-5704, *Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels* (Apr. 1999) (Attach. 8) to determine F_{en} values for stainless steel components. *See* NUREG-1800, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants*, Revision 1 at 4.3-5, 4.3-7 (Sept. 2005) ("SRP-LR") (Attach. 4); NUREG 1801, *Generic Aging Lessons Learned Report*, Rev. 1, Vol. 2 at X M-1 (Sept. 2005) ("GALL Report") (Attach. 5).

22. As discussed in the SRP-LR and GALL Report, one method for addressing EAF found acceptable by the NRC Staff is the conduct of CUF_{en} analyses for the six critical locations identified in NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components* (Mar. 1995) (Attach. 9). Those locations include: (1) the reactor vessel shell and lower head, (2) the reactor vessel inlet and outlet nozzles, (3) the pressurizer surge line (including hot leg and pressurizer nozzles), (4) RCS piping charging system nozzle, (5) RCS piping safety injection nozzle, and (6) residual heat removal ("RHR") Class 1 piping. Attach. 4 at 4.3-5; Attach. 5 at X M-1; Attach. 5 at 5-62.

III. IPEC'S AGING MANAGEMENT PROGRAM FOR METAL FATIGUE

A. The IPEC Fatigue Monitoring Program as Described in the LRA

23. Chapter 4 of the IPEC LRA summarizes Entergy's evaluation of the effects of metal fatigue and EAF on in-scope piping and components for the PEO. Section 4.3.1 (Class 1 Fatigue) addresses components (and subcomponents) that were designed in accordance with ASME Code, Section III. Broadly speaking, those components include the reactor vessel, reactor vessel internals, pressurizer, steam generators, reactor coolant pumps, and control rod drive mechanisms, Class 1 heat exchangers, and Class 1 piping and components. Section 4.3.2 (Non-Class 1 Fatigue) addresses ANSI Code B31.1 and ASME Code Section III Class 2 and 3 piping systems. Section 4.3.3 (Effects of Reactor Water Environment on Fatigue) addresses EAF with respect to the critical component locations identified in NUREG/CR-6260.

24. The appendices to the LRA contain a description of Entergy's Fatigue Monitoring Program. Appendix A presents the information required by 10 C.F.R. § 54.21(d) relating to the AMP for fatigue monitoring that supplements the updated FSAR ("UFSAR") for IPEC. The supplement to the UFSAR, presented in Sections A.2 and A.3 of Appendix A for IP2 and IP3, respectively, contains a summary description of the program and activities for managing the effects of metal fatigue during the PEO. Appendix A states that the Fatigue Monitoring Program will be implemented prior to the PEO. LRA app. A at A-1, A-22, A-49, *available at* ADAMS Accession No. ML071210520.

25. Appendix B to the LRA describes those AMPs credited by Entergy to manage the effects of aging. Section B.1.12 describes the IPEC Fatigue Monitoring Program, which is designed to track the number of transients for selected RCS components. LRA app. B at B-1, B-44-B-46, B-54, *available at* ADAMS Accession No. ML071210523. Section B.1.12

indicates that it is consistent with the program described in GALL Report Section X.M1, but takes one exception to the "detection of aging effects" program element. *Id.* Entergy, however, later removed this exception via an amendment to the LRA. NL-08-092, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "Amendment 5 to License Renewal Application" (June 11, 2008) (Attach. 13); 2 NUREG-1930, *Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3, Docket Nos. 50-247 and 50-286, Entergy Nuclear Operations, Inc.* at 3-79 to 3-80 (Aug. 2009) ("SER"), *available at* ADAMS Accession No. ML093170671.

26. As discussed in LRA Section 4.3.1, design basis fatigue evaluations for Class 1 components produce calculated CUFs for components and subcomponents based on selected numbers and magnitudes of design transient cycles. For RCS components, Entergy projected the numbers of cycles accrued to date to the end of the PEO; *i.e.*, the numbers of cycles expected at the end of 60 years of plant operation. LRA Tables 4.3-1 and 4.3-2 show the specific transient conditions, the analyzed numbers of cycles, and the projected numbers of cycles for 60 years of operation for IP2 and IP3, respectively. LRA Tables 4.3-3 to 4.3-12 list the CUFs for the various Class 1 components and subcomponents based on the numbers of cycles assumed in the analyses. The numbers of cycles expected at the end of 60 years of plant operation are less than the numbers considered in the analyses for all but a few of the transients listed in LRA Tables 4.3-1 and 4.3-2.

27. As explained in LRA Section 4.3.1.8, IPEC RCS piping was designed to ANSI B31.1, *Power Piping Code*, which did not require a detailed fatigue usage factor evaluation.[1]

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[1] 10 C.F.R. § 50.55a(c)(1) states that components that are part of the reactor coolant pressure boundary must meet the requirements for Class 1 components in Section III of the ASME Boiler and Pressure Vessel Code,

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Instead, a stress analysis was performed on the main primary coolant piping, in accordance with the criteria set forth in ANSI B31.1, to ensure that the stress range was within the prescribed limits. Stress range reduction factors were used to account for anticipated transients for RCS piping (a stress range reduction factor of 1.0 is acceptable in the stress analyses for up to 7000 cycles). Therefore, Entergy evaluated the projected thermal cycles for 60 years of plant operation at IP2 and IP3 and determined that the total cycles in 60 years of operation are well below the 7000 cycles allowed by ANSI B31.1 for a stress range reduction factor of 1.0.

28. In preparing Section 4.3.3 of its April 2007 LRA, Entergy addressed EAF effects on those IPEC-specific components corresponding to the six critical locations identified in NUREG/CR-6260 by either projecting the analyses to the end of the PEO, per Section 54.21(c)(1)(ii), or demonstrating that aging effects will be adequately managed, per Section 54.21(c)(1)(iii). LRA at 4.3-20 to 4.3-25, *available at* ADAMS Accession No. ML071210517.

29. Consistent with the GALL Report, Entergy applied F_{en}s calculated as described in NUREG/CR-6583 and NUREG/CR-5704 to the 60-year-projected CUFs to determine CUF_{en} values. LRA at 4.3-21; tbls. 4.3.13 (IP2) & 4.3-14 (IP3). (As explained above, CUFs were not available for all RCS piping and RHR Class 1 piping locations because those locations were designed to ANSI B.31.1.). Entergy's F_{en} calculation methods are documented

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except for the components described in 10 C.F.R. § 50.55a(c)(2), (c)(3), and (c)(4). As relevant here, Section 50.55a(c)(4) states: "For a nuclear power plant whose construction permit was issued prior to May 14, 1984 the applicable Code Edition and Addenda for a component of the reactor coolant pressure boundary continue to be that Code Edition and Addenda that were required by Commission regulations for such component at the time of issuance of the construction permit." The IP2 and IP3 RCS piping and associated nozzles were designed and fabricated to meet ANSI B31.1 requirements.

in a response to a Staff request for additional information. NL-08-084, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "Reply to Request for Additional Information Regarding License Renewal Application – Time-Limited Aging Analyses and Boraflex" attach. 1, at 3-4 (May 16, 2008) (Attach. 12).

30. As indicated in LRA Tables 4.3.13 and 4.3-14, the projected CUF_{en} values for the three NUREG/CR-6260 reactor vessel locations were less than 1.0 for both IP2 and IP3. These three reactor vessel locations included the (1) bottom head to shell, (2) reactor vessel inlet nozzle, and (3) reactor vessel outlet nozzle. In addition, the IP2 (but not IP3) pressurizer surge line nozzle had a CUF_{en} less than 1.0. Accordingly, because these CUF_{en} values fell below 1.0, the EAF analyses for those components were projected to the end of the PEO (in the LRA) in accordance with Section 54.21(c)(1)(ii) and no further action was necessary. LRA at 4.3-22.

31. The component locations in LRA Tables 4.3-13 and 4.3-14 without projected CUF_{en} values less than 1.0 include the pressurizer surge line piping for IP2 and IP3; the RCS piping charging system nozzles for IP2 (for which a CUF was available) and for IP3; and the pressurizer surge line nozzles for IP3.[2] LRA tbls. 4.3.13 & 4.3-14. To address these locations, Entergy committed to take one of the following actions: (1) refine the fatigue analyses, at least two years before entering the PEO, to determine valid CUF_{en} values less than 1.0; (2) manage the effects of aging due to fatigue at the affected locations by an inspection program that has been reviewed and approved by the NRC; or (3) repair or replace

[2] As explained above and in LRA Section 4.3.3, the IP2 pressurizer surge nozzle had a CUF_{en} less than 1.0, whereas the IP3 pressurizer surge nozzle had CUF_{en} greater than 1.0, because the IP3 surge nozzle calculation includes the effects of the insurges/outsurges experienced by these nozzles, while the IP2 analysis did not include these effects. As a result, Entergy committed to re-analyze the pressurizer surge line nozzle for both units to include insurge/outsurge and environmental effects. LRA at 4.3-21 to 4.3-22.

the affected locations before exceeding CUF of 1.0 (collectively referred to as "Commitment 33"). *Id.* at 4.3-22 to 4.3-23.

32. Subsequently, in January 2008, as a part of LRA Amendment 2, Entergy amended the Fatigue Monitoring Program to provide information on cycle counting and the methodology used to determine stresses and fatigue usage, including environmental effects, in accordance with the NRC-endorsed ASME Code. NL-08-021, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "License Renewal Application Amendment 2," (Jan. 22, 2008) ("LRA Amendment 2") (Attach. 10). Entergy also amended the LRA to place Commitment 33 within the scope of the Fatigue Monitoring Program, by stating that it will use that AMP to manage the effects of reactor water environment on fatigue life, in accordance with § 54.21(c)(1)(iii). *Id.* Pursuant to revised Commitment 33, if Entergy does not demonstrate valid CUF_{en} values below 1.0 (Option 1) by conducting refined CUF_{en} analyses, then it must implement Option 2 of the commitment. Option 2 requires Entergy to repair or replace the affected locations before their refined CUF_{en}s exceed 1.0, consistent with the Fatigue Monitoring Program. *See* Attach. 10, attach. 2, at 15.

B. <u>Summary of the NRC Staff's Safety Evaluation Report Findings</u>

33. As documented in its final Safety Evaluation Report ("SER"), the NRC Staff, after conducting a detailed technical review of the LRA and performing related site audits, found that IPEC's Fatigue Monitoring Program includes acceptable program elements that are consistent with criteria in GALL Report Section X.M1. 2 SER at 3-81. The NRC Staff further concluded that Entergy has demonstrated that the effects of aging will be adequately managed so that the component intended functions will be maintained consistent with the CLB for the PEO, as required by 10 C.F.R. § 54.21(a)(3). *Id.* In addition to evaluating IPEC's Fatigue Monitoring Program, the Staff reviewed LRA Section 4.3.3 regarding EAF.

As part of its review, the NRC Staff confirmed that IPEC had correctly determined the environmental factors used as inputs for calculating F_{en} factors. It also found that Commitment 33 is consistent with 10 C.F.R. § 54.21(a)(1)(iii). *Id.* at 4-43 to 4-45.

C. Overview of the 2010 Westinghouse Environmental Fatigue Evaluations

34. To meet Commitment 33 and address the admitted contentions, Entergy retained Westinghouse Electric Company LLC ("Westinghouse") in 2008 to perform refined fatigue analyses to determine environmentally-adjusted CUFs. Westinghouse completed the refined fatigue analyses in late June 2010. Westinghouse Electric Co., WCAP-17199-P, Rev. 0, *Environmental Fatigue Evaluation for Indian Point Unit 2* (June 2010) (proprietary) (Attach. 15); Westinghouse Electric Co., WCAP-17200-P, Rev. 0, *Environmental Fatigue Evaluation for Indian Point Unit 3* (June 2010) (proprietary) (Attach. 16). Entergy approved the analyses on July 29, 2010, and shortly thereafter, notified the Staff of the results of the refined EAF analyses. NL-10-082, Letter from Fred R. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "License Renewal Application – Completion of Commitment #33 Regarding the Fatigue Monitoring Program" (Aug. 9, 2010) (Attach. 14).

35. As documented in WCAP-17199-P and WCAP-17200-P, and discussed further below, Westinghouse performed 60-year EAF evaluations using F_{en} factors calculated as described in NUREG/CR-5704 for stainless steel components and in NUREG/CR-6583 for those components containing carbon and low-alloy steel components. Westinghouse calculated the F_{en} factors, which it then applied to the ASME Code fatigue results.

36. As documented in WCAP-17199-P and WCAP-17200-P, the Westinghouse evaluations included the following four major steps: (1) determination of the limiting components for each location; (2) determination of the transients for each limiting component; (3) performance of ASME Code Section III stress and fatigue evaluations for

each limiting component; and (4) determination of the CUF_{en}, for each limiting component. Attach. 15, at 2-2 to 2-4; Attach. 16, at 2-2 to 2-4. As necessary, plant-specific projections of numbers of accrued cycles at the end of the PEO replaced conservative assumptions for numbers of cycles in the analyses. Attach. 15, at 2-4; Attach. 16, at 2-4. The fatigue evaluations followed the procedures given in ASME Code Section III, NB-3200. Attach. 15, at 5-20; Attach. 16, at 5-20.

37. Westinghouse developed transients using plant data and/or Westinghouse standard design transients, assuming the total period of operation to be 60 years. For the charging, safety injection, and RHR locations selected, Westinghouse developed stress inputs to the fatigue analyses using detailed finite element analysis models, and ran those models to create unit load transfer functions for each type of mechanical and thermal transient condition loading considered in the analyses. It also developed piping mechanical load functions to produce time history moment loads as a function of system parameters during a transient. The transfer functions and load functions were used in a proprietary Westinghouse computer code to determine detailed stress histories for each applicable transient, considering all applicable mechanical and thermal transient loads during each transient, and to calculate fatigue usage.

38. Section 5 of each Westinghouse EAF report details the manner in which Westinghouse evaluated EAF for the NRC-specified component locations at IP2 and IP3. As noted therein, Westinghouse's evaluations were based upon CUF analyses using transients that envelop the 60-year cycle projections. Sections 5.1 and 5.2, in particular, describe Westinghouse F_{en} calculations using the methodologies set forth in NUREG/CR-5704 for stainless steel and NUREG/CR-6583 for carbon steel.

39. The 60-year fatigue results for the critical component locations are provided in Tables 5-8 through 5-14 of WCAP-17199-P and WCAP-17200-P. Westinghouse determined that, for IP2 and IP3, the refined CUF_{en} values for pressurizer surge line piping, RCS piping charging system nozzle, RCS piping safety injection nozzle, and RHR Class 1 piping all are below 1.0 when projected to the end of the PEO. *See* Attach. 14, attach. 1, at 2-4; Attach. 15, at 6-1; Attach. 16, at 6-1.

IV. ISSUES RAISED IN NYS-26/26A AND RIVERKEEPER TC-1/1A

40. I understand that, as admitted by the Board, NYS-26/26A asserts that: (1) the LRA is incomplete without the calculations of the CUF_{en} values as threshold values necessary to assess the need for an AMP; (2) Entergy's AMP is inadequate for lack of final CUF_{en} values; and (3) the LRA must specify actions to be carried out by Entergy during the PEO to manage the aging of key reactor components susceptible to metal fatigue. *Entergy Nuclear Operations, Inc.* (Indian Point Nuclear Generating Units 2 & 3), LBP-08-13, 68 NRC 43, 116 (2008).

41. I have reviewed NYS-26/26A and NYS's supporting arguments. *See* New York State's Notice of Intention to Participate and Petition to Intervene at 227-233 (Nov. 30, 2007) ("NYS Petition"); Declaration of Dr. Richard T. Lahey, Jr. [in Support of New York State Contention 26], at 8-10 (Nov. 30, 2007); New York State Reply in Support of Petition to Intervene, at 124-130 (Feb. 22, 2008); Petitioner State of New York's Request for Admission of Supplemental Contention No. 26-A (Metal Fatigue) (Apr. 7, 2008); Declaration of Dr. Richard T. Lahey, Jr. in Support of the State of New York's Supplemental Contention 26-A (Apr. 7, 2008). I further understand that NYS alleges that Entergy, by not assessing certain projected CUF_{en} values exceeding 1.0, has not adequately shown that the TLAAs for metal fatigue are valid for the PEO. NYS further argues that Entergy has not provided sufficient

details concerning the analytical methods and assumptions to be used to calculate revised CUF_{en} values.

42. As admitted by the Board, I understand that Riverkeeper TC-1/1A asserts that issues exist relative to: (1) the extent to which an applicant must expand the scope of its TLAAs to meet the recommendations of the GALL Report and NUREG/CR-6260; (2) the extent, if any, that refinement of the CUF_{en} values is a valid corrective action and what relationship it has to the repair and replacement options; (3) the scope of the commitments needed to monitor, manage, and correct age-related degradation to meet NRC regulations; and (4) the degree of detail and specificity with which the repair or replacement decision criteria must be defined. *Indian Point,* LBP-08-13, 68 NRC at 167-68.

43. I have reviewed Riverkeeper TC-1/1A and its supporting arguments. *See* Riverkeeper, Inc.'s Request for Hearing and Petition to Intervene in Indian Point License Renewal Proceedings at 7-15 (Nov. 30, 2007) ("Riverkeeper Petition"); Declaration of Dr. Joram Hopenfeld in Support of Riverkeeper's Contentions TC-1 and TC-1 (Nov. 28, 2007); Riverkeeper, Inc.'s Reply to Entergy's and the NRC Staff's Responses to Hearing Request and Petition to Intervene, at 2-12 (Feb. 15, 2008); Riverkeeper, Inc.'s Request for Admission of Amended Contention (Mar. 5, 2008). Riverkeeper alleges, in principal part, that:

- Entergy's TLAA is "facially non-compliant" with 10 C.F.R. § 54.21(c)(1)(i)-(ii) because four representative components of the RCS have projected CUF_{en} values greater than 1.0. Entergy cannot avoid the "legal requirement" that the LRA is "required to demonstrate" under § 54.21(c)(1)(iii) that CUF_{en} values are less than 1.0. Any refined CUF_{en} analyses cannot be deferred until after a renewed license is granted. Riverkeeper Petition at 7-13.

- Entergy must submit a list of all components with CUF larger than unity, as well as an AMP that includes "clear criteria for determining when a defect in any one of these components is acceptable, when it is acceptable but requires monitoring, and when it is unacceptable and requires repairs." *Id.* at 13.

- Entergy must "broaden its TLAA analysis" beyond the scope of the representative components identified in Tables 4.3-13 and 4.3-14 to identify other components whose CUF may be greater than one. *Id.* at 7.

- Entergy's list of components with CUF_{en}s less than 1.0 in LRA Tables 4.3-13 and 4.3-14 is incomplete because Entergy's methods and assumptions for identifying those components are "unrealistic and inadequate." *Id.* at 7, 14. Entergy used an unrealistically low number of 2.45 for an F_{en} (instead of an F_{en} of 17); relied on the "CUF of Record" (40 years) instead of projecting the number of cycles to 60 years; and failed to calculate several NUREG-CR/6260 limiting locations because they are designed to ANSI B31.1, despite the availability of "generic CUF values" from NUREG/CR-6260. *Id.*

V. RESPONSE TO ISSUES RAISED IN CONTENTIONS NYS-26/26A AND RIVERKEEPER TC-1/1A

44. As a technical matter, NYS's and Riverkeeper's claim that Entergy's approach to addressing EAF is "vague" or "inadequate" is unfounded. The specific methodology and assumptions used by Westinghouse to determine refined CUF_{en} values for the relevant IPEC components are fully documented in WCAP-17199-P and WCAP-17200-P. The refined CUF_{en} analyses were performed in accordance with Westinghouse's Quality Assurance ("QA") Program, as approved by Entergy, and included design input verification and independent reviews to ensure that valid assumptions, transients, cycles, external loadings, analysis methods, and F_{en} factors were used in the EAF analyses for IP2 and IP3.

45. In accordance with Commitment 33, and at Entergy's request, Westinghouse prepared detailed stress models of the (1) surge line hot leg nozzle, (2) pressurizer surge nozzle, (3) RCS piping charging system nozzle, (4) RCS piping safety injection nozzle, and (5) RHR system class 1 piping locations for each unit using standard methods of the ASME Code, Section III.[3] As discussed above and in LRA Section 4.3.3, and reflected in LRA

[3] The pressurizer surge line nozzles were evaluated for environmental fatigue to address the surge line locations identified in NUREG/CR-6260. As explained in the Westinghouse's refined EAF analyses, the IP2 and IP3 surge lines were previously evaluated (in WCAP-12937) for the effects of thermal stratification and plant-specific transients. For IP2 and IP3, the controlling fatigue location was the surge line weld to the

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Tables 4.3.13 and 4.3.14, Entergy previously applied bounding F_{en} values to the CUFs for the bottom head to shell region, the reactor vessel inlet nozzle, and the reactor vessel outlet nozzle and determined that the projected CUF_{en} values for these three locations are all below 1.0 for both units Therefore, refined CUF_{en} analyses were not necessary for these three reactor vessel locations. LRA at 4.3-21, 4.3-24 to 4.3-25; Attach. 14, attach. 1, at 2-4). As the Staff concluded in its SER, "the analyses performed for these components were projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii)." 2 SER at 4-42.

46. Westinghouse developed detailed stress history inputs for all transients relevant to fatigue of the affected component locations enumerated above and used those inputs in the EAF evaluations for those five locations As explained previously, these ASME Code evaluations were performed for the piping components to reduce excess conservatisms in the analyses, or because the CLB qualification for the component was to the ANSI B31.1 Power Piping Code, which did not require a fatigue usage factor (*i.e.*, CUF) calculation at the time of plant design.

47. In determining the refined CUF_{en} values for the component locations listed above, Westinghouse used F_{en} factors calculated using the NRC-approved methodology in NUREG/CR-5704 for the stainless steels in the pressurizer surge line, the reactor coolant piping charging and safety injection system nozzles, and the RHR system Class 1 piping. In addition, Westinghouse applied F_{en} factors calculated using the NRC-approved methodology in NUREG/CR-6583 for the carbon steel associated with the pressurizer surge nozzle

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pressurizer surge nozzle in WCAP-12937. The surge line hot leg nozzle was the location evaluated in NUREG/CR-6260. Therefore, the IP2 and IP3 pressurizer surge nozzles and surge line hot leg nozzles were both evaluated to account for EAF effects. Attach. 15, at 2-1, 3-1; Attach. 16, at 2-1, 3-1.

dissimilar metal weld. The F_{en} factors were calculated using the detailed inputs described in the applicable NUREG and were applied directly to the ASME Code fatigue results. Attach. 15, at 1-2; Attach. 16, at 1-2. This approach is consistent with that described in both the SRP-LR and GALL Report.

48. There is no technical basis for Riverkeeper's claim that Entergy must use F_{en} values derived from NUREG/CR-6909, *Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials* (Feb. 2007). As noted above, Entergy's use of the formulae for calculating refined CUF_{en}s in NUREG/CR-6583 and NUREG/CR-5704 is consistent with NRC license renewal guidance. Additionally, use of the F_{en} factors derived from NUREG/CR-6909, as suggested by Riverkeeper, would actually yield less conservative CUF_{en} values, because the ASME Code design air curves for carbon steel and low-alloy steels contained in air in NUREG/CR-6583 and NUREG/CR-5704 are more conservative than the newer air curves in NUREG/CR-6909, as noted on page 81 of the latter document.

49. Riverkeeper's assertion that Entergy improperly relied on the "CUF of Record" (*i.e.*, 40 years) instead of projecting the number of cycles to 60 years is incorrect. As discussed above and clearly stated in LRA Section 4.3, Entergy projected the numbers of cycles accrued to date to determine the numbers of cycles expected at the end of 60 years of operation. LRA at 4.3-2. Furthermore, in its refined EAF analyses, Westinghouse performed 60-year fatigue analyses. Attach. 15 at 2-1, 6-1; Attach. 16 at 2-1, 6-1.

50. In the case of IPEC-specific components designed to ANSI B31.1, Entergy did not improperly refuse to use "generic CUF values." No such requirement exists. In any case, Westinghouse determined plant-specific CUF_{en} values for both IP2 and IP3 ANSI B31.1 piping that, without exception, are less than 1.0.

51. Contrary to Riverkeeper's claim, there is no requirement or need for Entergy to "broaden" its EAF analyses beyond the components identified in LRA Tables 4.3-13 and 4.3-14. As noted above, the refined CUF_{en} values for the critical locations listed in those tables, corresponding to the NUREG/CR-6260 locations, all are less than the ASME code limit of 1.0. Therefore, no further analyses are required for purposes of assessing EAF in the context of an LRA. As NUREG/CR-6260 explains, the six component locations identified therein "were chosen to give a representative overview of components that had higher CUFs and/or were important from a risk perspective." Attach. 9, at 4-1. Similarly, MRP-47, Revision 1 (which NYS and Riverkeeper reference multiple times) states that the six NRC-specified locations "were considered representative enough that the effects of LWR environment on fatigue could be assessed." Attach. 6, at 3-4. It further states: "For cases where acceptable fatigue results are demonstrated for these locations for 60 years of plant operation including environmental effects, additional evaluations or locations need not be considered." *Id.* (emphasis added).

52. Riverkeeper alleges that if the CUF_{en}s exceed 1.0, the second part of Commitment 33, or Option 2 (*i.e.*, repair or replace the affected locations), is too vague to meet the demonstration requirement of Part 54. As an initial matter, because the refined CUF_{en} values are less than 1.0 when projected to the end of the PEO (*i.e.*, 60 years of plant operation), there is no present need for corrective actions. Regardless, the second portion of Commitment 33 is consistent with the GALL Report and 10 C.F.R. § 54.21(c)(1)(iii), as the NRC Staff concluded in its SER. 2 SER at 4-44 to 4-47. GALL Report Section X.M1 states, in relevant part: "Acceptable corrective actions include repair of the component, replacement of the component, and a more rigorous analysis of the component to demonstrate that the

design code limit will not be exceeded during the extended period of operation." Attach. 5 at X M-2.

53. During the PEO, Entergy will continue to monitor the condition of piping and components at the locations of interest under IPEC's ISI Program, which includes periodic visual, surface, and volumetric examination of Classes 1, 2, and 3 pressure-retaining components, their attachments, and supports. If a flaw is detected during an ISI Program inspection or by other means, then the component may be replaced or repaired, or evaluated for continued service in accordance with the criteria contained in ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components." Acceptance standards for examination evaluations, repair procedures, inservice test requirements, and replacements for ASME Class 1 components are defined in ASME Section XI paragraphs IWB-3000, IWB-4000, IWB-5000 and IWB-7000, respectively.

54. Under the Fatigue Monitoring Program, Entergy will manage the effects of fatigue by monitoring cycles incurred and ensuring they do not exceed the analyzed numbers of cycles, such that the CUF_{en} analyses remain valid. LRA at 4.3-2 to 4.3-3; 2 SER at 4-44 to 4-45; NL-08-084, Attach. 12, attach. 1, at 3-4.

55. As required by the Fatigue Monitoring Program, Entergy tracks actual plant transients and evaluates these against the design transients. LRA at 4.3-2 to LRA 4.3-3; SER at 4-45; Attach. 10, attach. 1, at 6); Attach. 12, attach. 1, at 4. The plant transient counts will be updated at least once each operating cycle, which is an acceptable frequency because the evaluation during each update determines if the number of design transients could be exceeded prior to the next update. Attach. 12, attach. 1, at 4); SER at 3-79, 4-44.

56. Consistent with GALL Report Section X.M1, the Fatigue Monitoring Program requires that corrective actions be implemented in accordance with the IPEC Corrective

Action Program before the plant exceeds the analyzed number of transient cycles. Attach. 10, attach. 1, at 5-6; Attach. 12, attach. 1, at 4); SER at 4-44 to 4-45. IPEC procedures contain specific "alert levels" that trigger the initiation of corrective actions under the Fatigue Monitoring Program. SER at 4-44. Specifically, IPEC calculates alert levels by adding twice the number of cycles that occurred in the last fuel cycle to the total number of cycles to date. Corrective action is initiated if this alert level exceeds the number of analyzed transients. If the number of cycles is projected to remain at or below the analyzed level for two additional fuel cycles, then no corrective action is required. *Id.*. ; Attach. 10, attach. 1, at 5-6; Attach. 12, attach. 1, at 4.

57. Any further reanalysis (*i.e.*, future analysis updates), if necessary, would be governed by applicable QA procedures, as discussed above. Attach. 10, attach. 1, at 6; Attach. 12, attach. 1, at 4. Any repair or replacement of a component, if necessary, would be done in accordance with established plant procedures that must comply with Entergy's QA program and meet the applicable repair or replacement requirements of ASME Code Section XI. Attach. 10, attach. 1, at 6; Attach. 12, attach. 1, at 4. SER Sections 3.0.3.2.6, 4.1 and 4.3 discuss the IPEC Fatigue Monitoring Program in detail. 2 SER at 3-78 to 3-81, 4-19 to 4-39, 4-41 to 4-47.

58. Accordingly, Entergy's ISI Program, Fatigue Monitoring Program, and related ASME Code rules provide detailed and sufficient information concerning IPEC's future monitoring and corrective action activities related to environmentally-assisted fatigue during the PEO.

In accordance with 28 U.S.C. § 1746, I declare under penalty of perjury that the foregoing is true and correct.

Executed on AUGUST 20TH, 2010.

Nelson F. Azevedo

NYS-26/26A & RIVERKEEPER TC-1/1A: ATTACHMENT 3

Nelson Azevedo

Entergy, Indian Point Energy Center

Supervisor, Code Programs — January 2001 to present

Responsible for the Engineering Section that implements the American Society of Mechanical Engineers (ASME) Code programs, including Inservice Inspection (ISI), Inservice Testing (IST), Flow Accelerated Corrosion (FAC), Snubber Testing, Boric Acid Corrosion Control, Non-destructive Examination (NDE), Fatigue Monitoring, Steam Generators, Buried Piping, Alloy 600 cracking, Reactor Vessel Embrittlement, Welding and 10CFR50, Appendix J containment leakage program. Also responsible to ensure compliance with the ASME XI Code requirements for all repair and replacement activities at IPEC. Represent IPEC at industry organizations including the PWR Owners Group Management Committee.

Northeast Utilities

Department Manager, Materials Eng. & Code Programs — Jan. 1999 to Jan. 2001

Managed five Sections, with a staff of 38 engineers and other technical support staff responsible for implementation of engineering programs and structural integrity issues, at the Millstone Station. Examples of these programs include the ASME XI Inservice Inspection (ISI), Inservice Testing (IST), Non-Destructive Examinations (NDE), Steam Generator structural integrity (NEI 97-06), Reactor Vessel Embrittlement Management, Flow Accelerated Corrosion (FAC), Paints and Coatings, Fatigue Management and Welding programs. Represented the Seabrook and Millstone Units in the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) Senior Representatives Committee and in other ASME XI and Industry organizations. Responsible for development and management of department budgets, resources and implementation of policies and procedures. Part of the management team that recovered the Millstone Units after they were shutdown for safety and regulatory non-compliance issues.

Supervisor, Structural and Design Engineering — Sept. 1993 to Jan. 1999

Supervised a Section responsible for ensuring compliance with ASME XI, ASME III and ANSI B31.1 Code requirements. Technical areas of responsibility included flaw evaluations, piping stress analyses, ASME III piping and component fatigue evaluations, leak before break (LBB) implementation, finite element analysis, civil structures, FAC program and Reactor Vessel embrittlement. Responsible for structural integrity of large components including Low Pressure turbines, reactor vessels and steam generators as well as implementation of the A-46, seismic verification program for older units. Represented Seabrook, Connecticut Yankee and Millstone Units in several industry organizations including BWR Vessel Internals Project (BWRVIP), ASME Section XI and EPRI.

Engineer/Senior Engineer March 1982 to Sept. 1993

Responsible for resolving a variety of structural and equipment performance issues within the NU, Power Generation Division, which supported the Nuclear, Fossil and Hydro generating stations. Some of the issues included both high pressure and low pressure steam turbine creep and cracking issues, boiler tube failures, pump and valve performance issues and pipe support issues. Also responsible for performing pipe stress analyses, finite element analysis of large components, ASME XI Flaw evaluations, BWR intergranular stress corrosion cracking (IGSCC) evaluations, and reactor vessel structural integrity and embrittlement issues. Performed ASME III, Class 1 fatigue analyses, developed 10CFR50, Appendix G heat up and cooldown curves, designed weld overlays to mitigate IGSCC and performed upper shelf energy analysis for Connecticut Yankee to respond to GL 92-01 issues. Responsible for implementation of several large projects, including steam generator tube sleeving and plugging at Millstone 2 and the Connecticut Yankee FAC program following pipe ruptures. Active participant in the industry resolution of Pressurized Thermal Shock issues (PTS rule) and implementation of the LBB methodology that was published by the NRC in NUREG-1061. A member of the EPRI team that developed the SAFER computer program, which was designed to perform run/repair/retire evaluations of degraded High and Low Pressure turbines.

