August 20, 2012

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
ENTERGY NUCLEAR OPERATIONS, INC.	)	Docket Nos. 50-247/286-LR
	)	
(Indian Point Nuclear Generating	)	
Units 2 and 3)	)	

NRC STAFF TESTIMONY OF DR. ALLEN L. HISER, MR. ON YEE, AND DR. CHING NG
CONCERNING PORTIONS OF CONTENTION NYS-38

Q1. Please state your names, occupations, and by whom you are employed.

A1(a). My name is Dr. Allen Hiser, Jr. I am employed as the Senior Technical Advisor for License Renewal Aging Management in the Division of License Renewal, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission (NRC), in Washington, District of Columbia (DC). I received Bachelor of Science and Master of Science degrees in Mechanical Engineering from the University of Maryland at College Park. I also received a Ph.D. in Materials Science and Engineering from Johns Hopkins University. I have been a participant in American Society of Mechanical Engineers (ASME) Working Groups on Flaw Evaluation and Pipe Flaw Evaluation dating back to the early 1980s. For some of this time, I was the voting member and the NRC representative of these working groups. Currently, I am a member of the Special Working Group on Nuclear Plant Aging Management. A statement of my professional qualifications has been provided. *See* Hiser Professional Qualifications (ex. NRC000103).

A1(b). My name is Mr. On Yee. I am employed as a Mechanical Engineer in the Aging Management of Reactor Systems Branch, Division of License Renewal, Office of Nuclear Reactor Regulation, U.S. NRC, in Washington, DC. I received a Bachelor of Science degree in

Mechanical Engineering from Polytechnic University, which is located in Brooklyn, NY. A statement of my professional qualifications has been provided. *See* Yee Professional Qualifications (ex. NRC000104)

A1(c). My name is Dr. Ching Ng. I am employed as a Mechanical Engineer in the Aging Management of Reactor Systems Branch, Division of License Renewal, Office of Nuclear Reactor Regulation, U.S. NRC, in Washington, DC. I have earned Bachelor of Science, Master of Science, and Doctoral degrees in Mechanical Engineering from the University of California, Berkeley. A statement of my professional qualifications has been provided. *See* Ng Professional Qualifications (ex. NRC000105).

Q2. Please describe the nature of your current responsibilities.

A2 (a). (AH) My responsibilities include providing technical advice and assistance to the Division of License Renewal on a variety of technical, regulatory and policy issues related to aging management of nuclear power plant systems, structures, and components. I have worked at the agency for 22 years in the Office of Nuclear Regulatory Research and the Office of Nuclear Reactor Regulation. My responsibilities include serving as a lead technical expert for aging management evaluation and assisting other NRC Staff as they implement their review of license renewal applications.

A2 (b). (OY) I have been working at the agency over six years. I am currently a technical reviewer in the Aging Management of Reactor Systems Branch, which provides mechanical and materials engineering technical expertise in the review of the reactor coolant system of license renewal applications. I am responsible for conducting technical reviews of license renewal aging management programs (AMPs), aging management review (AMRs) items and time-limited aging analyses (TLAAs). Specifically, I review the area of metal fatigue, which includes on-site audits of the underlying detailed documents that support the Applicant's technical basis. I review and assess the relevant information in license renewal applications (LRAs), and craft requests for additional information when the LRA lacks information. I document the agency's findings in the area of metal fatigue and incorporate them into the safety evaluation report and audit reports. These two documents together form the bases of the agency's decision for license renewal.

A2(c). (CN) I have been working at the agency over six years. I am currently a technical reviewer in the Aging Management of Reactor Systems Branch, which provides mechanical and materials engineering technical expertise in the review of the reactor coolant system of license renewal applications. I am responsible for conducting technical reviews of license renewal aging management programs, aging management review items and time-limited aging analyses. Specifically, I review the area of metal fatigue, which includes on-site audits of the underlying detailed documents that support the Applicant's technical basis. I review and assess the relevant information in LRAs, and craft requests for additional information when the LRA lacks information. I document the agency's findings in the area of metal fatigue and incorporate them into the safety evaluation report and audit reports. These two documents together form the bases of the agency's decision for license renewal.

Q3. Please describe your duties in connection with the NRC Staff's review of the LRA submitted by Entergy Nuclear Operations, Inc. ("Entergy" or "Applicant") for Indian Point Nuclear Generating Units 2 and 3 ("IP2" and "IP3," or "Indian Point").

A3(a). (AH) I was the Chief of the Steam Generator Tube Integrity and Chemical Engineering Branch in the Office of Nuclear Reactor Regulation when the Indian Point LRA was received.

My branch was responsible for the review of several parts of the Indian Point LRA. I provided leadership to the technical reviewers for the aging management programs related to Steam Generator Tube Integrity, Flow Accelerated Corrosion, Containment Protective Coatings, Steam Generator Blowdown System, Charging and Volume Control System, and Boraflex and Boral in the Spent Fuel Pool. I reviewed and approved the requests for additional information and safety evaluation report input produced by my branch. I also provided feedback on these work products that were developed by the technical reviewers in my branch. During my work in the Division of License Renewal, I assisted and guided the Staff in its review of information submitted by the Applicant on environmentally-assisted fatigue analyses, which was used to develop NUREG-1930, Supplement 1, *Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3*, (August 2011) (ADAMS Accession No. ML11242A215) (Ex. NYS000160) ("SER Supp. 1"). As a part of this work I reviewed the license renewal application and the Staff safety evaluation report. Further, I reviewed and provided advice to the author for the request for additional information related to environmentally-assisted fatigue analyses and the consideration of additional locations to address the effects of reactor water environment on metal fatigue. I previously provided testimony to present the Staff's view with respect to the consolidated New York Contention 26B and Riverkeeper Contention TC-1B (NYS-26B/RK-TC-1B) (Metal Fatigue).

A3(b). (OY). I have been part of the IP2 and IP3 LRA review since October 2007.
From October 2007 to November 2009, as part of my formal qualification process as a License Renewal Technical Auditor/Team Leader, I assisted in the review of the existing Fatigue Monitoring program, metal fatigue TLAA and environmentally-assisted fatigue analyses for the IP2 and IP3 LRA. In this regard, I assisted in the review of the Applicant's on-site technical documentation that described its existing Fatigue Monitoring program, which will also be used

as its aging management program for license renewal. I also assisted in the review of the Applicant's existing metal fatigue analyses, which are time-limited aging analyses as defined in Title 10 of the *Code of Federal Regulations* (CFR) Section 54.3, and the Applicant's environmentally-assisted fatigue analyses, which are not time-limited aging analyses as defined in 10 C.F.R. 54.3. I worked with the principal reviewer in the preparation of the review in these areas, which is documented in NUREG-1930, Vol. 1 and Vol. 2, *Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3*, (November 2009) (ADAMS Accession Nos. ML093170451 and ML093170671) (Ex. NYS00326A-F) (together, "SER"). I was the responsible Staff member for the area of metal fatigue during the Advisory Committee on Reactor Safeguards Sub-Committee and Full-Committee meetings on March 4, 2009 and September 10-12, 2009, respectively. Subsequently, I was qualified as a License Renewal Technical Auditor/Team Leader on December 1, 2009. From January 2011 to August 2011, I was a peer reviewer for the environmentally-assisted fatigue analyses for the IP2 and IP3 LRA. In this regard, I peer reviewed and provided technical feedback on the environmentally-assisted fatigue section of the Staff's SER Supp. 1. Also as part of my responsibilities, I submitted an affidavit on behalf of the Staff in response to the Applicant's motion for summary disposition of New York Contention 26/26A and Riverkeeper Contention TC-1/1A Applicant's Motion for Summary Disposition of New York State Contentions 26/26A & Riverkeeper Technical Contentions 1/1A (Metal Fatigue of Reactor Components) (Aug. 25, 2010), which the Staff filed on September 14, 2010. NRC Staff's Answer to Applicant's Motion for Summary Disposition of New York Contention 26/26A and Riverkeeper Contention TC-1/1A -- Metal Fatigue (Sept. 14, 2010). I previously provided testimony to present the Staff's view with respect to NYS-26B/RK-TC-1B (Metal Fatigue).

A3(c). (CN). From June 2010 to August 2011, I served as a reviewer for the environmentally-assisted fatigue analyses for the IP2 and IP3 LRA. In that capacity, I reviewed the Applicant's response to the Staff's request for additional information (RAI), NRC Letter, *Request for Additional Information for the Review of the Indian Point Nuclear Generating Unit Numbers 2 and 3, License Renewal Application,* (February 10, 2011) (ADAMS Accession No. ML110190809) (Ex. NYS000150) ("RAI Letter"), related to effects of reactor water environment on metal fatigue. Based on the Staff's guidance and my engineering experience, I developed the updated environmentally-assisted fatigue section in Section 4.3.3 of the Staff's SER Supp. 1. I previously provided testimony to present the Staff's view with respect to NYS-26B/RK-TC-1B (Metal Fatigue).

Q4. What is the purpose of your testimony?

A4. (AH, OY, CN) The purpose of our testimony is to present the NRC Staff's (Staff's) views with respect to NYS-38/RK-TC-5, specifically those aspects of the contention related to metal fatigue. As directed by the Board, we are also providing rebuttal testimony to NYS-38/RK-TC-5. Our testimony is being used to support the Staff's Statement of Position concerning the New York Contention 38 and Riverkeeper Contention TC-5 (NYS-38/RK-TC-5), which the Staff is filing simultaneously with our testimony.

Q5. Are you familiar with the New York Contention 38 and Riverkeeper Contention TC-1B?

A5. (AH, OY, CN) Yes. As stated in the Board Memorandum and Order (Admitting New Contention NYS-38/RK-TC-5), (November 10, 2011) (ADAMS Accession No. ML11314A211) ("Order") and Board Order (Granting Entergy's Motion for Clarification of Licensing Board Memorandum and Order Admitting Contention NYS-38/RK-TC-5), (December 6, 2011) (ADAMS Accession No. ML11340A088) ("Memorandum and Order"), NYS-38/RK-TC-5 questions whether Entergy has a program that will manage the effects of aging of several critical components or systems and whether the proffered programs provide an adequate record and rational basis to the NRC upon which it can determine whether to grant a renewed license to Entergy. Order at 2. The Intervenors' contention, which relied on multiple bases, included the claim that there is insufficient information in Entergy's recent commitments that were addressed in the SER Supp. 1. Order at 3. We also have read the Intervenors' statement of position, *State of New York and Riverkeeper, Inc. Initial Statement of Position in Support of Joint Contention NYS-38/RK-TC-5 (*Ex. NYS000371) ("NYS-38/RK-TC-5 SOP") and the supporting expert testimony from Dr. Richard T. Lahey Jr., Pre-Filed Written Testimony of Dr. Richard T. Lahey, Jr. Regarding Contention NYS-38/RK-TC-5 *(*Ex. NYS000374) ("Lahey Pre-Filed Testimony") and Dr. Joram Hopenfeld, *Prefiled Written Testimony of Dr. Joram Hopenfeld Regarding Contention NYS-38/RK-TC-5 (*Ex. RIV000102) ("Hopenfeld Pre-Filed Testimony")submitted on June 19, 2012.

Q6. What are the "multiple bases" that the Intervenors referred to?

A6. (AH, OY, CN) As described in the State of New York and Riverkeeper's New Joint Contention NYS-38/RK-TC-5, (September 30, 2011) (ADAMS Accession No. ML11273A196) ("NYS-38/RK-TC-5"), the Intervenors' bases are: Basis (1) that Entergy has deferred defining the process to be used to determine the most limiting locations for environmentally-assisted metal fatigue calculations (CUF_{en} calculations) and selection of those locations; Basis (2) that Entergy has not specified the criteria it will use and assumptions upon which it will rely for modifying the WESTEMS computer model for CUF_{en} calculations; Basis (3) that Entergy has not adequately defined how it will manage primary water stress corrosion cracking (PWSCC) for the steam generator divider plates because it will rely on an industry "report which is not expected to be available until 2013 and, in the meantime to institute an unspecified inspection program to ascertain, long after commencement of the license renewal period, whether stress corrosion cracking is actually occurring in the divider plates of the steam generators;" and, Basis (4) that Entergy "has offered an AMP for reactor vessel internals which it will not actually follow and has promised to follow an AMP the details of which are not disclosed." See NYS-38/RK-TC-5 at 1-3. The Intervenors' expert testimony and statement of position submitted on June 19, 2012 are associated with Basis (1), Basis (2) and Basis (3), as described above.

Q7 Which bases of NYS-38/RK-TC-5 does your testimony address?

A7. (AH, OY, CN) This testimony addresses Basis (1) and Basis (2), which relate to determining the most limiting locations for CUF_{en} calculations, and criteria and assumptions for modifying the WESTEMS computer model. Basis (1) is related to the applicant's Commitment No. 43 and Basis (2) is related to the applicant's Commitment No. 44.

Q8. Do you agree with NYS-38/RK-TC-5 as related to Basis (1) and Basis (2)?

A8. (AH, OY, CN) No, we do not agree with NYS-38/RK-TC-5 as related to Basis (1) and Basis (2).

Q9. What are "CUF_{en} calculations" that are the subject of Basis (1) and Basis (2) of NYS-38/RK-TC-5?

A9. (AH, OY, CN) CUF_{en} calculations are metal fatigue calculations that have considered the effects of reactor water environment. These CUF_{en} calculations are directly related to metal fatigue calculations required by the Commission's regulations.

Q10. What are the Commission's requirements pertaining to metal fatigue?

A10. (AH, OY, CN) Title 10 of the Code of Federal Regulations (10 CFR) §§ 50.55a(c)(1) requires that components of the reactor coolant pressure boundary meet the metal fatigue requirements for Class 1 components in the *American Society of Mechanical Engineers Boiler & Pressure Vessel Code (ASME Code) Section III, Rules for Construction of Nuclear Power Plant Components* (ADAMS Accession No ML11356A334) (Ex. NYS000349). The ASME Code provides the methodology for calculating the cumulative usage factors (CUF) for nuclear power plant components, and specifies a design limit of 1.0 for the CUF of any given component, including any additional stress cycles that may occur during the period of extended operation. Fatigue evaluations for ASME Code Class 1 components will be referred hereinafter as "ASME Code Class 1 fatigue evaluations."

In the context of license renewal, 10 CFR 54.33 and 54.35 require that a licensee comply with the requirements of 10 C.F.R. Part 50 during the period of extended operation, which include the provisions requiring compliance with the metal fatigue requirements of ASME Code, Section III. The ASME Code is developed by industry experts, with active participation from the NRC Staff. Engineers in the nuclear, nonnuclear, and fossil industries routinely rely on these documents in their operations. The ASME Code is updated to reflect operating experience and ongoing research and development activities in materials science and analytical areas.

