NRC000168
Submitted: August 10, 2015

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 Hiser, Dr. Ching Ng, Mr. Gary Stevens, P.E., and Mr. On Yee, Concerning Contentions NYS-26B/RK-TC-1B and NYS-38/RK-TC-5

Witness Background

Q1 Please state your name, occupation, and by whom you are employed.

A1 [GS] My name is Mr. Gary L. Stevens. I have been employed by the NRC for more than five years. I am currently employed as a Senior Materials Engineer in the Vessel and Internals Integration Branch in the Division of Engineering, Office of Nuclear Reactor Regulation (NRR), U.S. Nuclear Regulatory Commission (NRC), in Rockville, Maryland (MD). Prior to March 2015, I was employed as a Senior Materials Engineer in the Component Integrity Branch, Division of Engineering, Office of Nuclear Regulatory Research (RES), NRC, in Rockville, MD. I received a Bachelor of Science degree in Mechanical Engineering from California Polytechnic State University in San Luis Obispo, CA, and a Master of Science degree in Mechanical Engineering from San Jose State University. I have been a participating member in American Society of Mechanical Engineers (ASME) Code, Section XI Committees for more than 25 years. My statement of qualifications is attached hereto (Ex. NRC000227).

[AH] My name is Dr. Allen Hiser, Jr. I have worked at the NRC for 25 years in the Office of Nuclear Regulatory Research and NRR. I am employed as the Senior Technical Advisor for License Renewal Aging Management in the Division of License Renewal, NRR, NRC, in Rockville, MD. 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 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 is attached hereto (Ex. NRCR00103).

[OY] My name is Mr. On Yee. I have been working at the NRC for approximately ten years. I am currently employed as a Reactor Systems Engineer in the Containment & Balance of Plant Branch, Japan Lessons-Learned Division, NRR, NRC, in Rockville, MD. I was employed as a Mechanical Engineer in the Aging Management of Reactor Systems Branch, Division of License Renewal, NRR, NRC, in Rockville, MD. 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 is attached hereto (Ex. NRCR00104).

[CN] My name is Dr. Ching Ng. I have been working at the NRC for more than eight years. I am currently employed as a Reliability and Risk Analyst in the Probabilistic Risk Assessment Operations and Human Factors Branch, Division of Risk Assessment, NRR, NRC, in Rockville, MD. I was employed as a Mechanical

Engineer in the Aging Management of Reactor Systems Branch, Division of License Renewal, NRR, NRC, in Rockville, MD. I received Bachelor of Science, Master of Science, and Doctoral degrees in Mechanical Engineering from the University of California, Berkeley. A statement of my professional qualifications is attached hereto (Ex. NRCR00105).

Q2 Please describe the nature of your current responsibilities.

A2 [GS] My current responsibilities include technical reviews of licensing applications, topical reports, and other submittals related to the integrity of the reactor vessel and reactor vessel internals of commercial nuclear pressurized water reactors (PWRs) and boiling water reactors (BWRs). I have performed numerous analyses and reviews related to reactor vessels and reactor vessel internals for license amendments related to power uprates, relief requests, surveillance capsule schedule changes, and license renewal applications (LRAs). I have reviewed or performed research and analyses related to of the technical bases for existing and proposed rulemaking, generic communications, and guidance documents related to reactor vessel integrity.

I am also the NRC's current subject matter expert on environmentally-assisted fatigue (EAF), and I am leading the NRC's research activities to update and revise Regulatory Guide (RG) 1.207, *Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors*, (March 2007) (ADAMS Accession No. ML070380586) (Ex. NRC000179) ("RG 1.207") and the associated update and revision of the supporting technical basis document, NUREG/CR-6909, *Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials*, (February 2007) (Ex. NYS000357) ("NUREG/CR-6909"). I have led these research activities for the NRC since 2010.

[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. My duties include serving as a lead technical expert for aging management evaluation and assisting other NRC Staff (Staff) as they implement their review of LRAs.

I am chairman of the Steering Committee of the International Atomic Energy Agency (IAEA) program to develop aging management standards for international use and the International Generic Aging Lessons Learned program. In addition, I have been a team member on several IAEA missions to evaluate the aging management for international plants pursuing license renewal, and I have been a trainer in international workshops for regulators and plants on aging management for license renewal.

[OY] I am currently a technical reviewer in the Containment & Balance of Plant Branch, which provides mechanical engineering and reactor systems technical expertise in the review of balance-of-plant aspects of all licensee actions directly related to licensee compliance with NRC Order EA-12-049, *Mitigating Strategies*. This involves reviewing licensee's diverse and flexible mitigation strategies that will increase defense-in-depth for beyond-design-basis scenarios to address an extended loss of alternating current power and loss of normal access to the ultimate heat sink occurring simultaneously at all units on a site.

[CN] I am currently a Reliability and Risk Analyst who develops risk assessment tools used for the Significance Determination Process in the Reactor Oversight Process. I also perform risk assessment for emergent safety issues including reactor events, Notice of Enforcement Discretion requests, and other special requests from Regional Offices.

Q3 Please describe your duties in connection with the 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 [GS] I was not directly involved in the Staff's review of Entergy's, *Indian Point Energy Center License Renewal Application*, (April 2007) (Ex. ENT000015A-B) ("LRA") for IP2 and IP3. I am providing testimony because I am the NRC's current subject matter expert on EAF, and because of my knowledge as an author of the revision of RG 1.207, the revision of NUREG/CR-6909, and the MPA Stuttgart Seminar Technical Paper, all of which are subjects included in these proceedings. I was also a participant in the Staff's audit and review of the Salem Nuclear Generating Station's use of the WESTEMS™ fatigue software during the license renewal process. The WESTEMS™ fatigue software is a subject of these proceedings.

[OY] I was part of the IP2 and IP3 LRA review from October 2007 until September 2013. 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 time-limited aging analyses (TLAAs) and environmentally-assisted fatigue analyses associated with the IP2 and IP3 LRA. As part of those activities, I assisted in the review of Entergy'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 Entergy's existing metal fatigue analyses, which are TLAAs as defined in Title 10 of the *Code of Federal Regulations* (10 C.F.R.) Section 54.3, and Entergy's environmentally-assisted fatigue analyses, which are not TLAAs as defined in 10 C.F.R. 54.3. I worked with the principal reviewer in the preparation in the review of these areas, the results of 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) (Ex. NYS00326A-F) (together, "SER"). I was the responsible Staff member for the area of metal fatigue during the Advisory Committee on Reactor Safeguards (ACRS) 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 IP2 and IP3. In this regard, I peer reviewed and provided technical feedback on the environmentally-assisted fatigue section of NUREG-1930, Supplement 1, *Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3*, (August 2011) (Ex. NYS000160) ("SER Supp. 1"). Also as part of my responsibilities, I submitted an affidavit on behalf of the Staff in response to Entergy'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). The NRC's response, 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*, was filed on September 14, 2010.