Education

BS, Mechanical and Materials Engineering
University of Connecticut

MS, Mechanical Engineering
Rensselaer Polytechnic Institute

MBA, General Management
Rensselaer Polytechnic Institute

NYS-26/26A & RIVERKEEPER TC-1/1A: ATTACHMENT 4

NUREG-1800, Rev. 1

Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants

Manuscript Completed: September 2005
Date Published: September 2005

Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

3.0 INTRODUCTION TO STAFF REVIEW OF AGING MANAGEMENT

The NRC project manager (PM) responsible for the safety review of the license renewal application (LRA) is responsible for assigning to appropriate NRC Office of Nuclear Reactor Regulation (NRR) divisions the review or audit of aging management reviews (AMRs) or aging management programs (AMPs) identified in the applicant's LRA. The PM should document to which organization each AMR or AMP is assigned. The assigned AMRs and AMPs should be reviewed per the criteria described in Sections 3.1 through 3.6 of this standard review plan (SRP-LR, NUREG-1800) for review of license renewal applications, as directed by the scope of each of these sections.

The NRC divisions that are usually assigned responsibility for the review of AMRs and AMPs are the Division of Engineering (DE), Division of System Safety Analysis (DSSA), and the Division of Regulatory Improvement Program (DRIP) License Renewal and Environmental Impacts Program (RLEP). Typically, the PM will assign DRIP/RLEP to review the AMRs and AMPs that the LRA identifies as being consistent with the GALL Report or NRC-approved precedents. As common exceptions to this assignment, the PM will assign to DE those AMRs and AMPs that address issues identified as emerging technical issues. Usually, AMRs and AMPs that are not in one of the aforementioned categories are assigned to DE.

Review of the AMPs requires assessment of ten program elements as defined in this SRP-LR. The NRC divisions assigned the AMP should review the ten program elements to verify their technical adequacy. For three of the ten program elements (corrective actions, confirmation process, and administrative controls) the NRC division responsible for quality assurance should verify that the applicant has documented a commitment in the FSAR Supplement to expand the scope of its 10 CFR Part 50, Appendix B program to address the associated program elements for each AMP. If the applicant chooses alternate means of addressing these three program elements (e.g., use of a process other than the applicant's 10 CFR Part 50, Appendix B program), the NRC divisions assigned to review the AMP should request that the Division responsible for quality assurance review the applicant's proposal on a case-by-case basis.

3.0.1 Background on the Types of Reviews

10 CFR 54.21(a)(3) requires that the LRA must demonstrate, for systems, structures, and components (SSCs) identified in the scope of license renewal and subject to an AMR pursuant to 10 CRF 54.21(a)(1), that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis (CLB) for the period of extended operation. This AMR consists of identifying the material, environment, aging effects, and the AMP(s) credited for managing the aging effects.

Sections 3.1 through 3.6 of this SRP-LR describe how the AMRs and AMPs are reviewed. One method that the applicant may use to conduct its AMRs is to satisfy the NUREG-1801 (GALL Report) recommendations. The applicant may choose to use methodology other than that in the GALL Report to demonstrate compliance with 10 CFR 54.21(a)(3).

As stated in the GALL Report:

> The GALL Report is a technical basis document to the SRP-LR, which provides the staff with guidance in reviewing a license renewal application. The GALL Report should be treated in the same manner as an approved topical report that is generically applicable. An applicant may reference the GALL Report in a license renewal application to

demonstrate that the programs at the applicant's facility correspond to those reviewed and approved in the GALL Report and that no further staff review is required, as described in the next paragraph. If the material presented in the GALL Report is applicable to the applicant's facility, the staff should find the applicant's reference to the GALL Report acceptable. In making this determination, the staff should consider whether the applicant has identified specific programs described and evaluated in the GALL Report. The staff, however, should not conduct a re-review of the substance of the matters described in the GALL Report. Rather, the staff should ensure that the applicant verifies that the approvals set forth in the GALL Report for generic programs apply to the applicant's programs. The focus of the staff review should be on augmented programs for license renewal. The staff should also review information that is not addressed in the GALL Report or is otherwise different from that in the GALL Report.

If an applicant takes credit for a program in the GALL Report, it is incumbent on the applicant to ensure that the plant program contains all the elements of the referenced GALL Report program. In addition, the conditions at the plant must be bounded by the conditions for which the GALL Report program was evaluated. The above verifications must be documented on-site in an auditable form. The applicant should include a certification in the license renewal application that the verifications have been completed and are documented on-site in an auditable form.

The GALL Report contains one acceptable way to manage aging effects for license renewal. An applicant may propose alternatives for staff review in its plant-specific license renewal application. Use of the GALL Report is not required, but its use should facilitate both preparation of a license renewal application by an applicant and timely, uniform review by the NRC staff.

In addition, the GALL Report does not address scoping of structures and components for license renewal. Scoping is plant-specific, and the results depend on the plant design and current licensing basis. The inclusion of a certain structure or component in the GALL Report does not mean that this particular structure or component is within the scope of license renewal for all plants. Conversely, the omission of a certain structure or component in the GALL Report does not mean that this particular structure or component is not within the scope of license renewal for any plants.

The GALL Report contains an evaluation of a large number of structures and components that may be in the scope of a typical LRA. The evaluation results documented in the GALL Report indicate that many existing, typical generic aging management programs are adequate to manage aging effects for particular structures or components for license renewal without change. The GALL Report also contains recommendations on specific areas for which generic existing programs should be augmented (require further evaluation) for license renewal and documents the technical basis for each such determination. In addition, the GALL Report identifies certain SSCs that may or may not be subject to particular aging effects, and for which industry groups are developing generic aging management programs or investigating whether aging management is warranted. To the extent the ultimate generic resolution of such an issue will need NRC review and approval for plant-specific implementation, as indicated in a plant-specific FSAR supplement, and reflected in the SER associated with a particular LR application, an amendment pursuant to 10 CFR 50.90 will be necessary.

In this SRP-LR, Subsection 3.X.2 (where X denotes number 1-6) presents the acceptance criteria describing methods to determine whether the applicant has met the requirements of NRC's regulations in 10 CFR 54.21. Subsection 3.X.3 presents the review procedures to be followed. Some rows (line-items) in the AMR tables (in Chapters II through VIII of the GALL Report, Vol. II) establish the need to perform "further evaluations." The acceptance criteria for satisfying these "further evaluations" are found in Subsections 3.X.2.2. The related review procedures are provided in Subsections 3.X.3.2.

In Regulatory Guide 1.188, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses," the NRC has endorsed an acceptable methodology for applicants to structure license renewal applications. Using the guidance described in the aforementioned Regulatory Guide, the applicant documents in the LRA whether its AMR line-item is consistent or not consistent with the GALL Report.

A portion of the AMR includes the assessment of the AMPs in the GALL Report. The applicant may choose to use an AMP that is consistent with the GALL Report AMP, or may choose a plant-specific AMP.

If a GALL Report AMP is selected to manage aging, the applicant may take one or more exceptions to specific GALL Report AMP program elements. However, any deviation or exception to the GALL Report AMP should be described and justified. Exceptions are portions of the GALL Report AMP that the applicant does not intend to implement.

In some cases, an applicant may choose an existing plant program that does not currently meet all the program elements defined in the GALL Report AMP. If this is the situation, the applicant may make a commitment to augment the existing program to satisfy the GALL Report AMP element prior to the period of extended operation. This commitment is an AMP enhancement.

Enhancements are revisions or additions to existing aging management programs that the applicant commits to implement prior to the period of extended operation. Enhancements include, but are not limited to, those activities needed to ensure consistency with the GALL Report recommendations. Enhancements may expand, but not reduce, the scope of an AMP.

An audit and review is conducted at the applicant's facility to evaluate those AMRs or AMPs that the applicant claims to be consistent with the GALL Report. An audit also includes technical assessments of exceptions or enhancements to the GALL Report AMP program elements. Reviews are performed to address those AMRs or AMPs related to emergent issues, stated to be not consistent with the GALL Report, or based on an NRC-approved precedent (e.g., AMRs and AMPs addressed in an NRC SER of a previous LRA). As a result of the criteria established in 10 CFR Part 54, and the guidance provided in SRP-LR, GALL Report, Regulatory Guide 1.188, and the applicant's exceptions and/or enhancements to a GALL Report AMP, the following types of AMRs and AMPs should be audited or reviewed by the NRC staff.

AMRs

- AMR results consistent with the GALL Report
- AMR results for which further evaluation is recommended by the GALL Report
- AMR results not consistent with or not addressed in the GALL Report

AMPs

- Consistent with GALL Report AMPs

- Plant-specific AMPs

FSAR Supplement

- Each LRA AMP will provide an FSAR Supplement which defines changes to the FSAR that will be made as a condition of a renewed license. This FSAR Supplement defines the aging management programs the applicant is crediting to satisfy 10 CFR 54.21(a)(3).
- The FSAR Supplement should also contain a commitment to implement the LRA AMP enhancement prior to the period of extended operation.

3.0.2 Applications with approved Extended Power Uprates

Extended power uprates (EPU) are licensing actions that some licensees have recently requested the NRC staff to approve. This can affect aging management. In a NRC staff letter to the Advisory Committee on Reactor Safeguards, dated October 26, 2004, (ADAMS Accession ML042790085), the NRC Executive Director for Operation states that, "All license renewal applications with an approved EPU will be required to perform an operating experience review and its impact on [aging] management programs for structures, and components before entering the period of extended operation." One way for an applicant with an approved EPU to satisfy this criterion is to document its commitment to perform an operating experience review and its impact on aging management programs for systems, structures, and components (SSCs) before entering the period of extended operation as part of its license renewal application. Such licensee commitments should be documented in the NRC staff's SER written in support of issuing a renewed license. The staff expects to impose a license condition on any renewed license to ensure that the applicant will complete these activities no later than the committed date. EPU impact on SSCs should be part of the license renewal review. If necessary, the PM will assign a responsible group to address EPU.

4.3 METAL FATIGUE ANALYSIS

Review Responsibilities

Primary - Branch responsible for the TLAA issues
Secondary - None

4.3.1 Areas of Review

A metal component subjected to cyclic loading at loads less than the static design load may fail because of fatigue. Metal fatigue of components may have been evaluated based on an assumed number of transients or cycles for the current operating term. The validity of such metal fatigue analysis is reviewed for the period of extended operation.

The metal fatigue analysis review includes, as appropriate, a review of in service flaw growth analyses, reactor vessel underclad cracking analysis, reactor vessel internals fatigue analysis, postulated high energy line break, leak-before-break, RCP flywheel, and metal bellows.

4.3.1.1 Time-Limited Aging Analysis

Metal components may be designed or analyzed based on requirements in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code or the American National Standards Institute (ANSI) guidance. These codes contain explicit metal fatigue or cyclic considerations based on TLAAs.

4.3.1.1.1 ASME Section III, Class 1

ASME Class 1 components, which include core support structures, are analyzed for metal fatigue. ASME Section III (Ref. 1) requires a fatigue analysis for Class 1 components that considers all transient loads based on the anticipated number of transients. A Section III Class 1 fatigue analysis requires the calculation of the "cumulative usage factor" (CUF) based on the fatigue properties of the materials and the expected fatigue service of the component. The ASME Code limits the CUF to a value of less than or equal to one for acceptable fatigue design. The fatigue resistance of these components during the period of extended operation is an area of review.

4.3.1.1.2 ANSI B31.1

ANSI B31.1 (Ref. 2) applies only to piping. It does not call for an explicit fatigue analysis. It specifies allowable stress levels based on the number of anticipated thermal cycles. The specific allowable stress reductions due to thermal cycles are listed in Table 4.3-1. For example, the allowable stress would be reduced by a factor of 1.0, i.e., no reduction, for piping that is not expected to experience more than 7,000 thermal cycles during plant service, but would be reduced to half of the maximum allowable static stress for 100,000 or more thermal cycles. The fatigue resistance of these components during the period of extended operation is an area of review.

4.3.1.1.3 Other Evaluations Based on CUF

The codes also contain metal fatigue analysis criteria based on a CUF calculation [the 1969 edition of ANSI B31.7 (Ref. 3) for Class 1 piping, ASME NC-3200 vessels, ASME NE-3200

Class MC components, and metal bellows designed to ASME NC-3649.4(e)(3), ND-3649.4(e)(3), or NE-3366.2(e)(3)]. For these components, the discussion relating to ASME Section III, Class 1 in Subsection 4.3.1.1.1 of this review plan section applies.

4.3.1.1.4 ASME Section III, Class 2 and 3

ASME Section III, Class 2 and 3 piping cyclic design requirements are similar to the guidance in ANSI B31.1. The discussion relating to B31.1 in Subsection 4.3.1.1.2 of this review plan section applies.

4.3.1.2 Generic Safety Issue

The fatigue design criteria for nuclear power plant components have changed as the industry consensus codes and standards have developed. The fatigue design criteria for a specific component depend on the version of the design code that applied to that component, i.e., the code of record. There is a concern that the effects of the reactor coolant environment on the fatigue life of components were not adequately addressed by the code of record.

The NRC has decided that the adequacy of the code of record relating to metal fatigue is a potential safety issue to be addressed by the current regulatory process for operating reactors (Refs. 4 and 5). The effects of fatigue for the initial 40-year reactor license period were studied and resolved under Generic Safety Issue (GSI)-78, "Monitoring of Fatigue Transient Limits for reactor coolant system," and GSI-166, "Adequacy of Fatigue Life of Metal Components" (Ref. 6). GSI-78 addressed whether fatigue monitoring was necessary at operating plants. As part of the resolution of GSI-166, an assessment was made of the significance of the more recent fatigue test data on the fatigue life of a sample of components in plants where Code fatigue design analysis had been performed. The efforts on fatigue life estimation and ongoing issues under GSI-78 and GSI-166 for 40-year plant life were addressed separately under a staff generic task action plan (Refs. 7 and 8). The staff documented its completion of the fatigue action plan in SECY-95-245 (Ref. 9).

SECY-95-245 was based on a study described in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components" (Ref. 10). In NUREG/CR-6260, sample locations with high fatigue usage were evaluated. Conservatisms in the original fatigue calculations, such as actual cycles versus assumed cycles, were removed, and the fatigue usage was recalculated using a fatigue curve considering the effects of the environment. The staff found that most of the locations would have a CUF of less than the ASME Code limit of 1.0 for 40 years. On the basis of the component assessments, supplemented by a 40-year risk study, the staff concluded that a backfit of the environmental fatigue data to operating plants could not be justified. However, because the staff was less certain that sufficient excessive conservatisms in the original fatigue calculations could be removed to account for an additional 20 years of operation for renewal, the staff recommended in SECY-95-245 that the samples in NUREG/CR-6260 should be evaluated considering environmental effects for license renewal. GSI-190, "Fatigue Evaluation of Metal Components for 60-year Plant Life," was established to address the residual concerns of GSI-78 and GSI-166 regarding the environmental effects on fatigue of pressure boundary components for 60 years of plant operation.

The scope of GSI-190 included design basis fatigue transients. It studied the probability of fatigue failure and its effect on core damage frequency (CDF) of selected metal components for 60-year plant life. The results showed that some components have cumulative probabilities of

crack initiation and through-wall growth that approach one within the 40- to 60-year period. The maximum failure rate (through-wall cracks per year) was in the range of 10^{-2} per year, and those failures were generally associated with high cumulative usage factor locations and components with thinner walls, i.e., pipes more vulnerable to through-wall cracks. In most cases, the leakage from these through-wall cracks is small and not likely to lead to core damage. It was concluded that no generic regulatory action is necessary and that GSI-190 is resolved based on results of probabilistic analyses and sensitivity studies, interactions with the industry (NEI and EPRI), and different approaches available to licensees to manage the effects of aging (Refs. 11 and 12).

However, the calculations supporting resolution of this issue, which included consideration of environmental effects, indicate the potential for an increase in the frequency of pipe leaks as plants continue to operate. Thus, the staff concluded that licensees are to address the effects of coolant environment on component fatigue life as aging management programs are formulated in support of license renewal.

The applicant's consideration of the effects of coolant environment on component fatigue life for license renewal is an area of review.

4.3.1.3 FSAR Supplement

Detailed information on the evaluation of TLAAs is contained in the renewal application. A summary description of the evaluation of TLAAs for the period of extended operation is contained in the applicant's FSAR supplement. The FSAR supplement is an area of review.

4.3.2 Acceptance Criteria

The acceptance criteria for the areas of review described in Subsection 4.3.1 of this review plan section delineate acceptable methods for meeting the requirements of the NRC's regulations in 10 CFR 54.21(c)(1).

4.3.2.1 Time-Limited Aging Analysis

Pursuant to 10 CFR 54.21(c)(1)(i) - (iii), an applicant must demonstrate one of the following:

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

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

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

Specific acceptance criteria for metal fatigue are:

4.3.2.1.1 ASME Section III, Class 1

For components designed or analyzed to ASME Class 1 requirements, the acceptance criteria, depending on the applicant's choice of 10 CFR 54.21(c)(1)(i), (ii), or (iii), are:

4.3.2.1.1.1 10 CFR 54.21(c)(1)(i)

The existing CUF calculations remain valid because the number of assumed transients would not be exceeded during the period of extended operation.

4.3.2.1.1.2 10 CFR 54.21(c)(1)(ii)

The CUF calculations have been reevaluated based on an increased number of assumed transients to bound the period of extended operation. The resulting CUF remains less than or equal to unity for the period of extended operation.

4.3.2.1.1.3 10 CFR 54.21(c)(1)(iii)

In Chapter X of the GALL report (Ref. 13), the staff has evaluated a program for monitoring and tracking the number of critical thermal and pressure transients for the selected reactor coolant system components. The staff has determined that this program is an acceptable aging management program to address metal fatigue of the reactor coolant system components according to 10 CFR 54.21(c)(1)(iii). The GALL report may be referenced in a license renewal application and should be treated in the same manner as an approved topical report. In referencing the GALL report, the applicant should indicate that the material referenced is applicable to the specific plant involved and should provide the information necessary to adopt the finding of program acceptability as described and evaluated in the report. The applicant should also verify that the approvals set forth in the GALL report for the generic program apply to the applicant's program.

4.3.2.1.2 ANSI B31.1

For piping designed or analyzed to B31.1, the acceptance criteria, depending on the applicant's choice of 10 CFR 54.21(c)(1)(i), (ii), or (iii), are:

4.3.2.1.2.1 10 CFR 54.21(c)(1)(i)

The existing fatigue strength reduction factors remain valid because the number of cycles would not be exceeded during the period of extended operation.

4.3.2.1.2.2 10 CFR 54.21(c)(1)(ii)

The fatigue strength reduction factors have been reevaluated based on an increased number of assumed thermal cycles and the stress reduction factors (e.g., Table 4.3-1) given in the applicant's code of record to bound the period of extended operation. The adjusted fatigue strength reduction factors are such that the component design basis remains valid during the period of extended operation.

4.3.2.1.2.3 10 CFR 54.21(c)(1)(iii)

The effects of aging on the intended function(s) will be adequately managed for the period of extended operation. The component could be replaced and the allowable stresses for the replacement will be sufficient as specified by the code during the period of extended operation.

Alternative acceptance criteria under 10 CFR 54.21(c)(1)(iii) have yet to be developed. They will be evaluated on a case-by-case basis to ensure that the aging effects will be managed such that the intended functions(s) will be maintained during the period of extended operation.

4.3.2.1.3 Other Evaluations Based on CUF

The acceptance criteria in Subsection 4.3.2.1.1 of this review plan section apply.

4.3.2.1.4 ASME Section III, Class 2 and 3

The acceptance criteria in Subsection 4.3.2.1.2 of this review plan section apply.

4.3.2.2 Generic Safety Issue

The staff recommendation for the closure of GSI-190 is contained in a December 26, 1999 memorandum from Ashok Thadani to William Travers (Ref. 11). The staff recommended that licensees address the effects of the coolant environment on component fatigue life as aging management programs are formulated in support of license renewal. One method acceptable to the staff for satisfying this recommendation is to assess the impact of the reactor coolant environment on a sample of critical components. These critical components should include, as a minimum, those selected in NUREG/CR-6260 (Ref. 10). The sample of critical components can be evaluated by applying environmental correction factors to the existing ASME Code fatigue analyses. Formulas for calculating the environmental life correction factors for carbon and low-alloy steels are contained in NUREG/CR-6583 (Ref. 14) and those for austenitic SSs are contained in NUREG/CR-5704 (Ref. 15).

4.3.2.3 FSAR Supplement

The specific criterion for meeting 10 CFR 54.21(d) is:

> The summary description of the evaluation of TLAAs for the period of extended operation in the FSAR supplement is appropriate such that later changes can be controlled by 10 CFR 50.59. The description should contain information associated with the TLAAs regarding the basis for determining that the applicant has made the demonstration required by 10 CFR 54.21(c)(1).

4.3.3 Review Procedures

For each area of review described in Subsection 4.3.1, the following review procedures should be followed:

4.3.3.1 Time-Limited Aging Analysis

4.3.3.1.1 ASME Section III, Class 1

For components designed or analyzed to ASME Class 1 requirements, the review procedures, depending on the applicant's choice of 10 CFR 54.21(c)(1)(i), (ii), or (iii), are:

4.3.3.1.1.1 10 CFR 54.21(c)(1)(i)

The operating transient experience and a list of the assumed transients used in the existing CUF calculations for the current operating term are reviewed to ensure that the number of assumed transients would not be exceeded during the period of extended operation.

4.3.3.1.1.2 10 CFR 54.21(c)(1)(ii)

The operating transient experience and a list of the increased number of assumed transients projected to the end of the period of extended operation are reviewed to ensure that the transient projection is adequate. The revised CUF calculations based on the projected number of assumed transients are reviewed to ensure that the CUF remains less than or equal to one at the end of the period of extended operation.

The code of record should be used for the reevaluation, or the applicant may update to a later code edition pursuant to 10 CFR 50.55a. In the latter case, the reviewer verifies that the requirements in 10 CFR 50.55a are met.

4.3.3.1.1.3 10 CFR 54.21(c)(1)(iii)

The applicant may reference the GALL report in its license renewal application, as appropriate. The review should verify that the applicant has stated that the report is applicable to its plant with respect to its program that monitors and tracks the number of critical thermal and pressure transients for the selected reactor coolant system components. The reviewer verifies that the applicant has identified the appropriate program as described and evaluated in the GALL report. The reviewer also ensures that the applicant has stated that its program contains the same program elements that the staff evaluated and relied upon in approving the corresponding generic program in the GALL report. No further staff evaluation is necessary.

4.3.3.1.2 ANSI B31.1

For piping designed or analyzed to ANSI B31.1 guidance, the review procedures, depending on the applicant's choice of 10 CFR 54.21(c)(1)(i), (ii), or (iii), are:

4.3.3.1.2.1 10 CFR 54.21(c)(1)(i)

The operating cyclic experience and a list of the assumed thermal cycles used in the existing allowable stress determination are reviewed to ensure that the number of assumed thermal cycles would not be exceeded during the period of extended operation.

4.3.3.1.2.2 10 CFR 54.21(c)(1)(ii)

The operating cyclic experience and a list of the increased number of assumed thermal cycles projected to the end of the period of extended operation are reviewed to ensure that the thermal cycle projection is adequate. The revised allowable stresses based on the projected number of assumed thermal cycles and the stress reduction factors given in the applicant's code of record are reviewed to ensure that they remain sufficient as specified by the code during the period of extended operation. Typical stress reduction factors based on thermal cycles are given in Table 4.3-1.

The code of record should be used for the reevaluation, or the applicant may use the criteria of 10 CFR 50.55a. In the latter case, the reviewer verifies that the requirements in 10 CFR 50.55a are met.

4.3.3.1.2.3 10 CFR 54.21(c)(1)(iii)

The applicant's proposed program to ensure that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation is reviewed. If the applicant proposed a component replacement before it exceeds the assumed thermal cycles, the reviewer verifies that the allowable stresses for the replacement will remain sufficient as specified by the code during the period of extended operation. Other applicant-proposed programs will be reviewed on a case-by-case basis.

4.3.3.1.3 Other Evaluations Based on CUF

The review procedures in Subsection 4.3.3.1.1 of this review plan section apply.

4.3.3.1.4 ASME Section III, Class 2 and 3

The review procedures in Subsection 4.3.3.1.2 of this review plan section apply.

4.3.3.2 Generic Safety Issue

The reviewer verifies that the applicant has addressed the staff recommendation for the closure of GSI-190 contained in a December 26, 1999 memorandum from Ashok Thadani to William Travers (Ref. 11). The reviewer verifies that the applicant has addressed the effects of the coolant environment on component fatigue life as aging management programs are formulated in support of license renewal. If an applicant has chosen to assess the impact of the reactor coolant environment on a sample of critical components, the reviewer verifies the following:

1. The critical components include, as a minimum, those selected in NUREG/CR-6260 (Ref. 10).

2. The sample of critical components has been evaluated by applying environmental correction factors to the existing ASME Code fatigue analyses.

3. Formulas for calculating the environmental life correction factors are those contained in NUREG/CR–6583 (Ref. 14) for carbon and low-alloy steels, and in NUREG/CR-5704 (Ref. 15) for austenitic SSs, or an approved technical equivalent.

4.3.3.3 FSAR Supplement

The reviewer verifies that the applicant has provided information, to be included in the FSAR supplement that includes a summary description of the evaluation of the metal fatigue TLAA. Table 4.3-2 contains examples of acceptable FSAR supplement information for this TLAA. The reviewer verifies that the applicant has provided a FSAR supplement with information equivalent to that in Table 4.3-2.

The staff expects to impose a license condition on any renewed license to require the applicant to update its FSAR to include this FSAR supplement at the next update required pursuant to 10 CFR 50.71(e)(4). As part of the license condition, until the FSAR update is complete, the

applicant may make changes to the programs described in its FSAR supplement without prior NRC approval, provided that the applicant evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59. If the applicant updates the FSAR to include the final FSAR supplement before the license is renewed, no condition will be necessary.

As noted in Table 4.3-2, an applicant need not incorporate the implementation schedule into its FSAR. However, the reviewer should verify that the applicant has identified and committed in the license renewal application to any future aging management activities, including enhancements and commitments to be completed before the period of extended operation. The staff expects to impose a license condition on any renewed license to ensure that the applicant will complete these activities no later than the committed date.

4.3.4 Evaluation Findings

The reviewer determines whether the applicant has provided sufficient information to satisfy the provisions of this section and whether the staff's evaluation supports conclusions of the following type, depending on the applicant's choice of 10 CFR 54.21(c)(1)(i), (ii), or (iii), to be included in the staff's safety evaluation report:

> On the basis of its review, as discussed above, the staff concludes that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1), that, for the metal fatigue TLAA, [choose which is appropriate] (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. The staff also concludes that the FSAR Supplement contains an appropriate summary description of the metal fatigue TLAA evaluation for the period of extended operation as reflected in the license condition.

4.3.5 Implementation

Except in those cases in which the applicant proposes an acceptable alternative method, the method described herein will be used by the staff in its evaluation of conformance with NRC regulations.

4.3.6 References

1. ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Power Plant Components," American Society of Mechanical Engineers.

2. ANSI/ASME B31.1, "Power Piping," American National Standards Institute.

3. ANSI/ASME B31.7-1969, "Nuclear Power Piping," American National Standards Institute.

4. SECY-93-049, "Implementation of 10 CFR Part 54, 'Requirements for Renewal of Operating Licenses for Nuclear Power Plants,'" March 1, 1993.

5. Staff Requirements Memorandum from Samuel J. Chilk, dated June 28, 1993.

6. NUREG-0933, "A Prioritization of Generic Safety Issues," Supplement 20, July 1996.

7. Letter from William T. Russell of NRC to William Rasin of the Nuclear Management and Resources Council, dated July 30, 1993.

8. SECY-94-191, "Fatigue Design of Metal Components," July 26, 1994.

9. SECY-95-245, "Completion of The Fatigue Action Plan," September 25, 1995.

10. NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," March 1995.

11. Letter from Ashok C. Thadani of the Office of Nuclear Regulatory Research to William D. Travers, Executive Director of Operations, dated December 26, 1999.

12. NUREG/CR-6674, "Fatigue Analysis of Components for 60-Year Plant Life," June 2000.

13. NUREG-1801, "Generic Aging Lessons Learned (GALL)," U.S. Nuclear Regulatory Commission, March 2001.

14. NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low–Alloy Steels," March 1998.

15. NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," April 1999.

Table 4.3-1. Stress Range Reduction Factors

Number of Equivalent Full Temperature Cycles	Stress Range Reduction Factor
7,000 and less	1.0
7,000 to 14,000	0.9
14,000 to 22,000	0.8
22,000 to 45,000	0.7
45,000 to 100,000	0.6
100,000 and over	0.5

Table 4.3-2. Example of FSAR Supplement for Metal Fatigue TLAA Evaluation

10 CFR 54.21(c)(1)(iii) Example

TLAA	Description of Evaluation	Implementation Schedule*
Metal fatigue	The aging management program monitors and tracks the number of critical thermal and pressure test transients, and monitors the cycles for the selected reactor coolant system components. The aging management program will address the effects of the coolant environment on component fatigue life by assessing the impact of the reactor coolant environment on a sample of critical components that include, as a minimum, those components selected in NUREG/CR-6260. The sample of critical components can be evaluated by applying environmental correction factors to the existing ASME Code fatigue analyses. Formulas for calculating the environmental life correction factors are contained in NUREG/CR-6583 for carbon and low-alloy steels and in NUREG/CR-5704 for austenitic SSs.	Evaluation should be completed before the period of extended operation

* An applicant need not incorporate the implementation schedule into its FSAR. However, the reviewer should verify that the applicant has identified and committed in the license renewal application to any future aging management activities to be completed before the period of extended operation. The staff expects to impose a license condition on any renewed license to ensure that the applicant will complete these activities no later than the committed date.

Table A.1-1. Elements of an Aging Management Program for License Renewal

Element	Description
1. Scope of program	Scope of program should include the specific structures and components subject to an AMR for license renewal.
2. Preventive actions	Preventive actions should prevent or mitigate aging degradation.
3. Parameters monitored or inspected	Parameters monitored or inspected should be linked to the degradation of the particular structure or component intended function(s).
4. Detection of aging effects	Detection of aging effects should occur before there is a loss of structure or component intended function(s). This includes aspects such as method or technique (i.e., visual, volumetric, surface inspection), frequency, sample size, data collection and timing of new/one-time inspections to ensure timely detection of aging effects.
5. Monitoring and trending	Monitoring and trending should provide predictability of the extent of degradation, and timely corrective or mitigative actions.
6. Acceptance criteria	Acceptance criteria, against which the need for corrective action will be evaluated, should ensure that the structure or component intended function(s) are maintained under all CLB design conditions during the period of extended operation.
7. Corrective actions	Corrective actions, including root cause determination and prevention of recurrence, should be timely.
8. Confirmation process	Confirmation process should ensure that preventive actions are adequate and that appropriate corrective actions have been completed and are effective.
9. Administrative controls	Administrative controls should provide a formal review and approval process.
10. Operating experience	Operating experience of the aging management program, including past corrective actions resulting in program enhancements or additional programs, should provide objective evidence to support the conclusion that the effects of aging will be managed adequately so that the structure and component intended function(s) will be maintained during the period of extended operation.

NYS-26/26A & RIVERKEEPER TC-1/1A: ATTACHMENT 5

NUREG-1801, Vol. 2, Rev. 1

Generic Aging Lessons Learned (GALL) Report

Tabulation of Results

Manuscript Completed: September 2005
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Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

X.M1 METAL FATIGUE OF REACTOR COOLANT PRESSURE BOUNDARY

Program Description

In order not to exceed the design limit on fatigue usage, the aging management program (AMP) monitors and tracks the number of critical thermal and pressure transients for the selected reactor coolant system components.

The AMP addresses the effects of the coolant environment on component fatigue life by assessing the impact of the reactor coolant environment on a sample of critical components for the plant. Examples of critical components are identified in NUREG/CR-6260. The sample of critical components can be evaluated by applying environmental life correction factors to the existing ASME Code fatigue analyses. Formulae for calculating the environmental life correction factors are contained in NUREG/CR-6583 for carbon and low-alloy steels and in NUREG/CR-5704 for austenitic stainless steels.