Q11. What is an ASME Code Class 1 fatigue evaluation?

A11. (AH, OY, CN) An ASME Code Class 1 fatigue evaluation is a calculation that was performed by the applicant in accordance with the ASME Code, Section III; the calculation is part of an applicant's current licensing basis for the plant. An ASME Code fatigue analysis is one measure that identifies the likelihood of a component initiating a crack due to fatigue based on cyclic loading.

As described in the Section 4.3 of NUREG-1800, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants* (SRP-LR), Rev. 2, (December. 2010) (Accession No. ML103490041) (Ex. NYS000161) ("SRP-LR Rev. 2") a metal component may degrade due to fatigue when subjected to fluctuating stresses. This degradation can occur in components without flaws by the development of fatigue cracks during service. ASME Code, Section III, requires a fatigue analysis for Class 1 components unless exempted under applicable ASME Code, Section III, provisions. ASME Code, Section III, provides a specific process for this analysis, which considers the anticipated severity and number of thermal and pressure cycles for all transients, and includes the calculation of a parameter called cumulative usage factor (CUF). *See* SRP-LR Rev. 2 at 4.3-1 (Ex. NYS000161). CUF is evaluated by first determining the severity of each transient, considering both the temperatures and pressures for the cycle. Next the number of cycles that a transient of this severity would take to initiate a fatigue crack ("Ni") is determined. The usage factor for this one transient is then determined by dividing the number of anticipated cycles for this transient by "Ni." Summing up the usage factor from each transient for the component allows the cumulative usage factor, or CUF, to be determined.

The ASME Code limits the CUF to a value of less than or equal to 1.0 for acceptable fatigue design and provides assurance that no crack has been formed by fatigue. A cumulative usage factor above a value of 1.0 indicates an increased likelihood that a fatigue crack may form.

Q12. What does it mean when a CUF value is less than 1.0?

A12. (AH, OY, CN) When a CUF value is less than 1.0, it provides assurance that a fatigue crack has not formed or initiated in the material. *See* SRP-LR, Rev. 2 at page 4.3-1.

Q13. What does it mean that a crack has not formed?

A13. (AH, OY, CN) This means that when a CUF value is less than 1.0 there are no fatigue cracks assumed to be present in the location that is being evaluated. When a CUF value exceeds 1.0, it means that there is a possibility that a fatigue crack may have formed and is assumed to be present in the location that is being evaluated. *See* SRP-LR, Rev. 2 at page 4.3-1.

Q14. What is the Staff's definition of fatigue life?

A14. (AH, OY, CN) Fatigue life is the accumulation of fatigue usage from zero to 1.0 and is associated with the initiation of fatigue cracks.

Q15. What portion of the ASME Code addresses the initiation of fatigue cracks?

A15. (AH, OY, CN) ASME Code, Section III, Subsection NB addresses the initiation of fatigue cracks and provides the process to calculate a CUF value.

Q16. Does the Staff consider a fatigue crack is present and growing when the CUF value is less than 1.0?

A16. (AH, OY, CN) No, the Staff does not consider that a fatigue crack has initiated and is growing in the component when the CUF value is less than or equal to 1.0. At this point it is considered that the fatigue life of the component has not been exceeded and the component is capable of withstanding cyclic stress.

Q17. Does a CUF value of 1.0 indicate the immediate failure of a component?

A17. (AH, OY, CN) No, a CUF value of 1.0 is not indicative of immediate failure of the component.

Q18. What does it mean when CUF is greater than 1.0?

A18. (AH, OY, CN) A CUF above the value of 1.0 allows for the increasing possibility that a crack may form. *See* SRP-LR, Rev. 2 at Page 4.3-1.

Q19. What does it mean "a crack may form"?

A19. (AH, OY, CN) The Staff is not aware of any scientific evidence to indicate that a fatigue crack has formed when CUF value exceeds 1.0. The Staff conservatively assumes that a fatigue crack may have formed and is growing when a CUF value is greater than 1.0.

Q20. Once a fatigue crack has formed or initiated, what happens if the material is subjected to more cycles?

A20. (AH, OY, CN) Once a fatigue crack has initiated or formed, the fatigue crack will grow and propagate under further cyclic loading.

Q21. What part of the ASME Code governs the growth and propagation of a crack under cyclic loading?

A21. (AH, OY, CN) Appendix A and Appendix C of the ASME Code, Section XI, *Analysis of Flaws* (Ex. NRC000149) ("Appendix A") and *Evaluation of Flaws in Piping* (Ex. NRC000150) ("Appendix C"), respectively, provide the procedures to evaluate the growth of a crack, which was detected through inspection, under cyclic loading. An introduction to the purpose of Appendices A and C are outlined in Section A-1000 and C-1000, respectively. *See* Appendix A at 301 (Ex. NRC000149). *See* Appendix C at 325 (Ex. NRC000150). These appendices to ASME Section XI demonstrate that there are established methods to evaluate flaws that are identified through inspections to determine their acceptability for continued service and that a crack or flaw does not necessarily correlate to a failed component.

ASME Code, Section XI, Appendix L provides methods for performing fatigue assessments to determine acceptability of reactor coolant system and primary pressure boundary components and piping subjected to cyclic loadings for continued service. *See* Appendix L at 421 (Ex. NRC000113). One option provided by Appendix L is to perform fatigue usage factor evaluation, in accordance with ASME Code, Section III, for reactor coolant system primary pressure boundary components and piping in operating plants. *See* Appendix L at 422 (Ex. NRC000113). Another option provided by Appendix L is to perform a flaw tolerance evaluation for operating plant components and piping. This option involves analyzing the growth of a postulated fatigue crack under cyclic loading. *See* Appendix L at 423 to 427 (Ex. NRC000113). This appendix to ASME Section XI demonstrates that there is an established method to determine the growth of a postulated flaw and that a CUF greater than 1.0 does not necessarily correlate to a failed component.

Q22. Does the Staff prohibit the use of these appendices in the ASME Code, Section XI?

A22. (AH, OY, CN) ASME Code, Section XI is required by the Commission's regulations in 10 CFR 50.55a. This section of the Commission's regulations does not prohibit the use of these appendices in the ASME Code, Section XI, but does provide a condition in 10 CFR 50.55a(b)(2)(xxviii) for the use of ASME Nonmandatory Appendix A, "Analysis of Flaws."

Q23. How is CUFen calculated?

A23. (AH, OY, CN) A value of CUF_{en} is computed in two parts. The first part is to calculate the CUF using the methodology from ASME Code, Section III. The second part is to calculate the environmental adjustment factor (F_{en}) by using the guidance recommended in NUREG-1801, *GALL Report*, Rev. 1 (September 2005) (Accession Nos. ML052110005 and ML052110006) (Ex. NYS00146A-D) (together, "GALL Report Rev. 1") and NUREG-1801, *Generic Aging Lessons Learned (GALL) Report*, Rev. 2, (Dec. 2010) (ADAMS Accession No. ML103490036) (Ex. NYS000147A-D) ("GALL Report Rev. 2"). The CUF_{en} value is equal to the CUF multiplied by the F_{en} factor.

Q24. What are the Commission's requirements pertaining to aging management of metal fatigue for license renewal?

A24. (AH, OY, CN) 10 CFR 54.29 states that a renewed license may be issued if actions have been identified and have been or will be taken with respect to managing the effects of aging (§§ 54.29(a)(1)) and time-limited aging analyses (§§ 54.29(a)(2)) during the period of extended operation to ensure the functionality of structures and components within the scope of license renewal, such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis (CLB), and that any changes made to the plant's current licensing basis in order to comply with this paragraph are in accord with the Atomic Energy Act and the Commission's regulations. Cracking due to metal fatigue is one of the effects of aging that requires management during the period of extended operation.

Q25. What are the Commission's requirements pertaining to time-limited aging analyses?

A25. (AH, OY, CN) Time-limited aging analyses ("TLAAs") are defined in 10 CFR 54.3. 10 CFR 54.21(c)(1) requires that a license renewal application include an evaluation of TLAAs to demonstrate that:

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

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

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

Q26. Are metal fatigue evaluations required by NRC regulations classified as time-limited aging analyses?

A26. (AH, OY, CN) Yes, metal fatigue evaluations required by NRC regulations are TLAAs because they are a part of the current licensing basis, which is one of the provisions of the definition of a TLAA in 10 CFR 54.3.

Q27. What are the Commission's requirements pertaining to the effects of reactor water environment on metal fatigue?

A27. (AH, OY, CN) There are no explicit requirements in the Commission's regulations nor in the ASME Code, which has been approved for use by the Staff, for an assessment of the effects of reactor water environment on metal fatigue (hereinafter environmentally-assisted fatigue analyses).

Q28. What Commission documents identify the need to consider the effects of reactor water environment on metal fatigue?

A28. (AH, OY, CN) The need to evaluate environmentally-assisted fatigue is identified in license renewal guidance documents, specifically the GALL Report Rev. 1, GALL Report Rev. 2, NUREG-1800, *SRP-LR, Rev. 1*, (September, 2005) (Accession No. ML052110007) (Ex. NYS000195) ("SRP-LR Rev. 1") and the SRP-LR Rev.2.

Q29. What is the purpose of NUREG-1801, GALL Report?

A29. (AH, OY, CN) GALL Report, Rev. 2 and 1 are technical basis documents to SRP-LR Rev.

2 and SRP-LR Rev. 1, respectively. The GALL Report, Rev. 1 and Rev. 2, state that 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. The GALL Report contains one acceptable way to manage aging effects for license renewal. It also states that an applicant may propose alternatives for Staff review in its plant-specific license renewal application and use of the GALL Report is not required, but use of the GALL Report should facilitate both preparation of a license renewal application by an applicant and timely, uniform review by the NRC Staff. *See* GALL Report, Vol.1, Rev. 1 at 4 and GALL Report Rev. 2 at 8.

Q.30 What is the Staff's guidance in the GALL Report regarding metal fatigue?

A30. (AH, OY, CN) The GALL Report provides items for aging management review that are identified by component, material, environment and aging effect for metal fatigue. Section X.M1 of the GALL Report, Rev. 1 and Rev. 2, provides recommendations for an acceptable aging management program (AMP) to manage the effects of metal fatigue. *See* GALL Report Rev. 1 at X M-1 (Ex. NYS00146A-D) and GALL Report Rev. 2 at X M-1 (Ex. NYS00147A-D). The aging effect of metal fatigue is associated with the number of thermal transients that occur, such as when the plant heats up and cools down, and not with the passage of time alone. The design of the plant originally considered a certain number of occurrences for thermal and pressure transients that were expected to occur in the lifetime of the plant. Specifically, in order not to exceed the design limit for the plant, the aging management program described in both Revisions of GALL Report AMP X.M1 recommends monitoring and tracking the number of thermal and pressure transients.

Q31. How are the effects of reactor water environment on metal fatigue addressed in the GALL Report?

A31. (AH, OY, CN) The GALL Report, Rev. 1 and Rev. 2, includes the effects of reactor water environment on metal fatigue in the AMP described in Section X.M1 of the GALL Report, as if the effects of reactor water environment on metal fatigue were a TLAA. *See* GALL Report Rev. 1 at X M-1 (Ex. NYS00146A-D) and GALL Report Rev. 2 at X M-1 (Ex. NYS00147A-D). Including environmentally-assisted fatigue in the fatigue monitoring AMP is for convenience and for ease of review for the Staff, since the same AMP used for the TLAAs associated with metal fatigue CUF values also apply to CUFen. However, the effects of reactor water environment on metal fatigue are not TLAAs in accordance with 10 CFR 54.3(a), as described previously, because the analyses are not used in the original design nor the current licensing basis of plants, including IP2 and IP3.

Q32. Please describe the purpose of NUREG-1800, SRP-LR?

A32. (AH, OY, CN) The SRP-LR, Rev. 1 and Rev. 2, provides guidance to the Staff on how to perform safety reviews of applications to renew nuclear power plant licenses in accordance with 10 CFR Part 54. The principal purposes of the SRP-LR are to ensure the quality and uniformity of the Staff's review and to present a well-defined base from which to evaluate the applicant's programs and activities for the period of extended operation. *See* SRP-LR Rev. 1 and Rev. 2 at iii (Ex. NYS000195 and Ex. NYS000161, respectively). The SRP-LR notes that the Staff conducts an audit and review at the applicant's facility to evaluate AMPs that the applicant claims to be consistent with the GALL Report. *See* SRP-LR Rev. 1 and Rev. 2 at 3.0-1 (Exh. NYS000195 and NYS000161, respectively).

Q33. What is the Staff's guidance in the SRP-LR regarding metal fatigue?

A33. (AH, OY, CN) SRP-LR Rev. 1 and Rev. 2 address the review of metal fatigue as a TLAA in Section 4.3. This section states that metal fatigue of components may have been evaluated based on an assumed number of transients or cycles for the current operating term and the validity of such metal fatigue analysis is reviewed for the period of extended operation. This section of the SRP-LR also provides specific guidance for areas of review in metal fatigue, which include, in part, ASME Section III, Class 1, 2 and 3 analyses, American National Standards Institute (ANSI) B31.1 analyses and environmentally-assisted fatigue analyses. In addition, the SRP-LR provides the reviewer with specific acceptance criteria and review procedures for the identified areas of review. For those areas of review not specifically identified in SRP-LR, general acceptance criteria and review procedures are available. If the applicant proposes that the effects of metal fatigue will be adequately managed for the period of extended operation, the SRP-LR provides guidance to the Staff to verify that the applicant has identified a program for aging management as described and evaluated in Section X.M1 of the GALL Report. *See* SRP-LR Rev. 1 at 4.3-1 through 4.3-10 (Ex. NYS000195) and SRP-LR Rev. 2 at 4.3-1 through 4.3-15 (Ex. NYS000161).

Q34. How are the effects of reactor water environment on metal fatigue addressed in the SRP-LR?

A34. (AH, OY, CN) The SRP-LR provides specific guidance on addressing the effects of reactor water environment. SRP-LR, Rev. 1 and Rev. 2, recommend that the specific components identified in NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components*, (March 1995) (ADAMS Accession No. ML031480219) (Ex. NYS000355) ("NUREG/CR-6260"), as a minimum, should be considered for the effects of reactor environment on metal fatigue. *See* SRP-LR Rev. 1 at 4.3-7 (Ex.