[AH] I was the Chief of the Steam Generator Tube Integrity and Chemical Engineering Branch in NRR 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 Entergy on environmentally-assisted fatigue analyses, which was used to develop SER Supp. 1. As a part of this work, I reviewed the LRA and the SER. 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.

[CN] From June 2010 to January 2012, I served as a reviewer for the environmentally-assisted fatigue analyses associated with the IP2 and IP3 LRA. In that capacity, I reviewed Entergy'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) (Ex. NYS000150) ("RAI Letter"), related to the 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.

Q4 Please describe your professional experience related to the area of metal fatigue.

A4 [GS] I have worked in the nuclear industry for 34 years. I have been responsible for performing metal fatigue calculations for nuclear power plant components, including the reactor vessel, reactor vessel internals, and piping, in accordance with the ASME Boiler & Pressure Vessel Code (ASME Code) ASME Section III, *Rules for Construction of Nuclear Power Plant Components* (Ex. NYS000349) ("ASME Section III"), for more than 30 years. Over my career, I have collectively performed hundreds of cumulative usage factor (CUF) and environmentally-assisted cumulative usage factor (CUF_{en}) analyses. I am a registered Professional Engineer in three states, and I have used my PE certifications on many occasions to stamp and certify Design Reports that document stress and fatigue analyses in accordance with ASME Section III for nuclear piping and reactor vessel components. I have been involved in EAF calculations since 1995 when NUREG/CR-6260, *Application of NUREG/CR–5999 Interim Design Curves to Selected Nuclear Power Plant Components*, (February 1995) ("NUREG/CR-6260") (Ex. NYS000355) was published. I have been an industry leader in EAF activities since the late 1990s when EAF evaluations started with the first LRA for the Calvert Cliffs plant. I have participated in most of the ASME Code committees and industry technical groups established to address EAF requirements since 1996, I am a member of the ASME Code Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components* (Ex. NRC000195) ("ASME Section XI") Standards Committee on Nuclear Inservice Inspection, I am a member of the ASME Section XI Executive Committee, I have been the Secretary for the ASME Section XI Subgroup on Evaluation Standards since 2002, and I am the standing NRC representative to ASME Section XI. Prior to joining the NRC in February 2010, I was a technical business leader in the area of TLAA submittals for LRAs and reactor vessel integrity evaluations, and I performed many EAF CUF_{en} calculations over the previous 15 years. Since joining the NRC, I have been the NRC subject matter expert on EAF, I have led all of the NRC's research activities on EAF, and I have participated as a consultant to many of the Staff's reviews of LRAs.

[AH] During the early part of my career, I was involved in fatigue crack growth rate testing of reactor grade materials in a simulated reactor water environment. More recently, I have been a peer reviewer and have provided guidance to NRC reviewers who have evaluated LRA sections that have dealt with all aspects of metal fatigue, including fatigue monitoring AMPs, metal fatigue TLAAs and CUF_{en} calculations. I was a member of the audit team that reviewed Entergy's use of the WESTEMS™ software program in metal fatigue analyses in the Salem Nuclear Generating Station LRA, and developed Regulatory Issue Summary (RIS) 2011-14, *Metal Fatigue Analysis Performed by Computer Software*, (December 2011) (Ex. NRC000112) ("RIS-2011-14"). I have been a reviewer of draft documents and regulatory guidance on metal fatigue, including the draft revisions of NUREG/CR-6909 and RG 1.207, and the final revisions of NUREG-1800, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants* (SRP-LR), Rev. 2, (December. 2010) (Ex. NYS000161), and NUREG-1801, *Generic Aging Lessons Learned (GALL) Report, Rev. 2*, (Dec. 2010) (Ex. NYS000147A-D) ("GALL Report Rev. 2").

[OY] I performed reviews of multiple LRAs in the area of metal fatigue, which included auditing program basis documents, reviewing and assessing the relevant information in LRAs, and crafting requests for additional information when the LRA lacked sufficient information to complete my review.

My reviews included licensee's metal fatigue TLAAs of reactor vessel internals, metal fatigue TLAAs for ASME Code Class 1, 2 and 3 components, metal fatigue TLAAs for ANSI B31.1, *Power Piping,* components, and analyses of reactor coolant environment effects on the fatigue lives of reactor coolant pressure boundary components and piping. The reviews I performed enabled the NRC to determine whether licensees satisfactorily addressed aging of metal components caused by fatigue in their applications, how licensees proposed to address and manage aging of metal components caused by fatigue for future operation, and the adequacy of any proposed aging management programs (i.e., fatigue management programs).

As part of my reviews, I assessed how licensees used the guidance in NUREG/CR-6260, NUREG/CR-5704, *Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels* (April 1999) (Ex. NYS000354) ("NUREG/CR-5704"), NUREG/CR-6583, *Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels* (March 1998) (Ex. NYS000356) ("NUREG/CR-6583"), and NUREG/CR-6909. Further, my reviews also considered SRP-LR Rev. 2 and GALL Report Rev. 2 to determine if the LRA demonstrated that the effects of aging will be adequately managed and meet the rules in 10 C.F.R. Part 54.

[CN] I performed reviews of multiple LRAs, including Indian Point, Palo Verde Nuclear Generating Station, Hope Creek Generating Station, Salem Nuclear Generating Station, Vermont Yankee Nuclear Power Station, and Diablo Canyon Nuclear Power Plant.

The focus of my work in these reviews was in the area of metal fatigue, and it included auditing program basis documents, reviewing and assessing the relevant information in the LRA, and crafting requests for additional information when the LRA lacked sufficient information to complete my review. My reviews included the licensee's identification of TLAAs, plant-specific TLAAs, evaluations of the effects of reactor coolant environment on reactor coolant pressure boundary component fatigue lives, metal fatigue TLAAs of reactor vessel internals, and fatigue evaluations for ASME Code Class 1, 2 and 3 components. The reviews I performed enabled the Staff to determine adequacy of proposed TLAA dispositions and the adequacy of proposed aging management programs, including fatigue monitoring programs.