As evaluated below, this is an acceptable option for managing metal fatigue for the reactor coolant pressure boundary, considering environmental effects. Thus, no further evaluation is recommended for license renewal if the applicant selects this option under 10 CFR 54.21(c)(1)(iii) to evaluate metal fatigue for the reactor coolant pressure boundary.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program includes preventive measures to mitigate fatigue cracking of metal components of the reactor coolant pressure boundary caused by anticipated cyclic strains in the material.

2. ***Preventive Actions:*** Maintaining the fatigue usage factor below the design code limit and considering the effect of the reactor water environment, as described under the program description, will provide adequate margin against fatigue cracking of reactor coolant system components due to anticipated cyclic strains.

3. ***Parameters Monitored/Inspected:*** The program monitors all plant transients that cause cyclic strains, which are significant contributors to the fatigue usage factor. The number of plant transients that cause significant fatigue usage for each critical reactor coolant pressure boundary component is to be monitored. Alternatively, more detailed local monitoring of the plant transient may be used to compute the actual fatigue usage for each transient.

4. ***Detection of Aging Effects:*** The program provides for periodic update of the fatigue usage calculations.

5. ***Monitoring and Trending:*** The program monitors a sample of high fatigue usage locations. This sample is to include the locations identified in NUREG/CR-6260, as minimum, or propose alternatives based on plant configuration.

6. ***Acceptance Criteria:*** The acceptance criteria involves maintaining the fatigue usage below the design code limit considering environmental fatigue effects as described under the program description.

7. ***Corrective Actions:*** The program provides for corrective actions to prevent the usage factor from exceeding the design code limit during the period of extended operation.

Acceptable corrective actions include repair of the component, replacement of the component, and a more rigorous analysis of the component to demonstrate that the design code limit will not be exceeded during the extended period of operation. For programs that monitor a sample of high fatigue usage locations, corrective actions include a review of additional affected reactor coolant pressure boundary locations. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. ***Confirmation Process:*** Site quality assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of Appendix B to 10 CFR Part 50. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process and administrative controls.

9. ***Administrative Controls:*** See Item 8, above.

10. ***Operating Experience:*** The program reviews industry experience regarding fatigue cracking. Applicable experience with fatigue cracking is to be considered in selecting the monitored locations.

References

NUREG/CR-5704, *Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels,* U.S. Nuclear Regulatory Commission, April 1999.

NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components,* U.S. Nuclear Regulatory Commission, March 1995.

NUREG/CR-6583, *Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels,* U.S. Nuclear Regulatory Commission, March 1998.

NYS-26/26A & RIVERKEEPER TC-1/1A: ATTACHMENT 6

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Materials Reliability Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application (MRP-47, Revision 1)

Technical Report

Materials Reliability Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application (MRP-47 Revision 1)

1012017

Final Report, September 2005

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REPORT SUMMARY

For about the last 15 years, the effects of light water reactor environment on fatigue have been the subject of research in both the United States and abroad. Based on a risk study reported in NUREG/CR-6674, the NRC concluded that reactor water environmental effects were not a safety issue for a 60-year operating life, but that some limited assessment of its effect would be required for a license renewal extended operating period beyond 40 years. This guideline offers methods for addressing environmental fatigue in a license renewal submittal.

Background

Many utilities are currently embarking upon efforts to renew their operating licenses. One of the key areas of uncertainty in this process relates to fatigue of pressure boundary components. Although the NRC has determined that fatigue is not a significant contributor to core damage frequency, they believe that the frequency of pipe leakage may increase significantly with operating time and have requested that license renewal applicants perform an assessment to determine the effects of reactor water coolant environment on fatigue, and, where appropriate, manage this effect during the license renewal period. As the license renewal application process progressed starting in 1998, several utilities addressed this request using different approaches. In more recent years, a unified approach has emerged that has obtained regulator approval and allowed utilities to satisfactorily address this issue and obtain a renewed operating license for 60 years of plant operation.

Objectives

- To provide guidance for assessment and management of reactor coolant environmental effects
- To minimize the amount of plant-specific work necessary to comply with NRC requirements for addressing this issue in a license renewal application
- To provide "details of execution" for applying the environmental fatigue approach currently accepted by the NRC in the license renewal application process.

Approach

The project team reviewed previous work by EPRI and utilities related to fatigue environmental effects and license renewal including reports on this subject created by EPRI, NRC, and NRC contractors. Recent license renewal applications, NRC Requests for Additional Information, and the commitments made by the past license renewal applicants provided insight into NRC expectations. After evaluation of all this information, the project team developed alternatives for addressing fatigue environmental effects. This revision provides guidelines based on industry experience, consensus, and insight gained from more than six years of experience with this issue and the license renewal approval process.

Results

The report describes a fatigue environmental effect license renewal approach that can be applied by any license renewal applicant. It provides guidelines for performing environmental fatigue assessments using fatigue environmental factors from currently accepted F_{en} methodology.

EPRI Perspective

Utilities have committed significant resources to license renewal activities related to fatigue. Based on input from applicants to-date, NRC requirements for addressing fatigue environmental effects continued to change for the first few applicants, but more recently have become more unified. These guidelines were developed to provide stability, refined guidance, and assurance of NRC acceptance and include an approach that may be taken to address fatigue environmental effects in a license renewal application. Use of the approach provided in this document should limit the amount of effort necessary by individual license renewal applicants in addressing this requirement and putting activities in place for the extended operating period to manage reactor water environmental effects on fatigue.

Keywords

Fatigue
License Renewal
Reactor Water Environmental Fatigue Effects

ABSTRACT

For about the last 15 years. the effects of light water reactor environment on fatigue have been the subject of research in both the United States and abroad. The conclusions from this research are that the reactor water temperature and chemical composition (particularly oxygen content or ECP) can have a significant effect on the fatigue life of carbon, low alloy, and austenitic stainless steels. The degree of fatigue life reduction is a function of the tensile strain rate during a transient, the specific material, the temperature, and the water chemistry. The effects of other than moderate environment were not considered in the original development of the ASME Code Section III fatigue curves.

This issue has been studied by the Nuclear Regulatory Commission (NRC) for many years. One of the major efforts was a program to evaluate the effects of reactor water environment for both early and late vintage plants designed by all U.S. vendors. The results of that study, published in NUREG/CR-6260, showed that there were a few high usage factor locations in all reactor types, and that the effects of reactor water environment could cause fatigue usage factors to exceed the ASME Code-required fatigue usage limit of 1.0. On the other hand, it was demonstrated that usage factors at many locations could be shown acceptable by refined analysis and/or fatigue monitoring of actual plant transients.

Based on a risk study reported in NUREG/CR-6674, the NRC concluded that reactor water environmental effects were not a safety issue for a 60-year operating life, but that some limited assessment of its effect would be required for a license renewal extended operating period beyond 40 years. Thus, for all license renewal submittals to-date, there have been formal questions raised on the topic of environmental fatigue and, in all cases, utility commitments to address the environmental effects on fatigue in the extended operating period. Many plants have already performed these commitments.

This guideline offers methods for addressing environmental fatigue in a license renewal submittal. It requires that a sampling of the most affected fatigue sensitive locations be identified for evaluation and tracking in the extended operating period. NUREG/CR-6260 locations are considered an appropriate sample for F_{en} evaluation as long as none exceed the acceptance criteria with environmental effects considered. If this occurs, the sampling is to be extended to other locations. For these locations, evaluations similar to those conducted in NUREG/CR-6260 are required. In the extended operating period, fatigue monitoring is used for the sample of locations to show that ASME Code limits are not exceeded. If these limits are exceeded, corrective actions are identified for demonstrating acceptability for continued operation.

Using the guidance provided herein, the amount of effort needed to justify individual license renewal submittals and respond to NRC questions should be minimized, and a more unified, consistent approach should be achieved throughout the industry. More importantly, this revision provides "details of execution" for applying the environmental fatigue approach currently accepted by the NRC in the license renewal application process.

9. ADMINISTRATIVE CONTROLS

   The Thermal Fatigue Licensing Basis Monitoring Guideline actions are implemented by plant work processes.

10. OPERATING EXPERIENCE

    Refer to Sections 1.1 and 2.5.2.3 of Reference [23] for a discussion of how operating experience becomes part of the Thermal Fatigue Licensing Basis Monitoring Guideline implementation.

3.2 Method for Evaluation of Environmental Effects

There are several methods that have been published to assess the effects of reactor water environment on fatigue for each specific location to be considered. In this document, guidance is provided for performing evaluations in accordance with NUREG/CR-6583 [3] for carbon and low alloy steels and NUREG/CR-5704 [4] for austenitic stainless steels, since these are the currently accepted methodologies for evaluating environmental fatigue effects. Other methods that have been published, including those currently being used in Japan, are documented in References [18] and [22].

Figure 3-1 is a flowchart that shows an overview of the assessment approach.

- The first step is to identify the locations to be used in the assessment. This step is discussed in Section 3.2.1
- The second step is to perform an assessment of the effects of environmental fatigue on the locations identified in Step 1. This includes an assessment of the actual expected fatigue usage factor including the influence of environmental effects. Inherent conservatisms in design transients may be removed to arrive at realistic CUFs that include environmental effects. This approach is most applicable to locations where the design transients significantly envelope actual operating conditions in the plant. Further discussion is provided in Section 3.2.2. Specific guidance on performing such evaluation is provided in Section 4.0.
- The bottom of Figure 3-1 indicates that fatigue management occurs after the evaluation from Step 2 is performed for each location. This may be as simple as counting the accumulated cycles and showing that they remain less than or equal to the number of cycles utilized in the assessment performed in Step 2. On the other hand, it may not be possible to show continued acceptance throughout the extended operating period such that additional actions are required. Such options are discussed in Section 3.3. Refer also to Reference [23] for a discussion of cycle counting.

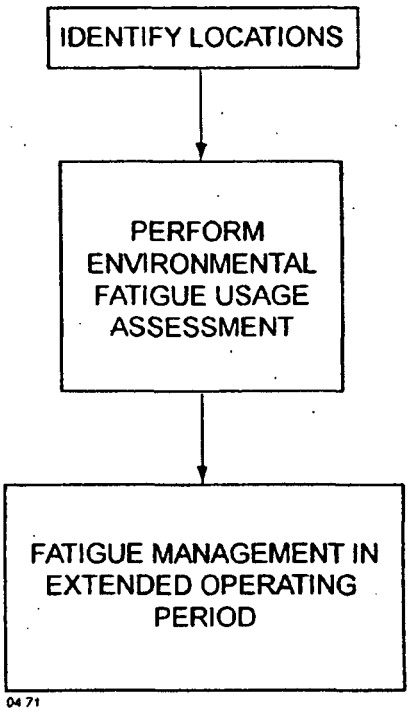

Figure 3-1
Overview of Fatigue Environmental Effects Assessment and Management

3.2.1 Identification of Locations for Assessment of Environmental Effects

A sampling of locations is chosen for the assessment of environmental effects. The purpose of identifying this set of locations is to focus the environmental assessment on just a few components that will serve as leading indicators of fatigue reactor water environmental effects. Figure 3-2 shows an overview of the approach identified for selecting and evaluating locations.

For both PWR and BWR plants, the locations chosen in NUREG/CR-6260 [2] were deemed to be representative of locations with relatively high usage factors for all plants. Although the locations may not have been those with the highest values of fatigue usage reported for the plants evaluated, they were considered representative enough that the effects of LWR environment on fatigue could be assessed.

The locations evaluated in NUREG/CR-6260 [2] for the appropriate vendor/vintage plant should be evaluated on a plant-unique basis. For cases where acceptable fatigue results are demonstrated for these locations for 60 years of plant operation including environmental effects, additional evaluations or locations need not be considered. However, plant-unique evaluations may show that some of the NUREG/CR-6260 [2] locations do not remain within allowable limits for 60 years of plant operation when environmental effects are considered. In this situation, plant specific evaluations should expand the sampling of locations accordingly to include other locations where high usage factors might be a concern.

In original stress reports, usage factors may have been reported in many cases that are unrealistically high, but met the ASME Code requirement for allowable CUF. In these cases, revised analysis may be conducted to derive a more realistic usage factor or to show that the revised usage factor is significantly less than reported.

If necessary, in identifying the set of locations for the expanded environmental assessment, it is important that a diverse set of locations be chosen with respect to component loading (including thermal transients), geometry, materials, and reactor water environment. If high usage factors are presented for a number of locations that are similar in geometry, material, loading conditions, and environment, the location with the highest expected CUF, considering typical environmental fatigue multipliers, should be chosen as the bounding location to use in the environmental fatigue assessment. Similar to the approach taken in NUREG/CR-6260 [2], the final set of locations chosen for expanded environmental assessment should include several different types of locations that are expected to have the highest CUFs and should be those most adversely affected by environmental effects. The basis of location choice should be described in the individual plant license renewal application.

In conclusion, the following steps should be taken to identify the specific locations that are to be considered in the environmental assessment:

- Identify the locations evaluated in NUREG/CR-6260 [2] for the appropriate vintage/vendor plant.
- Perform a plant-unique environmental fatigue assessment for the NUREG/CR-6260 locations.
- If the CUF results for all locations above are less than or equal to the allowable (typically 1.0) for the 60-year operating life, the environmental assessment may be considered complete; additional evaluations or locations need not be considered.
- If the CUF results for any locations above are greater than the allowable for the 60-year operating life, expand the locations evaluated, considering the following:
  - Identify all Class 1 piping systems and major components. For the reactor pressure vessel, there may be multiple locations to consider.
  - For each system or component, identify the highest usage factor locations. By reasons of geometric discontinuities or local transient severity, there will generally be a few locations that have the highest usage factors when considering environmental effects.
  - From the list of locations that results from the above steps, choose a set of locations that are a representative sampling of locations with the highest expected usage factors when considering environmental effects. Considerations for excluding locations can include: (1) identification of excess conservatism in the transient grouping or other aspects of the design fatigue analysis, or (2) locations that have similar loading conditions, geometry, material, and reactor water environment compared to another selected location.

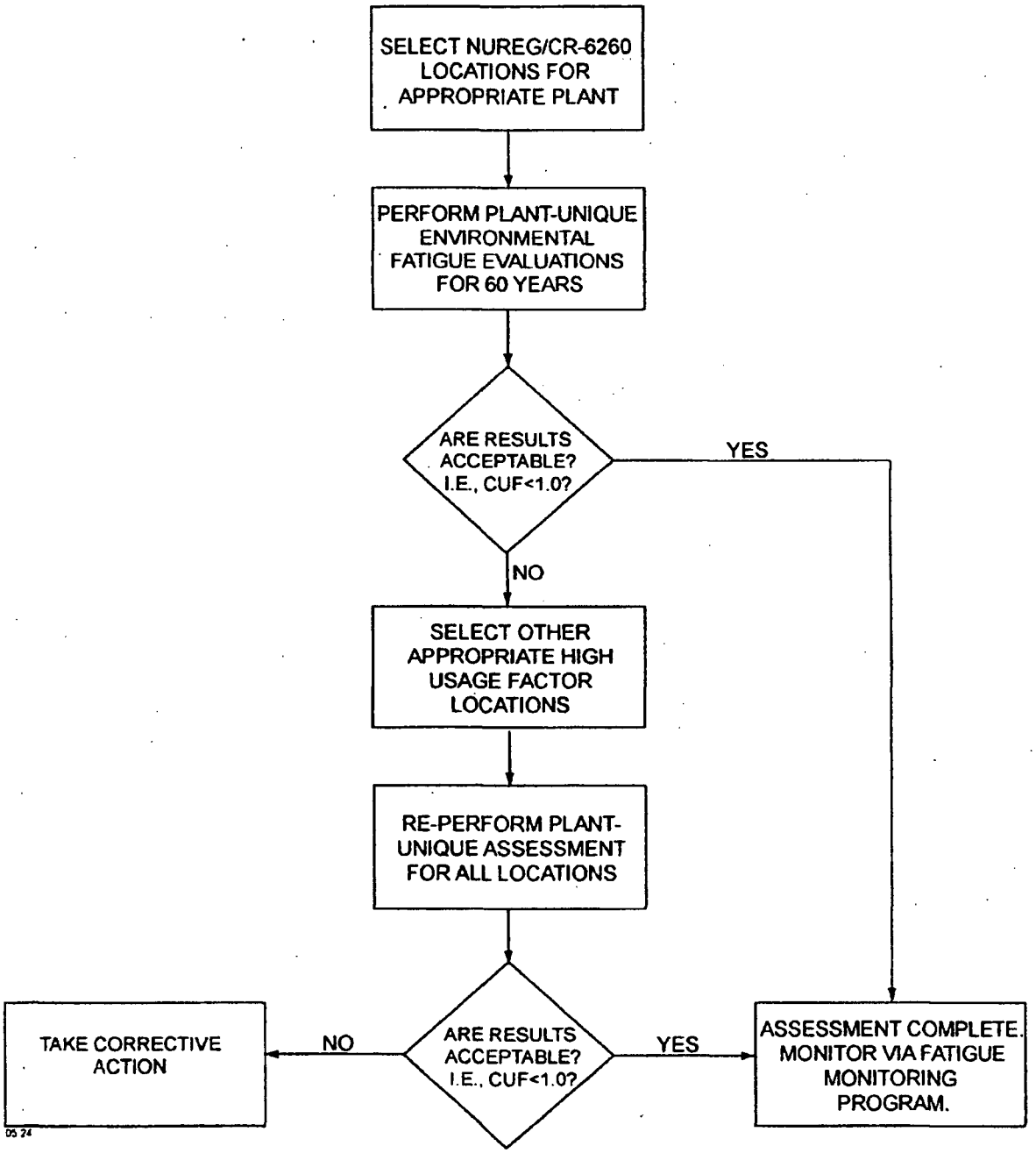

Figure 3-2
Identification of Component Locations and Fatigue Environmental Effects Assessment

3.2.2 Fatigue Assessment Using Environmental Factors

In performing an assessment of environmental fatigue effects, factors to account for environmental effects are incorporated into an updated fatigue evaluation for each selected location using the F_{en} approach documented in NUREG/CR-6583 [3] for carbon and low alloy steels and NUREG/CR-5704 [4] for austenitic stainless steels. Excess conservatism in the loading definitions, number of cycles, and the fatigue analyses may be considered. Figure 3-3 shows the approach for performing the assessment and managing fatigue in the extended operating period.

Determination of Existing Licensing Basis

Existing plant records must be reviewed to determine the cyclic loading specification (transient definition and number of cycles) and stress analysis for the location in question. Review of the analysis may or may not show that excess conservatism exists. Reference [23] provides guidance on reviewing the original design basis, the operating basis, and additions imposed by the regulatory oversight process, to determine the fatigue licensing basis events for which the component is required to be evaluated.

Consideration of Increased Cycles for Extended Period

As a part of the license renewal application process, the applicant must update the projected cycles to account for 60 years of plant operation. The first possible outcome is that the number of expected cycles in the extended operating period will remain at or below those projected for the initial 40-year plant life. In this case, the governing fatigue analyses will not require modification to account for the extended period of operation.

The second possibility is that more cycles are projected to occur for 60 years of plant operation than were postulated for the first 40 years. In this case, an applicant must address the increased cycle counts. One possible solution is to perform a revised fatigue analysis to confirm that the increased number of cycles will still result in a CUF less than or equal to the allowable. A second possibility is to determine the number of cycles at which the CUF would be expected to reach the allowable. This cycle quantity then becomes the allowable against which the actual operation is tracked. Section 3.3 discusses options to be employed if this lower allowable is projected to be exceeded.

Fatigue Assessment

Fatigue assessment includes the determination of CUF considering environmental effects. This may be accomplished conservatively using information from design documentation and bounding F_{en} factors from NUREG/CR-6583 [3] and NUREG/CR-5704 [4], or it may require a more extensive approach (as discussed in Section 4.0).

A revised fatigue analysis may or may not be required. Possible reasons for updating the fatigue analysis could include:

- Excess conservatism in original fatigue analysis with respect to modeling, transient definition, transient grouping and/or use of an early edition of the ASME Code.

- For piping, use of an ASME Code Edition prior to 1979 Summer Addenda, which included the ΔT_1 term in Equation (10) of NB-3650. Use of a later code reduces the need to apply conservative elastic-plastic penalty factors.
- Re-analysis may be needed to determine strain rate time histories possibly not reported in existing component analyses, such that bounding environmental multipliers (i.e., very low or "saturated" strain rates) would not have to be used.

A simplified revised fatigue analysis may be performed using results from the existing fatigue analysis, if sufficient detail is available. Alternatively, a new complete analysis could be conducted to remove additional conservatisms. Such an evaluation would not necessarily need the full pedigree of a certified ASME Code Section III analysis (i.e.. Certified Design Specification, etc.), but it should utilize all of the characteristic methods from Section III for computing CUF. In the environmental fatigue assessment, the environmental fatigue usage may be calculated using the following steps:

- For each load set pair in the fatigue analysis, determine an environmental factor F_{en}. This factor should be developed using the equations in NUREG/CR-6583 [3] or NUREG/CR-5704 [4]. (Section 4.0 provides specific guidance on performing an F_{en} evaluation)
- The environmental partial fatigue usage for each load set pair is then determined by multiplying the original partial usage factor by F_{en}. In no case shall the F_{en} be less than 1.0.
- The usage factor is the sum of the partial usage factors calculated with consideration of environmental effects.

Fatigue Management Approach

As shown in Figure 3-3, the primary fatigue management approaches for the extended operating period consist of tracking either the CUF or number of accumulated cycles.

- For cycle counting, an updated allowable number of cycles may be needed if the fatigue assessment determined the CUF to be larger than allowable. One approach is to derive a reduced number of cycles that would limit the CUF to less than or equal to the allowable value (typically 1.0). On the other hand, if the assessed CUF was shown to be less than or equal to the allowable, the allowable number of cycles may remain as assumed in the evaluation, or increased appropriately. As long as the number of cycles in the extended operating period remains within this allowed number of cycles, no further action is required.
- For CUF tracking, one approach would be to utilize fatigue monitoring that accounts for the actual cyclic operating conditions for each location. This approach would track the CUF due to the actual cycle accumulation, and would take credit for the combined effects of all transients. Environmental factors would have to be factored into the monitoring approach or applied to the CUF results of such monitoring. No further action is required as long as the computed usage factor remains less than or equal to the allowable value.

Prior to such time that the CUF is projected to exceed the allowable value, or the number of actual cycles is projected to exceed the allowable number of cycles, action must be taken such that the allowable limits will not be exceeded. If the cyclic or fatigue limits are expected to be exceeded during the license renewal period, further approaches to fatigue management would be required prior to reaching the limit, as described in Section 3.3. Further details on guidelines for thermal fatigue monitoring and compliance/mitigation options are provided in Reference [23].

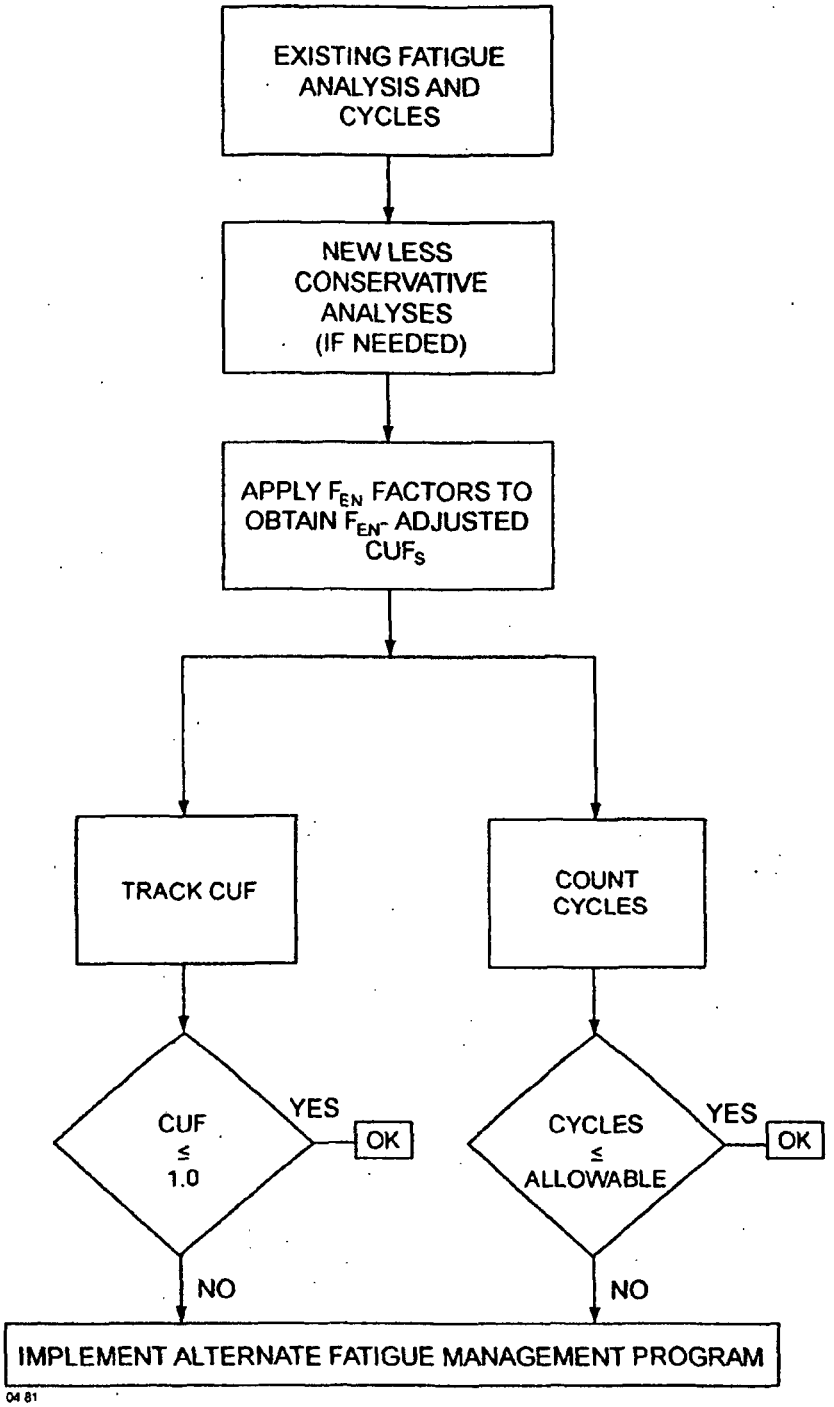

Figure 3-3
Fatigue Management if Environmental Assessment Conducted

A separate section follows for each parameter utilized in the F_{en} expressions, that is transformed sulfur content (S^*), transformed temperature (T^*), transformed dissolved oxygen (O^*), and transformed strain rate ($\dot{\varepsilon}^*$). For the transformed strain rate, temperature, and oxygen parameters, the three approaches are discussed. Transformed sulfur does not vary over the three approaches. A single approach should be utilized for all of the transformed parameters in a single load-pair F_{en} determination, although different approaches may be utilized for different load-pair F_{en}s.

First, the typical content of a fatigue calculation is presented.

4.2.1 Contents of a Typical Fatigue Evaluation

This section provides the content of a typical fatigue calculation. Whereas fatigue calculations have varied over the years, their basic content is the same. With the advent of computer technology, the calculations have basically maintained the same content, but computations have become more refined and exhaustive. For example, 30 years ago it was computationally difficult for a stress analyst to evaluate 100 different transients in a fatigue calculation. Therefore, the analyst would have grouped the transients into as few as one transient grouping and performed as few incremental fatigue calculations as possible. With today's computer technology and desire to show more margin, it is relatively easy for the modern-day analyst to evaluate all 100 incremental fatigue calculations for this same problem. Also, older technology would have likely utilized conservative shell interaction hand solutions for computing stress, whereas today finite element techniques are commonly deployed. This improvement in technology would not have changed the basic inputs to the fatigue calculation (i.e., stress), but it would have typically yielded significantly more representative input values.

The discussion here is limited to the general content of most typical fatigue calculations. Discussions of removing excess conservatisms from the input (stress) values of these calculations are not included, as it is assumed that those techniques are generally well understood by engineers performing these assessments throughout the industry.

Two typical fatigue calculations are shown in Figures 4-1 through 4-4. Figure 4-1 reflects an "old" calculation, i.e., one that is typical from a stress report from a plant designed in the 1960s. Figures 4-2 through 4-4 reflect a "new" calculation, i.e., one that is typical from a 1990s vintage stress report. A description of the content of these two calculations is provided below.

The same basic content is readily apparent in both CUF calculations shown in Figures 4-1 through 4-4. However, it is also apparent that much more detail is present in Figures 4-2 through 4-4 for the "new" calculation compared to Figure 4-1 for the "old" calculation. Therefore, with respect to applying F_{en} methodology to a CUF calculation, the guidance provided in the following sections equally applies to both vintages of calculations. The main difference is in assumptions that need to be made for the F_{en} transformed variables due to a lack of detail backing up the calculations in the stress report. Guidance for these assumptions is described in Sections 4.2.2 through 4.2.5, with appropriate reference to the calculations shown in Figures 4-1 through 4-4.

NYS-26/26A & RIVERKEEPER TC-1/1A: ATTACHMENT 7

NUREG/CR-6583
ANL-97/18

Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels

Argonne National Laboratory

U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, DC 20555-0001

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Argonne National Laboratory

Prepared for
U.S. Nuclear Regulatory Commission

EFFECTS OF LWR COOLANT ENVIRONMENTS ON FATIGUE DESIGN CURVES OF CARBON AND LOW-ALLOY STEELS

by

O. K. Chopra and W. J. Shack

Abstract

The ASME Boiler and Pressure Vessel Code provides rules for the construction of nuclear power plant components. Figures I-9.1 through I-9.6 of Appendix I to Section III of the Code specify fatigue design curves for structural materials. While effects of reactor coolant environments are not explicitly addressed by the design curves, test data indicate that the Code fatigue curves may not always be adequate in coolant environments. This report summarizes work performed by Argonne National Laboratory on fatigue of carbon and low-alloy steels in light water reactor (LWR) environments. The existing fatigue S-N data have been evaluated to establish the effects of various material and loading variables such as steel type, dissolved oxygen level, strain range, strain rate, temperature, orientation, and sulfur content on the fatigue life of these steels. Statistical models have been developed for estimating the fatigue S-N curves as a function of material, loading, and environmental variables. The results have been used to estimate the probability of fatigue cracking of reactor components. The different methods for incorporating the effects of LWR coolant environments on the ASME Code fatigue design curves are presented.

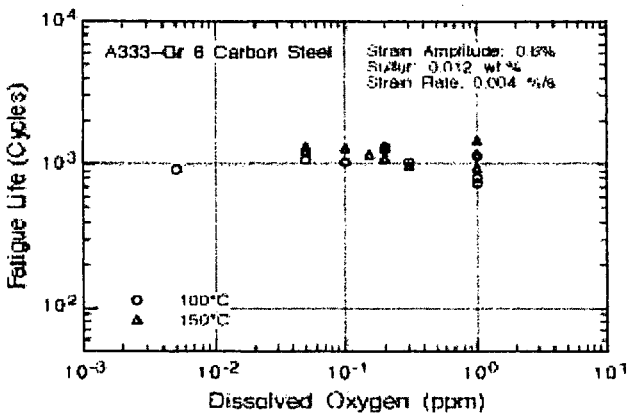

Figure 68.
Fatigue life of A333-Gr 6 carbon steel as a function of dissolved oxygen in water at 100 and 150°C

5 Statistical Model

5.1 Modeling Choices

In attempting to develop a statistical model from incomplete data and where physical processes are only partially understood, care must be taken to avoid overfitting the data. Different functional forms of the predictive equations (e.g., different procedures for transforming the measured variables into data used for fitting equations) were tried for several aspects of the model. Fatigue S-N data are generally expressed by Eq. 1.1, which may be rearranged to express fatigue life N in terms of strain amplitude ε_a as

$$\ln(N) = [\ln B - \ln(\varepsilon_a - A)]/b. \tag{5.1}$$

Additional terms may be added to the model that would improve agreement with the current data set. However, such changes may not hold true in other data sets, and the model would typically be less robust, i.e., it would not predict new data well. In general, complexity in a statistical model is undesirable unless it is consistent with accepted physical processes. Although there are statistical tools that can help manage the tradeoff between robustness and detail in the model, engineering judgment is required. Model features that would be counter to known effects are excluded. Features that are consistent with previous studies use such results as guidance, e.g., defining the threshold or saturation values for an effect, but where there are differences from previous findings, the reasons for the differences are evaluated and an appropriate set of assumptions is incorporated into the model.