NYS000195) and SRP-LR Rev. 2 at 4.3-9 (Ex. NYS000161). NUREG/CR-6260 is a report that assesses the significance of the interim fatigue curves developed in the early 1990's by performing environmental fatigue evaluations for a sample of the components in the reactor coolant pressure boundary for the four U.S. nuclear steam supply system vendors (i.e., Westinghouse, General Electric, Combustion Engineering, and Babcock & Wilcox). The significance of this sample of components was chosen to provide a representative overview of components that had higher CUFs and/or were important from a risk perspective, from facilities designed by each of the four U.S. nuclear steam supply system vendors. *See* NUREG/CR-6260 at iii (Ex. NYS000355)

SRP-LR, Rev. 1 and Rev. 2, identify several technical reports as acceptable for use in evaluating the effects of reactor water environment on metal fatigue. SRP-LR, Rev. 1, identifies NUREG/CR-5704, *Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels* (April 1999) (ADAMS Accession No. ML031480394) (Ex. NYS000354) ("NUREG/CR-5704"), for use in determining the environmental effects for austenitic stainless steel components. SRP-LR, Rev. 1, also identifies NUREG/CR-6583, *Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels* (March 1998) (ADAMS Accession No. ML031480391) (Ex. NYS000356) ("NUREG/CR-6583"), for use in determining the environmental effects for carbon and low-alloy steel components. *See* SRP-LR Rev. 1 at 4.3-7 (Ex. NYS000195).

In addition to the use of NUREG/CR-5704 and NUREG/CR-6583, SRP-LR, Rev. 2, provides an additional option to use NUREG/CR-6909, *Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials*, (February 2007) (ADAMS Accession No. ML070660620) (Ex. NYS000357) ("NUREG/CR-6909"), as an acceptable alternative for austenitic stainless steel, and carbon and low-alloy steel, with additional guidance on the appropriate fatigue curve to be used with NUREG/CR-6909. In addition, this technical report is the only report cited in the Staff's guidance as acceptable for use in determining the environmental effects for nickel alloy components. *See* SRP-LR Rev. 2 at 4.3-9 through 4.3-10 (Ex. NYS000161).

Q35. How does the Staff use the guidance to review a license renewal application related to metal fatigue and fatigue monitoring?

A35. (AH, OY, CN) To follow this guidance, the Staff performs an on-site review of the applicant's program basis documents, plant procedures, and detail documentation related to metal fatigue and the AMP for fatigue monitoring. The Staff also performs review of the applicant's LRA and its responses to any requests for additional information related to metal fatigue. The Staff uses the GALL Report and SRP-LR as guidance throughout the review.

Q36. How does the Staff use the guidance to review a license renewal application related to the effects of reactor water environment on metal fatigue?

A36. (AH, OY, CN) The Staff reviews information provided in the license renewal application to assess consistency with the guidance provided in the SRP-LR, which identifies the effects of reactor water environment on metal fatigue to be considered as a TLAA. Although the Staff reviews these analyses similar to the approach used with TLAAs, analyses to evaluate the effects of reactor water environment on metal fatigue are not TLAAs because the effects are not considered in the current licensing basis (CLB), which is one aspect of the definition of a TLAA in 10 CFR 54.3.

Q37. Does Indian Point's LRA identify environmentally-assisted fatigue analyses as TLAAs?

A37. (AH, OY, CN) Yes, Indian Point's LRA identifies environmentally-assisted fatigue analyses as TLAAs.

Q38. Does the NRC Staff agree that the environmentally-assisted fatigue analyses for Indian Point are TLAAs?

A38. No, the Staff does not agree that the environmentally-assisted fatigue analyses for Indian Point are TLAAs. Because environmentally-assisted fatigue analyses are not contained in Indian Point's current licensing basis, they are not TLAAs as defined in 10 CFR 54.3, consistent with the Commission's holding in *Entergy Nuclear Vermont Yankee, L.L.C. and Entergy Nuclear Operations, Inc.* (Vermont Yankee Nuclear Power Station), CLI-10-17, 72 N.R.C. 1 (July 8, 2010). Because environmentally-assisted fatigue analyses do not apply to the current licensing basis for Indian Point, evaluation of environmentally-assisted fatigue analyses is not a prerequisite to issuance of a renewed license, consistent with the Commission's holding in *Entergy Nuclear Vermont Yankee, L.L.C. and Entergy Nuclear Operations, Inc.* (Vermont Yankee Nuclear Power Station), CLI-10-17, 72 N.R.C. 1 (July 8, 2010).

Q39. What information was provided in the Indian Point LRA on metal fatigue and CUF_{en} that is pertinent to NYS-38/RK-TC-5?

A39. (AH, OY, CN) The Indian Point LRA described the Fatigue Monitoring program as a pertinent aging management program. The LRA also included TLAA evaluations for metal fatigue and evaluations for environmentally-assisted fatigue.

The Fatigue Monitoring program is described in LRA Section B.1.12 as 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.

LRA Section 4.3 describes Metal Fatigue TLAAs, with Section 4.3.3 addressing effects of reactor water environment on fatigue life.

Q40. As they relate to NYS-38/RK-TC-5, what are the Staff's conclusions concerning the adequacy of the metal fatigue TLAAs, environmentally-assisted fatigue analyses, and Fatigue Monitoring program related to license renewal of IP2 and IP3?

A40. (AH, OY, CN) The Staff has determined that the metal fatigue TLAAs, the AMP for fatigue monitoring, and the environmentally-assisted fatigue analyses for IP2 and IP3 are acceptable, because the aging effect of metal fatigue due to transient loads, such as temperature and pressure changes, will be managed.

The Staff's conclusions and bases for its conclusions for the Applicant's Fatigue Monitoring program are documented in Section 3.0.3.2.6 of the Staff's SER. *See* SER at 3-76 through 3-79 (Ex. NYS00326A-F). The Staff's conclusions and bases for its conclusions for the Applicant's metal fatigue TLAAs are documented in SER Section 4.3. *See* SER at 4-18 through 4-41 (Ex. NYS00326A-F). Finally, the Staff's conclusions and bases for its conclusions on the Applicant's environmentally-assisted fatigue analyses are documented in Section 4.3.3 of the Staff's SER and Section 4.3.3 of the Staff's SER Supp. 1. *See* SER at 4-41 through 4-46 (Ex. NYS00326A-F) and SER Supp. 1 at 4-1 through 4-3 (Ex. NYS000160). In addition, the Staff's aging management program audit report, *Audit Report for Plant Aging Management Programs and*

Reviews, Indian Point Nuclear Generating Unit Nos. 2 and 3, (January 13, 2009) (ADAMS Accession No. ML083540662) (Ex. NRC000108) ("AMP Audit Report"), and the Staff's scoping and screening audit report, *Scoping and Screening Methodology Audit Trip Report For Indian Point, Units 2 and 3*, (January 13, 2009) (ADAMS Accession No. ML083540648) (Ex. NRC0000102) ("Scoping Audit Report") provide additional information that supports the Staff's conclusions in the Staff's SER and SER Supp. 1.

Q41. With respect to the Fatigue Monitoring program, the metal fatigue TLAAs and environmentally-assisted fatigue analyses, what are the issues related to NYS-38/RK-TC-5?

A41. (AH, OY, CN) There are two aspects to the Intervenors' contention:

(1) The Intervenors claim that Entergy does not demonstrate that it has a program that will manage the effects of aging. *See* Order at 2.

(2) The Intervenors claim that the Staff does not have a record and a rational basis upon which it can determine whether to grant a renewed license to Entergy. *See* Order at 2.

Q42. Does the Staff agree with the Intervenors' claim that Entergy does not have a program that will manage the effects of aging related to metal fatigue and environmentally-assisted fatigue analyses?

A42. (AH, OY, CN) The Staff does not agree with the Intervenors that Entergy has not demonstrated that it has a program that will manage the effects of metal fatigue on critical components and systems. To the contrary, Entergy has provided an aging management program in LRA Section B.1.12. Entergy has described that its program is consistent with the

GALL Report AMP X.M1 (see LRA at B-44 to B-46). It should be noted that the aging management program described in LRA Section B.1.12 that was provided by Entergy has been amended based on the Staff's audit questions and requests for additional information. The Staff's SER and SER Supp. 1 document the Staff's review including any amendments made by Entergy to its aging management programs.

Q43. Please summarize and describe the Applicant's Fatigue Monitoring program.

A43. (AH, OY, CN) The Applicant's Fatigue Monitoring program, for which Entergy described as consistent with GALL Report AMP X.M1, monitors actual plant transients and evaluates their severity against that of the design transients used in the design basis fatigue calculations.

The scope of the Applicant's Fatigue Monitoring program includes those reactor coolant system components that have metal fatigue TLAAs and environmentally-assisted fatigue analyses that were explicitly analyzed for a specified number of fatigue transients and are dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

The program also includes the sample of locations identified in NUREG/CR-6260, which include analyses that were explicitly analyzed for a specified number of fatigue transients that are dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

Entergy submitted these amendments to the LRA in Letter NL-08-057, *Letter from Entergy Nuclear Operations Inc., to NRC, Indian Point Nuclear Generating Units 2 and 3, License Renewal Application Amendment 3*, (March 24, 2008) (ADAMS Accession No. ML081070255) (Ex. NRC000109) ("NL-08-057"). In Letter NL-11-032, *Letter from Entergy Nuclear Operations, Inc. to NRC, Indian Point, Units 2 & 3, Response to Request for Additional Information on Aging*

Management Programs, (March 28, 2011) (ADAMS Accession No. ML110960360) (Ex. NRC000110) ("NL-11-032"), the Applicant provided Commitment No. 43 to review its design basis ASME Code Class 1 fatigue evaluations to confirm whether the NUREG/CR-6260 locations that have been evaluated for the effects of the reactor water environment on fatigue usage are the most limiting locations for the IP2 and IP3 configurations. If more limiting locations are identified, the most limiting location will be evaluated for the effects of the reactor water environment on fatigue usage. The applicant stated that it will complete these actions prior to September 28, 2013 for IP2 and prior to December 12, 2015 for IP3. *See* NL-11-032, Attachment 1 at 26 and Attachment 2 at 17 (Ex. NRC000110).

Q44. How did the Staff review the Applicant's Fatigue Monitoring program?

A44. (AH, OY, CN) The Staff reviewed the Applicant's Fatigue Monitoring program consistent with SRP-LR Section 4.3 and GALL Section X.M1. *See* SRP-LR Rev.1 and Rev. 2 at Section 4.3 (Ex. NYS000195 and NYS000161, respectively) Also *see* GALL Report Rev.1 and Rev. 2 at Section X.M1 (Ex. NYS00146A-D and Ex. NYS00147A-D, respectively) In addition, the Staff performed an on-site inspection and an on-site audit of the Fatigue Monitoring program.

Q45. What is the Staff's guidance for inspections related to license renewal?

A45. (AH, OY, CN) The policy and guidance for the performance of inspections at the applicant's facility are provided in Inspection Manual Chapter (IMC) 2516, *Policy and Guidance for the License Renewal Inspection Programs* (February 18, 2005) (ADAMS Accession No ML050660153) (Ex. ENT000252) ("IMC 2516"). The purpose of IMC 2516 is to document policy and guidance for review and inspection activities associated with the License Renewal Inspection Program, which is the process used by the NRC Staff, region, and consultants to verify the accuracy of the aging management programs and activities associated with an applicant's request for a renewed license for a commercial nuclear power plant beyond the initial licensing period under 10 C.F.R. Part 54. *See* IMC 2516 at 1 (Ex. ENT000252).

Q46. What guidance did the Staff use for performing the on-site audits and inspections for the Indian Point License Renewal Application?

A46. (AH, OY, CN) For the on-site audit of Indian Point Units 2 and 3, the Staff developed an audit plan, *Audit and Review Plan for Plant Aging Management Reviews and Programs – Indian Point Generating Units Nos. 2 and 3, Docket Nos. 50-247 and 50-286*, (September, 2007) (ADAMS Accession No ML072290180) (Ex. NRC0000101) ("Audit Plan"). The scope of work is defined in this audit plan and the project team sought to verify that the Applicant's aging management activities and programs will adequately manage the effects of aging on structures and components, so that their intended functions will be maintained consistent with the IP2 and IP3 CLB for the period of extended operation. *See* Audit Plan at 1 (Ex. NRC0000101).

The first on-site inspection of Indian Point Units 2 and 3 related to license renewal occurred at the Applicant's site starting January 28, 2008, and was completed on June 18, 2008. For that inspection the Staff followed NRC Inspection Manual, Inspection Procedure 71002, *License Renewal Inspection* (November 2011) (ADAMS Accession No. ML11238A010) (Ex. NRC000106) ("IP71002"). In particular, the inspection verified that the Applicant's license renewal program, including supporting activities, are planned or will be implemented consistent with the requirements of 10 C.F.R. Part 54 and the Applicant's license renewal application. The inspection also verified the documentation, implementation, and effectiveness of the aging management programs and activities associated with the Applicant's license renewal program. In addition, it verified that the Applicant had adequate programs planned or in place to implement aging management for the systems, structures and components (SSCs) that require an aging management review, such that these SSCs will be adequately maintained consistent with the rule, the Staff's existing safety evaluations, and the Applicant's license renewal program.

Details about the scope and results of the IP71002 inspection performed for Indian Point Units 2 and 3 are contained in *NRC Inspection Report 05000247/2008006 and 05000286/2008006* (August 1, 2008) (ADAMS Accession No ML082140149) (Ex. NRC000107) ("IP71002 Report").

The second on-site inspection of Indian Point Units 2 and 3 related to license renewal will be performed prior to entering the period of extended operation. For this inspection, the Staff will follow the NRC Inspection Manual, Inspection Procedure 71003, *Post-Approval Site Inspection For License Renewal* (October 2008) (ADAMS Accession No. ML082830294) (Ex. ENT000251) ("IP71003"). This inspection will verify completion of license renewal commitments and license conditions that have been added as part of the renewed license, and will ensure that selected aging management programs are implemented in accordance with the license renewal regulations. *See* IP71003 at 1 (Ex. ENT000251). Inspection results will be processed and documented using processes that are similar to those established for the ongoing oversight process.

The Staff noted that the IP71003 was structured to be completed after the license was renewed and the Staff was expected to verify that the license renewal applicant had implemented license renewal commitments before it entered the period of extended operation (i.e., the post-40-year license period). For IP2, the NRC was aware that there is the possibility that a renewed license may not be issued prior to entering the period of extended operation. Therefore, the NRC issued Temporary Instruction 2516/001, “Review of License Renewal Activities.” (March 30, 2011) (ADAMS Accession No. ML110620255) (Ex. NRC000151) (“TI 2516/001”) TI 2516/001 states that it shall be performed in cases where holders of an operating license meet the criteria of 10 CFR 2.109, “Effect of Timely Renewal Application,” for timely renewal but IP71003 cannot be performed in a timely manner because the NRC’s final decision regarding the renewal of the operating license may not allow sufficient time to plan and conduct post license-renewal inspection before the period of extended operation. *See* TI 2516/001 at 2. One of the inspection objectives is to report the status of the applicant’s implementation of license renewal commitments, license conditions and selected aging management programs as described in a plant’s license renewal safety evaluation report. *See* TI 2516/001 at 1.