My primary work products included requests for additional information, audit reports and input to the draft and final safety evaluation reports. I was a member of the on-site audit teams that evaluated licensee aging management reviews and aging management programs. I was also a member of the audit team that reviewed the Salem Generating Station's use of the WESTEMS™ software program in metal fatigue analyses in the Salem Nuclear Generating Station LRA. The results of my reviews on identification of TLAAs, evaluation of plant-specific TLAAs, evaluation of the effects of reactor coolant environment on reactor coolant pressure boundary component fatigue lives, and evaluation of metal fatigue TLAAs, are documented in the Staff's safety evaluation report for each LRA.

Q5 Please describe your work at the Office of Nuclear Regulatory Research (RES) with respect to metal fatigue.

A5 [GS] I joined the NRC in February 2010. Beginning in August 2010, I initiated and led the NRC's research activities to update and revise RG 1.207 and NUREG/CR-6909. These research activities were requested by NRR and the Office of New

Reactors (NRO) to provide technical support to update the existing environmental fatigue evaluation method and develop techniques for applying this method for structural and component evaluations. The outcomes of this research assisted NRR and NRO review of licensee environmental fatigue submittals associated with license renewal and new reactor applications, and supported evaluation of proposed changes to ASME Section III fatigue analyses and subsequent adoption of these changes within 10 C.F.R. Part 50, 50.55a, *Codes and Standards* ("10 C.F.R. 50.55a"). As part of these research activities, RES staff and contractors reviewed additional available laboratory data collected subsequent to the initial publication of the fatigue curves and F_{en} technique documented in RG 1.207 and NUREG/CR-6909. The outcome of this research is being used to revise RG 1.207 and NUREG/CR-6909.

Since joining the NRC, I have also periodically provided consultation and technical support on EAF to the NRC's Division of License Renewal and the NRC's Office of New Reactors.

Although I moved to NRR in March 2015, I still retain the responsibilities to complete the revision of RG 1.207 and NUREG/CR-6909.

Overview of Contentions

NYS-26B/RK-TC-1B and NYS-38/RK-TC-5

Q6 What is the purpose of your testimony?

A6 [GS, AH, OY, CN] The purpose of our testimony is to present the Staff's views with respect to the consolidated New York Contention 26B and Riverkeeper Contention TC-1B ("NYS-26B/RK-TC-1B"). As directed by the Board, we are also providing rebuttal testimony to NYS-26B/RK-TC-1B. Our testimony is being used to support the Staff's Statement of Position concerning NYS-26B/RK-TC-1B, which the Staff is filing simultaneously with our testimony.

In addition, our testimony is to present the 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.

Q7 Are you familiar with the consolidated New York Contention 26B and Riverkeeper Contention TC-1B?

A7 [GS, AH, OY, CN] Yes, we are familiar with the consolidated New York Contention 26B and Riverkeeper Contention TC-1B. As stated in the Board Order (*Ruling on Motion for Summary Disposition of NYS-26/26A/Riverkeeper TC-1/1A (Metal Fatigue of Reactor Components)* and *Motion for Leave to File New Contention NYS-26B/Riverkeeper TC-1B*), (November 4, 2010) (ADAMS Accession No. ML103080987)) NYS-26B/RK-TC-1B characterizes Entergy's fatigue re-analyses as inadequate under NRC regulations and GALL Report Rev. 2 because these re-analyses (1) inappropriately limited the number of components subject to fatigue analyses, (2) neither explained the methodology used to conduct the CUF analyses nor included a detailed error analysis, (3) excluded "a fatigue evaluation of important structures and fittings within the reactor pressure vessel (RPV)", (4) excluded from evaluation "the potential failure of highly fatigued structures and fittings under" certain types of "large thermal/pressure shock-type loads," and (5) contained lower safety margins that create more risk because the new CUFs have been "reduced by more than an order of magnitude." The Intervenors also noted that "Entergy has not committed to repair or replace components when the CUF approaches unity (1.0)." Order at 8.

We also have read the Intervenors' the supporting expert testimony from Dr. Richard T. Lahey Jr Pre-Filed Written Testimony of Richard T. Lahey, Jr. Regarding Consolidated Contention NYS-26B/RK-TC-1B," dated December 27, 2011 (Ex. NYSR00344) ("Lahey") and the "Revised Pre-Filed Written Testimony of Dr. Richard T. Lahey, Jr., in Support of Consolidated Contention NYS-26B/RK-TC-1B," dated June 9, 2015 (Ex. NYS000530) ("Revised Lahey").

In addition we have read the Intervenors' supporting expert testimony from Dr. Joram Hopenfeld, Prefiled Written Testimony of Dr. Joram Hopenfeld Regarding NYS-26-B/RK-TC-1B-Metal Fatigue, (December 22, 2011) ("Hopenfeld") (Ex. RIV000034), Report of Dr. Joram Hopenfeld in Support of Contention Riverkeeper TC-1B-Metal Fatigue, (December 22, 2011) ("Hopenfeld Report") (Ex. RIV000035) and the Supplemental Report of Dr. Joram Hopenfeld In Support of Contention NYS-26/RK-TC-1B and Amended Contention NYS-38/RK-TC-5, dated June 8, 2015 ("Hopenfeld Supplemental Report") (Ex. RIV000144).

Q8 Do you agree with the merits of NYS-26B/RK-TC-1B, and has your review of NYS-26B/RK-TC-1B changed your view on the adequacy of Entergy's Fatigue Monitoring Program and its fatigue re-analyses?

A8 No, we do not agree that the issues identified in NYS-26B/RK-TC-1B have merit. Our opinion is that Entergy's Fatigue Monitoring Program and its fatigue re-analyses are sufficient to provide reasonable assurance of adequate aging management such that the intended functions of the structures and components affected by fatigue will be maintained, consistent with 10 CFR 54.29, which 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 TLAAs (§§ 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 CLB in order to comply with this paragraph are in accord with the Atomic Energy Act and the Commission's regulations.

Q9 Are you familiar with the New York Contention 38 and Riverkeeper Contention TC-5?