5.2 Least-Squares Modeling within a Fixed Structure

The parameters of the model are commonly established through least-squares curve-fitting of the data to either Eq. 1.1 or 5.1. An optimization program sets the parameters so as to minimize the sum of the square of the residual errors, which are the differences between the predicted and actual values of ε_a or ln(N). A predictive model based on least-squares fit on ln(N) is biased for low ε_a; in particular, runoff data cannot be included. The model also leads to probability curves that converge to a single value of threshold strain.

However, the model fails to address the fact that at low ε_a, most of the error in life is due to uncertainty associated with either measurement of stress or strain or variation in threshold strain caused by material variability. On the other hand, a least-squares fit on ε_a does not work well for higher strain amplitudes. The two kinds of models are merely transformations of each other, although the precise values of the coefficients differ.

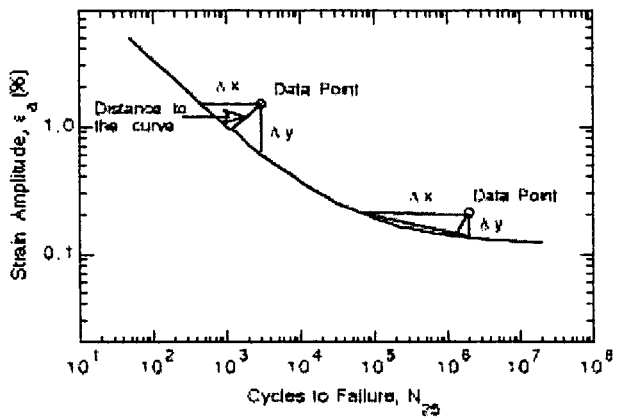

Figure 69.
Schematic of least-squares curve-fitting of data by minimizing sum of squared Cartesian distances from data points to predicted curve

The statistical models[27,28] were developed by combining the two approaches and minimizing the sum of squared Cartesian distances from the data points to the predicted curve (Fig. 69). For low ε_a, this is very close to optimizing the sum of squared errors in predicted ε_a; at high ε_a, it is very close to optimizing the sum of squared errors in predicted life; and at medium ε_a, this model combines both factors. However, because the model includes many nonlinear transformations of variables and because different variables affect different parts of the data, the actual functional form and transformations are partly responsible for minimizing the squares of the errors. The functional forms and transformation are chosen a priori, and no direct computational means exist for establishing them.

To perform the optimization, it was necessary to normalize the x and y axes by assigning relative weights to be used in combining the error in life and strain amplitude because x and y axes are not in comparable units. In this analysis, errors in strain amplitude (%) are weighted 20 times as heavily as errors in ln(N). A value of 20 was selected for two related reasons. First, this factor leads to approximately equal weighting of low- and high-strain-amplitude data in the least-squared error computation of model coefficients. Second, when applied to the model to generate probability curves, it yielded a standard deviation on strain amplitude comparable to that obtained from the best-fit of the high cycle fatigue data to Eq. 1.1. Because there is necessarily judgment applied in the selection of this value, a sensitivity analysis was performed, and it showed that the coefficients of the model do not change much for weight factors between 10 and 25. Distance from the curve was estimated as

$$D = \left\{(x - \hat{x})^2 + \left[k(y - \hat{y})\right]^2\right\}^{1/2}, \tag{5.2}$$

where $\hat{x}$ and $\hat{y}$ represent predicted values, and k = 20.

5.3 The Model

Based on the existing fatigue S-N data base, statistical models have been developed for estimating the effects of material and loading conditions on the fatigue lives of CSs and LASs.[27,28] The dependence of fatigue life on DO level has been modified because it was determined that in the range of 0.05-0.5 ppm, the effect of DO was more logarithmic than linear.[45,93] In this report, the models have been further optimized with a larger fatigue S-N data base. Because of the conflicting possibilities that with decreasing strain rate, fatigue life may either be unaffected, decrease for some heats, or increase for others, effects of strain rate in air were not explicitly considered in the model. The effects of orientation, i.e., size and distribution of sulfide inclusions, on fatigue life were also excluded because the existing data base does not include information on sulfide distribution and morphology. In air, the fatigue data for CSs are best represented by

$$\ln(N_{25}) = 6.595 - 1.975 \ln(\varepsilon_a - 0.113) - 0.00124\ T \tag{5.3a}$$

and for LASs by

$$\ln(N_{25}) = 6.658 - 1.808 \ln(\varepsilon_a - 0.151) - 0.00124\ T. \tag{5.3b}$$

In LWR environments, the fatigue data for CSs are best represented by

$$\ln(N_{25}) = 6.010 - 1.975 \ln(\varepsilon_a - 0.113) + 0.101\ S^* T^* O^* \dot{\varepsilon}^* \tag{5.4a}$$

and for LASs by

$$\ln(N_{25}) = 5.729 - 1.808 \ln(\varepsilon_a - 0.151) + 0.101\ S^* T^* O^* \dot{\varepsilon}^*, \tag{5.4b}$$

where S^*, T^*, O^*, and $\dot{\varepsilon}^*$ = transformed sulfur content, temperature, DO, and strain rate, respectively, defined as follows:

$$\begin{aligned} S^* &= S && (0 < S \le 0.015 \text{ wt.\%}) \\ S^* &= 0.015 && (S > 0.015 \text{ wt.\%}) \end{aligned} \tag{5.5a}$$

$$\begin{aligned} T^* &= 0 && (T < 150°C) \\ T^* &= T - 150 && (T = 150\text{–}350°C) \end{aligned} \tag{5.5b}$$

$$\begin{aligned} O^* &= 0 && (DO < 0.05 \text{ ppm}) \\ O^* &= \ln(DO/0.04) && (0.05 \text{ ppm} \le DO \le 0.5 \text{ ppm}) \\ O^* &= \ln(12.5) && (DO > 0.5 \text{ ppm}) \end{aligned} \tag{5.5c}$$

$$\begin{aligned} \dot{\varepsilon}^* &= 0 && (\dot{\varepsilon} > 1 \text{ \%/s}) \\ \dot{\varepsilon}^* &= \ln(\dot{\varepsilon}) && (0.001 \le \dot{\varepsilon} \le 1 \text{ \%/s}) \\ \dot{\varepsilon}^* &= \ln(0.001) && (\dot{\varepsilon} < 0.001 \text{ \%/s}) \end{aligned} \tag{5.5d}$$

The functional form and bounding values of the transformed parameters S^*, T^*, O^*, and $\dot{\varepsilon}^*$ were based upon experimental observations and data trends discussed in Section 4.2. Significant features of the model for estimating fatigue life in LWR environments are as follows:

(a) The model assumes that environmental effects on fatigue life occur primarily during the tensile-loading cycle; minor effects during the compressive loading cycle have been excluded. Consequently, the loading and environmental conditions, e.g., temperature, strain rate, and DO, during the tensile-loading cycle are used for estimating fatigue lives.

(b) When any one of the threshold condition is not satisfied, e.g., <0.05 ppm DO in water, the effect of strain rate is not considered in the model, although limited data indicate that heats of steel that are sensitive to strain rate in air also show a decrease in life in water with decreasing strain rate.

(c) The model assumes a linear dependence of S^* on S content in steel and saturation at 0.015 wt.% S.

The model is recommended for predicted fatigue lives of $\leq 10^6$ cycles. For fatigue lives of 10^6 to 10^8 cycles, the results should be used with caution because, in this range, the model is based on very limited data obtained from relatively few heats of material.

The estimated and experimental S-N curves for CS and LAS in air at room temperature and 288°C are shown in Fig. 70. The mean curves used in developing the ASME Code design curve and the average curves of Higuchi and Iida[7] are also included in the figure. The results indicate that the ASME mean curve for carbon steels is not consistent with the experimental data; at strain amplitudes <0.2%, the mean curve predicts significantly lower fatigue lives than those observed experimentally. The estimated curve for low-alloy steels is comparable with the ASME mean curve. For both steels, Eq. 5.3 shows good agreement with the average curves of Higuchi and Iida.

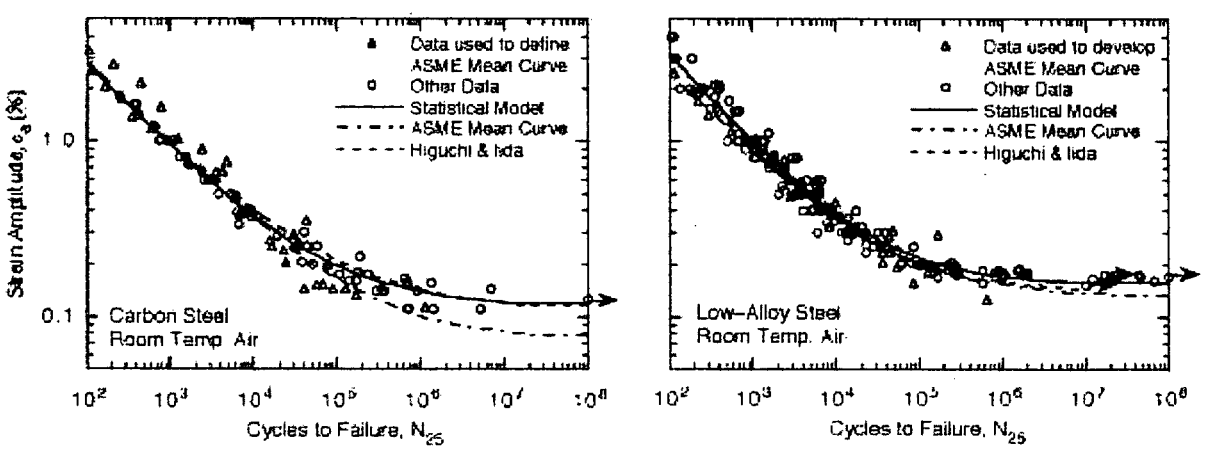

Figure 70. Fatigue S-N behavior for carbon and low-alloy steels estimated from model and determined experimentally in air at room temperature

5.4 Distribution of Fatigue Life

For a given steel type, the average distance of data points from the mean curve does not vary much for different environmental conditions. To develop a distribution on life, we start with the assumption that there are three sources of prediction error: (a) measurement errors

for the applied strain amplitude, (b) variations in the threshold strain amplitude due to material variability, and (c) errors due to uncertainty in test and material conditions or other unexplained variation. Because measurement errors are small at high strain amplitudes, the standard deviation of distance from the mean curve at high strain amplitudes is a good measure of the scatter in fatigue life due to unexplained variations. At low amplitudes where the S-N curve is almost horizontal, the errors (as measured by the distance from the mean curve) are dominated by the variation in strain amplitude. The standard deviation of the error in strain amplitude was taken to be equal to the standard deviation in the predicted fatigue life divided by a factor of 20 consistent with the weighting factor used for optimization. The standard deviation on life was 0.52 for CSs and LASs. These results can be combined with Eq. 5.3 to estimate the distribution in life for smooth test specimens. In air, the xth percentile of the distribution on life $N_{25}[x]$ for CSs is

$$\ln(N_{25}) = 6.595 + 0.52\, F^{-1}[x] - 1.975 \ln(\varepsilon_a - 0.113 + 0.026\, F^{-1}[1-x]) - 0.00124\, T \qquad (5.6a)$$

and for LASs it is

$$\ln(N_{25}) = 6.658 + 0.52\, F^{-1}[x] - 1.808 \ln(\varepsilon_a - 0.151 + 0.026\, F^{-1}[1-x]) - 0.00124\, T\,. \qquad (5.6b)$$

In LWR environments, the xth percentile of the distribution on life $N_{25}[x]$ for CSs is

$$\ln(N_{25}) = 6.010 + 0.52\, F^{-1}[x] - 1.975 \ln(\varepsilon_a - 0.113 + 0.026\, F^{-1}[1-x]) + 0.101\, S^*\, T^*\, O^*\, \dot{\varepsilon}^* \qquad (5.7a)$$

and for LASs it is

$$\ln(N_{25}) = 5.729 + 0.52\, F^{-1}[x] - 1.808 \ln(\varepsilon_a - 0.151 + 0.026\, F^{-1}[1-x]) + 0.101\, S^*\, T^*\, O^*\, \dot{\varepsilon}^*. \qquad (5.7b)$$

The parameters S^*, T^*, O^*, and $\dot{\varepsilon}^*$ are defined in Eqs. 5.5, and $F^{-1}[\cdot]$ denotes the inverse of the standard normal cumulative distribution function. The coefficients of distribution functions $F^{-1}[x]$ and $F^{-1}[1-x]$ represent the standard deviation on life and strain amplitude, respectively. For convenience, values of the inverse of standard normal cumulative distribution function in Eqs. 5.6 and 5.7 are given in Table 3. The standard deviation of 0.026 on strain amplitude obtained from the analysis may be an overly conservative value. A more realistic value for the standard deviation on strain could be obtained by analysis of the fatigue limits of different heats of material. The existing data are inadequate for such an analysis because (a) not enough heats of materials are included in the data base, and (b) there are very few high-cycle fatigue data for accurate estimations of the fatigue limit for specific heats.

The estimated probability curves for the fatigue life of CSs and LASs in an air and LWR environments in Figs. 71-73 show good agreement with experimental data; nearly all of the data are bounded by the 5% probability curve. Relative to the 50% probability curve, the 5% probability curve is a factor of ≈2.5 lower in life at strain amplitudes >0.3% and a factor of 1.4-1.7 lower in strain at <0.2% strain amplitudes. Similarly, the 1% probability curve is a factor of ≈3.7 lower in life and a factor of 1.7-2.2 lower in strain.

Table 3. Inverse of standard cumulative distribution function

Probability	$F^{-1}[x]$	$F^{-1}[1-x]$	Probability	$F^{-1}[x]$	$F^{-1}[1-x]$
0.01	-3.7195	3.7195	3.00	-1.8808	1.8808
0.02	-3.5402	3.5402	5.00	-1.6449	1.6449
0.03	-3.4319	3.4319	7.00	-1.4758	1.4758
0.05	-3.2905	3.2905	10.00	-1.2816	1.2816
0.07	-3.1947	3.1947	20.00	-0.8416	0.8416
0.10	-3.0902	3.0902	30.00	-0.5244	0.5244
0.20	-2.8782	2.8782	50.00	0.0000	0.0000
0.30	-2.7478	2.7478	65.00	0.3853	-0.3853
0.50	-2.5758	2.5758	80.00	0.8416	-0.8416
0.70	-2.4573	2.4573	90.00	1.2816	-1.2816
1.00	-2.3263	2.3263	95.00	1.6449	-1.6449
2.00	-2.0537	2.0537	98.00	2.0537	-2.0537

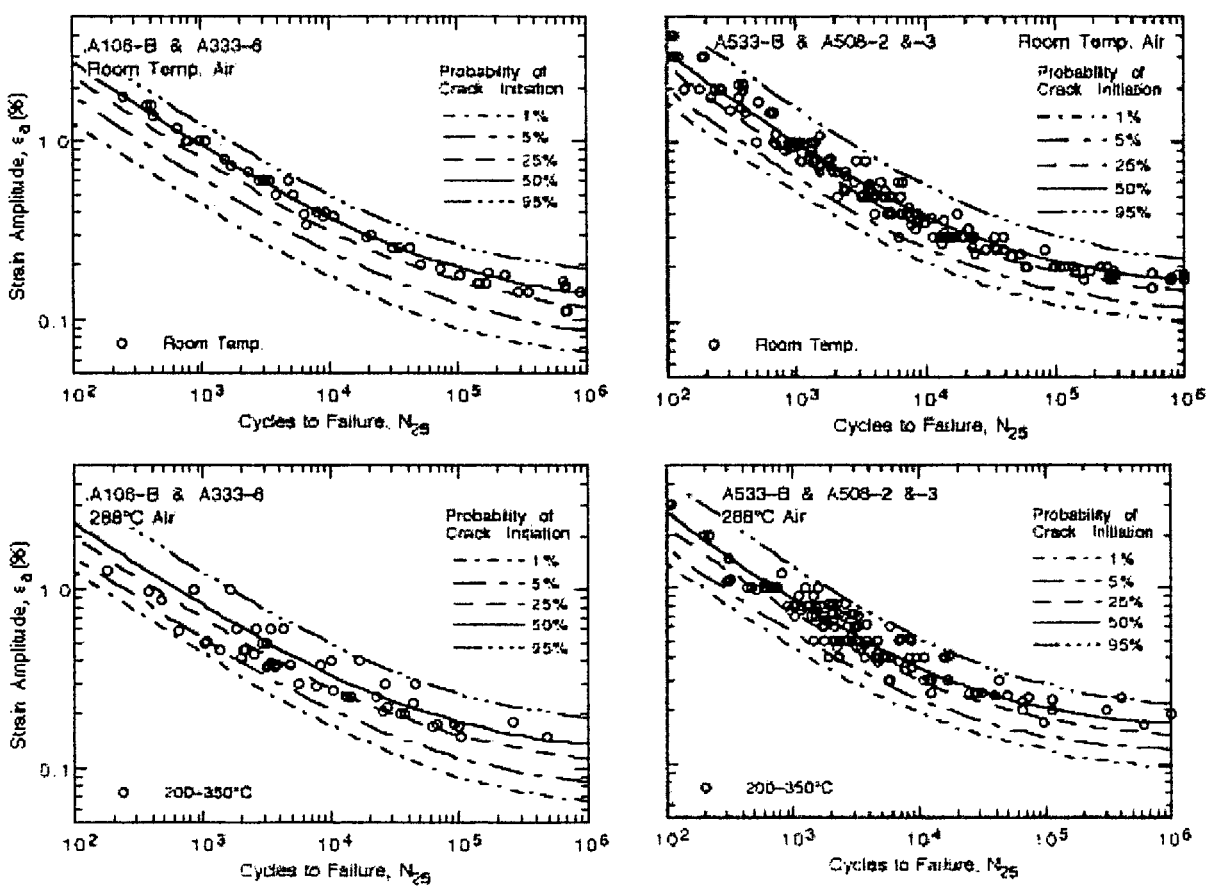

Figure 71. Experimental data and probability of fatigue cracking in carbon and low-alloy steel test specimens in air

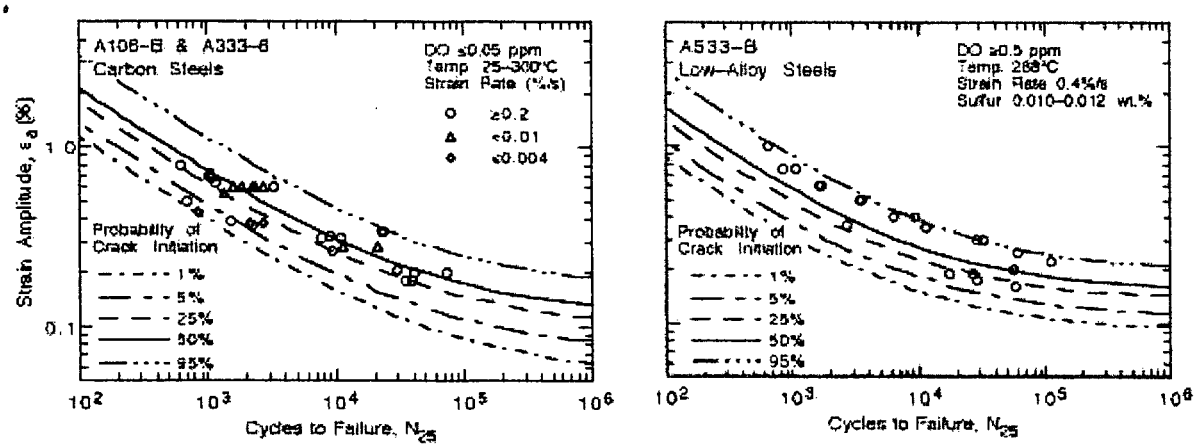

Figure 72. Experimental data and probability of fatigue cracking in carbon and low-alloy steel test specimens in simulated PWR environments

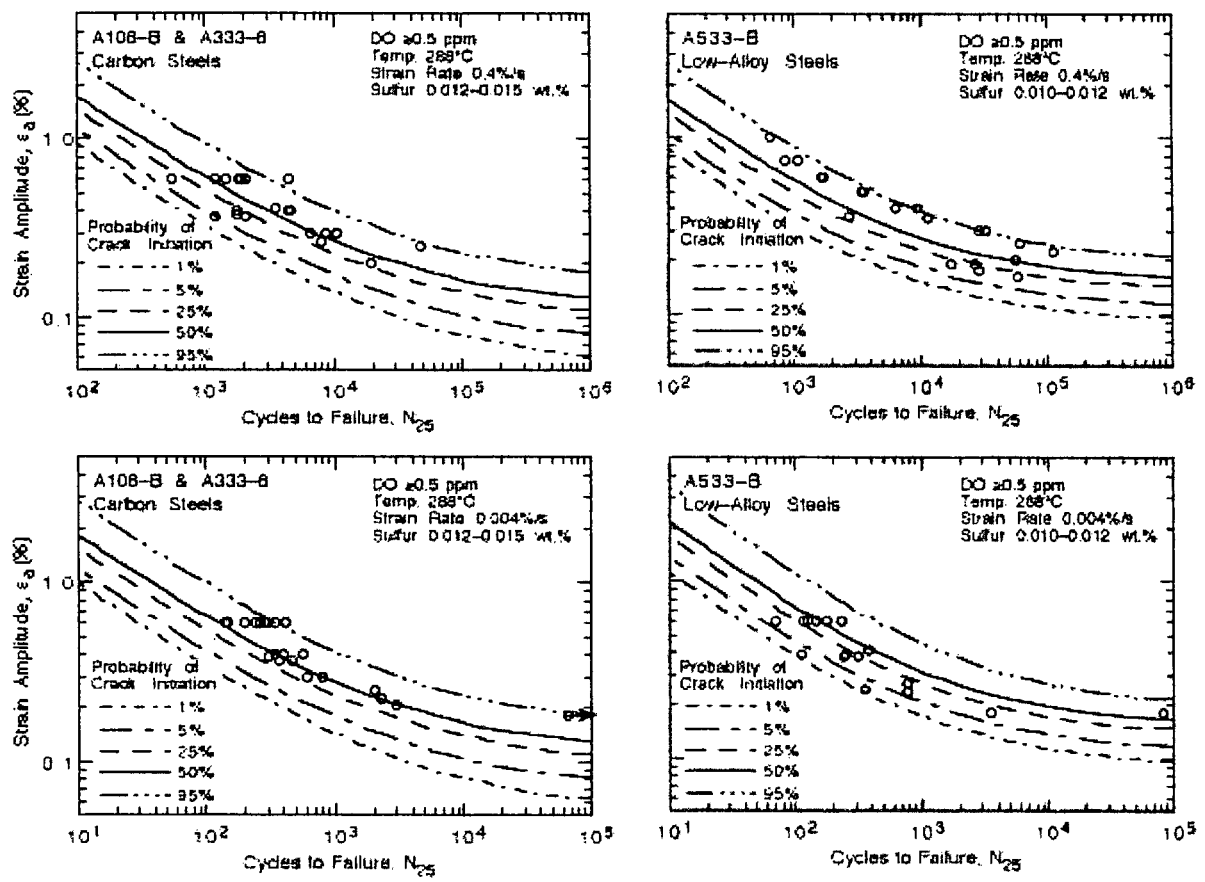

Figure 73. Experimental data and probability of fatigue cracking in carbon and low-alloy steel test specimens in high-dissolved-oxygen water

As with other aspects of this model, the estimates of the probability of cracking should not be extrapolated much beyond the data. The probabilities assume a normal distribution, which is consistent with the data for most of the range. The existing data are not sufficient to determine precise distributions because more data are required to estimate distributions than to estimate the mean curve. However, the assumption of normality is reasonable (and conservative) down to 0.1-1% probability of cracking and it is empirically verified by the number of data points that fall below the respective curves. The probability is not expected to deviate significantly from the normal curve for another order of magnitude (one more standard deviation) even if the probability distribution is not the same. Because estimates of extremely low or high probabilities are sensitive to the choice of distribution, the probability distribution curves should not be extrapolated beyond 0.02% probability.

6 Fatigue Life Correction Factor

An alternative approach for incorporating the effects of reactor coolant environments on fatigue S-N curves has been proposed by the Environmental Fatigue Data (EFD) Committee of the Thermal and Nuclear Power Engineering Society (TENPES) of Japan.* A fatigue life correction factor F_{en} is defined as the ratio of the life in air at room temperature to that in water at the service temperature. The fatigue usage for a specific load pair based on the current Code fatigue design curve is multiplied by the correction factor to account for the environmental effects. Note that the fatigue life correction factor does not account for any differences that might exist between the current ASME mean air curves and the present mean air curves developed from a larger data base. The specific expression for F_{en}, proposed initially by Higuchi and Iida,[7] assumes that life in the environment N_{water} is related to life in air N_{air} at room temperature through a power-law dependence on the strain rate

$$F_{en} = \frac{N_{air}}{N_{water}} = (\dot{\varepsilon})^{-P}, \tag{6.1a}$$

or

$$\ln(F_{en}) = \ln(N_{air}) - \ln(N_{water}) = -P\ln(\dot{\varepsilon}). \tag{6.1b}$$

In air at room temperature, the fatigue life N_{air} of CSs is expressed as

$$\ln(N_{air}) = 6.653 - 2.119\ \ln(\varepsilon_a - 0.108) \tag{6.2a}$$

and for LASs by

$$\ln(N_{air}) = 6.578 - 1.761\ \ln(\varepsilon_a - 0.140), \tag{6.2b}$$

where ε_a is the applied strain amplitude (%). Only the tensile loading cycle is considered to be important for environmental effects on fatigue life. The exponent P is a product of a environmental factor R_p, which depends on temperature T (°C) and DO level (ppm), and a material factor P_c, which depends on the ultimate tensile strength σ_u (MPa) and sulfur content S (wt/.%) of the steel. Thus

$$P = R_p\ P_c, \tag{6.3a}$$

* Presented at the Pressure Vessel Research Council Meeting, April 1996, Orlando, FL.

$$P_c = 0.864 - 0.00092\,\sigma_u + 14.6\,S, \tag{6.3b}$$

$$R_p = \frac{R_{pT} - 0.2}{2.64}\ln(DO) + 1.75R_{pT} - 0.035, \qquad 0.2 \le R_p \le R_{pT} \tag{6.3c}$$

and $R_{pT} = 0.198\exp(0.00557T)$. (6.3d)

The fatigue lives of carbon and low-alloy steels measured experimentally and those estimated from the statistical and EFD models are shown in Figs. 74–78. Although the EFD correlations for exponent P were based entirely on data for carbon steels, Eqs. 6.3a–6.3d were also used for estimating the fatigue lives of LASs. Also, σ_u in Eq. 6.3b was assumed to be 520 and 650 MPa, respectively, for CSs and LASs. The significant differences between the two models are as follows:

(a) The EFD correlations have been developed from data for CSs alone.

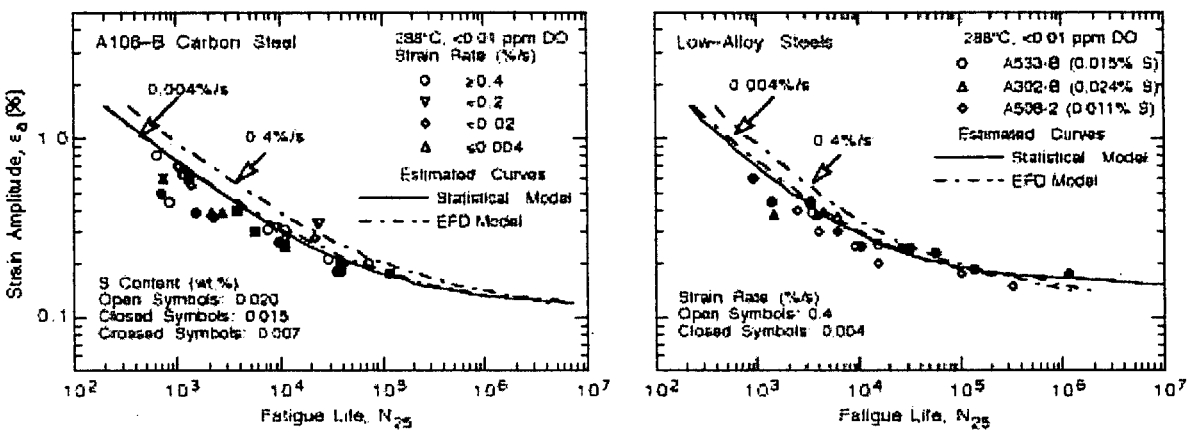

Figure 74. Experimental fatigue lives and those estimated from statistical and EFD models for carbon and low-alloy steels in simulated PWR water

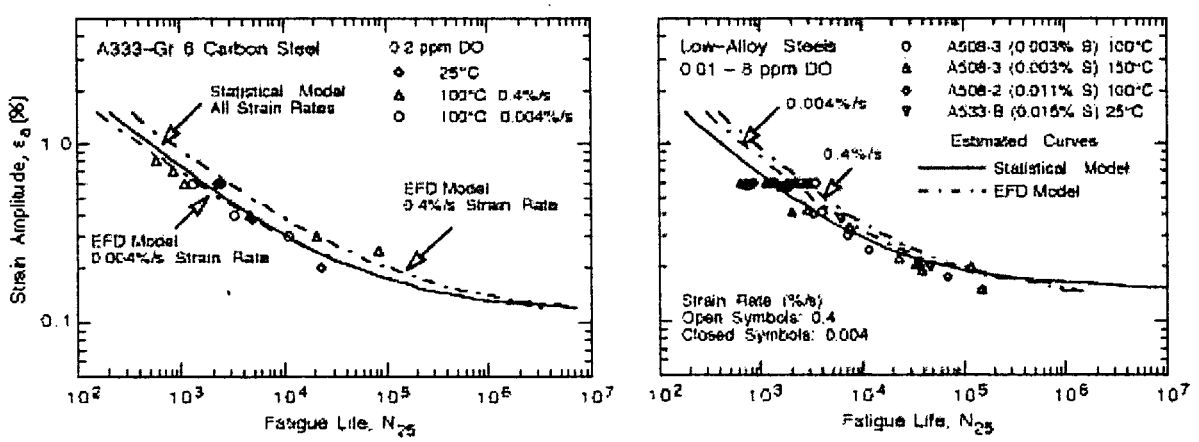

Figure 75. Experimental fatigue lives and those estimated from statistical and EFD models for carbon and low-alloy steels in water at temperatures below 150°C

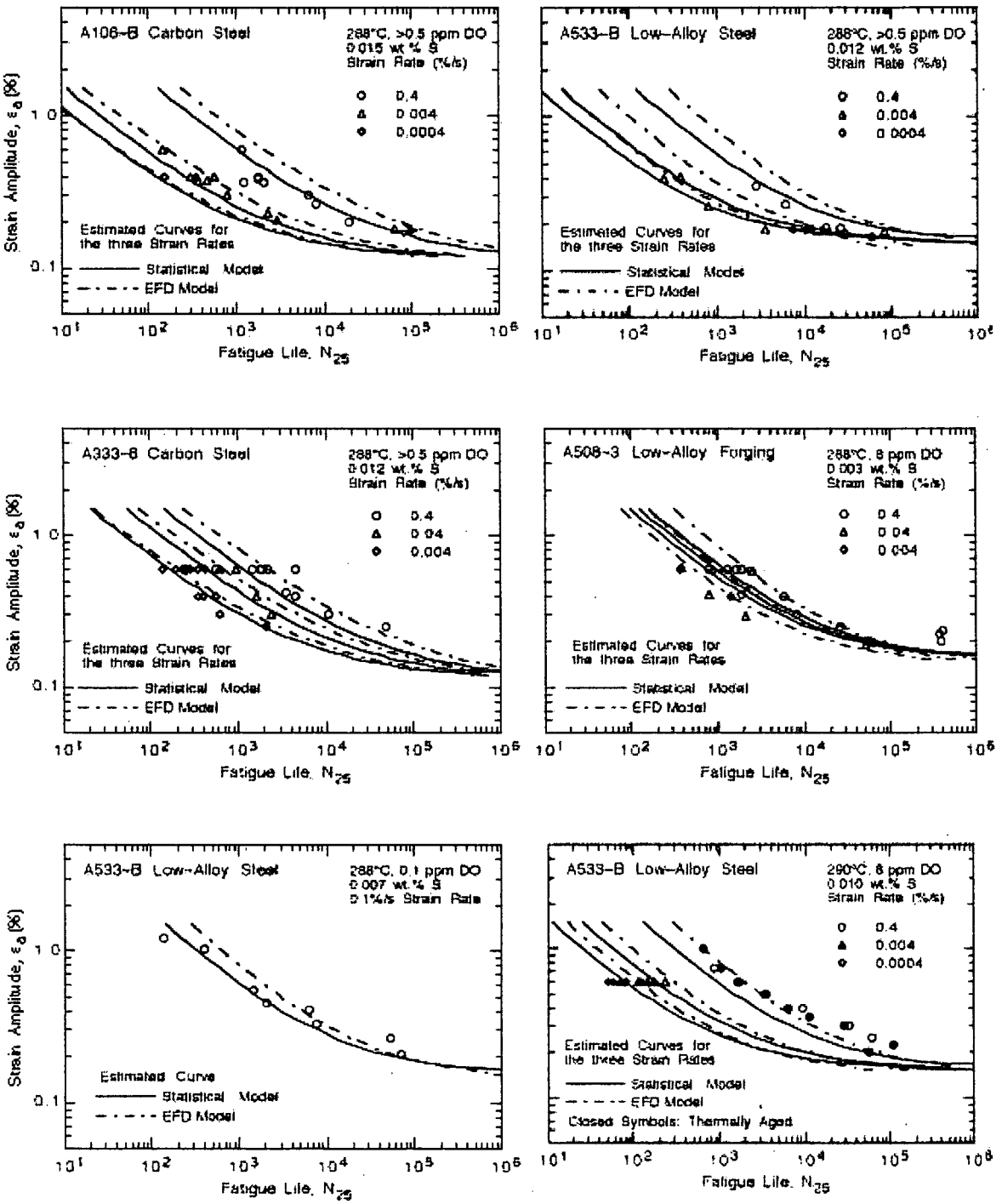

Figure 76. Experimental fatigue lives and those estimated from statistical and EFD models for carbon and low-alloy steels in high-dissolved-oxygen water