On March 8, 2012, the NRC completed an inspection at the Indian Point Nuclear Generating Unit 2, which was an examination of license renewal activities under TI 2516/001. The associated Inspection Report 05000247/2012008 was issued on April 19, 2012 and states that the inspection was directed toward those activities and facilities accessible during the refueling outage. The inspection also reviewed the completion of commitments made during the renewed license application process and compliance with the Commission's rules and regulations and the conditions of IP2’s operating license. Within these areas, the inspection involved examination of selected procedures and representative records, observations of activities, and interviews with personnel. (April 19, 2012) (ADAMS Accession No. ML12110A315) (Ex. NRC000152) (“Inspection Report 05000247/2012008”) *See* Inspection Report 05000247/2012008 at 1.

In addition, as documented in Inspection Report 05000247/2012008, the NRC inspectors determined that no findings were identified and that Entergy’s actions on four commitments

(Commitments 28, 29, 34, and 36) were complete and met regulatory expectations as reflected in the staff's safety evaluation report. Furthermore, the NRC inspectors determined that inspection of two commitments (Commitments 2 and 3) concluded that additional inspection was needed and is planned prior to the scheduled completion date of September 28, 2013. *See* Inspection Report 05000247/2012008, Enclosure at 5.

The summary of the inspection performed in accordance with TI 2516/001, documented in Inspection Report 05000247/2012008, demonstrates that Entergy has begun activities associated with license renewal at IP2 and completed four commitments. Once again, the NRC inspectors determined that Entergy for these four commitments met regulatory expectations as reflected in the staff's safety evaluation report.

Q47. Was Entergy's Fatigue Monitoring program a part of the Staff's IP71002 inspection?

A47. (AH, OY, CN) Yes.

Q48. What were the conclusions related to Entergy's Fatigue Monitoring program from the IP71002 inspection?

A48. (AH, OY, CN) As documented in the IP71002 Report, the inspectors reviewed the program elements and implementation. In addition, selected components were reviewed to determine the adequacy of the process used to maintain the transient count for the component. For the Fatigue Monitoring program, the inspectors concluded that Entergy had performed adequate evaluations, including reviews of industry experience and plant history, to determine appropriate aging effects. In addition, the inspectors concluded that Entergy provided adequate guidance to ensure the aging effects are appropriately identified and addressed. *See* IP71002 Report at 4

(Ex. NRC000107).

Q49. What are the bases for Staff's conclusion that the applicant's Fatigue Monitoring program will adequately manage metal fatigue during the period of extended operation?

A49. (AH, OY, CN) The bases for the Staff's conclusion come from its review of the LRA, responses to requests for additional information, findings from the IP71002 on-site inspection, and items from the on-site audit related to the Fatigue Monitoring program, the metal fatigue TLAAs, and the environmentally-assisted fatigue analyses for IP2 and IP3. Based on this cumulative information, the Staff concluded that the Fatigue Monitoring program will adequately manage the metal fatigue TLAAs and the environmentally-assisted fatigue analyses that the Applicant dispositioned in accordance with 10 C.F.R. 54.21(c)(1)(iii).

The basis for the Staff position is summarized as follows: Section 3.2.5 of the Staff's AMP Audit Report documents the Applicant's response to Audit Item 39, in which the Applicant describes that the program monitors the transients listed in Table 4.3-1 (IP2) and 4.3-2 (IP3) of the LRA and Table 4.1-8 of the IP2 UFSAR and Table 4.1-8 of the IP3 UFSAR, by performing a review of the site data to determine transients that have occurred since the last review, and then updates the list of total transients to date. *See* AMP Audit Report at 52 and 53 (Ex. NRC000108)

As described by the Applicant's RAI response in Entergy's Letter NL-08-084, *Letter from Entergy Nuclear Operations, Inc., to NRC, Indian Point Nuclear Generating Units 2 and 3, Reply to Request for Additional Information Regarding License Renewal Application Time-Limited Aging Analyses and Boraflex* (May 16, 2008) (ADAMS Accession No. ML081490317) (Ex. ENT000194) ("NL-08-084"), the Fatigue Monitoring program also includes preventive actions that (1) update the counting of plant transients at least once each operating cycle, (2) determine if the number of cycles for each transient may be exceeded before the next update, and (3) ensure that corrective actions will be taken prior to exceeding the analyzed transient cycles. *See* NL-08-084, Attachment 1 at 4 (Ex. ENT000194).

The Applicant's program for IP2 includes 'alert cycles', which are defined as the number of cycles which are projected to accumulate (based on a specific methodology) in the next two operating periods. In addition, the Applicant's program for IP3 does not allow plant operation if the analyzed number of cycles for a particular transient is exceeded, unless an appropriate engineering evaluation, developed under the corrective action program, has determined that plant operation is acceptable. Section 3.2.5 of the Staff's AMP Audit Report documents these action limits. See AMP Audit Report at 53 (Ex. NRC000108).

As described by the Applicant's RAI response in Entergy's Letter NL-08-084, the Fatigue Monitoring program periodically ensures that the number of analyzed cycles used in the metal fatigue TLAAs and environmentally-assisted fatigue analyses are not exceeded, thereby, ensuring that the calculated cumulative usage factor and the design code limit of 1.0 are not exceeded. *See* NL-08-084, Attachment 1 at 3 through 4 (Ex. ENT000194).

The Fatigue Monitoring program includes corrective actions that are initiated if the monitoring of transient cycles performed indicates the potential for a condition outside those analyzed in the fatigue evaluation. Corrective actions include performing a more rigorous analysis to assess the condition, and repair or replacement of affected components, before the cumulative usage factor exceeds 1.0. These corrective actions were described in the Letter NL-08-021, *Letter from Entergy Nuclear Operations, Inc., to NRC, Indian Point Nuclear Generating Units 2 and 3, License Renewal Application Amendment 2* (January 22, 2008) (ADAMS Accession No.

ML080230637) (Ex. NYS000351 at Attachment 1, 2 through 3). Also *see* NL-08-084, Attachment 1 at 4 (Ex. ENT000194). These corrective actions are consistent with the recommendation in the GALL Report for X.M1, which is an acceptable option for managing metal fatigue for the reactor coolant pressure boundary, considering environmental effects.

Accordingly, the Staff has concluded that the metal fatigue TLAAs, the environmentally-assisted fatigue analyses and the Fatigue Monitoring program for license renewal of IP2 and IP3 are acceptable, and there is no merit in the contention's assertion that Entergy has not demonstrated that it has a program that will manage the effects of metal fatigue on critical components and systems.

Q50. NYS-38/RK-TC-5 identifies issues that are related to the Applicant's Commitment No. 43 as part of its basis. When did Entergy provide Commitment No. 43?

A50. (AH, OY, CN) Entergy's letter NL-11-032, dated March 28, 2011, provided Commitment No. 43.

Q51. What did Entergy state that it would do in Commitment No. 43?

A51. (AH, OY, CN) First, Entergy stated that it will review its design basis ASME Code Class 1 fatigue evaluations to confirm that the NUREG/CR-6260 locations that have been evaluated for the effects of the reactor water environment on fatigue usage are the limiting locations for the IP2 and IP3 configurations. If more limiting locations are identified, the most limiting location will be evaluated for the effects of the reactor water environment on fatigue usage. Second, Entergy will use the NUREG/CR-6909 methodology, which currently represents the best methodology for nickel alloys, in the evaluation of the limiting locations consisting of nickel alloy, if any. *See* NL-11-032, Attachment 1 at 26 and Attachment 2 at 17 (Ex. NRC000110).

Q52. What is a "design basis” ASME Code Class 1 fatigue evaluation and why is this commitment limited to “design basis” fatigue evaluations?

A52. (AH, OY, CN) A “design basis” ASME Code Class 1 fatigue evaluation is an existing calculation that was performed by the licensee during the design of the plant in accordance with ASME Code, Section III, Subsection NB, and is a part of an applicant’s current licensing basis. Subsection NB of the ASME Code, Section III, is applicable to Class 1 components, which in this case refers to those components in the reactor coolant pressure boundary. These complex technical calculations were used to demonstrate the suitability of a component for service involving cyclic application of loads and thermal conditions. To assess this suitability, a cumulative usage factor of a Class 1 component was calculated to demonstrate that it was less than the limit of 1.0 as defined in ASME Code, Section III, Subsection NB.

The commitment to consider design basis fatigue evaluations is consistent with Commission policy in this area since these evaluations are the only relevant current licensing basis fatigue evaluations. Specifically, the Commission limited the scope of the license renewal review to the effects of age-related degradation unique to license renewal, stating that the on-going regulatory process provides reasonable assurance that the licensing bases of all currently operating plants provide and maintain an acceptable level of safety for operation during any renewal period. This is discussed by the Commission in the Statement of Considerations for the Final Rule (10 C.F.R. Part 54) in the Federal Register / Vol. 60, No. 88 / Monday, May 8, 1995. (Ex. NRC000117 at 22464). Therefore, based on this Commission policy for the license renewal rule, only those design basis fatigue evaluations that are part of the applicant’s current licensing basis are the subject of this commitment.

Q53. What is "NUREG/CR-6260" and what is its significance?

A53. (AH, OY, CN) NUREG/CR-6260 is a report that assessed the significance of environmental fatigue for a sample of the components in the reactor coolant pressure boundary for the four U.S. nuclear steam supply system vendors. This sample of components is significant because it was chosen to provide a representative overview of components that had higher CUFs and/or were important from a risk perspective. *See* NUREG/CR-6260 at xxi (Ex. NYS000355). The components in this report were meant to provide a representative overview of the effects of reactor water environment for the different components from facilities designed by each of the four U.S. nuclear steam supply system vendors (i.e., Westinghouse, General Electric, Combustion Engineering and Babcock & Wilcox). *See* NUREG/CR-6260 at iii (Ex. NYS000355).

For the purposes of Commitment No. 43 and the Indian Point plant configuration, NUREG/CR-6260 recommends that the following sample of components be evaluated for the effects of reactor water environment of metal fatigue:

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 system Class I piping

See NUREG/CR-6260 at 5-62 (Ex. NYS000355).

Q54. Which components "have been evaluated for the effects of reactor water environment on fatigue usage?"

A54. (AH, OY, CN) The following components have been evaluated for the effects of reactor water environment on metal fatigue for IP2 and IP3, as described In LRA Section 4.3.3:

Vessel shell and lower head

Vessel inlet nozzle

Vessel outlet nozzle

Pressurizer surge line nozzles

Pressurizer surge line piping

Reactor coolant system piping charging system nozzle

Reactor coolant system piping safety injection nozzle

Residual Heat Removal Class 1 piping

See LRA at 4.3-24 through 4.3-25 (Ex. ENT00015A and Ex. ENT00015B).

Q55. What is the purpose of Commitment No. 43?

A55. (AH, OY, CN) The purpose of this commitment is for the Applicant to confirm that the representative sample of components that were selected for an older vintage Westinghouse plant in NUREG/CR-6260 is sufficient for IP2 and IP3. If the Applicant identifies additional locations that should be managed, then it will also manage these additional locations with its Fatigue Monitoring program.

It should be noted that the recommendations in GALL Report, Rev. 1, state that the effects of the coolant environment on metal fatigue are addressed by assessing the impact of the reactor water environment on a sample of critical components for the plant, with an example of the critical components identified in NUREG/CR-6260. It further recommends that the sample is to include the locations identified in NUREG/CR-6260, as a minimum, or alternatives proposed by the Applicant based on its plant configuration. *See* GALL Report Rev. 1 at X M-1 (Ex. NYS00146A-D). This recommendation is clarified in GALL Report, Rev. 2, which states that this sample set should include the locations identified in NUREG/CR-6260 and additional plant-specific component locations in the reactor coolant pressure boundary if they may be more limiting than those considered in NUREG/CR-6260. *See* GALL Report Rev. 2 at X M-1 (Ex. NYS00147A-D).

Based on the Staff's experience related to the identification of locations to consider the effects of reactor water environment on metal fatigue for a different utility, the Staff's opinion is that it is prudent for license renewal applicants to confirm the critical locations that will be managed for the effects of reactor water environment on metal fatigue.

Q56. Why does Commitment No. 43 specifically refer to the use of the NUREG/CR-6909 methodology in the evaluation of the limiting locations consisting of nickel alloy?

A56. (AH, OY, CN) The Staff noted that none of the NUREG/CR-6260 locations for IP2 and IP3 were fabricated from nickel alloy; however, with the completion of Commitment No. 43 it may be possible that nickel alloy components are identified. Since the Staff did not previously identify in its guidance documents how to determine the environmental factor for nickel alloy components, the Staff specifically asked the Applicant to identify the methodology it would use for determining the Fen factor for nickel alloy components, if necessary. *See* RAI Letter at 11 through 13 (Ex. NYS000150). The Applicant specified the use of NUREG/CR-6909 for nickel alloy components in this commitment. *See* NL-11-032, Attachment 1 at 26 and Attachment 2 at 17 (Ex. NRC000110).

The most recent and appropriate methodology for determining the F_{en} factor for nickel alloys is documented in NUREG/CR-6909, which was published in 2007. Based on the equation in NUREG/CR-6909, the environmental factor for nickel alloys is dependent on temperature, strain-rate, and dissolved oxygen. *See* NUREG/CR-6909, Appendix A at A.2 (Ex. NYS000357). Commitment No. 43 specifically refers to the use of the NUREG/CR-6909 methodology in the evaluation of the potential additional locations consisting of nickel alloy to ensure that the most appropriate values for the environmental factor will be used for nickel alloy locations. *See* NL 11-032, Attachment 1 at 26 and Attachment 2 at 17 (Ex. NRC000110).

Q57. What is meant by "limiting locations for the IP2 and IP3 configurations" in Commitment No. 43? And what does Entergy mean by "more limiting locations?"