A9 [GS, AH, OY, CN] Yes, we are familiar with the New York Contention 38 and Riverkeeper Contention TC-5. 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 June") and Dr. Joram Hopenfeld, *Prefiled Written Testimony of Dr. Joram Hopenfeld Regarding Contention NYS-38/RK-TC-5 (*Ex. RIV000102) ("Hopenfeld June") submitted on June 19, 2012.

In addition, the Staff is familiar with the Board Memorandum and Order (Granting Motions for Leave to File Amendments to Contentions NYS-25 and NYS-38/RK-TC-5) (March 31, 2014) (ADAMS Accession No. ML15090A771), admitting the additional bases for NYS38/RK-TC-5 related to the contents and conclusions of Entergy's Amended and Revised RVI Plan, as well as the contents and conclusions of the SSER2, which approved Entergy's Amended and Revised RVI Plan.

We also have read the Intervenors' supporting expert testimony from Dr. Richard T. Lahey Jr., Revised Pre-Filed Written Testimony of Dr. Richard T. Lahey, Jr., in Support of Joint Contention NYS-38/RK-TC-5 dated June 9, 2015 (Ex. NYS000562) ("Lahey Revised June") and the Supplemental Report of Dr. Joram Hopenfeld In Support of Contention NYS-26/RK-TC-1B and Amended Contention NYS-38/RK-TC-5, dated June 8, 2015 ("Hopenfeld Supplemental Report") (Ex. RIV000144).

The Staff noted that Dr. Lahey's revised pre-filed written testimony in support of joint contention NYS-38/RK-TC-5, dated June 9, 2015 (Ex. NYS000562), is similar to the testimony provided in his revised pre-filed written testimony in support of consolidated contention NYS-26B/RK-TC-1B, dated June 9, 2015 (Ex. NYS000530). For the purposes its testimony related to metal fatigue, the Staff addresses Dr. Lahey's revised pre-filed written testimony in support NYS-26B/RK-TC-1B and NYS-38/RK-TC-5 separately.

Q10 What are the "multiple bases" that the Intervenors referred to in NYS38/RK-TC-5??

A10 [GS, 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." 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.

As discussed in Q9, the Board admitted additional bases to NYS-38/RK-TC-5. As described in the State Of New York And Riverkeeper's Joint Motion For Leave To Supplement Previously-Admitted Joint Contention NYS-38/RK-TC-5, (February 13, 2015) (ADAMS Accession No. ML15044A500) ("NYS-38/RK-TC-5"), the Intervenors' basis is: Basis (4) relates to the contents and conclusions of Entergy's Amended and Revised reactor vessel internals (RVI) Plan, as well as the contents and conclusions of the SSER2, which approved Entergy's Amended and Revised RVI Plan. The Intervenors' basis is that the amended and revised reactor vessel internals Plan substantially modified and replaced Entergy's previously-proposed approach relating to RVIs.

Q11 Which bases of NYS-38/RK-TC-5 does your testimony address?

A11 [GS, 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 Entergy's Commitment No. 43 and Basis (2) is related to Entergy's Commitment No. 44.

In addition, the Staff's testimony addresses Basis (4), only in part, as it relates to calculation of cumulative usage factors adjusted for environmental effects for various reactor vessel internals.

Q12 Do you agree with NYS-38/RK-TC-5 as related to Basis (1) and Basis (2)?

A12 [GS, AH, OY, CN] No, we do not agree with NYS-38/RK-TC-5 as related to Basis (1) and Basis (2).

Q13 What are "CUF_{en} calculations" that are the subject of Basis (1) and Basis (2) of NYS38/RK-TC-5?

A13 [GS, 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.

Regulations and Guidance Related to License Renewal

Q14 Please describe the Commission’s requirements pertaining to aging management.

A14 [GS, 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 TLAAs (§§ 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 CLB 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.

Q15 Please describe the Commission’s requirements pertaining to metal fatigue and the AMP for fatigue monitoring.

A15 [GS, AH, OY, CN] In the license renewal context, §§ 54.33 and 54.35 of 10 C.F.R. require that a licensee comply with 10 C.F.R. Part 50 regulations; this includes the provision of 10 C.F.R. 50.55a requiring compliance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (“ASME Code”) Section III (“ASME Section III”) and Section XI (“ASME Section XI”), during the period of extended operation.

The ASME Code is an industry consensus standard that is developed by industry experts, with active participation from the Staff. Engineers in the nuclear, non-nuclear, and fossil industries routinely rely on these documents in their operations.

The ASME Code is periodically updated to reflect operating experience and ongoing research and development activities in materials science and analytical areas. The NRC incorporates by reference consensus standards in 10 C.F.R. 50.55a.

In particular, 10 C.F.R. 50.55a(c)(1), *Standards Requirement for Reactor Coolant Pressure Boundary Components*, requires that components of the reactor coolant pressure boundary meet the metal-fatigue requirements for Class 1 components in ASME Section III. ASME Section III, in turn, provides the methodology for calculating the CUFs for nuclear power plant Class 1 components, and specifies a design allowable limit of 1.0 for the CUF for any such component. ASME Section III at 81 (Ex. NYS000349). Fatigue evaluations for ASME Code Class 1 components will be referred to hereinafter as, “ASME Code Class 1 fatigue evaluations.”

10 C.F.R. 50.55a(g), *Inservice Inspection Requirements*, requires that Class 1 components be designed and be provided with access to enable the performance of preservice and inservice inspection requirements and evaluations for Class 1 components in ASME Section XI. Subarticle IWB-3700, *Analytical Evaluation of Plant Operating Events*, of ASME Section XI provides procedures that may be used to assess the effects of thermal and mechanical fatigue concerns on component acceptability for continued service, and procedures that may also be used when the calculated fatigue usage exceeds the fatigue usage limit defined in the original Construction Code of the facility for any component. ASME Section XI at 140 (Ex. NRC000195).

Other regulations specifically address aging management. These regulations include 10 C.F.R. 54.29, which states:

> *A renewed license may be issued by the Commission up to the full term authorized by § 54.31 if the Commission finds that:*

(a) Actions have been identified and have been or will be taken with respect to the matters identified in paragraphs (a)(1) and (a)(2) of this section, 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], 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. These matters are:

1) managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under § 54.21(a)(1); and

2) *time-limited aging analyses that have been identified to require review under § 54.21(c).*

Also, § 54.21(a) requires, among other things, that each application contain an integrated plant assessment that must identify and list those structures and components subject to an aging management review that are within the scope of license renewal, as described in § 54.4. Section 54.21(a) also requires that, for each structure and component, the applicant must demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation. With respect to ASME Code Class 1 fatigue evaluations, this means that any additional stress cycles that may occur during the period of extended operation must be evaluated.