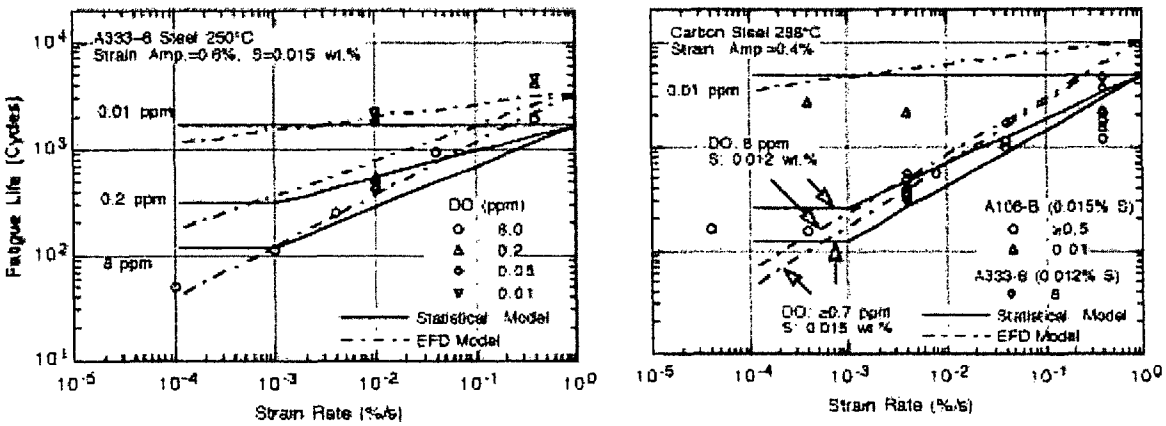

Figure 77. Dependence on strain rate of fatigue life of carbon steels observed experimentally and that estimated from statistical and EFD models

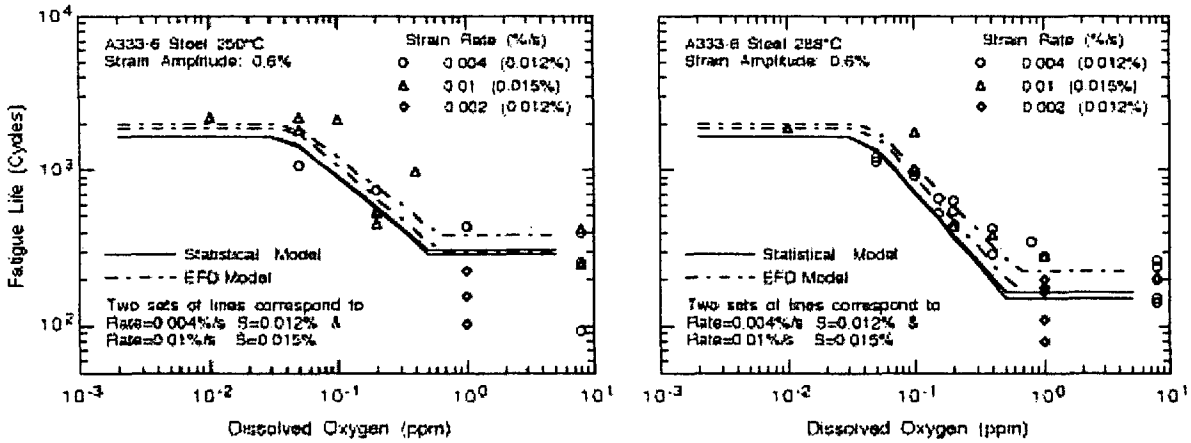

Figure 78. Dependence on dissolved oxygen of fatigue life of carbon steels observed experimentally and that estimated from statistical and EFD models

(b) The statistical model assumes that the effects of strain rate on fatigue life saturate below 0.001%/s, Fig. 77. Such a saturation is not considered in the EFD model.

(c) A threshold temperature of 150°C below which environmental effects on fatigue life are modest is incorporated in the statistical model but not in the EFD model.

(d) The EFD model includes the effect of tensile strength on fatigue life of CSs in LWR environments.

Another estimate of the fatigue life correction factor F_{en} can also be obtained from the statistical model. Since

$$\ln(F_{en}) = \ln(N_{air}) - \ln(N_{water}), \tag{6.4}$$

from Eqs. 5.3a and 5.4a, the fatigue life correction factor for CSs is given by

$$\ln(F_{en}) = 0.585 - 0.00124T - 0.101S^*T^*O^*\dot{\varepsilon}^* \tag{6.5a}$$

and from Eqs. 5.3b and 5.4b, the fatigue life correction factor for LASs is given by

$$\ln(F_{en}) = 0.929 - 0.00124T - 0.101S^*T^*O^*\dot{\varepsilon}^*, \tag{6.5b}$$

where the threshold and saturation values for S^*, T^*, O^*, and $\dot{\varepsilon}^*$ are defined in Eqs. 5.5. A value of 25°C is used for T in Eqs. 6.5a and 6.5b if the fatigue life correction factor is defined relative to RT air. Otherwise, both T and T^* represent the service temperature. A fatigue life correction factor F_{en} based on the statistical model has been proposed as part of a nonmandatory Appendix to ASME Section IX fatigue evaluations.[94,95]

7 Fatigue S–N Curves for Components

The current ASME Section III Code design fatigue curves were based on experimental data on small polished test specimens. The best-fit or mean curve to the experimental data used to develop the Code design curve, expressed in terms of stress amplitude S_a (MPa) and fatigue cycles N, for carbon steels is given by

$$S_a = 59{,}736/\sqrt{N} + 149.24 \tag{7.1a}$$

and for low-alloy steels by

$$S_a = 49{,}222/\sqrt{N} + 265.45. \tag{7.1b}$$

The stress amplitude S_a is the product of strain amplitude ε_a and elastic modulus E; the room temperature value of 206.8 GPa (30,000 ksi) for the elastic modulus for carbon and low-alloy steels was used in converting the experimental strain-versus-life data to stress-versus-life curves. To obtain design fatigue curves the best-fit curves (Eqs. 7.1a and 7.1b) were first adjusted for the effect of mean stress based on the modified Goodman relation

$$S'_a = S_a\left(\frac{\sigma_u - \sigma_y}{\sigma_u - S_a}\right) \qquad \text{for } S_a < \sigma_y, \tag{7.2a}$$

and

$$S'_a = S_a \qquad \text{for } S_a > \sigma_y, \tag{7.2b}$$

where S'_a is the adjusted value of stress amplitude, and σ_y and σ_u are yield and ultimate strengths of the material, respectively. The Goodman relation assumes the maximum possible mean stress and typically gives a conservative adjustment for mean stress at least when environmental effects are not significant. The design fatigue curves were then obtained by lowering the adjusted best-fit curve by a factor of 2 on stress or 20 on cycles, whichever was more conservative, at each point on the curve. The factor of 20 on cycles was intended to account for the uncertainties in fatigue life associated with material and loading conditions, and the factor of 2 on strain was intended to account for uncertainties in threshold strain caused by material variability. This procedure is illustrated for CSs and LASs in Fig. 79.

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Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels

Argonne National Laboratory

U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
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Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels

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EFFECTS OF LWR COOLANT ENVIRONMENTS ON FATIGUE DESIGN CURVES OF AUSTENITIC STAINLESS STEELS

by

O. K. Chopra

Abstract

The ASME Boiler and Pressure Vessel Code provides rules for the construction of nuclear power plant components. Figures I-9.1 through I-9.6 of Appendix I to Section III of the Code specify fatigue design curves for structural materials. While effects of reactor coolant environments are not explicitly addressed by the design curves, test data indicate that the Code fatigue curves may not always be adequate in coolant environments. This report summarizes work performed by Argonne National Laboratory on fatigue of austenitic stainless steels in light water reactor (LWR) environments. The existing fatigue S-N data have been evaluated to establish the effects of various material and loading variables such as steel type, dissolved oxygen level, strain range, strain rate, and temperature on the fatigue lives of these steels. Statistical models are presented for estimating the fatigue S-N curves as a function of material, loading, and environmental variables. Design fatigue curves have been developed for austenitic stainless steel components in LWR environments. The extent of conservatism in the design fatigue curves and an alternative method for incorporating the effects of LWR coolant environments into the ASME Code fatigue evaluations are discussed.

hydrogen-induced cracking. Fatigue striations should not be observed if enhancement of crack growth is caused by the slip oxidation/dissolution process.

5 Statistical Model

The fatigue S-N curves are generally expressed in terms of the Langer equation,[6] which may be used to represent either strain amplitude in terms of life or life in terms of strain amplitude. The parameters of the equation are commonly established through least-squares curve-fitting of the data to minimize the sum of the square of the residual errors for either fatigue life or strain amplitude. A predictive model based on least-squares fit on life is biased for low strain amplitude. The model leads to probability curves that converge to a single value of strain, and fails to address the fact that at low strain values, most of the error in life is due to uncertainty associated with either measurement of strain or variation in fatigue limit caused by material variability. On the other hand, a least-squares fit on strain does not work well for higher strain amplitudes. Statistical models have been developed at ANL[33,34] by combining the two approaches and minimizing the sum of the squared Cartesian distances from the data point to the predicted curve; the models were later updated with a larger fatigue S-N data base.[31] The functional forms and transformation for the different variables were based on experimental observations and data trends.

In air, the model assumes that fatigue life is independent of temperature and that strain rate effects occur at temperatures >250°C. It is also assumed that the effect of strain rate on life depends on temperature. One data set, obtained on Type 316 SS in room-temperature air, was excluded from the analysis. The tests in this data set were conducted in load-control mode at stress levels in the range of 190-230 MPa. The strain amplitudes were calculated only as elastic strains, i.e., strain amplitudes of 0.1-0.12% (the data are shown as circles in Fig. 5, with fatigue lives of 4×10^5 to 3×10^7). Based on cyclic stress vs. strain correlations for Type 316 SS (Eqs. 4a-4f), actual strain amplitudes for these tests should be 0.23-0.32%. In air, the fatigue life N of Types 304 and 316 SS is expressed as

$$\ln(N) = 6.703 - 2.030 \ln(\varepsilon_a - 0.126) + T^* \dot{\varepsilon}^* \tag{5a}$$

and that of Type 316NG, as

$$\ln(N) = 7.422 - 1.671 \ln(\varepsilon_a - 0.126) + T^* \dot{\varepsilon}^*, \tag{5b}$$

where ε_a is the strain amplitude (%) and T^* and $\dot{\varepsilon}^*$ are transformed temperature and strain rate, respectively, defined as follows:

$$\begin{aligned} T^* &= 0 && (T < 250°C) \\ T^* &= [(T - 250)/525]^{0.84} && (250 \le T < 400°C) \end{aligned} \tag{6a}$$

$$\begin{aligned} \dot{\varepsilon}^* &= 0 && (\dot{\varepsilon} > 0.4\%/s) \\ \dot{\varepsilon}^* &= \ln(\dot{\varepsilon}/0.4) && (0.0004 \le \dot{\varepsilon} \le 0.4\%/s) \\ \dot{\varepsilon}^* &= \ln(0.0004/0.4) && (\dot{\varepsilon} < 0.0004\%/s). \end{aligned} \tag{6b}$$

In LWR environments, the fatigue lives of austenitic SSs depends on strain rate, DO level, and temperature; the decrease in life is greater at low-DO levels and high temperatures. However, existing data are inadequate to establish the functional form for the dependence of fatigue life on DO level or temperature. Separate correlations have been developed for low- and high-DO levels (< or ≥ 0.05 ppm), and low and high temperatures (< or ≥ 200°C). Also, a threshold strain rate of

0.4%/s and saturation rate of 0.0004%/s is assumed in the model. Furthermore, for convenience in incorporating environmental effects into fatigue evaluations, the slope of the S-N curve in LWR environments was assumed to be the same as that in air although the best-fit of the experimental data in water yielded a slope for the S-N curve that differed from the slope of the curve that was obtained in air. In LWR environments, the fatigue life N of Types 304 and 316 SS is expressed as

$$\ln(N) = 5.768 - 2.030 \ln(\varepsilon_a - 0.126) + T^* \dot{\varepsilon}^* O^* \tag{7a}$$

and that of Type 316NG, as

$$\ln(N) = 6.913 - 1.671 \ln(\varepsilon_a - 0.126) + T^* \dot{\varepsilon}^* O^*, \tag{7b}$$

where the constants for transformed temperature, strain rate, and DO are defined as follows:

$$\begin{aligned} T^* &= 0 && (T < 200°C) \\ T^* &= 1 && (T \geq 200°C) \end{aligned} \tag{8a}$$

$$\begin{aligned} \dot{\varepsilon}^* &= 0 && (\dot{\varepsilon} > 0.4\%/s) \\ \dot{\varepsilon}^* &= \ln(\dot{\varepsilon}/0.4) && (0.0004 \leq \dot{\varepsilon} \leq 0.4\%/s) \\ \dot{\varepsilon}^* &= \ln(0.0004/0.4) && (\dot{\varepsilon} < 0.0004\%/s) \end{aligned} \tag{8b}$$

$$\begin{aligned} O^* &= 0.260 && (DO < 0.05 \text{ ppm}) \\ O^* &= 0.172 && (DO \geq 0.05 \text{ ppm}). \end{aligned} \tag{8c}$$

The model is recommended for predicted fatigue lives $\leq 10^6$ cycles. Recent test results indicate that for high-DO environments, conductivity of water is important for environmental effects on fatigue life of austenitic SSs. Therefore, the above correlations may be conservative for high-DO, i.e., ≥0.05 ppm DO, environments. The experimental values of fatigue life in air and water and those predicted from Eqs. 5-8 are plotted in Fig. 24. The estimated fatigue S-N curves for Types 304, 316, and 316NG SSs in air and LWR environments are shown in Figs. 5 and 25, respectively. The predicted fatigue lives show good agreement with the experimental data. Note that the ASME mean curve is not consistent with the existing fatigue S-N data (Fig. 5). Also, although the best-fit of the S-N data in LWR environments (Fig. 25) yields a steeper slope, the slope of the S-N curve in water was assumed to be the same as in air.

Upon completion of the modeling phase, the residual errors (i.e., the Cartesian distance from the prediction curve) should not show significant patterns, such as heteroskedasticity (changing variance), or a nonzero slope. The residual errors for each variable, grouped by steel type and environment (air or water), are plotted in Figs. 26-30. Most data subsets and plots

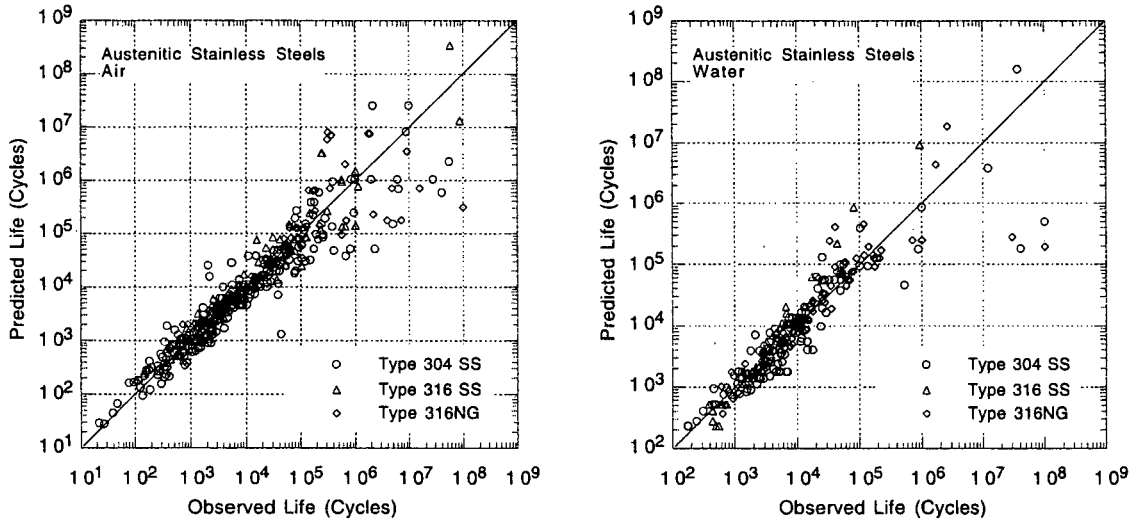

Figure 24. Experimental and predicted values of fatigue lives of austenitic SSs in air and water environments

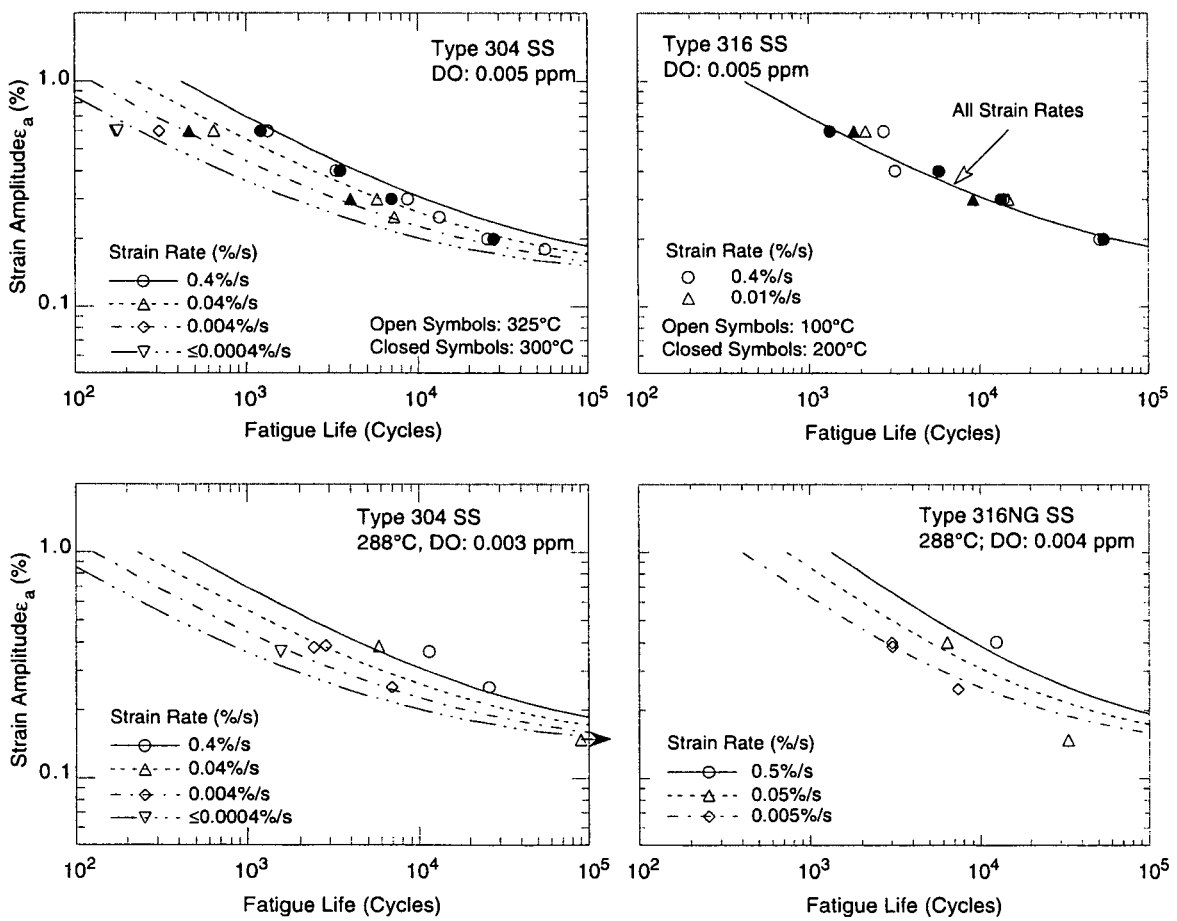

Figure 25. Experimental fatigue lives and those estimated from statistical models for austenitic SSs in water environments

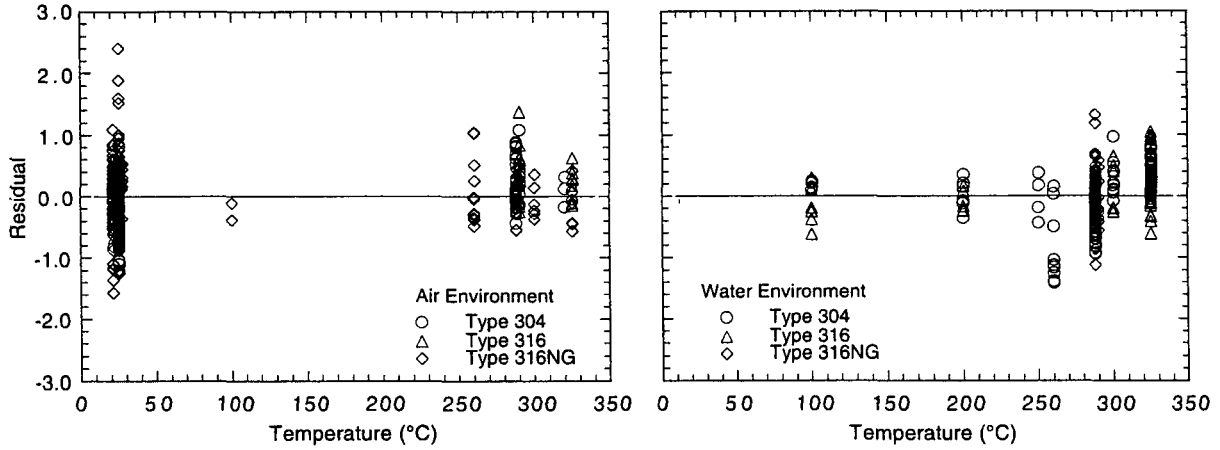

Figure 26. Residual error for austenitic SSs as a function of test temperature

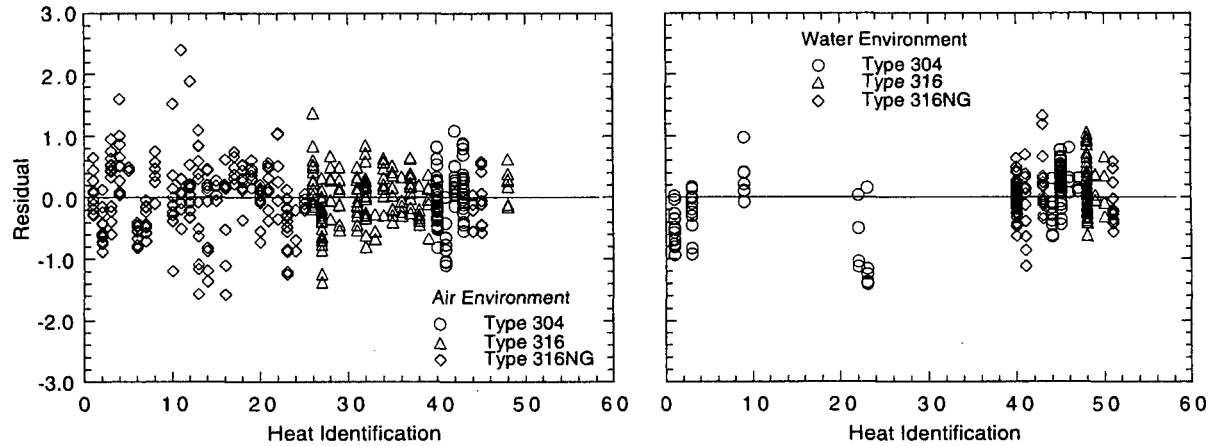

Figure 27. Residual error for austenitic SSs as a function of material heat

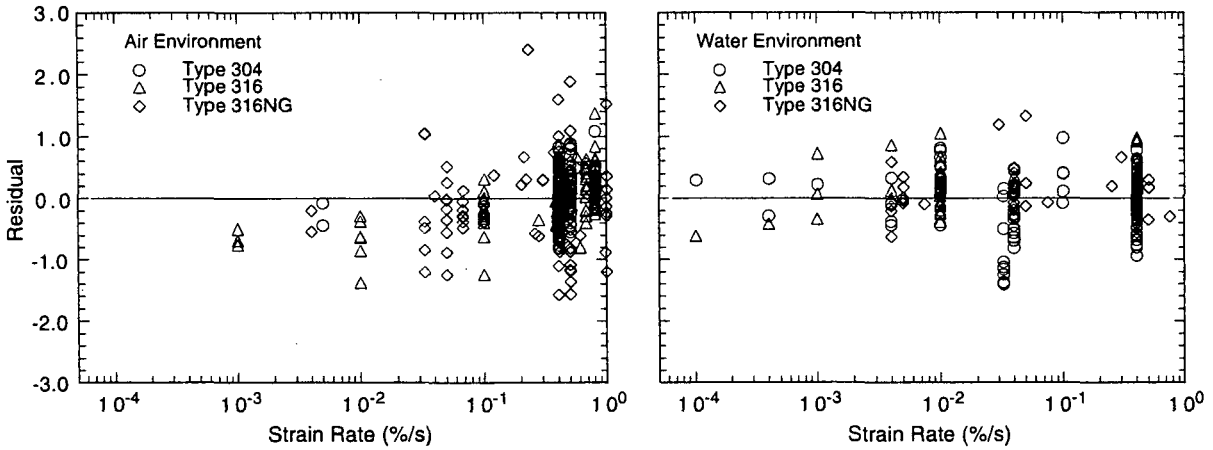

Figure 28. Residual error for austenitic SSs as a function of loading strain rate

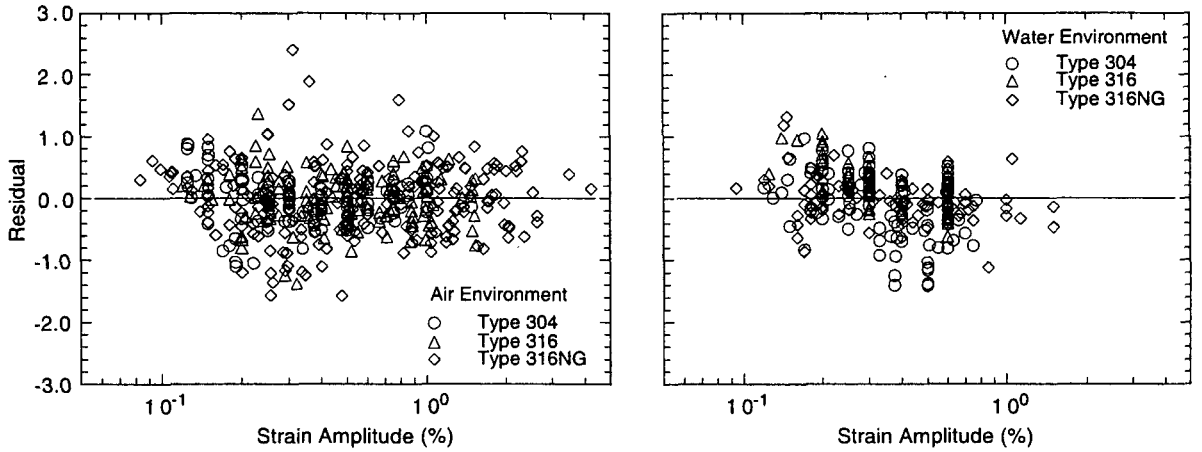

Figure 29. Residual error for austenitic SSs as a function of applied strain amplitude

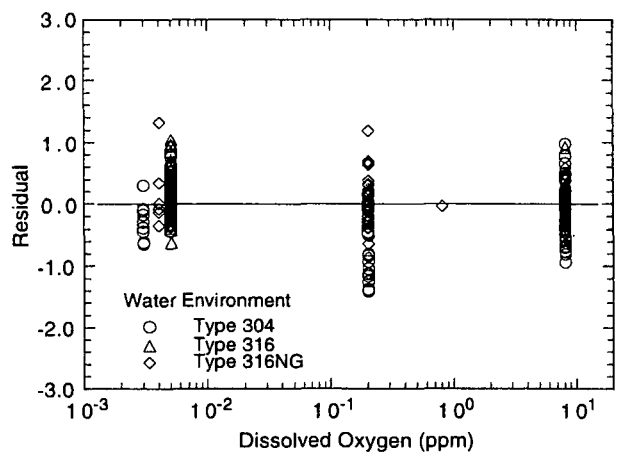

Figure 30.
Residual error for austenitic SSs as a function of dissolved oxygen in water

do not show patterns. In general, high variance tends to be associated with longer lives and lower strain amplitudes. Furthermore, biases seem to be traceable to heat-to-heat variation.

6 Design Fatigue Curves

The design fatigue curves in the current ASME Section III Code were based on experimental data on small polished test specimens. The curves were obtained by adjusting the best-fit curve for the effect of mean stress and then lowering the adjusted curve by a factor of 2 on stress or 20 on life, whichever was more conservative, at each point of the curve. The best-fit curve to the experimental data,[51] expressed in terms of strain amplitude ε_a (%) and fatigue cycles N, for austenitic SSs is given by

$$\ln[N] = 6.954 - 2.0 \ln(\varepsilon_a - 0.167). \tag{9}$$

The mean curve, expressed in terms of stress amplitude S_a (MPa), which is the product of ε_a and elastic modulus E, is given by

$$S_a = 58020/(N)^{1/2} + 299.92. \tag{10}$$

The strain-vs.-life data were converted to stress-vs.-life curves by using the room-temperature value of 195.1 GPa (28300 ksi) for the elastic modulus. The best-fit curves were adjusted for the effect of mean stress by using the modified Goodman relationship[46]

$$S'_a = S_a\left(\frac{\sigma_u - \sigma_y}{\sigma_u - S_a}\right) \qquad \text{for } S_a < \sigma_y, \tag{10a}$$

and $$S'_a = S_a \qquad \text{for } S_a > \sigma_y, \tag{10b}$$

where S'_a is the adjusted value of stress amplitude, and σ_y and σ_u are yield and ultimate strengths of the material, respectively. The Goodman relationship assumes the maximum possible mean stress and typically gives a conservative adjustment for mean stress, at least when environmental effects are not significant. The design fatigue curves were then obtained by lowering the adjusted best-fit curve by a factor of 2 on stress or 20 on cycles, whichever was more conservative, to account for differences and uncertainties in fatigue life associated with material and loading conditions.

The same procedure has been used to develop design fatigue curves for LWR environments. However, because of the differences between the ASME mean curve and the best-fit curve to existing fatigue data (Fig. 5), the margin on strain for the current ASME Code design fatigue curve is closer to 1.5 than 2. Therefore, to be consistent with the current Code design curve, a factor of 1.5 rather than 2 was used in developing the design fatigue curves from the updated statistical models in air and LWR environments.