A57. (AH, OY, CN) The locations that were selected in the NUREG/CR-6260 report were chosen to provide a representative sample of components that had higher CUFs and/or were important from a risk perspective for each nuclear steam supply system (NSSS) vendor; in the case of IP2 and IP3 this is for an older vintage Westinghouse plant. Since the locations evaluated in NUREG/CR-6260 represent a generic evaluation, the commitment will cause Entergy to consider the plant-specific configurations at IP2 and IP3 to ensure that the Entergy evaluation covers the locations that are more susceptible to fatigue when considering environmental effects. Entergy's evaluation in Commitment No. 43 will confirm that the representative sample of components that were selected for an older vintage Westinghouse plant in NUREG/CR-6260 is sufficient for IP2 and IP3. If the Applicant identifies additional locations that should be managed, then it will also manage these additional locations with its Fatigue Monitoring program.

"More limiting locations" refers to those locations that may be more susceptible to fatigue when considering environmental effects than those that have been previously evaluated for IP2 and IP3 consistent with NUREG/CR-6260.

Q58. What does the Staff expect Entergy will find once it has completed this evaluation?

A58. (AH, OY, CN) Once the evaluation has been completed, there are two possible outcomes. The first outcome is Entergy will confirm that the representative sample in NUREG/CR-6260 is adequate for the IP2 and IP3 plant-specific configurations. The second outcome is that Entergy will identify additional locations that will be managed for the effects of the reactor water environment on metal fatigue by the Fatigue Monitoring program during the period of extended operation. Either one of the outcomes will provide additional assurance that the Fatigue Monitoring program is adequate for managing the effects of the reactor water environment metal fatigue.

Q59. Explain how Entergy will determine the additional locations?

A59. (AH, OY, CN) Since Entergy has not yet completed the evaluations, the specific steps or procedure to determine the additional locations is not defined in the LRA. However, Commitment No. 43 provides certainty in the objective of the evaluation and the type of evaluation Entergy will complete.

The scope of Commitment No. 43 is defined as ASME Code Class 1 fatigue evaluations and as previously discussed, based on the Commission policy for the license renewal rule, only those design basis fatigue evaluations that are part of the applicant's current licensing basis are the subject of this commitment. Additionally, in letter dated June 14, 2012, (Ex. NRC000153) (ADAMS Accession No. ML12184A037) ("NL-12-089") Entergy stated, in response to RAIs, that under Commitment No. 43 the review will include fatigue evaluations for reactor vessel internals. *See* NL-12-089, Attachment 1 at 18.

The objective of Commitment No. 43 is to manage the most limiting locations of the plant for environmental effects on fatigue usage (i.e., CUF_{en}). To achieve this goal, Entergy can (1) re-evaluate the entire plant for environmentally-assisted fatigue or alternatively, (2) use a method of binning similar systems and/or components and then determining the bounding location(s) from each bin based on CUF_{en}. However both options are equally effective in meeting the objective of Commitment No. 43. Since any binning and determination of bounding locations will be dependent on CUF_{en}, the factors that contribute/affect this parameter will also need to be considered. These factors should include but are not limited to environmental conditions (e.g., water chemistry), material fabrication, the configuration of the system/component/piping and the associated transients. The methods for calculating the CUF_{en} value are defined and well-documented. Specifically, the methods to calculate CUF are documented in ASME Code, Section III and the methods to calculate F_{en} are documented in several NUREG/CR reports that are endorsed by the GALL Report.

Any additional locations that are determined as a result of Commitment No. 43 will be managed by the Fatigue Monitoring program, which the Staff has concluded is consistent with the GALL Report. In addition, the IP71003 inspection provides the Staff an opportunity to verify the completion of Commitment No. 43. Thus, the Staff has reasonable assurance that Entergy will adequately manage environmentally-assisted fatigue during period of extended operation.

Q60. Has Entergy deferred defining the methods used for determining the most limiting locations for metal fatigue calculations and selection of those locations is related to Commitment No. 43?

A60. (AH, OY, CN) Entergy has deferred stating whether it will use item (1) of re-evaluating the entire plant, item (2) of binning components, or some other alternative, but the method to evaluate the CUF_{en} values is well known as described in A59. However, the Staff's opinion is that this is not important because the completion of this commitment is not necessary to demonstrate that the aging effect of metal fatigue is managed. Entergy is managing metal fatigue and EAF with its Fatigue Monitoring program that (1) tracks actual plant transients, (2) evaluates these actual transients against design transient definitions to ensure the actual severity is not greater than the design severity, and (3) ensures that the number of cycles experienced by the plant remains within the analyzed number of cycles in the fatigue evaluations. These steps ensure that the accumulated fatigue usage, including environmental effects when applicable, will not exceed the Code design limit of 1.0, during the period of extended operation.

Q61. How will the NRC verify that the applicant has completed this commitment in an acceptable manner?

A61. (AH, OY, CN) The completion of this commitment is subject to the inspection performed in accordance with Inspection Procedure 71003, "Post-Approval Site Inspection for License Renewal." *See* IP71003 at 1 (Ex. ENT000251).

In addition, if questions arise about the implementation of license renewal commitments, the Staff can further pursue resolution as part of the ongoing oversight process. Furthermore, the Applicant's analyses, like all other records, will be available at the Applicant's site for the Staff during routine inspections as part of the ongoing oversight process.

Q62. What is Commitment No. 44?

A62. (AH, OY, CN) On March 28, 2011, Entergy submitted Commitment No. 44 in letter NL-11-032. Commitment No. 44 states that IPEC will include written explanation and justification of any user intervention in future evaluations using the WESTEMS "Design CUF" module.

Q63. Did the NRC request relevant information related to Commitment No. 44?

A63. (AH, OY, CN) No, the Staff did not request the information related Commitment No. 44 from Entergy. In letter NL-11-032, the applicant voluntarily provided Commitment No. 44 in Attachment 1 of the letter. The applicant stated on page 1 of the NL-11-032 letter that

> Attachment 1 includes a response to questions asked of other license renewal applicants regarding fatigue analysis software.

See NL-11-032 at 1 (Ex. NRC000110).

Q64. What is WESTEMS?

A64. (AH, OY, CN) WESTEMS is Westinghouse proprietary computer software. The software is used by Westinghouse engineers to perform ASME Code Section III design stress and fatigue analyses. The inputs to this computer program include plant operating data such as temperature and pressure through various systems in the plant. The available data is dependent on the measurements available at a nuclear power plant based on the installed instrumentation. It can also be used to calculate CUF values, which then in turn can be used to determine CUF_{en} values.

Q65. Does the Applicant mention the use of WESTEMS in its LRA?

A65. (AH, OY, CN) The LRA as submitted on April 23, 2007, does not provide information regarding the use of WESTEMS. The Applicant's amendments to the LRA during the course of the Staff's review also did not provide information regarding the use of WESTEMS. The applicant also did not request the review and approval of WESTEMS in the LRA.

Q66. QUESTION NOT USED

Q67. Why did the applicant offer Commitment No. 44?

A67. (AH, OY, CN) Entergy indicated to the Staff that it submitted Commitment No. 44 because it became aware of questions that are being asked to other license renewal applicants regarding WESTEMS. *See* NL-11-032 at 1 (Ex. NRC000110).

Q68. Did the Staff request additional information from other license renewal applicants related to the use of WESTEMS?

A68. (AH, OY, CN) Yes, the Staff requested additional information in November 2010 regarding the use of WESTEMS during the review of the license renewal application for the Salem Nuclear Generating Station.

Q69. Did Entergy respond to the concerns raised by the Staff for this other license renewal applicant?

A69. (AH, OY, CN) Yes, Entergy responded by voluntarily providing Commitment No. 44. *See* NL-11-032 at 1 (Ex. NRC000110).

Q70. Did the NRC Staff make the information and issues related to WESTEMS widely available to the public?

A70. (AH, OY, CN) At a public meeting on March 11, 2011, the Staff made a presentation related to the audit performed on Salem Nuclear Generating Station's use of WESTEMS™ fatigue software during the license renewal process (March 2011) (ADAMS Accession No. ML110760581) (Ex. NRC000119). During this presentation, the Staff discussed the concerns and results of this audit and informed all meeting participants that an audit report will be issued. In that presentation, the Staff also indicated that options were currently being considered on how to generically communicate the concerns and results of this audit. A summary of this public meeting is documented in letter from Evelyn Gettys to Trent Wertz, *Summary Of The Meeting Between The U.S. Nuclear Regulatory Commission Staff And The Nuclear Energy Institute To Discuss Current License Renewal Topics,* (April 11, 2011) (ADAMS Accession No. ML110950443) (Ex. NRC000154) ("Meeting Summary"). It should also be noted that there were several meeting participants with an affiliation to Entergy and Westinghouse. *See* Meeting Summary at Enclosure 1 (Ex. NRC000154).

Following this public meeting, the Staff issued an audit report, *Audit Report on the Use of WESTEMS™ Software in the Salem Nuclear Generating Station, Units 1 and 2, License Renewal Application (TAC NOS. ME1834 and ME1836),* (March 30, 2011) (ADAMS Accession No. ML110871243) (Ex. NRC000155) ("Salem Audit Report"), that discusses the details of the Staff's concerns, the Staff's activities during the audit, the documents reviewed by the Staff, the Staff's questions to Salem Nuclear Generating Station and the Staff's results from the audit. This information was publically available to members of the public and the nuclear industry.

Q71. Did the Staff make this operating experience related to the use of computer software to perform fatigue evaluations publicly available and widely distributed?

A71. (AH, OY, CN) Yes, subsequent to this March 11, 2011, public meeting, the Staff issued NRC Regulatory Issue Summary (RIS)-2011-14, *Metal Fatigue Analysis Performed By Computer Software*, (December 2011) (ADAMS Accession No. ML11143A035) (Ex. NRC000112) ("RIS-2011-14"). A RIS is a type of generic communication that the NRC routinely uses to communicate with the nuclear industry and the public on a broad spectrum of matters having generic applicability. The Staff published a notice of opportunity for public comment on this Regulatory Issue Summary in the *Federal Register* (76 FR60939) on September 30, 2011. The intent of this RIS was to remind addressees of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code) requirements in accordance with 10 CFR 50.55a, "Codes and Standards," and of the quality assurance requirements for design control in accordance with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50.

In addition, the RIS informed addressees of concerns with using computer software packages to demonstrate compliance with the ASME Code Section III and of the NRC's findings from license renewal and new reactor audits on applicants' analyses and methodologies that used the WESTEMS computer software to demonstrate compliance with Section III of the ASME Code. *See* RIS-2011-14 at 1

Q72. Has the Staff identified any conclusions on the operating experience aspect of the Applicant's Fatigue Monitoring program based on how the Applicant has addressed the WESTEMS issues that were identified for another applicant?

A72. (AH, OY, CN) Yes, in the Staff's opinion, Entergy's actions based on the Staff's concerns related to other license renewal applicants suggests that the operating experience aspect of the applicant's Fatigue Monitoring program is effective. One purpose of the "operating experience" program element of an aging management program is to ensure that plant-specific and industry operating experience is evaluated to ascertain the need to make enhancements to aging management programs or to develop new programs. *See* SRP-LR Rev. 2 at A.1-7 (Ex. NYS000161). The applicant identified the issue as applicable to its plant even before the publication of the RIS and voluntarily provided Commitment No. 44 addressing the Staff's concern.

Q73. In the Staff's opinion, did Commitment No. 44 provide any missing information in the application?

A73. (AH, OY, CN) No, Commitment No. 44 does not provide any information that is missing in the application. Commitment No. 44 is not used to demonstrate the adequacy of the Applicant's Fatigue Monitoring program.

Q74. What is the purpose of Commitment No. 44?

A74. (AH, OY, CN) Commitment No. 44 is related to how fatigue analysts document the use of engineering judgment and user intervention when conducting future fatigue analysis with the WESTEMS™ code. Entergy committed to document future use of the WESTEMS™ code, which the Staff noted is in accordance with Appendix B to 10 CFR Part 50.

As described in NRC RIS-2011-14, the Staff's review of WESTEMS™ for another license renewal applicant (but the same vendor) did not identify issues with the engineering judgment and user intervention exercised for that particular applicant's fatigue evaluations; the Staff review only identified concerns with the documentation of the user intervention. Thus the Staff did not question the accuracy or validity of the fatigue evaluations for that applicant. *See* RIS-2011-14 at 3 (Ex. NRC000112).

To clarify, Entergy does not rely on Commitment No. 44 to demonstrate that its Fatigue Monitoring program is capable of managing metal fatigue and environmentally-assisted fatigue and the Staff did not rely on this commitment in finding the Applicant's program acceptable.

Q75. What will Entergy do to implement Commitment No. 44?

A75. (AH, OY, CN) The Staff expects the applicant to revise any applicable implementation procedures, if necessary, indicating that engineering judgment and user intervention exercised when using the WESTEMS software will be documented.

Q76. What is the implementation schedule for Commitment No. 44?

A76. (AH, OY, CN) The implementation date, as proposed by the applicant in Letter NL-11-032 (March 28, 2011) (Ex. NRC000110) and amended in Letter NL-11-101, *Indian Point, Units 2 & 3 - Clarification for Request for Additional Information (RAI) Aging Management Programs.* (August 22, 2011) (ADAMS Accession No. ML11243A085) (Ex. NRC000156) ("NL-11-101"), is prior to the entrance of the period of extended operation for each unit. Thus, the Staff expects that the administrative changes to the Entergy's procedures will be implemented prior to the period of extended operation. *See* NL-11-101, Attachments 1 and 2 at 2 and 18, respectively.

Q77. What is the Staff's rationale for finding this implementation schedule reasonable?

A77. (AH, OY, CN) The implementation schedule of Commitment No. 44 provides Entergy sufficient time and a means of tracking until completion of needed actions, such as updating or revising its implementing procedures. Since Entergy is currently required by Appendix B to

10 CFR Part 50 to implement a Quality Assurance program that includes measures for design control and maintaining quality assurance records, the implementation or completion of the actions identified in Commitment No. 44 does not obviate Entergy's obligations to meet the requirements for a Quality Assurance program. Therefore, regardless of the applicant's implementation schedule for this commitment, all licensees must meet the requirements in Appendix B to 10 CFR Part 50 and the oversight process routinely ensures licensees are in compliance.

Q78. Entergy finished WCAP-17199-P and WCAP-17200-P and submitted these reports to the NRC. Does the Staff have any concern for these WCAP reports regarding the documentation of engineering judgment and user intervention?

A78. (AH, OY, CN) The Staff reviewed the WCAP reports only for the purposes of preparing testimony for Contentions NYS-26B/RK-TC-1B and NYS-38/RK-TC-5. The Staff did not specifically review these WCAP reports for its documentation of engineering judgment and user intervention. Entergy is currently required by Appendix B to 10 CFR Part 50 to implement a Quality Assurance program, which provides the means for ensuring that these analyses and calculations are sufficiently detailed such that a person technically qualified in the subject area can review and understand the analyses and verify the adequacy of the results without recourse to the originator.