Specific to TLAAs for license renewal, § 54.21(c)(1) requires that an LRA include an evaluation of TLAAs demonstrating that one of the following applies to each TLAA:

(i) The analysis remains valid for the period of extended operation;

(ii) The analysis has 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.

There are no regulatory requirements in the Commission's regulations nor the ASME Code endorsed by 10 C.F.R. 50.55a for an assessment of the effects of reactor water environment on metal fatigue (hereinafter environmentally-assisted fatigue analyses) in the form of calculating an environmental adjustment factor (F_{en}). The regulations at 10 C.F.R. § 54.3 define TLAAs as being contained in the CLB. Because environmentally-assisted fatigue analyses are not contained in Indian Point's CLB, they are not TLAAs and hence evaluation of environmentally-assisted fatigue analyses is not a prerequisite to issuance of a renewed license. The Staff's consideration of environmentally-assisted fatigue analyses in its review of the Indian Point LRA does not render an environmentally-assisted fatigue analysis a requirement under the Commission's regulations for TLAAs. This position is 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) .

Q16 Please describe the Staff's guidance on what to include in an LRA and how the Staff should review the application.

A16 [GS, AH, OY, CN] GALL Report Rev. 2, and NUREG-1801, *Generic Aging Lessons Learned (GALL) Report*, Rev. 1 (September 2005) (Ex. NYS00146A-C) ("GALL Report Rev. 1"), are technical basis documents to SRP-LR Rev. 2 and NUREG-1800, SRP-LR, Rev. 1, (September, 2005) (Ex. NYS000195) ("SRP-LR Rev. 1"), respectively. GALL Report Rev. 1 and GALL Report Rev. 2 state that an applicant may reference the GALL Report in an LRA 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 LRA and use of the GALL Report is not required, but use of the GALL Report should facilitate both preparation of an LRA by an applicant and timely, uniform review by the Staff. GALL Report Rev. 1 at 4 (Ex. NYS00146A-C) and GALL Report Rev. 2 at 8 (Ex. NYS00147A-D).

SRP-LR Rev. 1 and SRP-LR Rev. 2 provide guidance to the Staff on how to perform safety reviews of applications to renew nuclear power plant licenses in accordance with 10 C.F.R. Part 54. The principal purposes of SRP-LR Rev. 1 and SRP-LR Rev. 2 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. SRP-LR Rev. 1 and SRP-LR Rev. 2 at iii (Ex. NYS000195 and Ex. NYS000161, respectively). SRP-LR Rev. 1 and SRP-LR Rev. 2 note 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. SRP-LR Rev. 1 and SRP-LR Rev. 2 at 3.0-1 (Ex. NYS000195 and Ex. NYS000161, respectively).

It should be noted that GALL Report Rev. 1 and GALL Report Rev. 2 do not address scoping of structures and components for license renewal. Scoping is plant specific, and the results depend on the plant design and CLB. The inclusion of a certain structure or component in the GALL Report does not mean that this particular structure or component is within the scope of license renewal for all plants.

Conversely, the omission of a certain structure or component in the GALL Report does not mean that this particular structure or component is not within the scope of license renewal for any plants. GALL Report Rev. 1 at 4 (Ex. NYS00146A-C) and GALL Report Rev. 2 at 8 (Ex. NYS00147A-D).

<u>Staff's Review of Entergy's</u>

<u>Fatigue Monitoring Program, Metal Fatigue TLAAs and EAF Analyses</u>

Q17 How did the Staff review Entergy's Fatigue Monitoring program?

A17 [GS, AH, OY, CN] The Staff reviewed Entergy's Fatigue Monitoring program consistent with SRP-LR Section 4.3 and GALL Section X.M1. SRP-LR Rev.1 and SRP-LR Rev. 2 at Section 4.3 (Ex. NYS000195 and NYS000161, respectively), and GALL Report Rev.1 and Rev. 2 at Section X.M1 (Ex. NYS00146A-D and Ex. NYS00147A-D, respectively). In addition, the Staff performed on-site inspections and an on-site audit of the Fatigue Monitoring program.

Q18 Please describe the Staff's guidance on performing on-site audits for the Indian Point LRA.

A18 [GS, AH, OY, CN] For the on-site audit of IP2 and IP3, 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 27, 2007) (Ex. NRC000123) ("Audit Plan"). The scope of work is defined in this audit plan and the project team sought to verify that Entergy'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. Audit Plan at 1 (Ex. NRC000123).

Q19 Please describe the Staff's guidance on performing on-site inspections for the Indian Point LRA.

A19 [GS, 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 Program*, (August 13, 2013) (ADAMS Accession No. ML13092A015) (Ex. NRC000180) ("Revised 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 Staff to verify the adequacy of aging management programs and other activities associated with an applicant's request to renew the operating license of a commercial nuclear power plant beyond the initial licensing period under 10 C.F.R. Part 54, *Requirements for the Renewal of Operating Licenses for Nuclear Power Plants* ("The Rule"). Revised IMC 2516 at 1 (Ex. NRC000180).

The first on-site inspection of IP2 and IP3 related to license renewal occurred at the Indian Point site starting on January 28, 2008, and was completed on June 18, 2008. For that inspection, the Staff followed NRC Inspection Procedure 71002, *License Renewal Inspection*, (November 2011) (Ex. NRC000106) ("IP71002"). In particular, the inspection verified that Entergy'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 Entergy's LRA.

The inspection also verified the documentation, implementation, and effectiveness of the aging management programs and activities associated with Entergy's license renewal program. In addition, it verified that Entergy had adequate programs planned or in place to implement aging management for the structures and components that require an aging management review, such that these structures and components will be adequately maintained consistent with The Rule, the Staff's existing safety evaluations, and Entergy's license renewal program.

Details about the scope and results of the IP71002 inspection performed for IP2 and IP3 are contained in NRC Inspection Reports 05000247/2008006 and 05000286/2008006 (August 1, 2008) (Ex. NRC000107) ("IP71002 Report").