The design fatigue curves based on the statistical model for Types 304 and 316 SS in air and low- and high-DO water are shown in Figs. 31-33. A similar set of curves can be obtained for Type 316NG SS. Because the fatigue life of Type 316NG is superior to that of Types 304 or 316 SS, Figs. 31-33 may be used conservatively for Type 316NG SS. Also, as mentioned earlier, recent test results indicate that the conductivity of water is important for environmental effects on fatigue life of austenitic SSs in high-DO environments. Therefore, the design fatigue curves for Type 304 and 316 SS in water with ≥0.05 ppm DO (Fig. 33) may be conservative.

Although, in air at low stress levels, the differences between the current ASME Code design curve and the design curve obtained from the updated statistical model at temperatures <250°C have been reduced or eliminated by reducing the margin on stress from 2 to 1.5, significant differences still exist between the two curves. For example, at stress amplitudes >300 MPa, estimates of life from the updated design curve are a factor of ≈2 lower than those from the ASME Code curve. Therefore, the actual margins on stress and life for the current ASME Code design fatigue curve are 1.5 and 10, respectively, instead of 2 and 20.

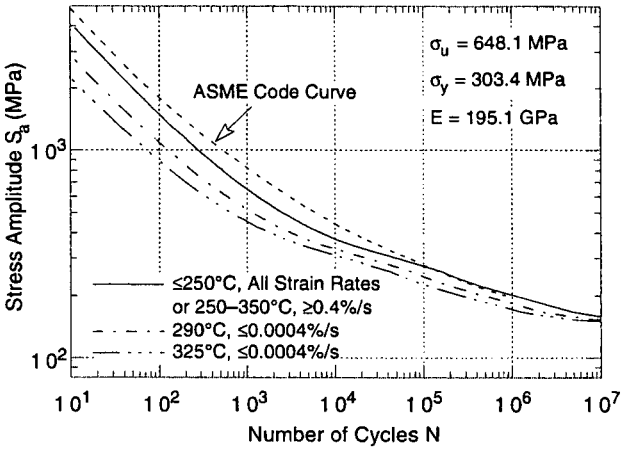

Figure 31.
ASME and statistical-model design fatigue curves for Types 304 and 316 SS in air

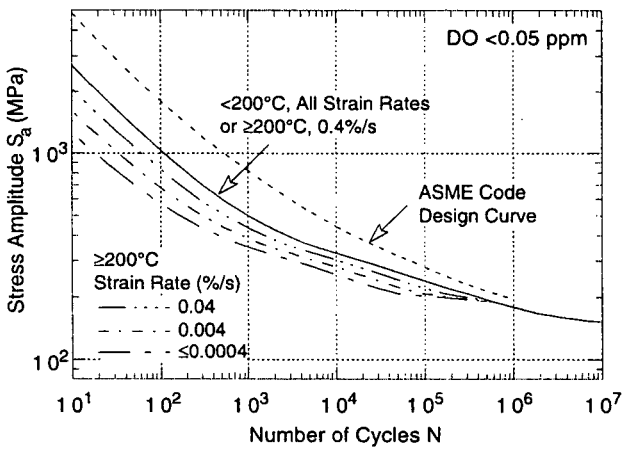

Figure 32.
ASME and statistical-model design fatigue curves for Types 304 and 316 SS in water with <0.05 ppm DO

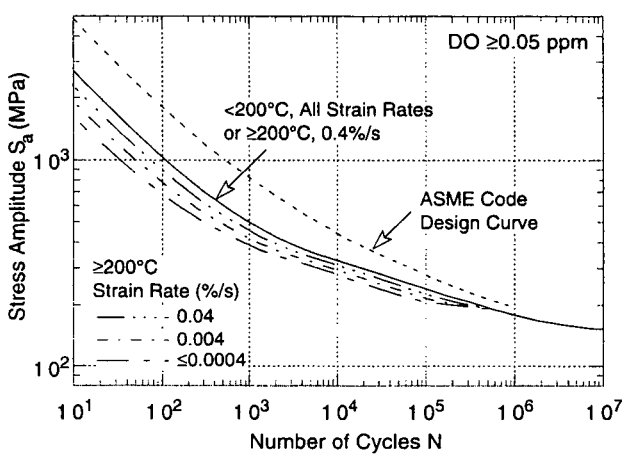

Figure 33.
ASME and statistical-model design fatigue curves for Types 304 and 316 SS in water with ≥0.05 ppm DO

As discussed above, the existing fatigue data indicate a threshold strain range of ≈0.32%, below which environmental effects on the fatigue life of austenitic SSs either do not occur or are insignificant. This value must be adjusted for the effects of mean stress and uncertainties due to material and loading variability. Threshold strain amplitudes are decreased by ≈10% to account for mean stress effects and by a factor of 1.5 to account for uncertainties in fatigue life associated with material and loading variability. Thus, a threshold strain amplitude of 0.097% (stress amplitude of 189 MPa) was selected, below which environmental effects on life are modest and are represented by the design curve for temperatures <200°C (shown by the solid line in Figs. 31 and 32).

These curves can be used to perform ASME Code fatigue evaluations of components that are in service in LWR environments. For each set of load pairs, a partial usage factor is obtained from the appropriate design fatigue curve. Information about the service conditions, such as temperature, strain rate, and DO level, are required for the evaluations. The procedure for obtaining these parameters depends on whether the elapsed-time-vs.-temperature information for the transient is available. The maximum values of temperature and DO level and the slowest strain rate during the transient may be used for a conservative estimate of life. Note that the design curves in LWR environments not only account for environmental effects on life but also include the difference between the current Code design curve and the updated design curve in air, i.e., the difference between the solid and dashed curves in Fig. 31.

7 Fatigue Life Correction Factor

The effects of reactor coolant environments on fatigue life have also been expressed in terms of a fatigue life correction factor F_{en}, which is the ratio of the life in air at room temperature to that in water at the service temperature.[11,52,53] To incorporate environmental effects into the ASME Code fatigue evaluation, a fatigue usage for a specific load pair, based on the current Code fatigue design curve, is multiplied by the correction factor. A fatigue life correction factor F_{en} can also be obtained from the statistical model, where

$$\ln(F_{en}) = \ln(N_{air}) - \ln(N_{water}). \tag{12}$$

From Eqs. 5a and 7a, the fatigue life correction factor relative to room-temperature air for Types 304 and 316 SSs is given by

$$F_{en} = \exp(0.935 - T^* \dot{\varepsilon}^* O^*), \tag{13}$$

where the threshold and saturation values for T^*, $\dot{\varepsilon}^*$, and O^* are defined in Eqs. 8a-8c. At temperatures ≥200°C and strain rates ≤0.0004%/s, Eq. 13 yields an F_{en} of ≈15 in low-DO PWR water (<0.05 ppm DO) and ≈8 in high-DO water (≥0.05 ppm DO). At temperatures <200°C, F_{en} is ≈2.5 in both low- and high-DO water at all strain rates.

8 Conservatism in Design Fatigue Curves

The overall conservatism in ASME Code fatigue evaluations has also been demonstrated in fatigue tests on piping welds and components.[54] In air, the margins on the number of cycles to failure for austenitic SS elbows and tees were 40-310 and 104-510, respectively. The margins for girth butt welds were significantly lower at 6-77. In these tests, fatigue life was expressed as the number of cycles for the crack to penetrate through the wall, which ranged in thickness from 6 to 18 mm (0.237 to 0.719 in). The fatigue design curves represent the number of cycles that are necessary to form a 3-mm-deep crack. Consequently, depending on wall thickness, the actual margins to failure may be lower by a factor of >2.

Deardorff and Smith[55] have discussed the types and extent of conservatisms present in the ASME Section III fatigue evaluations and the effects of LWR environments on fatigue margins. The sources of conservatism include design transients considerably more severe than those experienced in service, grouping of transients, and simplified elastic-plastic analysis. Environmental effects on two components, the BWR feedwater nozzle/safe end and PWR steam generator feedwater nozzle/safe end, both constructed from LAS and known to be affected by severe thermal transients,

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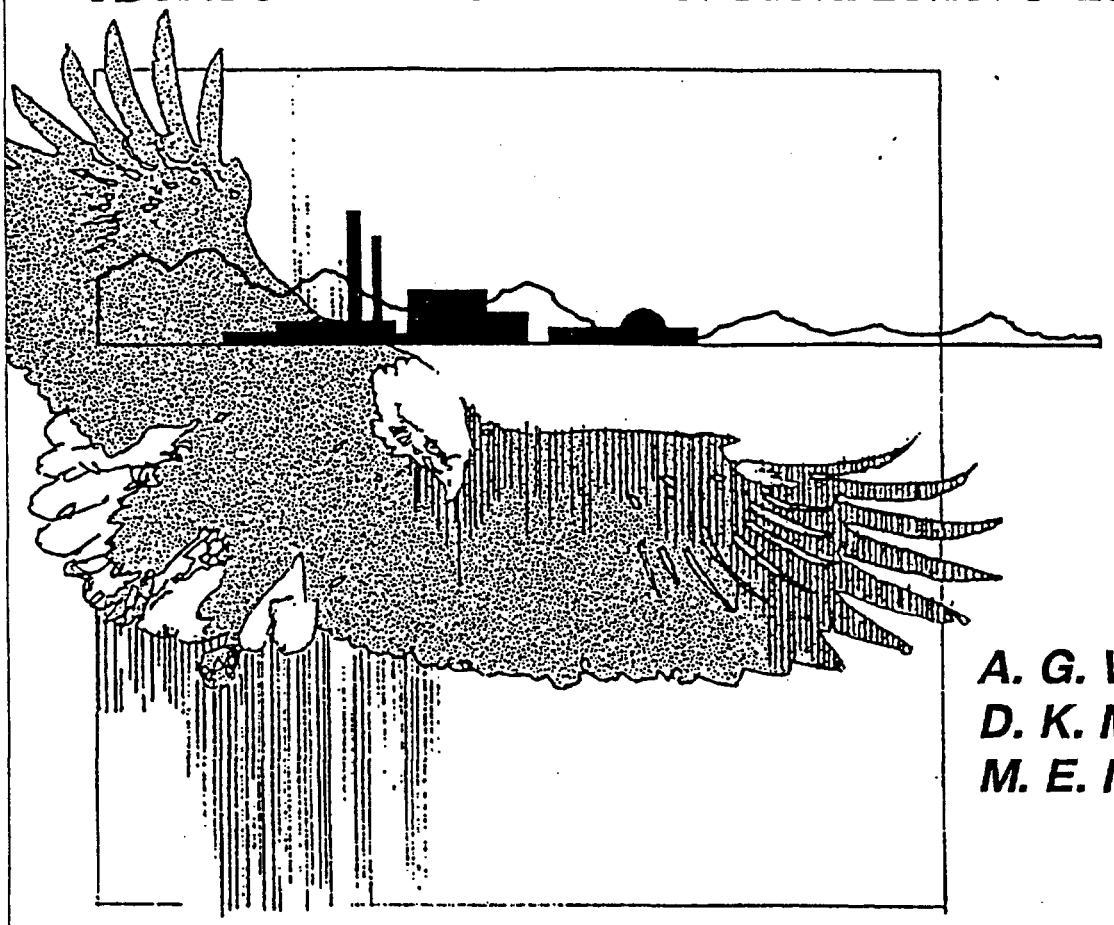

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Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components

Idaho National Engineering Laboratory

Lockheed Idaho Technologies Company

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FIN J2081

ABSTRACT

Recent test data indicate that the effects of the light water reactor (LWR) environment could significantly reduce the fatigue resistance of materials used in the reactor coolant pressure boundary components of operating nuclear power plants. Argonne National Laboratory has developed interim fatigue curves based on test data simulating LWR conditions, and published them in NUREG/CR-5999. In order to assess the significance of these interim fatigue curves, fatigue evaluations of a sample of the components in the reactor coolant pressure boundary of LWRs were performed. The sample consists of components from facilities designed by each of the four U.S. nuclear steam supply system vendors. For each facility, six locations were studied, including two locations on the reactor pressure vessel. In addition, there are older vintage plants where components of the reactor coolant pressure boundary were designed to codes that did not require an explicit fatigue analysis of the components. In order to assess the fatigue resistance of the older vintage plants, an evaluation was also conducted on selected components of three of these plants. This report discusses the insights gained from the application of the interim fatigue curves to components of seven operating nuclear power plants.

4. APPROACH

4.1 Selection of Components for Analysis

The components chosen for the evaluation of the five PWR plants [B&W, Combustion Engineering (one older vintage and one newer vintage), and Westinghouse (one older vintage and one newer vintage)] are as follows:

1. Reactor vessel shell and lower head.
2. Reactor vessel inlet and outlet nozzles.
3. Pressurizer surge line (including hot leg and pressurizer nozzles).
4. Reactor coolant piping charging system nozzle.
5. Reactor coolant piping safety injection nozzle.
6. Residual heat removal (RHR) system Class 1 piping.

The terminology used above is for Westinghouse plants. The first three components are the same for Combustion Engineering and B&W plants, but the latter three components for the three PWR nuclear steam supply system (NSSS) vendors are different either simply in name or in the routing of the piping. For cases where there is no direct one-for-one correspondence, the location that most nearly corresponded to the Westinghouse component was chosen. These locations are described in Section 5.

The components chosen for the evaluation of the two BWR plants [General Electric (one older vintage and one newer vintage)] are as follows:

1. Reactor vessel shell and lower head.
2. Reactor vessel feedwater nozzle.
3. Reactor recirculation piping (including inlet and outlet nozzles).
4. Core spray line reactor vessel nozzle and associated Class 1 piping.
5. RHR Class 1 piping.
6. Feedwater line Class 1 piping.

For both PWR and BWR plants, these components are not necessarily the locations with the highest design CUFs in the plant, but were chosen to give a representative overview of components that had higher CUFs and/or were important from a risk perspective. For example, the reactor vessel shell and lower head was chosen for its risk importance.

4.2 Application of NUREG/CR-5999 Fatigue Curves

NUREG/CR-5999 includes one fatigue curve for stainless steel, but several curves for carbon/low-alloy steels which are based on the sulfur content of the steel and the oxygen level in the coolant. For the five PWR plants, the curves for high-sulfur steel and a low-oxygen environment (typical for PWRs) were used. For the two BWR plants, the curves for high-sulfur steel and a high-oxygen environment were used. The high-oxygen (greater than 100 ppm) environment considered in the selected curves is consistent with the water chemistry in BWRs without hydrogen water chemistry. Neither of the two BWR plants evaluated have used hydrogen water chemistry.

4.2.1 Interior and Exterior Surfaces. The highest CUFs for components in the seven plants evaluated in this fatigue assessment study generally occur on the interior surfaces which experience the full effects of thermal shocks from fluid temperature changes. In a few cases the highest CUF was found to occur on the exterior surface (because of stress concentration effects), and in other cases no differentiation between interior and exterior surfaces was made in the licensee's calculations. Since it is expected that the interior

5.5 Older Vintage Westinghouse Plant

A comparison of the design CUFs from the licensee's design basis calculations and CUFs using the NUREG/CR-5999 interim fatigue curves was carried out for the locations of highest design CUF for the six components listed below:

1. Reactor vessel shell and lower head
2. Reactor vessel inlet and outlet nozzles
3. Pressurizer surge line (including hot leg and pressurizer nozzles)
4. Reactor coolant piping charging system nozzle (representative design basis fatigue calculation performed by INEL)
5. Reactor coolant piping safety injection nozzle (representative design basis fatigue calculation performed by INEL)
6. Residual Heat Removal system Class 1 piping (representative design basis fatigue calculation performed by INEL).

As of late 1993, the plant has been operated approximately 20 of the 40 years currently approved in its operating license. Table 5-83 shows the design basis cycles for transients that are important from a fatigue standpoint for the six components that were evaluated. The numbers of transients to date have been extrapolated to 40 years by multiplying by 40/20.

The results of a generic Westinghouse plant study of thermal stratification in surge lines was included in the licensee's fatigue analysis of the surge line. There were no plant specific data to remove conservatism assumptions for this particular plant.

5.5.1 Reactor Vessel Shell and Lower Head.

The highest CUF on the lower shell and head is 0.290 for the inside surface of the lower head near the shell-to-head transition, where core support guides are welded to the interior of the shell. The SA-302 Grade B head is protected from the coolant by a layer of stainless steel and Alloy 600 cladding. No fatigue analysis is performed for the cladding.

5.5.1.1 NUREG/CR-5999 CUF Based on Licensee's Design Calculation Stresses.

The licensee's CUF calculations used the ASME Code, Section III, 1965 edition, through Summer 1966 addenda.

The effect of the NUREG/CR-5999 interim fatigue curve is shown in Table 5-84. As previously discussed, the results shown in Table 5-84 assume that the coolant is in contact with the low-alloy steel base metal underneath the cladding. The S_{alt} values were adjusted for the effect of the modulus of elasticity by multiplying by 30/27, the ratio of the modulus of elasticity on the fatigue curve in the current edition of the Code to the value at 500°F for SA-302 Grade B low-alloy steel. The 1965 Code edition did not require an adjustment for the effect of the modulus of elasticity.

Table 5-83. Number of selected design basis cycles compared to anticipated number of cycles over 40-year license life.

Transient	Design basis cycles	Anticipated cycles for 40 years
Heatup/cooldown	200	172
Reactor trip	400	426
Hydrotest	5	2
5% power change	14500	512
10% power change (up/down)	2000/7000	42/86
50% power change	200	136

NYS-26/26A & RIVERKEEPER TC-1/1A: ATTACHMENT 10

Entergy Nuclear Northeast
Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
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Tel (914) 788-2055

Fred Dacimo
Vice President
License Renewal

January 22, 2008

Re: Indian Point Units 2 & 3
Docket Nos. 50-247 & 50-286
NL-08-021

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

SUBJECT: Entergy Nuclear Operations Inc.
Indian Point Nuclear Generating Unit Nos. 2 & 3
Docket Nos. 50-247 and 50-286
License Renewal Application Amendment 2

REFERENCES:

1. Entergy Letter dated April 23, 2007, F. R. Dacimo to Document Control Desk, "License Renewal Application" (NL-07-039)
2. Entergy Letter dated April 23, 2007, F. R. Dacimo to Document Control Desk, "License Renewal Application Boundary Drawings" (NL-07-040)
3. Entergy Letter dated April 23, 2007, F. R. Dacimo to Document Control Desk, "License Renewal Application Environmental Report References" (NL-07-041)

Dear Sir or Madam:

In the referenced letters, Entergy Nuclear Operations, Inc. (Entergy) applied for renewal of the Indian Point Energy Center operating licenses for Unit 2 and 3.

Based on discussions during license renewal audits, clarification to the LRA is provided in Attachment 1. This information clarifies the relationship between Commitment 33 regarding environmentally assisted fatigue and the Fatigue Monitoring Program described in LRA Section B.1.12. The Fatigue Monitoring Program includes the actions identified in Commitment 33 to address the evaluation of the effects of environmentally assisted fatigue in accordance with 10 CFR 54.21(c)(1)(iii). The revised Regulatory Commitment List is provided in Attachment 2.

If you have any questions, or require additional information, please contact Mr. Robert Walpole at 914-734-6710.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 1-22-08.

Sincerely,

Patricia W Conroy for per telecon

Fred R. Dacimo
Vice President
License Renewal

Attachments:

1. Fatigue Monitoring Program Clarification
2. Regulatory Commitment List, Revision 3

cc: Mr. Samuel J. Collins, Regional Administrator, NRC Region I
Mr. Kenneth Chang, NRC Branch Chief, Engineering Review Branch I
Mr. Bo M. Pham, NRC Environmental Project Manager
Mr. John Boska, NRR Senior Project Manager
Mr. Paul Eddy, New York State Department of Public Service
NRC Resident Inspector's Office
Mr. Paul D. Tonko, President, New York State Energy, Research, & Development Authority

ATTACHMENT 1 TO NL-08-021

Fatigue Monitoring Program Clarification

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 & 3
DOCKET NOS. 50-247 AND 50-286

License Renewal Application
Amendment 2

Fatigue Monitoring Program Clarification

LRA and commitment list revisions are provided below. (underline - added, strikethrough – deleted)

LRA Table 4.1-1, List of IP2 TLAA and Resolution, line item titled "Effects of reactor water environment on fatigue life", is revised as follows.

Effects of reactor water environment on fatigue life	~~Analyses remain valid 10 CFR 54.21(c)(1)(i)~~ ~~OR~~ Aging effect managed 10 CFR 54.21(c)(1)(iii)	4.3.3

LRA Table 4.1-2, List of IP3 TLAA and Resolution, line item titled "Effects of reactor water environment on fatigue life", is revised as follows.

Effects of reactor water environment on fatigue life	~~Analyses remain valid 10 CFR 54.21(c)(1)(i)~~ ~~OR~~ Aging effect managed 10 CFR 54.21(c)(1)(iii)	4.3.3

LRA Section 4.3.3, paragraph 10 is revised as follows.

At least 2 years prior to entering the period of extended operation, for the locations identified in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3)~~NUREG/CR-6260 for Westinghouse PWRs of the IPEC vintage~~, under the Fatigue Monitoring Program IPEC will implement one or more of the following.

(1) Consistent with the Fatigue Monitoring Program, Detection of Aging Effects, update the fatigue usage calculations using ~~R~~refined ~~the~~ fatigue analyses to determine valid CUFs less than 1.0 when accounting for the effects of reactor water environment. This includes applying the appropriate F_{en} factors to valid CUFs determined in accordance with one of the following.

For locations in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3)~~including NUREG/CR-6260 locations~~, with existing fatigue analysis valid for the period of

extended operation, use the existing CUF ~~to determine the environmentally adjusted CUF~~.

~~More limiting IPEC~~ Additional plant-specific locations with a valid CUF may be ~~added in addition to the NUREG/CR-6260 locations~~evaluated. In particular, the pressurizer lower shell will be reviewed to ensure the surge nozzle remains the limiting component.

Representative CUF values from other plants, adjusted to or enveloping the IPEC plant-specific external loads may be used if demonstrated applicable to IPEC.

An analysis using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case) may be performed to determine a valid CUF.

~~(2) Manage the effects of aging due to fatigue at the affected locations 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 acceptable to the NRC).~~

(~~3~~2) Consistent with the Fatigue Monitoring Program, Corrective Actions, ~~R~~repair or replace the affected locations before exceeding a CUF of 1.0.

~~Should IPEC select the option to manage the aging effects due to environmental assisted fatigue during the period of extended operation, details of the aging management program such as scope, qualification, method, and frequency will be submitted to the NRC at least 2 years prior to the period of extended operation.~~

~~Depending on the option chosen, which may vary by component, this TLAA will be projected through the period of extended operation per 10CFR54.21(c)(1)(ii) or t~~The effects of environmentally assisted fatigue will be managed per 10CFR54.21(c)(1)(iii).

LRA Section A.2.1.11, Fatigue Monitoring Program, is revised as follows.

The Fatigue Monitoring Program is an existing program that tracks the number of critical thermal and pressure transients for selected reactor coolant system components. The program ensures the validity of analyses that explicitly analyzed a specified number of fatigue transients by assuring that the actual effective number of transients does not exceed the analyzed number of transients. The program provides for update of the fatigue usage calculations to maintain a CUF of < 1.0 for the period of extended operation. For the locations identified in Section A.2.2.2.3, updated calculations will account for the effects of the reactor water environment. These calculation updates are governed by Entergy's 10 CFR 50 Appendix B Quality Assurance (QA) program and include design input verification and independent reviews ensuring that valid assumptions, transients, cycles, external loadings, analysis methods, and environmental fatigue life correction factors will be used in

the fatigue analyses. The program requires corrective actions including repair or replacement of affected components before fatigue usage calculations determine the CUF exceeds 1.0. Specific corrective actions are implemented in accordance with the IPEC corrective action program. Repair or replacement of the affected component(s), if necessary, will be in accordance with established plant procedures governing repair and replacement activities. These established procedures are governed by Entergy's 10 CFR 50 Appendix B QA program and meet the applicable repair or replacement requirements of the ASME Code Section XI.

LRA Section A.2.2.2.3, Environmental Effects on Fatigue, is revised as follows.

The effects of reactor water environment on fatigue were evaluated for license renewal. Projected cumulative usage factors (CUFs) were calculated for the limiting locations ~~identified in~~ based on NUREG/CR-6260. The identified IP2 locations are those listed in the license renewal application, Table 4.3-13. ~~For the locations with CUFs less than 1.0, the TLAA has been projected through the period of extended operation per 10 CFR 54.21(c)(1)(ii).~~ Several locations may exceed a CUF of 1.0 with consideration of environmental effects during the period of extended operation. The Fatigue Monitoring Program requires that A~~a~~t least two years prior to entering the period of extended operation, the site will implement one or more of the following:

(1) Consistent with the Fatigue Monitoring Program, Detection of Aging Effects, update the fatigue usage calculations using ~~R~~refined ~~the~~fatigue analyses to determine valid CUFs less than 1.0 when accounting for the effects of reactor water environment. This includes applying the appropriate F_{en} factors to valid CUFs determined in accordance with one of the following.

For locations~~, including NUREG/CR-6260 locations,~~ with existing fatigue analysis valid for the period of extended operation, use the existing CUF~~ to determine the environmentally adjusted CUF~~.

~~In addition to the NUREG/CR-6260 locations, more limiting~~Additional plant-specific locations with a valid CUF may be evaluated. In particular, the pressurizer lower shell will be reviewed to ensure the surge nozzle remains the limiting component.

Representative CUF values from other plants, adjusted to or enveloping the plant-specific external loads may be used if demonstrated applicable.

An analysis using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case) may be performed to determine a valid CUF.

~~(2) Manage the effects of aging due to fatigue at the affected locations 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 acceptable to the NRC).~~

(~~3~~2) Consistent with the Fatigue Monitoring Program, Corrective Actions, ~~R~~repair or replace the affected locations before exceeding a CUF of 1.0.

~~Should IPEC select the option to manage the aging effects due to environmental-assisted fatigue during the period of extended operation, details of the aging management program such as scope, qualification, method, and frequency will be submitted to the NRC at least 2 years prior to the period of extended operation.~~

LRA Section A.3.1.11, Fatigue Monitoring Program, is revised as follows.

The Fatigue Monitoring Program is an existing program that tracks the number of critical thermal and pressure transients for selected reactor coolant system components. The program ensures the validity of analyses that explicitly analyzed a specified number of fatigue transients by assuring that the actual effective number of transients does not exceed the analyzed number of transients. The program provides for update of the fatigue usage calculations to maintain a CUF of < 1.0 for the period of extended operation. For the locations identified in Section A.3.2.2.3, updated calculations will account for the effects of the reactor water environment. These calculation updates are governed by Entergy's 10 CFR 50 Appendix B Quality Assurance (QA) program and include design input verification and independent reviews ensuring that valid assumptions, transients, cycles, external loadings, analysis methods, and environmental fatigue life correction factors will be used in the fatigue analyses. The program requires corrective actions including repair or replacement of affected components before fatigue usage calculations determine the CUF exceeds 1.0. Specific corrective actions are implemented in accordance with the IPEC corrective action program. Repair or replacement of the affected component(s), if necessary, will be in accordance with established plant procedures governing repair and replacement activities. These established procedures are governed by Entergy's 10 CFR 50 Appendix B QA program and meet the applicable repair or replacement requirements of the ASME Code Section XI.

LRA Section A.3.2.2.3, Environmental Effects on Fatigue, is revised as follows.

The effects of reactor water environment on fatigue were evaluated for license renewal. Projected cumulative usage factors (CUFs) were calculated for the limiting locations ~~identified in~~ based on NUREG/CR-6260. The identified IP3 locations are those listed in the license renewal application, Table 4.3-14. ~~For the locations with CUFs less than 1.0, the TLAA has been projected through the period of extended operation per 10 CFR 54.21(c)(1)(ii).~~ Several locations may exceed a CUF of 1.0 with consideration of environmental effects during the period of extended operation. The Fatigue Monitoring Program requires that ~~A~~at least two years prior to entering the period of extended operation, ~~for the locations identified in NUREG/CR-6260 for Westinghouse PWRs of this vintage,~~ the site will implement one or more of the following:

(1) Consistent with the Fatigue Monitoring Program, Detection of Aging Effects, update the fatigue usage calculation using ~~R~~refined ~~the~~ fatigue analyses to determine valid CUFs less than 1.0 when accounting for the effects of reactor water environment. This includes applying the appropriate F_{en} factors to valid CUFs determined in accordance with one of the following.

For locations~~, including NUREG/CR-6260 locations,~~ with existing fatigue analysis valid for the period of extended operation, use the existing CUF ~~to determine the environmentally adjusted CUF~~.

~~In addition to the NUREG/CR-6260 locations,more limiting~~Additional plant-specific locations with a valid CUF may be evaluated. In particular, the pressurizer lower shell will be reviewed to ensure the surge nozzle remains the limiting component.

Representative CUF values from other plants, adjusted to or enveloping the plant-specific external loads may be used if demonstrated applicable.

An analysis using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case) may be performed to determine a valid CUF.

~~(2) Manage the effects of aging due to fatigue at the affected locations 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 acceptable to the NRC).~~

(~~3~~2) Consistent with the Fatigue Monitoring Program, Corrective Actions; ~~R~~repair or replace the affected locations before exceeding a CUF of 1.0.

~~Should IPEC select the option to manage the aging effects due to environmental-assisted fatigue during the period of extended operation, details of the aging management program such as scope, qualification, method, and frequency will be submitted to the NRC at least 2 years prior to the period of extended operation.~~

LRA Section B.1.12, Fatigue Monitoring, Program Description, is revised as follows.

The Fatigue Monitoring Program is an existing program that tracks the number of critical thermal and pressure transients for selected reactor coolant system components. The program ensures the validity of analyses that explicitly analyzed a specified number of fatigue transients by assuring that the actual effective number of transients does not exceed the analyzed number of transients.

The program provides for update of the fatigue usage calculations to maintain a CUF of < 1.0 for the period of extended operation. For the locations identified in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3), updated calculations will account for the effects of the reactor water environment. These calculation updates are governed by Entergy's 10 CFR 50 Appendix B Quality Assurance (QA) program and include design input verification and

independent reviews ensuring that valid assumptions, transients, cycles, external loadings, analysis methods, and environmental fatigue life correction factors will be used in the fatigue analyses.

The analysis methods for determination of stresses and fatigue usage will be in accordance with an NRC endorsed Edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III Rules for Construction of Nuclear Power Plant Components Division 1 Subsection NB, Class 1 Components, Sub articles NB-3200 or NB-3600 as applicable to the specific component. IPEC will utilize design transients from IPEC Design Specifications to bound all operational transients. The numbers of cycles used for evaluation will be based on the design number of cycles and actual IPEC cycle counts projected out to the end of the license renewal period (60 years).

Environmental effects on fatigue usage will be assessed using methodology consistent with the Generic Aging Lessons Learned Report, NUREG-1801, Rev. 1, (GALL) that states; "The sample of critical components can be evaluated by applying environmental life correction factors to the existing ASME Code fatigue analyses. Formulae for calculating the environmental life correction factors are contained in NUREG/CR-6583 for carbon and low-alloy steels and in NUREG/CR-5704 for austenitic stainless steels."

The Fatigue Monitoring Program tracks actual plant transients and evaluates these against the design transients. Cycle counts show no limits are expected to be approached for the current license term. The Fatigue Monitoring Program will ensure that the numbers of transient cycles experienced by the plant remain within the analyzed numbers of cycles and hence, the component CUFs remain below the values calculated in the design basis fatigue evaluations. If ongoing monitoring indicates the potential for a condition outside that analyzed above, IPEC may perform further reanalysis of the identified configuration using established configuration management processes as described above.

The program requires corrective actions including repair or replacement of affected components before fatigue usage calculations determine the CUF exceeds 1.0. Specific corrective actions are implemented in accordance with the IPEC corrective action program. Repair or replacement of the affected component(s), if necessary, will be in accordance with established plant procedures governing repair and replacement activities. These established procedures are governed by Entergy's 10 CFR 50 Appendix B QA program and meet the applicable repair or replacement requirements

ATTACHMENT 2 TO NL-08-021

Regulatory Commitment List, Revision 3

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 & 3
DOCKET NOS. 50-247 AND 50-286

List of Regulatory Commitments

Rev. 3

The following table identifies those actions committed to by Entergy in this document.

Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
1	Enhance the Aboveground Steel Tanks Program for IP2 and IP3 to perform thickness measurements of the bottom surfaces of the condensate storage tanks, city water tank, and fire water tanks once during the first ten years of the period of extended operation. Enhance the Aboveground Steel Tanks Program for IP2 and IP3 to require trending of thickness measurements when material loss is detected.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.1 A.3.1.1 B.1.1
2	Enhance the Bolting Integrity Program for IP2 and IP3 to clarify that actual yield strength is used in selecting materials for low susceptibility to SCC and clarify the prohibition on use of lubricants containing MoS_2 for bolting. The Bolting Integrity Program manages loss of preload and loss of material for all external bolting.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.2 A.3.1.2 B.1.2 Audit Items 201, 241, 270
3	Implement the Buried Piping and Tanks Inspection Program for IP2 and IP3 as described in LRA Section B.1.6. This new program will be implemented consistent with the corresponding program described in NUREG-1801 Section XI.M34, Buried Piping and Tanks Inspection.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.5 A.3.1.5 B.1.6 Audit Item 173

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
4	Enhance the Diesel Fuel Monitoring Program to include cleaning and inspection of the IP2 GT-1 gas turbine fuel oil storage tanks, IP2 and IP3 EDG fuel oil day tanks, IP2 SBO/Appendix R diesel generator fuel oil day tank, and IP3 Appendix R fuel oil storage tank and day tank once every ten years. Enhance the Diesel Fuel Monitoring Program to include quarterly sampling and analysis of the IP2 SBO/Appendix R diesel generator fuel oil day tank, IP2 security diesel fuel oil day tank, and IP3 Appendix R fuel oil storage tank. Particulates, water and sediment checks will be performed on the samples. Filterable solids acceptance criterion will be less than or equal to 10mg/l. Water and sediment acceptance criterion will be less than or equal to 0.05%. Enhance the Diesel Fuel Monitoring Program to include thickness measurement of the bottom surface of the following tanks once every ten years. IP2: EDG fuel oil storage tanks, EDG fuel oil day tanks, SBO/Appendix R diesel generator fuel oil day tank, GT-1 gas turbine fuel oil storage tanks, and diesel fire pump fuel oil storage tank; IP3: EDG fuel oil day tanks, Appendix R fuel oil storage tank, and diesel fire pump fuel oil storage tank. Enhance the Diesel Fuel Monitoring Program to change the analysis for water and particulates to a quarterly frequency for the following tanks. IP2: GT-1 gas turbine fuel oil storage tanks and diesel fire pump fuel oil storage tank; IP3: Appendix R fuel oil day tank and diesel fire pump fuel oil storage tank. Enhance the Diesel Fuel Monitoring Program to specify acceptance criteria for thickness measurements of the fuel oil storage tanks within the scope of the program. Enhance the Diesel Fuel Monitoring Program to direct samples be taken near the tank bottom and include direction to remove water when detected. Enhance the Diesel Fuel Monitoring Program to direct the addition of chemicals including biocide when the presence of biological activity is confirmed.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.8 A.3.1.8 B.1.9 Audit items 128, 129, 132

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
5	Enhance the External Surfaces Monitoring Program for IP2 and IP3 to include periodic inspections of systems in scope and subject to aging management review for license renewal in accordance with 10 CFR 54.4(a)(1) and (a)(3). Inspections shall include areas surrounding the subject systems to identify hazards to those systems. Inspections of nearby systems that could impact the subject systems will include SSCs that are in scope and subject to aging management review for license renewal in accordance with 10 CFR 54.4(a)(2).	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.10 A.3.1.10 B.1.11
6	Enhance the Fatigue Monitoring Program for IP2 to monitor steady state cycles and feedwater cycles or perform an evaluation to determine monitoring is not required. Review the number of allowed events and resolve discrepancies between reference documents and monitoring procedures. Enhance the Fatigue Monitoring Program for IP3 to include all the transients identified. Assure all fatigue analysis transients are included with the lowest limiting numbers. Update the number of design transients accumulated to date.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.11 A.3.1.11 B.1.12, Audit Item 164

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
7	Enhance the Fire Protection Program to inspect external surfaces of the IP3 RCP oil collection systems for loss of material each refueling cycle. Enhance the Fire Protection Program to explicitly state that the IP2 and IP3 diesel fire pump engine sub-systems (including the fuel supply line) shall be observed while the pump is running. Acceptance criteria will be revised to verify that the diesel engine does not exhibit signs of degradation while running; such as fuel oil, lube oil, coolant, or exhaust gas leakage. Enhance the Fire Protection Program to specify that the IP2 and IP3 diesel fire pump engine carbon steel exhaust components are inspected for evidence of corrosion and cracking at least once each operating cycle. Enhance the Fire Protection Program for IP3 to visually inspect the cable spreading room, 480V switchgear room, and EDG room CO_2 fire suppression system for signs of degradation, such as corrosion and mechanical damage at least once every six months.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.12 A.3.1.12 B.1.13

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
8	Enhance the Fire Water Program to include inspection of IP2 and IP3 hose reels for evidence of corrosion. Acceptance criteria will be revised to verify no unacceptable signs of degradation. Enhance the Fire Water Program to replace all or test a sample of IP2 and IP3 sprinkler heads required for 10 CFR 50.48 using guidance of NFPA 25 (2002 edition), Section 5.3.1.1.1 before the end of the 50-year sprinkler head service life and at 10-year intervals thereafter during the extended period of operation to ensure that signs of degradation, such as corrosion, are detected in a timely manner. Enhance the Fire Water Program to perform wall thickness evaluations of IP2 and IP3 fire protection piping on system components using non-intrusive techniques (e.g., volumetric testing) to identify evidence of loss of material due to corrosion. These inspections will be performed before the end of the current operating term and at intervals thereafter during the period of extended operation. Results of the initial evaluations will be used to determine the appropriate inspection interval to ensure aging effects are identified prior to loss of intended function. Enhance the Fire Water Program to inspect the internal surface of foam based fire suppression tanks. Acceptance criteria will be enhanced to verify no significant corrosion.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153 NL-08-014	A.2.1.13 A.3.1.13 B.1.14 Audit Items 105, 106

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
9	Enhance the Flux Thimble Tube Inspection Program for IP2 and IP3 to implement comparisons to wear rates identified in WCAP-12866. Include provisions to compare data to the previous performances and perform evaluations regarding change to test frequency and scope. Enhance the Flux Thimble Tube Inspection Program for IP2 and IP3 to specify the acceptance criteria as outlined in WCAP-12866 or other plant-specific values based on evaluation of previous test results. Enhance the Flux Thimble Tube Inspection Program for IP2 and IP3 to direct evaluation and performance of corrective actions based on tubes that exceed or are projected to exceed the acceptance criteria. Also stipulate that flux thimble tubes that cannot be inspected over the tube length and cannot be shown by analysis to be satisfactory for continued service, must be removed from service to ensure the integrity of the reactor coolant system pressure boundary.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.15 A.3.1.15 B.1.16

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
10	Enhance the Heat Exchanger Monitoring Program for IP2 and IP3 to include the following heat exchangers in the scope of the program. • Safety injection pump lube oil heat exchangers • RHR heat exchangers • RHR pump seal coolers • Non-regenerative heat exchangers • Charging pump seal water heat exchangers • Charging pump fluid drive coolers • Charging pump crankcase oil coolers • Spent fuel pit heat exchangers • Secondary system steam generator sample coolers • Waste gas compressor heat exchangers • SBO/Appendix R diesel jacket water heat exchanger (IP2 only) Enhance the Heat Exchanger Monitoring Program for IP2 and IP3 to perform visual inspection on heat exchangers where non-destructive examination, such as eddy current inspection, is not possible due to heat exchanger design limitations. Enhance the Heat Exchanger Monitoring Program for IP2 and IP3 to include consideration of material-environment combinations when determining sample population of heat exchangers. Enhance the Heat Exchanger Monitoring Program for IP2 and IP3 to establish minimum tube wall thickness for the new heat exchangers identified in the scope of the program. Establish acceptance criteria for heat exchangers visually inspected to include no unacceptable signs of degradation.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.16 A.3.1.16 B.1.17, Audit Item 52

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
11	Enhance the ISI Program for IP2 and IP3 to provide periodic visual inspections to confirm the absence of aging effects for lubrite sliding supports used in the steam generator and reactor coolant pump support systems.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.17 A.3.1.17 B.1.18 Audit item 59
12	Enhance the Masonry Wall Program for IP2 and IP3 to specify that the IP1 intake structure is included in the program.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.18 A.3.1.18 B.1.19
13	Enhance the Metal-Enclosed Bus Inspection Program to add IP2 480V bus associated with substation A to the scope of bus inspected. Enhance the Metal-Enclosed Bus Inspection Program for IP2 and IP3 to visually inspect the external surface of MEB enclosure assemblies for loss of material at least once every 10 years. The first inspection will occur prior to the period of extended operation and the acceptance criterion will be no significant loss of material. Enhance the Metal-Enclosed Bus Inspection Program for IP2 and IP3 to inspect bolted connections at least once every five years if performed visually or at least once every ten years using quantitative measurements such as thermography or contact resistance measurements. The first inspection will occur prior to the period of extended operation. The plant will process a change to applicable site procedure to remove the reference to "re-torquing" connections for phase bus maintenance and bolted connection maintenance.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.19 A.3.1.19 B.1.20 Audit Item 124 Audit Item 133
14	Implement the Non-EQ Bolted Cable Connections Program for IP2 and IP3 as described in LRA Section B.1.22.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.21 A.3.1.21 B.1.22

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
15	Implement the Non-EQ Inaccessible Medium-Voltage Cable Program for IP2 and IP3 as described in LRA Section B.1.23. This new program will be implemented consistent with the corresponding program described in NUREG-1801 Section XI.E3, Inaccessible Medium-Voltage Cables Not Subject To 10 CFR 50.49 Environmental Qualification Requirements.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.22 A.3.1.22 B.1.23 Audit item 173
16	Implement the Non-EQ Instrumentation Circuits Test Review Program for IP2 and IP3 as described in LRA Section B.1.24. This new program will be implemented consistent with the corresponding program described in NUREG-1801 Section XI.E2, Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.23 A.3.1.23 B.1.24 Audit item 173
17	Implement the Non-EQ Insulated Cables and Connections Program for IP2 and IP3 as described in LRA Section B.1.25. This new program will be implemented consistent with the corresponding program described in NUREG-1801 Section XI.E1, Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.24 A.3.1.24 B.1.25 Audit item 173

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
18	Enhance the Oil Analysis Program for IP2 to sample and analyze lubricating oil used in the SBO/Appendix R diesel generator consistent with oil analysis for other site diesel generators. Enhance the Oil Analysis Program for IP2 and IP3 to sample and analyze generator seal oil and turbine hydraulic control oil. Enhance the Oil Analysis Program for IP2 and IP3 to formalize preliminary oil screening for water and particulates and laboratory analyses including defined acceptance criteria for all components included in the scope of this program. The program will specify corrective actions in the event acceptance criteria are not met. Enhance the Oil Analysis Program for IP2 and IP3 to formalize trending of preliminary oil screening results as well as data provided from independent laboratories.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.25 A.3.1.25 B.1.26
19	Implement the One-Time Inspection Program for IP2 and IP3 as described in LRA Section B.1.27. This new program will be implemented consistent with the corresponding program described in NUREG-1801, Section XI.M32, One-Time Inspection.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.26 A.3.1.26 B.1.27 Audit item 173
20	Implement the One-Time Inspection – Small Bore Piping Program for IP2 and IP3 as described in LRA Section B.1.28. This new program will be implemented consistent with the corresponding program described in NUREG-1801, Section XI.M35, One-Time Inspection of ASME Code Class I Small-Bore Piping.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.27 A.3.1.27 B.1.28 Audit item 173
21	Enhance the Periodic Surveillance and Preventive Maintenance Program for IP2 and IP3 as necessary to assure that the effects of aging will be managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis through the period of extended operation.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.28 A.3.1.28 B.1.29

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
22	Enhance the Reactor Vessel Surveillance Program for IP2 and IP3 revising the specimen capsule withdrawal schedules to draw and test a standby capsule to cover the peak reactor vessel fluence expected through the end of the period of extended operation. Enhance the Reactor Vessel Surveillance Program for IP2 and IP3 to require that tested and untested specimens from all capsules pulled from the reactor vessel are maintained in storage.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.31 A.3.1.31 B.1.32
23	Implement the Selective Leaching Program for IP2 and IP3 as described in LRA Section B.1.33. This new program will be implemented consistent with the corresponding program described in NUREG-1801, Section XI.M33 Selective Leaching of Materials.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.32 A.3.1.32 B.1.33 Audit item 173
24	Enhance the Steam Generator Integrity Program for IP2 and IP3 to require that the results of the condition monitoring assessment are compared to the operational assessment performed for the prior operating cycle with differences evaluated.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.34 A.3.1.34 B.1.35
25	Enhance the Structures Monitoring Program to explicitly specify that the following structures are included in the program. • Appendix R diesel generator foundation (IP3) • Appendix R diesel generator fuel oil tank vault (IP3) • Appendix R diesel generator switchgear and enclosure (IP3) • city water storage tank foundation • condensate storage tanks foundation (IP3) • containment access facility and annex (IP3) • discharge canal (IP2/3) • emergency lighting poles and foundations (IP2/3) • fire pumphouse (IP2) • fire protection pumphouse (IP3) • fire water storage tank foundations (IP2/3) • gas turbine 1 fuel storage tank foundation • maintenance and outage building-elevated passageway (IP2) • new station security building (IP2)	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.35 A.3.1.35 B.1.36 Audit item 86 Audit item 88 Audit Item 87

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
	• nuclear service building (IP1) • primary water storage tank foundation (IP3) • refueling water storage tank foundation (IP3) • security access and office building (IP3) • service water pipe chase (IP2/3) • service water valve pit (IP3) • superheater stack • transformer/switchyard support structures (IP2) • waste holdup tank pits (IP2/3) Enhance the Structures Monitoring Program for IP2 and IP3 to clarify that in addition to structural steel and concrete, the following commodities (including their anchorages) are inspected for each structure as applicable. • cable trays and supports • concrete portion of reactor vessel supports • conduits and supports • cranes, rails and girders • equipment pads and foundations • fire proofing (pyrocrete) • HVAC duct supports • jib cranes • manholes and duct banks • manways, hatches and hatch covers • monorails • new fuel storage racks • sumps, sump screens, strainers and flow barriers Enhance the Structures Monitoring Program for IP2 and IP3 to inspect inaccessible concrete areas that are exposed by excavation for any reason. IP2 and IP3 will also inspect inaccessible concrete areas in environments where observed conditions in accessible areas exposed to the same environment indicate that significant concrete degradation is occurring. Enhance the Structures Monitoring Program for IP2 and IP3 to perform inspections of elastomers (seals, gaskets, seismic joint filler, and roof elastomers) to identify cracking and change in material properties and for inspection of aluminum vents and louvers to identify loss of material.			

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
	Enhance the Structures Monitoring Program for IP2 and IP3 to perform an engineering evaluation of groundwater samples to assess aggressiveness of groundwater to concrete on a periodic basis (at least once every five years). IPEC will obtain samples from at least 5 wells that are representative of the ground water surrounding below-grade site structures. Samples will be monitored for sulfates, pH and chlorides. Enhance the Structures Monitoring Program for IP2 and IP3 to perform inspection of normally submerged concrete portions of the intake structures at least once every 5 years.			
26	Implement the Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program for IP2 and IP3 as described in LRA Section B.1.37. This new program will be implemented consistent with the corresponding program described in NUREG-1801, Section XI.M12, Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.36 A.3.1.36 B.1.37 Audit item 173
27	Implement the Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program for IP2 and IP3 as described in LRA Section B.1.38. This new program will be implemented consistent with the corresponding program described in NUREG-1801 Section XI.M13, Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steel (CASS) Program.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.37 A.3.1.37 B.1.38 Audit item 173
28	Enhance the Water Chemistry Control – Closed Cooling Water Program to maintain water chemistry of the IP2 SBO/Appendix R diesel generator cooling system per EPRI guidelines. Enhance the Water Chemistry Control – Closed Cooling Water Program to maintain the IP2 and IP3 security generator cooling water system pH within limits specified by EPRI guidelines.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.39 A.3.1.39 B.1.40

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
29	Enhance the Water Chemistry Control – Primary and Secondary Program for IP2 to test sulfates monthly in the RWST with a limit of <150 ppb.	IP2: September 28, 2013	NL-07-039	A.2.1.40 B.1.41
30	For aging management of the reactor vessel internals, IPEC will (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	IP2: September 28, 2011 IP3: December 12, 2013	NL-07-039	A.2.1.41 A.3.1.41
31	Additional P-T curves will be submitted as required per 10 CFR 50, Appendix G prior to the period of extended operation as part of the Reactor Vessel Surveillance Program.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.2.1.2 A.3.2.1.2 4.2.3
32	As required by 10 CFR 50.61(b)(4), IP3 will submit a plant-specific safety analysis for plate B2803-3 to the NRC three years prior to reaching the RT_{PTS} screening criterion. Alternatively, the site may choose to implement the revised PTS (10 CFR 50.61) rule when approved, which would permit use of Regulatory Guide 1.99, Revision 3.	IP3: December 12, 2015	NL-07-039	A.3.2.1.4 4.2.5

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
33	At least 2 years prior to entering the period of extended operation, for the locations identified in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3), under the Fatigue Monitoring Program, IP2 and IP3 will implement one or more of the following: (1) Consistent with the Fatigue Monitoring Program, Detection of Aging Effects, update the fatigue usage calculations using ~~R~~refined ~~the~~ fatigue analyses to determine valid CUFs less than 1.0 when accounting for the effects of reactor water environment. This includes applying the appropriate Fen factors to valid CUFs determined in accordance with one of the following: 1. For locations in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3) ~~including NUREG/CR-6260 locations~~, with existing fatigue analysis valid for the period of extended operation, use the existing CUF. ~~to determine the environmentally adjusted CUF.~~ 2. ~~In addition to the NUREG/CR-6260 locations, more limiting~~ Additional plant-specific locations with a valid CUF may be evaluated. In particular, the pressurizer lower shell will be reviewed to ensure the surge nozzle remains the limiting component. 3. Representative CUF values from other plants, adjusted to or enveloping the IPEC plant specific external loads may be used if demonstrated applicable to IPEC. 4. An analysis using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case) may be performed to determine a valid CUF. ~~(2) Manage the effects of aging due to fatigue at the affected locations 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 acceptable to the NRC).~~ ~~(3)~~ (2) Consistent with the Fatigue Monitoring Program, Corrective Actions, ~~R~~repair or replace the affected locations before exceeding a CUF of 1.0. ~~Should IPEC select the option to manage the aging effects due to environmental assisted fatigue during the period of extended operation, details of the aging management program such as scope, qualification, method, and frequency will be submitted to the NRC at least 2 years prior to the period of extended operation.~~	IP2: September 28, 2011 IP3: December 12, 2013	NL-07-039 NL-07-153 NL-08-021	A.2.2.2.3 A.3.2.2.3 4.3.3 Audit item 146

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
34	IP2 SBO / Appendix R diesel generator will be installed and operational by April 30, 2008. This committed change to the facility meets the requirements of 10 CFR 50.59(c)(1) and, therefore, a license amendment pursuant to 10 CFR 50.90 is not required.	April 30, 2008	NL-07-078	2.1.1.3.5

NYS-26/26A & RIVERKEEPER TC-1/1A: ATTACHMENT 11

Entergy

Entergy Nuclear Northeast
Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
Buchanan, NY 10511-0249
Tel (914) 788-2055

Fred Dacimo
Vice President
License Renewal

March 24, 2008

Re: Indian Point Units 2 & 3
Docket Nos. 50-247 & 50-286
NL-08-057

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

SUBJECT: Entergy Nuclear Operations Inc.
Indian Point Nuclear Generating Unit Nos. 2 & 3
Docket Nos. 50-247 and 50-286
Amendment 3 to License Renewal Application (LRA)

REFERENCES:

1. Entergy Letter dated April 23, 2007, F. R. Dacimo to Document Control Desk, "License Renewal Application" (NL-07-039)
2. Entergy Letter dated April 23, 2007, F. R. Dacimo to Document Control Desk, "License Renewal Application Boundary Drawings (NL-07-040)
3. Entergy Letter dated April 23, 2007, F. R. Dacimo to Document Control Desk, "License Renewal Application Environmental Report References (NL-07-041)
4. Entergy Letter dated October 11, 2007, F. R, Dacimo to Document Control Desk, "License Renewal Application (LRA)" (NL-07-124)
5. Entergy Letter November 14, 2007, F. R, Dacimo to Document Control Desk, "Supplement to License Renewal Application (LRA) Environmental Report References" (NL-07-133)

Dear Sir or Madam:

In the referenced letters, Entergy Nuclear Operations, Inc. applied for renewal of the Indian Point Energy Center operating license.

This letter contains Amendment 3 of the License Renewal Application (LRA), which consists of five attachments. Attachment 1 consists of an amendment to the LRA to address Regional Inspection items. Attachment 2 consists of an amendment to address Audit Time Limited Aging

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NRR

Analyses (TLAA) and other LRA amendment items. Attachment 3 consists of a revision to the list of regulatory commitments associated with the LRA. Attachment 4 provides the responses to the questions raised by the NRC team during the TLAA portion of the LRA. Attachment 5 provides the responses to the questions raised by the NRC team during the Aging Management Programs (AMP) portion of the LRA.

If you have any questions, or require additional information, please contact Mr. Robert Walpole at 914-734-6710.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 3/24/08.

Sincerely,

Fred R. Dacimo
Vice President
License Renewal

Attachments:

1. Regional Inspection LRA Amendment
2. Audit TLAA and other LRA Amendment
3. IPEC LRA List of Regulatory Commitments, Revision 4
4. TLAA Audit Database Report
5. AMP Audit Database Report

cc: Mr. Samuel J. Collins, Regional Administrator, NRC Region I
Mr. Sherwin E. Turk, NRC Office of General Counsel, Special Counsel
Mr. Kenneth Chang, NRC Branch Chief, Engineering Review Branch I
Mr. Bo M. Pham, NRC Environmental Project Manager
Mr. John Boska, NRR Senior Project Manager
Mr. Paul Eddy, New York State Department of Public Service
NRC Resident Inspector's Office
Mr. Paul D. Tonko, President, New York State Energy, Research, & Development Authority

ATTACHMENT 3 TO NL-08-057

IPEC LRA List of Regulatory Commitments, Revision 4

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 & 3
DOCKET NOS. 50-247 AND 50-286

33	At least 2 years prior to entering the period of extended operation, for the locations identified in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3), under the Fatigue Monitoring Program, IP2 and IP3 will implement one or more of the following: (1) Consistent with the Fatigue Monitoring Program, Detection of Aging Effects, update the fatigue usage calculations using refined fatigue analyses to determine valid CUFs less than 1.0 when accounting for the effects of reactor water environment. This includes applying the appropriate Fen factors to valid CUFs determined in accordance with one of the following: 1. For locations in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3), with existing fatigue analysis valid for the period of extended operation, use the existing CUF. 2. Additional plant-specific locations with a valid CUF may be evaluated. In particular, the pressurizer lower shell will be reviewed to ensure the surge nozzle remains the limiting component. 3. Representative CUF values from other plants, adjusted to or enveloping the IPEC plant specific external loads may be used if demonstrated applicable to IPEC. 4. An analysis using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case) may be performed to determine a valid CUF. (2) Consistent with the Fatigue Monitoring Program, Corrective Actions, repair or replace the affected locations before exceeding a CUF of 1.0.	IP2: September 28, 2011 IP3: December 12, 2013	NL-07-039 NL-07-153 NL-08-021	A.2.2.2.3 A.3.2.2.3 4.3.3 Audit item 146
34	IP2 SBO / Appendix R diesel generator will be installed and operational by April 30, 2008. This committed change to the facility meets the requirements of 10 CFR 50.59(c)(1) and, therefore, a license amendment pursuant to 10 CFR 50.90 is not required.	April 30, 2008	NL-07-078	2.1.1.3.5

NYS-26/26A & RIVERKEEPER TC-1/1A: ATTACHMENT 12

Entergy

Entergy Nuclear Northeast
Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
Buchanan, NY 10511-0249
Tel (914) 788-2055

Fred R. Dacimo
Vice President
License Renewal

May 16, 2008

Re: Indian Point Units 2 & 3
Docket Nos. 50-247 & 50-286

NL-08-084

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

SUBJECT: **Reply to Request for Additional Information Regarding License Renewal Application – Time-Limited Aging Analyses and Boraflex**

Reference: NRC letter dated April 18, 2008; "Request for Additional Information for the Review of the Indian Point Nuclear Generating Unit Nos. 2 and 3, License Renewal Application – Time-Limited Aging Analyses and Boraflex"

Dear Sir or Madam:

Entergy Nuclear Operations, Inc is providing, in Attachment I, the additional information requested in the referenced letter pertaining to NRC review of the License Renewal Application for Indian Point 2 and Indian Point 3. The additional information provided in this transmittal addresses staff questions for Time-Limited Aging Analyses and Boraflex.

There are no new commitments identified in this submittal. If you have any questions or require additional information, please contact Mr. R. Walpole, Manager, Licensing at (914) 734-6710.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 5-16-08.

Sincerely,

Peter W. Conroy for
Fred R. Dacimo
Vice President
License Renewal

A128
NRR

Attachment:

1. Reply to NRC Request for Additional Information Regarding License Renewal Application – Time-Limited Aging Analyses and Boraflex

cc: Mr. Bo M. Pham, NRC Environmental Project Manager
Ms. Kimberly Green, NRC Safety Project Manager
Mr. John P. Boska, NRC NRR Senior Project Manager
Mr. Samuel J. Collins, Regional Administrator, NRC Region I
Mr. Sherwin E. Turk, NRC Office of General Counsel, Special Counsel
IPEC NRC Senior Resident Inspectors Office
Mr. Paul D. Tonko, President, NYSERDA
Mr. Paul Eddy, New York State Dept. of Public Service

ATTACHMENT I TO NL-08-084

REPLY TO NRC REQUEST FOR ADDITIONAL INFORMATION

REGARDING

LICENSE RENEWAL APPLICATION

Time-Limited Aging Analyses and Boraflex

ENTERGY NUCLEAR OPERATIONS, INC
INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 and 3
DOCKETS 50-247 and 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3
LICENSE RENEWAL APPLICATION (LRA)
REQUESTS FOR ADDITIONAL INFORMATION (RAI)
REGARDING TIME-LIMITED AGING ANALYSES AND BORAFLEX

Time-Limited Aging Analyses

RAI 4.3.1.8-1

License renewal application (LRA) Section 4.3.1 states "[c]urrent design basis fatigue evaluations calculate cumulative usage factors (CUFs) for components or sub-components based on design transient cycles." For CUF values listed in LRA Tables 4.3-13 and 4.3-14, please describe the details of how various environmental effects are factored into the calculation of the CUF using F_{en} values.

Response to RAI 4.3.1.8-1

For CUF values listed in LRA Tables 4.3-13 and 4.3-14, the F_{en} values were determined as described below.

NUREG-1801 calls for using formulas provided in NUREG/CR-5704 for austenitic stainless steel and NUREG/CR-6583 for carbon steel and low-alloy steel to calculate environmentally assisted fatigue correction factors (F_{en}). For IPEC, none of the locations identified in Tables 4.3-13 and 4.3-14 (NUREG/CR-6260 locations) are made of carbon steel, so calculation of F_{en} for carbon steel was not required.

The environmentally assisted fatigue correction factor (F_{en}) for **low alloy steel** was calculated as follows.

F_{en} =	$\exp(0.929-0.00124T-0.101\, S^*T^*O^*\dot{\varepsilon}^*)$		based on NUREG/CR-6583, Eq. 6.5b
T =	25°C	Reference temperature for original fatigue curves	
S^* =	S	($0 < S$ (Sulfur) $\leq$ 0.015 wt.%)	
S^* =	0.015	($S \geq$ 0.015 wt.%)	NUREG/CR-6583, Eq. 5.5a
T^* =	0	(T (Temperature) < 150°C)	
T^* =	T-150	(T = 150-350°C)	NUREG/CR-6583, Eq. 5.5b
O^* =	0	(DO (Dissolved Oxygen) < 0.05ppm)	
O^* =	$\ln(DO/0.04)$	(0.05 ppm $\leq$ DO $\leq$ 0.5 ppm)	
O^* =	$\ln(12.5) = 2.53$	(DO > 0.5 ppm)	NUREG/CR-6583, Eq. 5.5c
$\dot{\varepsilon}^*$ =	0	($\dot{\varepsilon}$ (strain rate) > 1%/s)	
$\dot{\varepsilon}^*$ =	$\ln(\dot{\varepsilon})$	($0.001 \leq \dot{\varepsilon} \leq$ 1%/s)	
$\dot{\varepsilon}^*$ =	$\ln(0.001)$	($\dot{\varepsilon}$ < 0.001%/s)	NUREG/CR-6583, Eq. 5.5d

There are four low alloy steel subcomponents for each unit in Tables 4.3-13 and 4.3-14 for the NUREG-6260 locations at IPEC. The F_{en} was calculated for each location on each unit as shown below.

IP2

$T_{(reference\ temperature\ °C)}$ = 25 Reference temperature for original fatigue curves

$O^{*}_{(RCS)}$ = 0.0 RCS dissolved oxygen is ≤ 50 ppb

Since $O^{*}_{(RCS)}$ equals 0.0, S*, T*, and ε^{*} terms are eliminated.

F_{en} = exp(0.929-(0.00124)(T))

$F_{en\ (bottom\ head\ to\ shell)}$ = exp(0.929-(0.00124)(25)) = 2.45
$F_{en\ (inlet\ nozzles)}$ = exp(0.929-(0.00124)(25)) = 2.45
$F_{en\ (outlet\ nozzles)}$ = exp(0.929-(0.00124)(25)) = 2.45
$F_{en\ (surge\ line\ nozzles)}$ = exp(0.929-(0.00124)(25)) = 2.45

IP3

$T_{(reference\ temperature\ °C)}$ = 25 Reference temperature for original fatigue curves

Since $O^{*}_{(RCS)}$ equals 0.0, S*, T*, and ε^{*} terms are eliminated.

F_{en} = exp(0.929-(0.00124)(T))
$F_{en\ (bottom\ head\ to\ shell)}$ = exp(0.929-(0.00124)(25)) = 2.45
$F_{en\ (inlet\ nozzles)}$ = exp(0.929-(0.00124)(25)) = 2.45
$F_{en\ (outlet\ nozzles)}$ = exp(0.929-(0.00124)(25)) = 2.45
$F_{en\ (surge\ line\ nozzles)}$ = exp(0.929-(0.00124)(25)) = 2.45

The environmentally assisted fatigue correction factor (F_{en}) for **austenitic stainless steel** was calculated as follows.

F_{en} = exp(0.935-T′O′ε′)		NUREG/CR-5704, Eq. 13
T′ = 0	(T < 200ºC)	
T′ = 1	(T ≥ 200ºC)	NUREG/CR-5704, Eq. 8a
O′ = 0.260	(DO < 0.05 ppm)	
O′ = 0.172	(DO ≥ 0.05 ppm)	NUREG/CR-5704, Eq. 8b
$\dot{\varepsilon}'$ = 0	($\dot{\varepsilon}$ > 0.4%/s)	
$\dot{\varepsilon}'$ = ln($\dot{\varepsilon}$ /0.4)	(0.0004 ≤ $\dot{\varepsilon}$ ≤ 0.4%/s)	
$\dot{\varepsilon}'$ = ln(0.0004/0.4)	($\dot{\varepsilon}$ < 0.0004%/s)	NUREG/CR-5704, Eq. 8c

There are four stainless steel subcomponents for each unit in Tables 4.3-13 and 4.3-14 for the NUREG-6260 locations at IPEC. The F_{en} will be calculated for each location on each unit as shown below.