Q79. Why weren't WCAP-17199-P and WCAP-17200-P reviewed by the Staff?

A79. (AH, OY, CN) The Staff's safety evaluation report in November 2009 was issued prior to the applicant's submittal of the CUF_{en} values resulting from the reanalyses in Letter NL-10-082, *Letter from Entergy Nuclear Operations, Inc. to NRC, Notification of Entergy's Submittal Regarding Completion of Commitment 33 for Indian Point Units 2 and 3*, (August 9, 2010)

(ADAMS Accession No. ML11356A335) (Ex. NYS000352) ("NL-10-082"). The reanalyses in WCAP-17199-P and WCAP-17200-P were not submitted and reviewed by the Staff as part of the LRA review process to ascertain whether the Fatigue Monitoring program is acceptable. WCAP-17199-P and WCAP-17200-P were submitted as part of the hearing process and not the LRA review process.

Entergy has proposed to manage the effects of metal fatigue and environmentally-assisted fatigue during the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii) by using its Fatigue Monitoring program. Entergy has not claimed that its reanalyses are valid during the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i) nor has it claimed that the reanalyses are projected to be valid during the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii). The Staff performed its review of Entergy's evaluation of metal fatigue and environmentally-assisted fatigue in accordance with 10 CFR 54.21(c)(1)(iii) and consistent with the review procedures in Section 4.3 of SRP-LR.

Q80. Has Entergy specified the criteria it will use and assumptions upon which it will rely for modifying the WESTEMS computer model for environmentally-adjusted cumulative usage factors (CUF_{en}) calculations?

A80. (AH, OY, CN) No, Entergy has not specified the criteria and assumptions that it will be relying on for modifying the WESTEMS computer model for its CUF_{en} calculations. However, the Staff's opinion is that this is not relevant because the completion of this commitment is not necessary for Entergy to demonstrate the aging effect of metal fatigue is managed. The requirements for documentation in any calculation are governed by the "design control" and "quality assurance records" elements of a Quality Assurance program that is mandated by Appendix B to 10 CFR Part 50. In other words, Entergy is currently required by Appendix B to

10 CFR Part 50 to implement a Quality Assurance program to ensure that any modification should be documented such that design analyses and calculations are sufficiently detailed that a person technically qualified in the subject area can review and understand the analyses and verify the adequacy of the results without recourse to the originator.

Q81. Will the NRC inspect the results of this commitment?

A81. (AH, OY, CN) The completion of this commitment is subject to inspection in accordance with IP71003. In addition, all applicant records and evaluations will be available at the Applicant's site for the Staff during routine inspections as part of the ongoing oversight process.

Q82. Does Entergy need to provide the results of Commitment No. 44 to demonstrate that the aging effects for metal fatigue and EAF will be managed?

A82. (AH, OY, CN) No, the completion of this commitment is not necessary to demonstrate that the aging effects of metal fatigue will be managed. Entergy is managing metal fatigue and EAF with its Fatigue Monitoring program that (1) tracks actual plant transients, (2) evaluates these actual transients against design transient definitions to ensure the actual severity is not greater than the design severity, and (3) ensures that the number of cycles experienced by the plant remain within the analyzed number of cycles in the fatigue evaluations. This ensures that the accumulated fatigue usage, including environmental effects when applicable, will not exceed the Code design limit of 1.0, during the current operating license term and during the period of extended operation.

Rebuttal to Dr. Lahey – June 18, 2012 –

Q83. Have you read the pre-filed written declaration of Dr. Richard T. Lahey, Jr. (Exhibit NYS000374) (ADAMS Accession No. ML12171A513) ("Lahey June") dated June 18, 2012?

A83. (AH, OY, CN) Yes, the Staff has read the declaration of Dr. Richard T. Lahey, Jr. dated June 18, 2012.

Q84. Has Dr. Lahey identified any errors in Entergy's license renewal application?

A84. (AH, OY, CN) No.

Q85. Has he identified any omissions of required information in Entergy's license renewal application?

A85. (AH, OY, CN) No.

Q86. If he has not identified errors or omissions of required information, what concerns are identified in Dr. Lahey's pre-filed testimony as it relates to metal fatigue and Contention NYS-38/RK-TC-5?

A86. (AH, OY, CN) Dr. Lahey has identified two concerns as it relates to metal fatigue and Contention NYS-38/RK-TC-5.

His first concern is Entergy has not disclosed the parameters surrounding user intervention in the previous runs of WESTEMS that provided the basis for what has been described as the refined metal fatigue analysis that was previously submitted in this proceeding. Dr. Lahey further contends that the absence of this information impedes and prevents a meaningful analysis of the metal fatigue analysis that Entergy has presented and the aging management program that Entergy has proposed. *See* Lahey June at 12 (Ex. NYS000374).

His second concern is that Entergy has not identified the additional limiting locations within the reactor coolant pressure boundary that are subject to fatigue, and the absence of this information also impedes and prevents a meaningful analysis of the metal fatigue analysis that Entergy has presented and the aging management program that Entergy has proposed. *See* Lahey June at 12 (Ex. NYS000374).

Q87. How does Dr. Lahey describe "user intervention" in his testimony?

A87. (AH, OY, CN) Dr. Lahey states that "[t]he term 'user intervention' refers to, among other things, the use of assumptions and engineering judgment in the process of calculating the CUFen values using codes such as WESTEMS." *See* Lahey June at 24 (Ex. NYS000374).

Q88. Is Dr. Lahey's characterization of "user intervention" consistent with the concerns identified with the use of WESTEMS?

A88. (AH, OY, CN) No, Dr. Lahey's characterization of "user intervention" in not consistent with the concerns identified with the use of WESTEMS. Dr. Lahey's characterization of "user intervention" is vague and, as stated, would call into question all assumptions and engineering judgment that is inherent in any fatigue analysis regardless if it is performed with or without computer software. Based on his general description of user intervention, it appears that Dr. Lahey misunderstands the term "user intervention" as it relates to WESTEMS and the Staff's concern, which is documented in RIS-2011-14.

Q89. Can you clarify what is meant by "user intervention" as it relates to WESTEMS and the Staff's concerns?

A89. (AH, OY, CN) As described in the RIS-2011-14,

> The staff also identified a concern in which, under certain circumstances, the use of this computer software package allows the user to manually modify stress peak and valley times in the total stress intensity time history used to calculate the cumulative usage factor during intermediate calculations.

See RIS-2011-14 at 2

This description of user intervention is clearly related to very specific steps and actions that a properly trained analyst performs during the calculation. This is in contrast to the vague description of "the use of assumptions and engineering judgment in the process" stated by Dr. Lahey. Furthermore, Dr. Lahey never specifies the effects that the "user intervention," as described in the RIS affects, has, if any, on the results from these fatigue calculations. The Staff's concern with "user intervention" as described in RIS 2011-14 was associated only with sufficient documentation of modifications of stress peaks and valleys by properly trained analysts using the WESTEMS software, and not with the engineering judgment exercised by the analyst or the results of the analyses.

Q90. What was the purpose of the Staff issuing RIS 2011-14?

A90. (AH, OY, CN) The purpose of this RIS was for the Staff to inform the nuclear industry and the public regarding specific potential concerns with using computer software packages, which may have generic applicability. The RIS encouraged addressees to review the documents discussed in the RIS and to consider actions, as appropriate, to ensure compliance with the requirements for ASME Code fatigue calculations and a quality assurance program, as described in 10 CFR 50.55a and Appendix B to 10 CFR Part 50, respectively.

Q91. Regarding WESTEMS, Dr. Lahey states [page 25, line 4] that certain assumptions could materially affect the results, thus it is necessary to have disclosed in advance the assumptions to be used in the analysis in order to ascertain whether the AMP is adequate. What is the Staff's opinion on this?

A91. (AH, OY, CN) The Staff does not agree with Dr. Lahey's statement made in his pre-filed testimony on June 18, 2012, in as far as it relates to user intervention identified in RIS-2011-14 and Commitment No. 44. As described in the RIS-2011-14, the Staff's concern is that this selection and modification of stress peaks and valleys should be documented such that design analyses and calculations are sufficiently described that a person technically qualified in the subject area can review, understand the analyses, and verify the adequacy of the results without recourse to consulting the originator.

The issue of documenting user intervention does not correlate to a conclusion that Entergy has failed to demonstrate that the aging effects of metal fatigue will be adequately managed because Entergy's documentation for any analysis, not just fatigue analyses, must be performed in accordance with a Quality Assurance program that is currently required to be implemented in accordance with Appendix B to 10 CFR Part 50. The Intervenors have not questioned or identified concerns regarding Entergy's current obligations to implement a Quality Assurance program in accordance with Appendix B to 10 CFR 50.

Further, the assumptions to be used in future analyses (that may be needed by the Applicant as a part of corrective actions) do not need to be, and in reality cannot be, disclosed in advance of a licensing decision in order to ascertain whether the AMP is adequate. The Staff's standards for determining whether an aging management program is adequate are provided in the GALL Report and the SRP-LR. The information provided by Entergy through the LRA process has demonstrated that its Fatigue Monitoring program is capable and sufficient to manage metal fatigue and environmentally-assisted fatigue. A description of how Entergy's Fatigue Monitoring program functions was provided in the Staff's testimony for NYS-26B/RK-TC-1B. *See* NRC000102 at 23 through 27.

Dr. Lahey's statement is very general in that it is obvious that the assumptions of the sort that he mentions can affect the results from a fatigue analysis. This is precisely the reason that an individual performing fatigue analyses must have specialized experience and be specifically trained. Entergy is currently required by Appendix B to 10 CFR Part 50 to implement a Quality Assurance program that takes into account the need for special controls, processes, and skills to attain the required quality, and the need for verification of quality when performing an analyses or calculation. In addition, Entergy is also currently required by Appendix B to 10 CFR Part 50 to implement a Quality Assurance program that provides for training of personnel performing activities affecting quality as necessary to assure that suitable proficiency is achieved and maintained.

Entergy is required to perform its activity in accordance with a Quality Assurance Program implemented in accordance with the current requirements of Appendix B to 10 CFR Part 50, which means that the EAF analyses documented in WCAP-17999-P and WCAP-17200-P, the EAF analyses that may be performed in the future and the evaluation that will be completed as part of Commitment No. 43 are governed by this aforementioned Quality Assurance program.

Entergy is currently required by Appendix B to 10 CFR Part 50 to implement a Quality Assurance program that ensures that, for the aforementioned analyses and evaluations, there are sufficient records and these records are maintained to document activities affecting quality. Furthermore, this Quality Assurance program required by Appendix B to 10 CFR Part 50 will provide measures for verifying or checking the adequacy of design, such as by the performance of design reviews. In addition, design analyses and calculations are to be sufficiently detailed such that a person technically qualified in the subject area can review and understand the analyses and verify the adequacy of the results without recourse to consulting the originator.

It appears that Dr. Lahey is concerned with the adequacy of Entergy's current requirements to implement a Quality Assurance program in accordance with Appendix B to 10 CFR Part 50, which is not the subject of this contention.

Q92. Dr. Lahey states on Page 26 of in his pre-filed testimony dated June 18, 2012 that Entergy has not disclosed the specific criteria it will use in deciding whether to make a user intervention and what standards will control the extent of these interventions. What is the Staff's opinion?

A92. (AH, OY, CN) Entergy is currently required by Appendix B to 10 CFR Part 50 to implement a Quality Assurance program that includes measures to provide for verifying or checking the adequacy of design, such as by the performance of design reviews. In addition, this Quality Assurance program that is required by Appendix B to 10 CFR Part 50 ensures that design analyses and calculations are to be sufficiently detailed such that a person technically qualified in the subject area can review and understand the analyses and verify the adequacy of the results without recourse to the originator. The involvement and judgment of the analyst is inherent when performing any calculation, not just fatigue calculations; thus these ASME Code fatigue evaluations must be completed by a specialized and trained analyst that understands the rules defined in ASME Code Section III for performing fatigue analyses.

In the Staff's opinion Dr. Lahey's concern with Entergy's disclosure of the specific criteria is associated with the adequacy of Entergy's current requirements to implement a Quality Assurance program in accordance with Appendix B to 10 CFR Part 50; however, this program is not the subject of this contention.

Q93. What does Dr. Lahey state in his pre-filed testimony dated June 18, 2012 regarding an error analysis? What is the Staff's opinion on this?

A93. (AH, OY, CN) Dr. Lahey believes that an error analysis must be done. *See* Lahey June at 27 (Ex. NYS000374). However, Dr. Lahey does not provide any basis either from the Commission's regulations or the ASME Code for his conclusion. The Staff noted that neither the Commission's regulations nor the ASME Code require an error analysis, or even hint at any situation or conditions for which an error analysis might be necessary. Thus the Staff disagrees with Dr. Lahey's conclusion.

The conservatism that is inherent in the fatigue calculation methodology dictated by the ASME Code is sufficient to account for the propagation of errors the Intervenors state is needed in calculating the cumulative usage factor. The conservatism in the analyses, which comes from multiple sources, makes an error analysis unnecessary. Two examples are as follows: Firstly, the transient severities for cycles that occur at the plant are typically not as severe as the severity of design transients used in the calculations. Although this is typical, Entergy's Fatigue Monitoring program (1) tracks actual plant transients, and (2) evaluates these actual transients against design transient definitions to ensure the actual severity is not greater than the design severity. Secondly, the Fatigue Monitoring program for both IP2 and IP3, as described by its implementing procedures, also provides for corrective actions when a single transient type (e.g., heat-up transient or cool-down transient) approaches its respective cycle or action limit, even if the remaining transients included in the analysis to determine CUF are below their respective limits in the analysis. *See* NL-07-153, Attachment 3 at 7 though 8 (Ex. NRC000111). Thus, the Staff believes an error analysis is not needed.

Q94. Let's move on to Dr. Lahey's second concern related to identification of limiting locations for fatigue analyses. Does Dr. Lahey further describe this concern?

A94. (AH, OY, CN) Yes, Dr. Lahey's description of his second concern is Entergy has agreed to reanalyze the locations it has previously identified in its LRA for environmentally-assisted fatigue and to determine if more limiting locations exist at other components.

However, he states that the exact time for reporting the results of future review and analysis, if additional locations are identified and detailed further analysis is required, was not specified other than it will be shortly before the period of extended operation. He further states, that this timing of reporting the results of the of the future analysis, just prior to the period of extended operation, will prevent those matters from being tested and resolved in these ASLB hearings and greatly handicaps, if not precludes, the State of New York from any meaningful role in their development and resolution.

In addition, Dr. Lahey states that this review will focus on structures, components and fittings outside the RPV and will thus not include a comprehensive consideration of the fatigue of important RPV internal structures, components and fittings.