The second on-site inspection related to license renewal is typically performed prior to entering the period of extended operation. In general, for this inspection, the Staff will follow the NRC Inspection Manual, Inspection Procedure 71003, *Post-Approval Site Inspection for License Renewal*, (October 2008) (Ex. ENT000251) ("IP71003"). The purpose of the inspection is to verify completion of license renewal commitments and license conditions that were added as part of the renewed license, and to ensure that selected aging management programs are implemented in accordance with the license renewal regulations.

For IP2, the NRC was aware during the review of the LRA that there was a 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) (Ex. NRC000151) ("TI 2516/001"). This temporary instruction is completed in cases where holders of an operating license meet the criteria for timely renewal in 10 C.F.R. 2.109, *Effect of Timely Renewal Application*, but an IP71003 inspection cannot be completed 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 a post license-renewal inspection before the period of extended operation. TI 2516/001 at 2 (Ex. NRC000151). 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. TI 2516/001 at 1 (Ex. NRC000151).

The Staff conducted three separate inspections in accordance with TI 2516/001, which collectively reviewed a total of 44 license renewal commitments for IP2. The first TI 2516/001 inspection for IP2 was conducted during a refueling outage from March 5-8, 2012, and the results are documented in NRC Inspection Report 05000247/2012008 (April 19, 2012) (Ex. NRC000152) ("Inspection Report 05000247/2012008"). The second TI 2516/001 inspection for IP2 was conducted from May 6-10 and May 20-23, 2013, and the results are documented in NRC Inspection Report 05000247/2013009 (July 5, 2013) (ADAMS Accession No. ML13186A179) (Ex. NRC000181) ("Inspection Report 05000247/2013009"). The third TI 2516/001 inspection for IP2 was conducted from September 9-12, 2013, and the results are documented in NRC Inspection Report 05000247/2013010 (September 19, 2013) (ADAMS Accession No. ML13263A020) (Ex. NRC000182) ("Inspection Report 05000247/2013010").

For IP3, in lieu of an IP71003 inspection, the second on-site inspection of IP3 related to license renewal will be performed prior to entering the period of extended operation in accordance with new Inspection Procedure 71013, *Site Inspection for Plants with a Timely Renewal Application*, (September 25, 2013) (ADAMS Accession No. ML13032A102) (Ex. NRC000183) ("IP71013"). IP71013 was issued on September 25, 2013, and states that TI 2516/001 expired on December 31, 2013, at which point timely renewal inspection activities would henceforth be conducted in accordance with IP71013. An inspection objective of IP71013 is to determine whether commitments made by the licensee to implement actions such as proposed license conditions, other regulatory commitments accepted by the Staff during the course of license renewal, selected AMPs, and TLAAs are implemented or completed. IP71013 at 1 (Ex. NRC000183). An inspection requirement of IP71013 is to verify that the licensee adequately completed the commitments made to implement actions such as proposed license conditions, other regulatory commitments accepted by the Staff during license renewal, AMPs and TLAAs. IP71013 at 1 (Ex. NRC000183).

Q20 Was Entergy's Fatigue Monitoring Program a part of the Staff's IP71002 inspection?

A20 [GS, AH, OY, CN] Yes Entergy's Fatigue Monitoring Program was a part of the IP71002 inspection.

Q21 What were the conclusions related to Entergy's Fatigue Monitoring Program from the IP71002 inspection?

A21 [GS, 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 each 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 were appropriately identified and addressed. IP71002 Report at 4 (Ex. NRC000107).

Q22 Were Entergy's Fatigue Monitoring Program and any associated commitments a part of the Staff's TI 2516/001 inspections for IP2?

A22 [GS, AH, OY, CN] Yes, Entergy's Fatigue Monitoring Program and any associated commitments were a part of the TI 2516/001 inspections for IP2. . The results related to Entergy's Fatigue Monitoring Program and any associated commitments from the Staff's TI 2516/001 inspections for IP2 are discussed in detail in our response to Q85.

Q23 Will Entergy's Fatigue Monitoring Program and any associated commitments be part of the Staff's IP71013 inspection for IP3?

A23 [GS, AH, OY, CN] Yes, Entergy's Fatigue Monitoring Program and any associated commitments will be a part of the IP71013 inspection for IP3.

Q24 What is an ASME Code Class 1 fatigue evaluation?

A24 [GS, AH, OY, CN] An ASME Code Class 1 fatigue evaluation is a calculation that was performed by an applicant in accordance with ASME Section III; the calculation is part of an applicant's CLB for the plant. An ASME Code Class 1 fatigue analysis is one part of a larger stress evaluation that is required by ASME to show compliance with ASME Section III for certification of the component for service. The fatigue evaluation part of the stress evaluation is a measure that identifies the likelihood of a component initiating a fatigue crack caused by cyclic loading.

As described in the Section 4.3 of SRP-LR Rev. 2, a metal component may become degraded due to fatigue when subjected to fluctuating stresses. This degradation can occur in components without fabrication flaws due to the development of fatigue cracks during service. ASME Section III requires a fatigue analysis for all Class 1 components, and requires calculation of CUF unless exempted under applicable ASME Section III, provisions. ASME Section III provides a specific process for this analysis, which considers the anticipated severity and number of thermal and pressure cycles for all transients specified by the designer. SRP-LR Rev. 2 at 4.3-1 (Ex. NYS000161).

CUF is evaluated by first determining the stress cycles in a component caused by the severity of each transient and any other external loads, considering both the temperature and pressure variations for each transient. Next, the allowed number of cycles for each transient to initiate a fatigue crack (“N_i”) is determined based on the total stress cycle for each transient. The incremental usage factor for each transient is determined by dividing the number of anticipated cycles for the transient by the “N_i” determined for that transient. The sum of all of the individual transient incremental usage factors is the CUF for the component.

ASME Section III limits the CUF to a value of less than or equal to 1.0 for acceptable fatigue design to provide assurance that fatigue crack initiation is unlikely. A CUF greater than 1.0 indicates an increased likelihood of fatigue crack initiation, but it does not necessarily indicate that a fatigue cracks has formed.

Q25 What does it mean when a CUF value is less than 1.0?

A25 [GS, 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. SRP-LR, Rev. 2 at page 4.3-1 (Ex. NYS000161).

Q26 What does it mean that a crack has not formed?