IP2

$T'_{(Surge\ line)}$	= 1.0	(T ≥ 200ºC)
$T'_{(Charging\ nozzle)}$	= 1.0	(T ≥ 200ºC)
$T'_{(SI\ nozzle)}$	= 1.0	(T ≥ 200ºC)
$T'_{(RHR\ piping)}$	= 1.0	(T ≥ 200ºC)

$O'_{(All)}$	= 0.260	RCS dissolved oxygen is ≤ 50 ppb
$\dot{\varepsilon}'_{(all)}$	= -6.91	Assume bounding strain rate
$F_{en\ (Surge\ line)}$	= exp(0.935-(1.0)(0.260)(-6.91)) = 15.35	
$F_{en\ (Charging\ nozzle)}$	= exp(0.935-(1.0)(0.260)(-6.91)) = 15.35	
$F_{en\ (SI\ nozzle)}$	= exp(0.935-(1.0)(0.260)(-6.91)) = 15.35	
$F_{en\ (RHR\ piping)}$	= exp(0.935-(1.0)(0.260)(-6.91)) = 15.35	

IP3

$T'_{(Surge\ line)}$	= 1.0	(T ≥ 200ºC)
$T'_{(Charging\ nozzle)}$	= 1.0	(T ≥ 200ºC)
$T'_{(SI\ nozzle)}$	= 1.0	(T ≥ 200ºC)
$T'_{(RHR\ piping)}$	= 1.0	(T ≥ 200ºC)
$O'_{(All)}$	= 0.260	RCS dissolved oxygen is ≤ 50 ppb
$\dot{\varepsilon}'_{(all)}$	= -6.91	Assume bounding strain rate
$F_{en\ (Surge\ line)}$	= exp(0.935-(1.0)(0.260)(-6.91)) = 15.35	
$F_{en\ (Charging\ nozzle)}$	= exp(0.935-(1.0)(0.260)(-6.91)) = 15.35	
$F_{en\ (SI\ nozzle)}$	= exp(0.935-(1.0)(0.260)(-6.91)) = 15.35	
$F_{en\ (RHR\ piping)}$	= exp(0.935-(1.0)(0.260)(-6.91)) = 15.35	

RAI 4.3.1.8-2

Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants (SRP-LR) Section 4.3.2.1.1.3 provides the basis for the staff acceptance of an aging management program to address environmental fatigue. It states, "[t]he staff has evaluated a program for monitoring and tracking the number of critical thermal and pressure transients for the selected reactor coolant system components. The staff has determined that this program is an acceptable aging management program to address metal fatigue of the reactor coolant system components according to 10 CFR 54.21(c)(1)(iii)." The staff is unable to determine if the Fatigue Monitoring Program of Indian Point 2 and Indian Point 3 contain sufficient details to satisfy this criterion. Please provide adequate details of the Fatigue Monitoring Program such that the staff can make a determination based on the criterion set forth in SRP-LR Section 4.3.2.1.1.3. Also, please explain in detail the corrective actions and the frequency that such actions will be taken so the acceptance criteria will not be exceeded in the period of extended operation.

Response to RAI 4.3.1.8-2

The IPEC Fatigue Monitoring Program was compared to the program described in NUREG-1801 (GALL), Section X.M1, Metal Fatigue of Reactor Coolant Pressure Boundary. The program description in the GALL report is directly applicable to the IPEC units. As indicated in LRA Section B.1.12, during the period of extended operation the IPEC program will be consistent with the GALL program with one exception. The exception to GALL is that rather than performing periodic updates of CUF calculations, IPEC periodically assesses the number of transient cycles compared to calculation assumptions and updates the CUF calculations, if

necessary. Based on this comparison including evaluation of the exception, the approvals set forth in the GALL report apply to the IPEC Fatigue Monitoring Program.

The program description was modified per letter NL-08-21, Indian Point Nuclear Generating Units Nos. 2 & 3 License Renewal Application Amendment 2, dated January 22, 2008. This letter commits to complete CUF calculations for all areas identified in NUREG-6260 (LRA Table 4.3-13 for IP2 and LRA Table 4.3-14 for IP3), incorporating the effect of the reactor coolant environment, for IP2 and IP3. Once these CUF calculations are complete (at least 2 years prior to the period of extended operation), IPEC will ensure that the cycles analyzed in the new or updated calculations are included in the Fatigue Monitoring Program. IPEC will continue to manage the effects of fatigue throughout the period of extended operation by monitoring cycles incurred and assuring they do not exceed the analyzed numbers of cycles.

As required by IPEC technical specifications, the Fatigue Monitoring Program tracks actual plant transients and evaluates these against the design transients. The plant transient counts are updated at least once each operating cycle. This frequency is acceptable since the evaluation during each update determines if the number of design transients could be exceeded prior to the next update. The Fatigue Monitoring Program ensures that the numbers of transient cycles experienced by the plant remain within the analyzed numbers of cycles and hence, the component CUF calculations remain valid.

The program requires corrective actions before exceeding the analyzed number of transient cycles. The corrective actions are implemented in accordance with the IPEC corrective action program. IPEC may perform further reanalysis if cycle counts approach analyzed numbers. These calculation updates will be governed by Entergy's 10 CFR 50 Appendix B Quality Assurance (QA) program and include design input verification and independent reviews ensuring that valid assumptions, transients, cycles, external loadings, analysis methods, and environmental fatigue life correction factors will be used in the fatigue analyses. Repair or replacement of the affected component(s), if necessary, will be done prior to exceeding the allowable CUF in accordance with established plant procedures governing repair and replacement activities. These established procedures are governed by Entergy's 10 CFR 50 Appendix B QA program and meet the applicable repair or replacement requirements of the ASME Code Section XI.

RAI 3.0.3.2.3-1

Indian Point 2 Updated Final Safety Analysis Report, Revision 20, dated 2006, Section 14.2.1 on page 55 of 218, states in part that:

> "Northeast Technology Corporation report NET-173-01 and NET-173-02 are based on conservative projections of amount of boraflex absorber panel degradation assumed in each sub-region. These projections are valid through the end of the year 2006."

Please confirm that the Boraflex neutron absorber panels in the Indian Point Unit 2 spent fuel pool have been re-evaluated for service through the end of the current licensing period. Also, please discuss the plans for updating the Boraflex analysis during the period of extended operation.

Response to RAI 3.0.3.2.3-1

Boron-10 areal density gage for evaluating racks (BADGER) testing was performed in February 2000, July 2003, and again in July 2006. Using the latest test data and RACKLIFE code projections, the Boraflex neutron absorber panels in the Indian Point Unit 2 spent fuel pool will meet the Technical Specification requirements through the end of the current licensing period. The next BADGER test will be performed prior to the end of calendar year 2009. As required by the Boraflex Monitoring Program (LRA Section B.1.3), periodic BADGER testing and RACKLIFE code projections will continue through the period of extended operation to confirm acceptable Boraflex condition.

The referenced section of the Indian Point 2 Updated Final Safety Analysis Report will be updated in the next revision.

NYS-26/26A & RIVERKEEPER TC-1/1A: ATTACHMENT 13

Entergy

Entergy Nuclear Northeast
Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
Buchanan, NY 10511-0249
Tel (914) 788-2055

Fred Dacimo
Vice President
License Renewal

June 11, 2008

Re: Indian Point Units 2 & 3
Docket Nos. 50-247 & 50-286
NL-08-092

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

SUBJECT: Entergy Nuclear Operations Inc.
Indian Point Nuclear Generating Unit Nos. 2 & 3
Docket Nos. 50-247 and 50-286
Amendment 5 to License Renewal Application (LRA)

REFERENCES:

1. Entergy Letter dated April 23, 2007, F. R. Dacimo to Document Control Desk, "License Renewal Application" (NL-07-039)
2. Entergy Letter dated April 23, 2007, F. R. Dacimo to Document Control Desk, "License Renewal Application Boundary Drawings (NL-07-040)
3. Entergy Letter dated April 23, 2007, F. R. Dacimo to Document Control Desk, "License Renewal Application Environmental Report References (NL-07-041)
4. Entergy Letter dated October 11, 2007, F. R, Dacimo to Document Control Desk, "License Renewal Application (LRA)" (NL-07-124)
5. Entergy Letter November 14, 2007, F. R, Dacimo to Document Control Desk, "Supplement to License Renewal Application (LRA) Environmental Report References" (NL-07-133)

Dear Sir or Madam:

In the referenced letters, Entergy Nuclear Operations, Inc. applied for renewal of the Indian Point Energy Center operating license.

This letter contains Amendment 5 of the License Renewal Application (LRA) which consists of three attachments. Attachment 1 consists of the annual update amendment to the LRA.

A128
NRR

Attachment 2 consists of an amendment for (a)(2) clarification. Attachment 3 consists of an amendment for reactor vessel clarification.

If you have any questions, or require additional information, please contact Mr. Robert Walpole at 914-734-6710.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 6/11/08.

Sincerely,

Fred R. Dacimo
Vice President
License Renewal

Attachments:

1. Annual Update Amendment
2. (A)(2) Clarification Amendment
3. Reactor Vessel Clarification Amendment

cc: Mr. Samuel J. Collins, Regional Administrator, NRC Region I
Mr. Sherwin E. Turk, NRC Office of General Counsel, Special Counsel
Mr. Kenneth Chang, NRC Branch Chief, Engineering Review Branch I
Mr. Bo M. Pham, NRC Environmental Project Manager
Mr. John Boska, NRR Senior Project Manager
Mr. Paul Eddy, New York State Department of Public Service
NRC Resident Inspector's Office
Mr. Paul D. Tonko, President, New York State Energy, Research, & Development Authority

ATTACHMENT 1 TO NL-08-092

Annual Update Amendment

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 & 3
DOCKET NOS. 50-247 AND 50-286

LRA Section A.2.1.17, Inservice Inspection – Inservice Inspection (ISI) Program, third paragraph, is revised as follows.

> On ~~July~~ March 1, 2007~~1994~~, ~~the plant~~ IP2 entered the ~~third~~ fourth ISI interval. The ASME code edition and addenda used for the ~~third~~ fourth interval is the ~~1989~~2001 Edition with ~~no~~2003 addenda.

LRA Section B.1.8, Containment Inservice Inspection, Program Description, is revised as follows.

> The Containment Inservice Inspection (CII) Program is an existing program encompassing ASME Section XI Subsection IWE and IWL requirements as modified by 10 CFR 50.55a. The IP2 program uses the ASME Boiler and Pressure Vessel Code, Section XI, ~~1992~~ 2001 Edition, ~~1992~~ 2003Addenda. The IP3 program uses the ASME Boiler and Pressure Vessel Code, Section XI, 1998 Edition, no Addenda. Every 10 years, each unit's program is updated to the latest ASME Section XI code edition and addenda approved by the Nuclear Regulatory Commission in 10 CFR 50.55a.

LRA Section B.1.18, Inservice Inspection, Program Description, seventh paragraph, is revised as follows.

> On ~~July~~ March 1, 2007~~1994~~, IP2 entered the ~~third~~ fourth ISI interval and on July 21, 2000, IP3 entered the third ISI interval. The ASME code edition and addenda used for the IP2 fourth interval is the 2001 Edition with 2003 addenda. The ASME code edition and addenda used for the IP3 third interval ~~for both units~~ is the 1989 Edition with no addenda.

LRA Section B.1.18, Inservice Inspection, Element 4, eighth paragraph, is revised as follows.

> For ~~both~~ IP2 ~~and IP3~~, Article IWF of ASME Section XI, ~~1989 Edition~~ 2001 Edition and 2003 Addenda, does not contain any specific exemption criteria for component supports. For IP3, ~~C~~components exempt from examination are in accordance with the criteria contained in Code Case N-491-2, Alternate Rules for Examination of Class 1, 2, 3 and MC Component Supports of Light-Water Cooled Power Plants, Section XI, Division 1, IWF-1230.

Additional LRA Clarification

LRA Section 4.3.3, Effects of Reactor Water Environment on Fatigue Life, is revised to delete the tenth paragraph as follows.

> ~~For those locations with CUFs less than 1.0, the TLAA has been projected through the period of extended operation per 10CFR54.21(c)(1)(ii).~~

LRA Section A.2.1.20, Nickel Alloy Inspection Program, last paragraph, is revised as follows. (Refer to RAI 3.0.3.3.5-2 response in letter NL-08-051 dated March 12, 2008)

~~The site will continue to implement commitments associated with (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff accepted industry guidelines.~~

The site commits to comply with future applicable NRC Orders. In addition, IPEC commits to implement applicable (1) Bulletins and Generic Letters associated with nickel alloys and (2) staff-accepted industry guidelines associated with nickel alloys.

LRA Section A.3.1.20, Nickel Alloy Inspection Program, last paragraph, is revised as follows. (Refer to RAI 3.0.3.3.5-2 response in letter NL-08-051 dated March 12, 2008)

~~The site will continue to implement commitments associated with (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff accepted industry guidelines.~~

The site commits to comply with future applicable NRC Orders. In addition, IPEC commits to implement applicable (1) Bulletins and Generic Letters associated with nickel alloys and (2) staff accepted industry guidelines associated with nickel alloys.

LRA Section B.1.21, Nickel Alloy Inspection, Program Description, last paragraph, is revised as follows. (Refer to RAI 3.0.3.3.5-2 response in letter NL-08-051 dated March 12, 2008)

~~IPEC will continue to implement commitments associated with (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff accepted industry guidelines.~~

IPEC commits to comply with future applicable NRC Orders. In addition, IPEC commits to implement applicable (1) Bulletins and Generic Letters associated with nickel alloys and (2) staff accepted industry guidelines associated with nickel alloys.

LRA Section B.1.12, Fatigue Monitoring, Exceptions to NUREG-1801, is revised as follows.

The Fatigue Monitoring Program is consistent with the program described in NUREG-1801, Section X.M1, Metal Fatigue of Reactor Coolant Pressure Boundary~~, with the following exception~~.

~~Attributes Affected~~	~~Exceptions~~
~~4. Detection of Aging Effects~~	~~NUREG-1801 specifies periodic updates of fatigue usage calculations. The IPEC program updates fatigue usage calculations when the number of actual cycles approach the analyzed number of cycles.~~ [1]

~~Exception Notes~~

~~[1] Updates of fatigue usage calculations are not necessary unless the number of accumulated fatigue cycles approaches the number of analyzed design cycles. The IPEC program provides for periodic assessment of the number of accumulated cycles. If any transient approaches its number of analyzed cycles, corrective action is taken which may include update of the fatigue usage calculation.~~

NYS-26/26A & RIVERKEEPER TC-1/1A: ATTACHMENT 14

Entergy

Entergy Nuclear Northeast
Indian Point Energy Center
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Tel (914) 788-2055

Fred Dacimo
Vice President
License Renewal

NL-10-082

August 9, 2010

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

SUBJECT: License Renewal Application – Completion of Commitment #33 Regarding the Fatigue Monitoring Program
Indian Point Nuclear Generating Unit Nos. 2 and 3
Docket Nos. 50-247 and 50-286
License Nos. DPR-26 and DPR-64

REFERENCE

1. Entergy Letter dated April 23, 2007, F. R. Dacimo to Document Control Desk, "License Renewal Application" (NL-07-039)
2. Entergy Letter dated April 23, 2007, F. R. Dacimo to Document Control Desk, "License Renewal Application Boundary Drawings (NL-07-040)
3. Entergy Letter dated April 23, 2007, F. R. Dacimo to Document Control Desk, "License Renewal Application Environmental Report References (NL-07-041)
4. Entergy Letter dated October 11, 2007, F. R, Dacimo to Document Control Desk, "License Renewal Application (LRA)" (NL-07-124)
5. Entergy Letter November 14, 2007, F. R, Dacimo to Document Control Desk, "Supplement to License Renewal Application (LRA) Environmental Report References" (NL-07-133)

Dear Sir or Madam:

In the referenced letters, Entergy Nuclear Operations, Inc. applied for renewal of the Indian Point Energy Center operating license. This letter contains information supporting the completion of commitment 33 to the License Renewal Application regarding the Fatigue Monitoring Program.

There are no new commitments identified in this submittal. If you have any questions, or require additional information, please contact Mr. Robert Walpole at 914-734-6710.

Sincerely,

FD/mb

Attachments:
1. Environmental Fatigue Evaluations
2. List of Regulatory Commitments

cc: Mr. S. J. Collins, Regional Administrator, NRC Region I
Mr. J. Boska, Senior Project Manager, NRC, NRR, DORL
Mr. Sherwin E. Turk, NRC Office of General Counsel, Special Counsel
Ms. Kimberly Green, NRC Safety Project Manager
NRC Resident Inspectors Office, Indian Point
Mr. Paul Eddy, NYS Dept. of Public Service
Mr. Francis J. Murray, Jr., President and CEO, NYSERDA

ATTACHMENT 1 TO NL-10-082

Environmental Fatigue Evaluations

ENTERGY NUCLEAR OPERATIONS, INC
INDIAN POINT NUCLEAR GENERATING UNITS 2 AND 3
DOCKET NOS. 50-247 & 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3
LICENSE RENEWAL APPLICATION
ENVIRONMENTAL FATIGUE EVALUATIONS

Environmental Fatigue Evaluation for Indian Point Unit 2 and Unit 3

Entergy has applied for renewed operating licenses for Indian Point Nuclear Generating Unit 2 and Unit 3 (IP2 and IP3). In the license renewal application, Entergy committed to address environmentally assisted fatigue. Entergy's commitment, amended by letter, NL-08-021, dated January 22, 2008, reads as follows.

> At least 2 years prior to entering the period of extended operation, for the locations identified in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3), under the Fatigue Monitoring Program, IP2 and IP3 will implement one or more of the following:
>
> (1) Consistent with the Fatigue Monitoring Program, Detection of Aging Effects, update the fatigue usage calculations using refined fatigue analyses to determine valid CUFs less than 1.0 when accounting for the effects of reactor water environment. This includes applying the appropriate Fen factors to valid CUFs determined in accordance with one of the following:
>
> 1. For locations in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3), with existing fatigue analysis valid for the period of extended operation, use the existing CUF.
>
> 2. Additional plant-specific locations with a valid CUF may be evaluated. In particular, the pressurizer lower shell will be reviewed to ensure the surge nozzle remains the limiting component.
>
> 3. Representative CUF values from other plants, adjusted to or enveloping the IPEC plant specific external loads may be used if demonstrated applicable to IPEC.
>
> 4. An analysis using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case) may be performed to determine a valid CUF.
>
> (2) Consistent with the Fatigue Monitoring Program, Corrective Actions, repair or replace the affected locations before exceeding a CUF of 1.0.

Entergy has updated the fatigue usage calculations using refined fatigue analyses to determine CUFs when accounting for the effects of reactor water environment. This includes applying the appropriate Fen factors to valid CUFs for the locations shown in LRA Table 4.3-13 and LRA Table 4.3-14.
The tables from the LRA are repeated as follows with the updated CUF values inserted.

The results of the refined analyses are shown in the tables as <u>underlined values.</u>

Table 4.3-13
IP2 Cumulative Usage Factors for NUREG/CR-6260 Limiting Locations

NUREG-6260 Generic Location		IP2 Plant-Specific Location	Material Type	CUF of Record	Per NUREG/CR-6583 or NUREG/CR-5704	
					F_{en}	Environmentally Adjusted CUF
1	Vessel shell and lower head	Bottom head to shell	LAS	0.004	2.45	0.01
2	Vessel inlet and outlet nozzles	Reactor vessel inlet nozzle	LAS	0.05	2.45	0.12
2	Vessel inlet and outlet nozzles	Reactor vessel outlet nozzle	LAS	0.281	2.45	0.69
3	Pressurizer surge line nozzles	Pressurizer surge line nozzle	LAS	~~0.264~~ 0.109	~~2.45~~ 1.74	~~0.646~~ 0.188
3	Pressurizer surge line piping	Surge line piping to safe end weld	SS	~~0.6~~ 0.062	~~15.35~~ 13.26	~~9.21~~ 0.822
4	RCS piping charging system nozzle	Charging system nozzle	SS	~~0.99~~ 0.0323	~~15.35~~ 8.7	~~15.20~~ 0.2809
5	RCS piping safety injection nozzle	NA	SS	~~NA~~[2] 0.1083	~~15.35~~ 7.8975	~~NA~~[2] 0.8553
6	RHR Class 1 piping	NA	SS	~~NA~~[2] 0.0721	~~15.35~~ 13.08	~~NA~~[2] 0.9434

Table 4.3-14
IP3 Cumulative Usage Factors for NUREG/CR-6260 Limiting Locations

	NUREG-6260 Location	IP3 Plant-Specific Location	Material Type	CUF of Record	Per NUREG/CR-6583 or NUREG/CR-5704	
					F_{en}	Environmentally Adjusted CUF
1	Vessel shell and lower head	Bottom head to shell	LAS	0.02	2.45	0.05
2	Vessel inlet and outlet nozzles	Reactor vessel inlet nozzle	LAS	0.049	2.45	0.12
2	Vessel inlet and outlet nozzles	Reactor vessel outlet nozzle	LAS	0.259	2.45	0.64
3	Pressurizer surge line nozzles	Pressurizer surge line nozzle	LAS	~~0.0612~~ 0.0903	~~2.45~~ 1.74	~~2.35~~ 0.157
3	Pressurizer surge line piping	Surge line piping to safe end weld	SS	~~0.6~~ 0.0411	~~15.35~~ 14.45	~~9.21~~ 0.594
4	RCS piping charging system nozzle	NA	SS	~~NA~~[2] 0.1812	~~15.35~~ 3.98	~~NA~~[2] 0.722
5	RCS piping safety injection nozzle	NA	SS	~~NA~~[2] 0.1709	~~15.35~~ 5.0117	~~NA~~[2] 0.8565
6	RHR Class 1 piping	NA	SS	~~NA~~[2] 0.1279	~~15.35~~ 7.79	~~NA~~[2] 0.9961

As described in LRA Section 4.3.3 and shown in Tables 4.3-13 and 4.3-14, the CUFs for the reactor vessel locations were not changed. The evaluation documented in the LRA used bounding Fens applied to the CUFs of record for the bottom head to shell region, the reactor vessel inlet nozzle and the reactor vessel outlet nozzle. The tables show the Cumulative Usage Factors are all below 1.0 for these three locations for both units.

The stress and fatigue evaluations for the remaining piping components listed in Table 4.3-13 and 4.3-14 were performed using standard methods of the ASME Code, Section III. Detailed stress models of the surge line hot leg nozzle, pressurizer surge nozzle, reactor coolant piping charging system nozzle, reactor coolant piping safety injection nozzle, and the RHR system class 1 piping locations were prepared. The analyst developed detailed stress history inputs for all transients considered in the evaluation, which were subsequently used to provide detailed inputs used in the EAF evaluations. The ASME Code evaluations were performed for the piping components to reduce conservatism in the analyses or because the current licensing basis (CLB) qualification was to the ANSI B31.1 Power Piping Code, which did not require a fatigue usage factor calculation. The evaluations were limited to the stress qualifications related to the fatigue requirements of the ASME Code. The primary stress qualifications documented in the CLB remain applicable for the components evaluated.

The evaluation of EAF was accomplished through the application of Fen factors, as described in NUREG/CR-5704 for the stainless steels in the pressurizer surge line, the reactor coolant piping charging and safety injection system nozzles, and the RHR system Class 1 piping. In addition, Fen factors for the pressurizer surge nozzle dissimilar metal weld also considered the carbon steel factors in NUREG/CR-6583. The Fen factors are calculated based on detailed inputs described in the applicable NUREG and are directly applied to the ASME Code fatigue results.

The refined fatigue analyses of the Indian Point Unit 2 and Unit 3 piping locations corresponding to the locations identified in NUREG/CR-6260 for older vintage Westinghouse plants demonstrate that cumulative fatigue usage factors including consideration of reactor water environmental effects are below a value of 1.0 for transients postulated for 60 years of operation.

The results of this evaluation resolve commitment number 33 in the NRC Safety Evaluation Report on license renewal for Indian Point Nuclear Generating Unit 2 and Unit 3.

ATTACHMENT 2 TO NL-10-082

List of Regulatory Commitments

ENTERGY NUCLEAR OPERATIONS, INC
INDIAN POINT NUCLEAR GENERATING UNITS 2 AND 3
DOCKET NOS. 50-247 & 50-286

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
5	Enhance the External Surfaces Monitoring Program for IP2 and IP3 to include periodic inspections of systems in scope and subject to aging management review for license renewal in accordance with 10 CFR 54.4(a)(1) and (a)(3). Inspections shall include areas surrounding the subject systems to identify hazards to those systems. Inspections of nearby systems that could impact the subject systems will include SSCs that are in scope and subject to aging management review for license renewal in accordance with 10 CFR 54.4(a)(2).	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.10 A.3.1.10 B.1.11
6	Enhance the Fatigue Monitoring Program for IP2 to monitor steady state cycles and feedwater cycles or perform an evaluation to determine monitoring is not required. Review the number of allowed events and resolve discrepancies between reference documents and monitoring procedures. Enhance the Fatigue Monitoring Program for IP3 to include all the transients identified. Assure all fatigue analysis transients are included with the lowest limiting numbers. Update the number of design transients accumulated to date.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039 NL-07-153	A.2.1.11 A.3.1.11 B.1.12, Audit Item 164
7	Enhance the Fire Protection Program to inspect external surfaces of the IP3 RCP oil collection systems for loss of material each refueling cycle. Enhance the Fire Protection Program to explicitly state that the IP2 and IP3 diesel fire pump engine sub-systems (including the fuel supply line) shall be observed while the pump is running. Acceptance criteria will be revised to verify that the diesel engine does not exhibit signs of degradation while running; such as fuel oil, lube oil, coolant, or exhaust gas leakage. Enhance the Fire Protection Program to specify that the IP2 and IP3 diesel fire pump engine carbon steel exhaust components are inspected for evidence of corrosion and cracking at least once each operating cycle. Enhance the Fire Protection Program for IP3 to visually inspect the cable spreading room, 480V switchgear room, and EDG room CO_2 fire suppression system for signs of degradation, such as corrosion and mechanical damage at least once every six months.	IP2: September 28, 2013 IP3: December 12, 2015	NL-07-039	A.2.1.12 A.3.1.12 B.1.13

#	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE	RELATED LRA SECTION / AUDIT ITEM
33	At least 2 years prior to entering the period of extended operation, for the locations identified in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3), under the Fatigue Monitoring Program, IP2 and IP3 will implement one or more of the following: (1) Consistent with the Fatigue Monitoring Program, Detection of Aging Effects, update the fatigue usage calculations using refined fatigue analyses to determine valid CUFs less than 1.0 when accounting for the effects of reactor water environment. This includes applying the appropriate Fen factors to valid CUFs determined in accordance with one of the following: 1. For locations in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3), with existing fatigue analysis valid for the period of extended operation, use the existing CUF. 2. Additional plant-specific locations with a valid CUF may be evaluated. In particular, the pressurizer lower shell will be reviewed to ensure the surge nozzle remains the limiting component. 3. Representative CUF values from other plants, adjusted to or enveloping the IPEC plant specific external loads may be used if demonstrated applicable to IPEC. 4. An analysis using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case) may be performed to determine a valid CUF. (2) Consistent with the Fatigue Monitoring Program, Corrective Actions, repair or replace the affected locations before exceeding a CUF of 1.0.	IP2: September 28, 2011 IP3: December 12, 2013 <u>Complete</u>	NL-07-039 NL-07-153 NL-08-021 <u>NL-10-082</u>	A.2.2.2.3 A.3.2.2.3 4.3.3 Audit item 146
34	IP2 SBO / Appendix R diesel generator will be installed and operational by April 30, 2008. This committed change to the facility meets the requirements of 10 CFR 50.59(c)(1) and, therefore, a license amendment pursuant to 10 CFR 50.90 is not required.	April 30, 2008 Complete	NL-07-078 NL-08-074	2.1.1.3.5

NYS-26/26A & RIVERKEEPER TC-1/1A: ATTACHMENT 15

ATTACHMENT 15 EXCLUDED BECAUSE IT CONTAINS ENTERGY-DESIGNATED PROPRIETARY INFORMATION OWNED BY WESTINGHOUSE AND SUBJECT TO NONDISCLOSURE AGREEMENT AND ASLB SEPTEMBER 4, 2009 PROTECTIVE ORDER

NYS-26/26A & RIVERKEEPER TC-1/1A: ATTACHMENT 16

ATTACHMENT 16 EXCLUDED BECAUSE IT CONTAINS ENTERGY-DESIGNATED PROPRIETARY INFORMATION OWNED BY WESTINGHOUSE AND SUBJECT TO NONDISCLOSURE AGREEMENT AND ASLB SEPTEMBER 4, 2009 PROTECTIVE ORDER

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	Docket Nos. 50-247-LR and
	)	50-286-LR
ENTERGY NUCLEAR OPERATIONS, INC.	)	
	)	
(Indian Point Nuclear Generating Units 2 and 3)	)	
	)	August 25, 2010

CERTIFICATE OF SERVICE

I hereby certify that copies of the "Applicant's Motion for Summary Disposition of New York State Contentions 26/26A and Riverkeeper Technical Contentions 1/1A (Metal Fatigue of Reactor Components" and the Public Version of the Supporting Attachments (i.e., excluding Entergy-Designated Proprietary Attachments 15 and 16), dated August 25, 2010, were served this 25th day of August, 2010 upon (1) all persons listed below by e-mail, and (2) by first-class mail to those persons who are not receiving the Non-Public Confidential Proprietary Version of the Supporting Attachments, as designated below with a double asterisk (**).

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* Original and 2 copies provided to the Office of the Secretary.

Docket, Hearing

From: O'Neill, Martin [martin.o'neill@morganlewis.com]
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To: Docket, Hearing; McDade, Lawrence; Wardwell, Richard; Lathrop, Kaye; OCAAMAIL Resource; Kirstein, Josh; Turk, Sherwin; Mizuno, Beth; Roth(OGC), David; Harris, Brian; Jones, Andrea; 'gss1@westchestergov.com'; 'mannajo@clearwater.org'; 'driesel@sprlaw.com'; 'jsteinberg@sprlaw.com'; 'sfiller@nylawline.com'; 'Ross Gould'; 'mdelaney@nycedc.com'; 'phillip@riverkeeper.org'; 'Deborah Brancato'; 'vob@bestweb.net'; 'Robert.Snook@po.state.ct.us'; 'Mylan.Denerstein@oag.state.ny.us'; 'John Sipos'; 'Janice Dean'; 'jlmatthe@gw.dec.state.ny.us'; 'jlparker@gw.dec.state.ny.us'
Cc: Sutton, Kathryn M.; Bessette, Paul M.
Subject: Correction of Clerical Error -- Public Version of Entergy's Motion for Summary Disposition of NYS-26/26A & Riverkeeper TC-1/1A (Metal Fatigue) , Indian Point Units 2 and 3, Docket Nos. 50-247-LR and 50-286-LR
Attachments: Cover Sheet & Attach. List for Public Filing.pdf

Yesterday evening, counsel for Entergy made the filing described in the e-mail below. The hard copy of that filing, which was sent via first-class mail only to those persons not receiving the proprietary version, included a Cover Sheet to the Supporting Attachments and a List of the Attachments. The Cover Sheet and List of Attachments were inadvertently omitted from the electronic filing only. This e-mail transmits the Cover Sheet and List of Exhibits for the Public (Non-proprietary) version of the filing to correct this clerical error.

I apologize for any inconvenience.

Sincerely,

Martin J. O'Neill
Counsel for Entergy Nuclear Operations, Inc.

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Cc: Sutton, Kathryn M.; Bessette, Paul M.
Subject: Separate Public Version of Entergy's Motion for Summary Disposition of NYS-26/26A & Riverkeeper TC-1/1A (Metal Fatigue) , Indian Point Units 2 and 3, Docket Nos. 50-247-LR and 50-286-LR

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

Earlier this evening, Entergy filed the "Applicant's Motion for Summary Disposition of New York State Contentions 26/26A and Riverkeeper Technical Contentions 1/1A (Metal Fatigue of Reactor Components," dated August 25, 2010." Because two of the supporting attachments (Attachments 15 and 16) contained Entergy-Designated Confidential Proprietary Information, Entergy confined initial service of that pleading to the Licensing Board, its law clerk, the Office

of the Secretary, and representatives of Hearing Participants that are authorized to receive such information pursuant to the Board's September 4, 2009 Protective Order.

Attached to this e-mail is a public version of the above-identified filing that excludes Confidential Proprietary Attachments 15 and 16. It is being served this 25th day of August, 2010 upon (1) all persons listed in the attached certificate of service by e-mail, and (2) those persons who are not receiving the Non-Public Confidential Proprietary Version of the Supporting Attachments, as designated in the attached certificate of service with a double asterisk (**), by first-class mail.

Please contact me if you have any questions concerning this transmittal or believe that you have received this e-mail in error.

Sincerely,

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Counsel for Entergy Nuclear Operations, Inc.

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cocsxht01.morganlewis.net ([10.242.132.24]) with mapi; Thu, 26 Aug 2010
13:18:29 -0400
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Thread-Topic: Correction of Clerical Error -- Public Version of Entergy's Motion for Summary Disposition of NYS-26/26A & Riverkeeper TC-1/1A (Metal Fatigue) , Indian Point Units 2 and 3, Docket Nos. 50-247-LR and 50-286-LR
Thread-Index: ActEvGGYrm0rUcYpS2C5cXcXYNye4AAhN09A
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Accept-Language: en-US
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X-MS-Has-Attach: yes
X-MS-TNEF-Correlator:
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acceptlanguage: en-US
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