Q95. What is the Staff's opinion related to Dr. Lahey's concerns?

A95. (AH, OY, CN) The Staff's opinion is that the completion of Commitment No. 43 is not needed prior to a licensing decision. Per the requirements in 10 CFR 54.21(a)(3), Entergy must demonstrate that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the CLB for the period of extended operation. For the case of metal fatigue and environmentally-assisted fatigue, Entergy will use its Fatigue Monitoring program to provide adequate management. The Fatigue Monitoring program described in the application, as amended, is an existing program that was modified to meet all the program elements defined in the GALL Report AMP X.M1. Prior to the application of license renewal, Entergy already had an existing program and that program was improved by addressing the effects of environmentally-assisted fatigue for the purpose of license renewal. Thus, Entergy has provided commitments to augment its existing program prior to the period of extended operation in order to meet the GALL Report AMP X.M1 recommendations for environmentally-assisted fatigue and identifying additional locations. Therefore, consistent with the Staff's review guidance, Entergy is not required to identify the additional locations that may be more limiting *now*, before a licensing decision is made. In addition, Entergy has satisfied the requirements in 10 CFR 54.21(a)(3) by demonstrating that its Fatigue Monitoring program is capable and adequate to manage metal fatigue and environmentally-assisted fatigue.

However, the Staff notes that, Entergy informed the Board, in a letter dated May 15, 2012, that it has determined that the initial screening review of design basis ASME Code Class 1 fatigue evaluations, as described in Commitment No. 43, to determine whether the NUREG/CR-6260 locations are the limiting locations for IPEC, is expected to be completed within approximately the next four months from the May 2012 letter date. *See* Entergy May 2012 Letter at 1. Therefore, Entergy has stated that the review will be done before the period of extended operation and before the license renewal decision, as Dr. Lahey (and Dr. Hopenfeld) have stated is necessary.

As for Dr. Lahey's concern that this review will not include a comprehensive consideration of the fatigue of important RPV internal structures, components and fittings; this concern is no longer relevant based on Entergy's NL-12-089 letter dated June 14, 2012. In this letter, Entergy responded to the requests for additional information related to LRA Amendment No. 9 and the Reactor Vessel Internals Program. As a part of the response Entergy stated that, consistent with Section 3.5.1 of the safety evaluation for MRP-227-A, the existing RVI fatigue calculations will be reviewed to evaluate the effects of the reactor coolant system water environment on the CUF prior to entering the period of extended operation. Specifically, under Commitment No. 43, Entergy will review the IPEC design basis ASME Code Class 1 fatigue evaluations to determine whether the NUREG/CR-6260 locations that have been evaluated for the effects of the reactor coolant environment on fatigue usage are the limiting locations for the IP2 and IP3 configurations. This review includes ASME Code Class 1 fatigue evaluations for reactor vessel internals. If more limiting locations are identified, the most limiting location will be evaluated for the effects of the reactor coolant environment on fatigue usage. *See* NL-12-089, Attachment 1 at 18.

Thus, based on this letter from Entergy, Dr. Lahey's concern that the review will focus on the structures, components and fittings outside the RPV and will thus not include a comprehensive consideration of the fatigue of important RPV internal structures, components and fittings is no longer pertinent because Entergy will consider the fatigue evaluations for reactor vessel internals as part of Commitment No. 43.

<u>Rebuttal to Dr. Hopenfeld – June 19, 2012 –</u>

Q96. Have you read the pre-filed written Testimony of Dr. Joram Hopenfeld (Exhibit RIV000102 (ADAMS Accession No. ML12171A559) ("Hopenfeld June") dated June 19, 2012?

A96. Yes, the Staff has read the Testimony of Dr. Joram Hopenfeld, Jr. dated June 19, 2012.

Q97. What is your opinion on Dr. Hopenfeld's view that Entergy employed a flawed methodology in Entergy's "refined" analyses in August 2010?

A97. (AH, OY, CN) Dr. Hopenfeld addressed the question regarding the validity of the methodology used in Entergy's refined analyses on page 8 of his pre-filed testimony dated June 19, 2012. This claim was previously addressed by all the parties as part of the written testimony related to NYS-26B/RK-TC-1B. As described in the testimony, the Staff does not agree with Dr. Hopenfeld's statement that "Entergy employed a flawed methodology." *See* NRC000102 at 56 & 81.

Q98. What is your opinion on Dr. Hopenfeld's view that Entergy's methodology used for refined analyses failed to account for all relevant plant parameters?

A98. (AH, OY, CN) Dr. Hopenfeld addressed the question regarding "whether all relevant plant parameters are accounted for" in the methodology used in Entergy's refined analyses on page 8 of his pre-filed testimony dated June 19, 2012. The Staff does not agree with Dr. Hopenfeld's opinion that Entergy failed to account for all relevant plant parameters." *See* Hopenfeld June at 8 (Ex. RIV000102).

This claim was previously addressed by all the parties as part of the written testimony related to

NYS-26B/RK-TC-1B. As described in the testimony, the Staff does not agree with Dr. Hopenfeld's statement that "Entergy employed a flawed methodology." *See* NRC000102 at 57-59.

Q99. What is your opinion on Dr. Hopenfeld's view that Entergy is required to identify and investigate additional reactor locations?

A99. (AH, OY, CN) Dr. Hopenfeld stated that:

> "According to regulatory and industry guidance, since the CUF_{en} for various components were initially found to exceed the regulatory threshold of 1.0, as presented in original LRA Tables 4.3-13 and 4.3-14, Entergy is required to identify and investigate additional reactor locations for potential high susceptibility to metal fatigue.

See Hopenfeld June at 8 (Ex. RIV000102).

The Staff does not agree with Dr. Hopenfeld that Entergy needs to identify and investigate additional locations *because* the CUF_{en} for various components were initially found to exceed the regulatory threshold of 1.0. The Staff guidance specifies that the sample set of locations "should include the locations identified in NUREG/CR-6260 and additional plant-specific component locations in the reactor coolant pressure boundary…" The Staff's guidance does not require Entergy to identify and investigate additional reactor locations for potential high susceptibility to metal fatigue when CUF_{en} values are initially found to exceed 1.0; the recommendation to investigate beyond the locations identified in NUREG/CR-6260 is already incorporated into GALL Report AMP X.M1 and Entergy's Commitment No. 43.

Q100. Dr. Hopenfeld refers to MRP-47, Rev. 1 in his pre-filed testimony. *See* Hopenfeld June at 8 (Ex. RIV000102). What is MRP-47, Rev. 1?

A100. (AH, OY, CN) MRP-47 Rev. 1 "Materials Reliability Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application" is an industry guidance document published in 2005 by Electric Power Research Institute. Among other information, MRP-47, Rev.1 provides a summary of the NUREG reports related to environmentally-assisted fatigue. It is important to note that neither MRP-47 Rev. 0 nor Rev.1 have been reviewed and approved by the Staff, and neither report is required for use during the current operating license period or during the period of extended operation.

Q101. Is Dr. Hopenfeld correct that Entergy is required to identify additional locations based on the guidance provided in MRP-47, Rev. 1?

A101. (AH, OY, CN) Dr. Hopenfeld mischaracterized MRP-47, Rev. 1, as a requirement; Entergy also has not indicated in the LRA or provided a formal commitment in the LRA that it will follow the guidance in MRP-47, Rev. 1. Therefore, since Entergy has not indicated in the LRA or provided a formal commitment in the LRA that it has used or will be using the guidance in MRP-47, Rev. 1, and this document is not a part of the Commission's regulation, Entergy is not required to follow MRP-47, Rev. 1.

It should be noted that Entergy's testimony for NYS-26B/RK-TC-1B indicates that Westinghouse did apply the F_{en} factors consistent with industry recommendations in MRP-47, but that is the only indication of use of MRP-47 by Entergy. See ENT0000183 at 109. But as we have stated, the use of this document is not required because it is not in the Commission's regulations and Entergy has not provided a formal commitment in the LRA to use MRP-47. The use of MRP-47 is also not a recommendation in the staff's guidance documents for license renewal (i.e., GALL Report Rev. 1 and 2 and SRP-LR Rev. 1 and 2).

Q102. Do you agree with Dr. Hopenfeld's view that the components analyzed for fatigue will likely exceed unity?

A102. (AH, OY, CN) The Staff does not agree with Dr. Hopenfeld's opinion that Entergy's fatigue analyses to date demonstrate that the components analyzed will likely exceed unity. *See* Hopenfeld June at 9 (Ex. RIV000102).

The Staff noted that the CUF_{en} values reported by Entergy in revised LRA Table 4.3-13 and 4.3-14 in Letter NL-10-082 are less than 1.0 (See NL-10-082, Attachment 1 at 3-4) (Ex. NYS000352). Dr. Hopenfeld has not provided a detailed justification for his speculation that the components analyzed by Entergy will likely exceed unity.

WCAP Report 17199-P and 17200-P document the methodology Entergy used to calculate the F_{en} for the applicable material types, which was previously addressed as part of the Staff's written testimony related to NYS-26B/RK-TC-1B. *See* NRC000102 at 74.

Q103. What is your opinion on Dr. Hopenfeld's view that it is not appropriate for NRC Staff to accept Entergy's vague commitment?

A103. (AH, OY, CN) Dr. Hopenfeld cited the guidance related to identifying additional plant-specific component in GALL Report X.M1 and stated that "therefore, it was not appropriate for NRC Staff to accept Entergy's vague commitment to determine at some point in the future what additional locations must be analyzed" *See* Hopenfeld June at 11 (Ex. RIV000102).

Dr. Hopenfeld does not appear to understand how an applicant's aging management program is considered consistent with the GALL Report. The Fatigue Monitoring program described in the LRA, as amended, is an existing program that will be augmented to meet all the program elements defined in the GALL Report AMP X.M1. Prior to the application for license renewal, Entergy had an established existing program and that program will be improved by addressing the effects of environmentally-assisted fatigue for the purposes of license renewal. Therefore, Entergy has provided commitments to augment its existing program prior to the period of extended operation in order to be consistent with the GALL Report AMP X.M1 recommendations for environmentally-assisted fatigue and identifying additional locations. Thus, it is appropriate for the Staff to accept Entergy's commitments to augment its existing Fatigue Monitoring program. The practice of augmenting an existing program and providing commitments to do so is consistent with the Staff's guidance documented in the SRP-LR Rev.1 and Rev.2. *See* SRP-LR Rev. 1 and Rev. 2 at 3.0-3 (Ex. NYS000195 and Ex. NYS000161, respectively).

Q104. Do you agree with Dr. Hopenfeld's view that Commitment No. 43 or an actual analysis to determine the most limiting locations must be completed before a determination can be made about license renewal?

A104. (AH, OY, CN) No, the Staff does not agree with Dr. Hopenfeld's view. *See* Hopenfeld June at 11 (Ex. RIV000102). The results related to Commitment No. 43 need not be provided before a licensing decision is reached. However, Commitment No. 43 must be completed prior to entering the period of the extended operation, which is consistent with the implementation schedule of this commitment. *See* NL-11-032, Attachment 1 at 26 and Attachment 2 at 17 (Ex. NRC000110). Thus the license renewal application is complete and there is no missing information. Entergy has demonstrated that its Fatigue Monitoring program is capable and sufficient to manage metal fatigue and environmentally-assisted fatigue during the period of extended operation.

In any event, Entergy informed the Board, in a letter dated May 15, 2012, that it has determined that the initial screening review of design basis ASME Code Class 1 fatigue evaluations, as described in Commitment No. 43, to determine whether the NUREG/CR-6260 locations are the limiting locations for IPEC, is expected to be completed within approximately the next four months from the May 2012 letter date. *See* Entergy May 2012 Letter at 1. In other words, Entergy has indicated that the review will be complete before the period of extended operation and before the license renewal decision, as both Dr. Lahey and Dr. Hopenfeld have stated is necessary.

Q105. Do you agree with Dr. Hopenfeld's view that Entergy has simply failed to provide sufficient information in order to assess whether Entergy's AMP for metal fatigue is adequate?

A105. (AH, OY, CN) No, the Staff does not agree with Dr. Hopenfeld's view that Entergy has failed to provide sufficient information in order to assess whether Entergy's AMP for metal fatigue is adequate. *See* Hopenfeld June at 12 (Ex. RIV000102). The Fatigue Monitoring program described in the LRA, as amended, is an existing program that was modified to meet all the program elements defined in the GALL Report AMP X.M1. Entergy augmented its existing program by addressing the effects of environmentally-assisted fatigue for the purpose of license renewal prior to the period of extended operation in order to meet the GALL Report AMP X.M1 recommendations for environmentally-assisted fatigue and identifying additional locations.

Information provided by Entergy has demonstrated that its program is adequate to manage the aging effect of metal fatigue and EAF because the Fatigue Monitoring program (1) tracks actual plant transients, (2) evaluates these actual transients against design transient definitions to ensure the actual severity is not greater than the design severity, and (3) ensures that the number of cycles experienced by the plant remain within the analyzed number of cycles in the fatigue evaluations. It is the Staff's opinion that no additional information related to Entergy's Fatigue Monitoring program is needed to demonstrate that metal fatigue will be managed during the period of extended operation.

Q106. What is your opinion on Dr. Hopenfeld's view regarding assessment of actual experience at Indian Point as well as at other pressurized water reactor plants?

A106. (AH, OY, CN) Dr. Hopenfeld stated that "a determination of the most limiting locations should also include an assessment of actual experience at Indian Point as well as at other PWR plants." *See* Hopenfeld June at 12 (Ex. RIV000102).

It is the Staff's opinion that Commitment No. 43 addresses the exact point that Dr. Hopenfeld is demanding. Commitment No. 43 states that Entergy will include in its evaluation all Class 1 fatigue analyses. Class 1 fatigue analyses includes the analyses required during the original design of the plant as well as any fatigue analyses performed throughout the current licensed operation, which may be additional analyses performed during a power uprate license amendment or due to industry experience at other pressurized water reactor plants. Commitment No. 43 does not restrict Entergy's evaluation only to Class 1 fatigue analyses from the original design.

Q107. What is your opinion on Dr. Hopenfeld's view that thermal striping during stratification should be generally considered?

A107. (AH, OY, CN) Dr. Hopenfeld stated that "thermal striping during stratification should be generally considered as these affect fatigue life, and since the GALL Report requires that environmental effects be included in the calculations and does not exclude thermal striping from such requirements." *See* Hopenfeld June at 12 (Ex. RIV000102).