A26 [GS, AH, OY, CN] This means that when a CUF value is less than 1.0, it is highly unlikely that a fatigue crack has initiated (or formed) at 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. SRP-LR, Rev. 2 at page 4.3-1(Ex. NYS000161).

Q27 What is the Staff’s definition of fatigue life?

A27 [GS, 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.

Q28 What portion of the ASME Code addresses the initiation of fatigue cracks?

A28 [GS, AH, OY, CN] ASME Section III, Subsection NB addresses the initiation of fatigue cracks and provides the process to calculate a CUF value.

Q29 Does the Staff consider a fatigue crack to be present and growing when the CUF value is less than 1.0?

A29 [GS, 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 additional stress cycles.

Q30 Does a CUF value of 1.0 indicate the immediate failure of a component?

A30 [GS, AH, OY, CN] No, a CUF value of 1.0 is not indicative of immediate failure of the component.

Q31 What does it mean when CUF is greater than 1.0?

A31 [GS, AH, OY, CN] A CUF above the value of 1.0 allows for the increasing possibility that a crack may form. SRP-LR, Rev. 2 at Page 4.3-1 (Ex. NYS000161).

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Q32 What does it mean "a crack may form"?

A32 [GS, AH, OY, CN] The Staff is not aware of any scientific evidence to indicate that a fatigue crack has initiated or formed when the CUF value exceeds 1.0. The Staff conservatively assumes that a fatigue crack may have formed and may be growing when a CUF value is greater than 1.0.

Q33 Once a fatigue crack has formed or initiated, what happens if the material is subjected to more cycles?

A33 [GS, AH, OY, CN] Once a fatigue crack has initiated or formed, the fatigue crack will grow and propagate under further cyclic loading.

Q34 What part of the ASME Code governs the growth and propagation of a crack under cyclic loading?

A34 [GS, AH, OY, CN] Appendix A and Appendix C of the ASME 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. Appendix A at 301 (Ex. NRC000149). 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 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. ASME Appendix L at 421 (Ex. NRC000113). One option provided by ASME Appendix L is to perform fatigue usage factor evaluation, in accordance with ASME Section III, for reactor coolant system primary pressure boundary components and piping in operating plants. ASME Appendix L at 422 (Ex. NRC000113). Another option provided by ASME 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. ASME 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.

Q35 Does the Staff prohibit the use of these appendices in the ASME Section XI?

A35 [GS, AH, OY, CN] ASME 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 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."

Q36 How is the environmentally-assisted fatigue usage factor (CUF_{en}) calculated?

A36 [GS, AH, OY, CN] A value of CUF_{en} is computed in three parts. The first part is to calculate the CUF using the methodology from ASME Section III, as described in our response to Q24. The second part is to calculate the environmental adjustment factor (F_{en}) by using the guidance recommended in the GALL Report. The third part is to calculate the CUF_{en} as the product of the CUF and the F_{en} factor.

Q37 What is the Staff's guidance in the GALL Report regarding metal fatigue?

A37 [GS, 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 GALL Report Rev. 2 provide recommendations for an acceptable aging management program to manage the effects of metal fatigue. GALL Report Rev. 1 at X M-1 (Ex. NYS00146A-C) and GALL Report Rev. 2 at X M-1 (Ex. NYS00147A-D). The aging effect of metal fatigue is associated with the actual, accumulated numbers and severities of thermal transients that occur for each component, such as when the plant heats up and cools down, which are not dependent upon the passage of time alone. The design of the plant originally postulated a certain number of occurrences and severities for thermal and pressure transients that were expected to occur over the design lifetime of the plant. Specifically, in order for plant component fatigue limits to not be exceeded, the aging management program described in GALL Report AMP X.M1 recommends monitoring and tracking the numbers and severities of all applicable thermal and pressure transients.

Q38 What Commission documents identify the need to consider the effects of reactor water environment on metal fatigue?

A38 [GS, 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.

Q39 How are the effects of reactor water environment on metal fatigue addressed in the GALL Report?

A39 [GS, AH, OY, CN] The GALL Report Rev. 1 and GALL Report Rev. 2 include 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. GALL Report Rev. 1 at X M-1 (Ex. NYS00146A-C) and GALL Report Rev. 2 at X M-1 (Ex. NYS00147A-D).

This inclusion in the fatigue monitoring AMP is for convenience and for ease of review, since the same AMP used for the TLAAs associated with metal fatigue CUF values also applies to CUF_{en}.

However, the effects of reactor water environment on metal fatigue are not TLAAs in accordance with 10 C.F.R. 54.3(a), as described in our response to Q15, because the analyses were not developed in the original design and CLB of plants, including IP2 and IP3.

Q40 What do you mean that the GALL Report treats the effects of reactor water environment on metal fatigue as if it were a TLAA? Why aren't these analyses TLAAs?

A40 [GS, AH, OY, CN] The effect of reactor water environment on metal fatigue is not a TLAA because analyses to evaluate these effects are not a part of the plant's CLB, and thus do not meet the definition of a TLAA in 10 C.F.R. 54.3(a):

> *Time-limited aging analyses, for the purposes of this part, are those licensee calculations and analyses that:*
>
> *(1) Involve systems, structures, and components within the scope of license renewal, as delineated in § 54.4(a);*
>
> *(2) Consider the effects of aging;*

(3) Involve time-limited assumptions defined by the current operating term, for example, 40 years;

(4) Were determined to be relevant by the licensee in making a safety determination;

(5) Involve conclusions or provide the basis for conclusions related to the capability of the system, structure, and component to perform its intended functions, as delineated in § 54.4(b); and

(6) Are contained or incorporated by reference in the CLB.

This is consistent with the Commission's Order in Vermont Yankee, CLI-10-17, which states that the environmentally-assisted fatigue analyses need not be completed prior to the issuance of a renewed license because the analyses are not a part of the CLB. 72 N.R.C. 1 (July 8, 2010). The applicant must, however, demonstrate that it will adequately manage the effects of the reactor water environment on metal fatigue in accordance with 10 C.F.R. 54.21(a)(3).

Q41 What is the Staff's guidance in the SRP-LR regarding metal fatigue?

A41 [GS, AH, OY, CN] SRP-LR Rev. 1 and SRP-LR 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 assumed numbers and severities of transients for the current operating term, and the validity of such metal fatigue analyses 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. 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).