It is the Staff's opinion that Dr. Hopenfeld's statement is irrelevant because the aging management program for fatigue monitoring described in the GALL Report recommends monitoring all plant design transients that are important for the design consideration. Additionally, Dr. Hopenfeld has not described any systems or components at IP2 or IP3 in which this thermal striping transient would be a concern and is important for the design consideration. Dr. Hopenfeld has only made a general statement that thermal striping should be considered without providing supporting information or justification as to how it applies to IP2 and IP3.

Q108. What is your opinion on Dr. Hopenfeld's view that CUF values of the steam-generator divider plate may exceed unity when the effects of PWSCC and the environment are included?

A108. (AH, OY, CN) Dr. Hopenfeld stated that "[t]hese CUFs may exceed unity when they are corrected for the effects of PWSCC and the environment." *See* Hopenfeld June at 13 (Ex. RIV000102). Because Dr. Hopenfeld does not provide any data, research, or operating experience that would justify his assertions, the Staff believes that his statement is not well founded and the effects of PWSCC does not need to be considered by environmentally-assisted fatigue.

Primary water stress corrosion cracking is a cracking mechanism of certain microstructure due to high stress and high temperature. *See* GALL Report Rev. 2 at pg IX-36. When calculating CUF and F_{en} values, which are related to fatigue crack initiation, stress and temperature are part of the input. Further, the data that were used to develop the F_{en} values were developed in water environments that would also be conducive to PWSCC; thus the F_{en} values should incorporate environmental effects, such as those from PWSCC that could possibly affect the crack initiation behavior embodied in the F_{en} values. Thus, the environmental-assisted fatigue calculation includes any effects of PWSCC on the subject components and Dr. Hopenfeld is incorrect that the effect of PWSCC needs to be considered in addition to the environmental effects of reactor water.

Q109. Dr. Hopenfeld provided a table of locations that Entergy must consider at a minimum to determine the more limiting locations. What is your opinion of these locations that Dr. Hopenfeld provided?

A109. (AH, OY, CN) Regarding the question of what components Entergy should evaluate to determine whether they may be more limiting, Dr. Hopenfeld provided a table of sample locations that he thinks Entergy must consider at a minimum. *See* Hopenfeld June at 14 and 15 (Ex. RIV000102).

Although the Staff notes that these are some of the components at IP2 and IP3 with calculated CUF values, the Staff does not agree that these components “must” be considered as a minimum.

There are several locations (e.g., reactor pump outlet nozzle, RHR SI nozzle, mixing tees of RHR system, piping of the pressurizer spray line and piping of unisolable branched connected to RCS piping) indicated as "n/a." The Staff noted that this may mean that that there is no CUF value for these components identified in Entergy's current licensing basis for IP2 and IP3. The Staff disagrees that these locations should be included in the evaluation for identifying additional limiting locations for environmentally-assisted fatigue because Commitment No. 43 explicitly states that Entergy shall consider all of its current licensing basis Class 1 fatigue analyses. Commitment No. 43 is not intended to require an analysis of fatigue (i.e., CUF) for components that were not already analyzed for fatigue as part of Entergy's current licensing basis.

If a CUF value does not exist for a particular component in Entergy's current licensing basis for IP2 and IP3, this would indicate that fatigue was not deemed to be an issue as part of the design specification and the original design of the component. Further Dr. Hopenfeld does not identify any data, research, or operating experience that would justify his assertions that all of these locations "must" be considered.

As previously mentioned, per the requirements in 10 CFR 54.21(a)(3), Entergy must demonstrate that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the CLB for the period of extended operation. Since for these components fatigue crack initiation (i.e., CUF value) was not analyzed and is not a part of Entergy's CLB, Entergy is not required to calculate CUF values for these components or include them as part of the evaluation for Commitment No. 43.

Q110. Let's continue with Dr. Hopenfeld's concern related to Commitment No. 44, WESTEMS and user intervention. Does Dr. Hopenfeld provide his description of "user intervention"?

A110. (AH, OY, CN) No, Dr. Hopenfeld does not define what user intervention is but uses this term when describing his concern. Dr. Hopenfeld stated his concern, in part, as "Entergy must specify the criteria and assumptions upon which it will rely to modify the WESTEMS computer model for calculation of CUF_{en} prior to a decision on license renewal." *See* Hopenfeld June at 15. (Ex. RIV000102). In addition, Dr. Hopenfeld states that "without specifying the modifications to be made to the model, or the process for deciding when and how to have user intervention in the use of the model, Entergy has not demonstrated that the aging effects of metal fatigue will be adequately managed." *See* Hopenfeld June at 15-16. (Ex. RIV000102).

Q111. Is Dr. Hopenfeld's use of the term "user intervention" in the context of Commitment No. 44, WESTEMS and the Staff's concern accurate?

A111. (AH, OY, CN) Dr. Hopenfeld is vague in his description of the term "user intervention." He is also not clear about the "criteria and assumptions" that he is referencing because the use of assumptions and engineering judgment is inherent in any fatigue analysis regardless if it is performed with or without computer software. However, based on this generic description and use of the terms "user intervention" and "criteria and assumptions," it appears that Dr. Hopenfeld has misunderstood the term "user intervention" as it relates to Commitment No. 44, WESTEMS and the Staff's concern.

As we stated previously, the term "user intervention" is clearly defined in RIS 2011-14 is related to very specific steps and actions that a properly trained analyst does during the calculation. This is in contrast to Dr. Hopenfeld's vague description of the "criteria and assumptions upon which [Entergy] will rely to modify the WESTEMS computer model."

To restate: the Staff's concern with "user intervention" as described in RIS 2011-14 was associated only with sufficient <u>documentation</u> of modifications of stress peaks and valleys by properly trained analysts using the WESTEMS software, and not with the engineering judgment exercised by the analyst or the results of the analyses.

Q112. What is your opinion on Dr. Hopenfeld's statement that Entergy has not demonstrated that the aging effects of metal fatigue will be adequately managed?

A112. (AH, OY, CN) Dr. Hopenfeld stated that "[t]hus, without specifying the modifications to be made to the model, or the process for deciding when and how to have user intervention in the use of the model, Entergy has not demonstrated that the aging effects of metal fatigue will be adequately managed." *See* Hopenfeld June at 15 and 16 (Ex. RIV000102).

The Staff does not agree with Dr. Hopenfeld's statement. As described in the RIS-2011-14, the Staff concern is that the modification to the calculation should be documented such that design analyses and calculations are sufficiently detailed that a person technically qualified in the subject area can review, understand the analyses, and verify the adequacy of the results without recourse to the originator.

It is the Staff's opinion that this issue with documentation does not render that Entergy has failed to demonstrate that the aging effects of metal fatigue will be adequately managed because Entergy's documentation for any analysis, not just fatigue analyses, must be performed in accordance with a Quality Assurance program that is currently required by Appendix B to 10 CFR 50. The Intervenors have not questioned or identified concerns regarding the Entergy's current requirements to implement a Quality Assurance program in accordance with Appendix B to 10 CFR Part 50.

Entergy is required to perform its activity in accordance with a Quality Assurance Program implemented in accordance with the current requirements of Appendix B to 10 CFR Part 50, which means that the EAF analyses documented in WCAP-17999-P and WCAP-17200-P, the EAF analyses that may be performed in the future and the evaluation that will be completed as part of Commitment No. 43 are governed by this Quality Assurance program. Entergy is currently required by Appendix B to 10 CFR Part 50 to implement a Quality Assurance program that ensures that, for the aforementioned analyses and evaluations, there are sufficient records and these records are maintained to document activities affecting quality. Furthermore, this Quality Assurance program required by Appendix B to 10 CFR Part 50 will provide measures for verifying or checking the adequacy of design, such as by the performance of design reviews. In addition, design analyses and calculations are to be sufficiently detailed such that a person technically qualified in the subject area can review and understand the analyses and verify the adequacy of the results without recourse to consulting the originator.

Dr. Hopenfeld's statement is very general in that it is obvious that the assumptions used in an analysis can affect the results of an analysis. But that is the exact reason that an individual performing fatigue analyses must have specialized experience and be specifically trained. Entergy's current requirements to implement a Quality Assurance program in accordance with Appendix B to 10 CFR Part 50 takes into account the need for special controls, processes, and skills to attain the required quality, and the need for verification of quality when performing an analyses or calculation. In addition, Entergy is currently required to implement a Quality Assurance program in accordance with Appendix B to 10 CFR Part 50 that provides for indoctrination and training of personnel performing activities affecting quality as necessary to assure that suitable proficiency is achieved and maintained.

It appears that Dr. Hopenfeld is concerned with the adequacy of Entergy's current requirements to implement a Quality Assurance program in accordance with Appendix B to 10 CFR Part 50, which is not the subject of this contention.

Q113. What is your opinion on Dr. Hopenfeld's summary regarding whether or not Entergy has demonstrated that metal fatigue of reactor components will be adequately managed?

A113. (AH, OY, CN) Dr. Hopenfeld stated that "Entergy has failed to make the affirmative demonstration that it has a program to sufficiently monitor, manage, and correct metal fatigue related degradation at Indian Point." *See* Hopenfeld June at 16 (Ex. RIV000102).

The Staff does not agree with Dr. Hopenfeld statement because the Fatigue Monitoring program described in the application, as amended, is an existing program that was enhanced to meet all the program elements defined in the GALL Report AMP X.M1. Entergy augmented its existing program by addressing the effects of environmentally-assisted fatigue for the purpose of license renewal prior to the period of extended operation in order to meet the GALL Report AMP X.M1 recommendations for environmentally-assisted fatigue and identifying additional locations.

Information provided by the applicant shows that Entergy has demonstrated that its program is adequate to manage the aging effect of metal fatigue and EAF because the Fatigue Monitoring program (1) tracks actual plant transients, (2) evaluates these actual transients against design transient definitions to ensure the actual severity is not greater than the design severity, and (3) ensures that the number of cycles experienced by the plant remain within the analyzed number of cycles in the fatigue evaluations.

Letter from Morgan Lewis on May 15, 2012 - ML12136A420

Q114. Have you read the "Letter from Morgan Lewis," dated May 15, 2011 (Exhibit NYS000395) ("Entergy May 2012 Letter")?

A114. (AH, OY, CN) Yes.

Q115. When does Entergy believe it will be done with its initial screening review of design basis ASME Code Class 1 fatigue evaluations?

A115. (AH, OY, CN) By letter dated May 15, 2012, Entergy informed the Board that Entergy has determined that the initial screening review of design basis ASME Code Class 1 fatigue evaluations, as described in Commitment No. 43, to determine whether the NUREG/CR-6260 locations are the limiting locations for IPEC, is expected to be completed within approximately four months from the date of the letter. *See* Entergy May 2012 Letter at 1. Meeting this schedule would mean that the review will be done before PEO and before the license renewal decision, as both Dr. Lahey and Dr. Hopenfeld have stated is necessary.

Q116. How is the screening review performed?

A116. (AH, OY, CN) Entergy has not identified the screening review methodology. There are many different approaches that the Applicant could follow that would yield appropriate results.

The Staff's opinion is that the results related to Commitment No. 43 need not be provided before a licensing decision is reached; therefore, the screening review methodology that Entergy uses is not required before a licensing decision is reached. However, Commitment No. 43 must be completed prior to entering the period of the extended operation, which is consistent with the implementation schedule of this commitment. *See* NL-11-032, Attachment 1 at 26 and Attachment 2 at 17 (Ex. NRC000110). The Staff's opinion is that the license renewal application is complete and there is no missing information. Entergy has demonstrated that its Fatigue Monitoring program is capable and adequate to manage metal fatigue and environmentally-assisted fatigue during the period of extended operation. Entergy's Fatigue Monitoring program has been described in detail in our testimony prepared for NYS-26B/RK-TC-1B. *See* NRC000102 at 23 through 27.

Q117. How is the screening review related to the required general information and technical information in a license renewal application (i.e. 10 C.F.R. §§ 54.21 and 54.22)?

A117. (AH, OY, CN) The screening information is not the type of information that the applicant is required to include in a license renewal application.

Q118. What does Entergy do with the results?

A119. (AH, OY, CN) If more limiting locations are identified, then Entergy will include these additional locations to be managed by the Fatigue Monitoring program.

Q120. How does this May 15, 2012 letter relate to Contention NYS-38/RK-TC-5?

A120. (AH, OY, CN) The May 15, 2012, letter is associated with Entergy's planned completion of Commitment No. 43, which is one of the commitments the Intervenors call into question as part of Contention NYS-38/RK-TC-5. The completion of Commitment No. 43 would appear to satisfy part of the claim in NYS-38/RK-TC-5 related to identifying additional limiting locations.

Q121. Does this conclude your testimony?

A121. (AH, OY, CN) Yes.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
ENTERGY NUCLEAR OPERATIONS, INC.	)	Docket Nos. 50-247/286-LR
	)	
(Indian Point Nuclear Generating	)	
Units 2 and 3)	)	

AFFIDAVIT OF ALLEN L. HISER, JR.

CONCERNING CONTENTION NYS-38/ RK-TC5

I, Allen Hiser, do hereby declare under penalty of perjury that my statements in the foregoing testimony and my statement of professional qualifications are true and correct to the best of my knowledge and belief.

Executed in Accord with 10 C.F.R. § 2.304(d).

Allen Hiser, Jr.
Mechanical Engineer
Division of License Renewal
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
(301) 415-5650
Allen.Hiser@nrc.gov

August 20, 2012

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
ENTERGY NUCLEAR OPERATIONS, INC.	)	Docket Nos. 50-247-LR/ 50-286-LR
	)	
(Indian Point Nuclear Generating	)	
Units 2 and 3)	)	

AFFIDAVIT OF ON YEE

CONCERNING CONTENTION NYS-38/ RK-TC5

I, On Yee, do hereby declare under penalty of perjury that my statements in the foregoing testimony and my statement of professional qualifications are true and correct to the best of my knowledge and belief.

Executed in Accord with 10 C.F.R. § 2.304(d).

On Yee
Mechanical Engineer
Division of License Renewal
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
(301) 415-1905
On.Yee@nrc.gov

August 20, 2012

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
ENTERGY NUCLEAR OPERATIONS, INC.	)	Docket Nos. 50-247-LR/ 50-286-LR
	)	
(Indian Point Nuclear Generating	)	
Units 2 and 3)	)	

AFFIDAVIT OF CHING NG

CONCERNING CONTENTION NYS-38

I, Ching Ng, do hereby declare under penalty of perjury that my statements in the foregoing testimony and my statement of professional qualifications are true and correct to the best of my knowledge and belief.

Executed in Accord with 10 C.F.R. § 2.304(d).

Ching Ng
Mechanical Engineer
Division of License Renewal
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
(301) 415-8054
Ching.Ng@nrc.gov

August 20, 2012