It should be pointed out that Section 4 of the SRP-LR Rev. 1 and SRP-LR Rev. 2, includes environmentally-assisted fatigue because of its close relation with metal fatigue and CUF analyses. However, the effect of reactor water environment on metal fatigue is not a TLAA as defined in 10 C.F.R. 54.3(a) because EAF analyses are not a part of the plant's CLB; thus not meeting the definition of a TLAA. This is consistent with the Commission's Order in Vermont Yankee, CLI-10-17, which states that the environmentally-assisted fatigue analyses need not be completed prior to the issuance of a renewed license because the analyses are not a part of the CLB.

Q42 How are the effects of reactor water environment on metal fatigue addressed in the SRP-LR?

A42 [GS, AH, OY, CN] The SRP-LR provides specific guidance on addressing the effects of reactor water environment. SRP-LR Rev. 1 and SRP-LR Rev. 2 recommend that the specific components identified in NUREG/CR-6260, as a minimum, should be considered for the effects of reactor environment on metal fatigue. 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 assessed the significance of the interim fatigue curves developed in the early 1990's (and that pre-date the F_{en} method) 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 (NSSS) vendors (i.e., Westinghouse, General Electric, Combustion Engineering, and Babcock & Wilcox). The sample of components in NUREG/CR-6260 was chosen to provide a representative overview of components that had higher CUFs and/or were important from a risk perspective, from representative older and newer-vintage facilities designed by each of the four U.S. NSSS vendors. NUREG/CR-6260 at iii (Ex. NYS000355).

SRP-LR Rev. 1 and SRP-LR 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 for use in determining the environmental effects for austenitic stainless steel components and NUREG/CR-6583 for use in determining the environmental effects for carbon and low-alloy steel components. SRP-LR Rev. 1 at 4.3-7 (Ex. NYS000195). These two reports define the F_{en} methodology for carbon, low-alloy, and austenitic stainless steels.

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 as an acceptable alternative for austenitic stainless, carbon, and low-alloy steels, with additional guidance on the appropriate fatigue curves to be used with the F_{en} method. 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. SRP-LR Rev. 2 at 4.3-9 through 4.3-10 (Ex. NYS000161). NUREG/CR-6909 provides updated F_{en} methodology compared to NUREG/CR-5704 and NUREG/CR-6583, and it provided the technical basis for RG 1.207, which contains guidance for applying the F_{en} methodology to new reactor components.

Q43 How does the Staff use the guidance to review an LRA related to metal fatigue and fatigue monitoring?

A43 [GS, AH, OY, CN] To follow this guidance, the Staff performs an on-site review of the applicant's program basis documents, plant procedures, and detailed documentation related to metal fatigue and the AMP for fatigue monitoring. The Staff also performs review of the applicant's LRA and their 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.

Q44 How does the Staff use the guidance to review an LRA related to the effects of reactor water environment on metal fatigue?

A44 [GS, AH, OY, CN] The Staff reviews information provided in the LRA to assess consistency with the guidance provided in the SRP-LR, which identifies the review guidance for effects of reactor water environment on metal fatigue in the same section as the review guidance for TLAAs. 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 CLB, which is one aspect of the definition of a TLAA in 10 C.F.R. 54.3, as described in the response to Q40.

Q45 Does Indian Point's LRA identify environmentally-assisted fatigue analyses as TLAAs?

A45 [GS, AH, OY, CN] Yes, Indian Point's LRA identifies environmentally-assisted fatigue analyses in LRA Section 4, where TLAAs are located.

Q46 Does the Staff agree that the environmentally-assisted fatigue analyses for Indian Point are TLAAs?

A46 [GS, AH, OY, CN] Consistent with the Staff's guidance in SRP-LR Rev. 1 and SRP-LR Rev. 2, EAF analyses were identified as TLAAs in Section 4 of the LRA. The Staff noted that EAF was included in Section 4 of the SRP-LR Rev. 1 and SRP-LR Rev. 2, because of its close relation with metal fatigue and CUF analyses. The Staff notes that including EAF analyses in Section 4 of the LRA is appropriate because of its connection with metal fatigue analyses in the CLB and the Fatigue Monitoring Program.

However, the effect of reactor water environment on metal fatigue is not a TLAA as defined in 10 C.F.R. 54.3(a) because EAF analyses are not a part of the plant's CLB; thus not meeting the definition of a TLAA. This is consistent with the Commission's Order in Vermont Yankee, CLI-10-17, which states that the environmentally-assisted fatigue analyses need not be completed prior to the issuance of a renewed license because the analyses are not a part of the CLB. 72 N.R.C. 1 (July 8, 2010). However, Entergy must demonstrate that it will adequately manage the effects of the reactor water environment on metal fatigue in accordance with 10 C.F.R. 54.21(a)(3).

Q47 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, as they relate to New York Contention 26B and Riverkeeper Contention TC-1B?

A47 [GS, 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 they fulfill the applicable regulatory criteria in Part 54 of the Commission's regulations.

The Staff's conclusions and bases for its conclusions for Entergy's Fatigue Monitoring Program are documented in Section 3.0.3.2.6 of the Staff's SER. SER at 3-76 through 3-79 (Ex. NYS00326A-F). The Staff's conclusions and bases for its conclusions for Entergy's metal fatigue TLAAs are documented in SER Section 4.3. SER at 4-18 through 4-41 (Ex. NYS00326A-F). Finally, the Staff's conclusions and bases for its conclusions on Entergy'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. 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) (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) (Ex. NRC000124) ("Scoping Audit Report") provide additional information that supports the Staff's conclusions in the Staff's SER and SER Supp. 1. Furthermore, details about the results of the IP71002 inspection performed for IP2 and IP3 regarding the Fatigue Monitoring Program are contained in IP71002 Report. IP71002 Report at 4 (Ex. NRC000107).

Q48 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?

A48 [GS, 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. 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. Order at 2.

Q49 What information was provided in the Indian Point LRA on metal fatigue and CUF_{en} that is pertinent to NYS-38/RK-TC-5?

A49 [GS, 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. LRA at B-45 (Ex. ENT00015A and Ex. ENT00015B). 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. LRA at 4.3-24 through 4.3-25 (Ex. ENT00015A and Ex. ENT00015B).

Q50 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?

A50 [GS, 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 Entergy's Fatigue Monitoring program are documented in Section 3.0.3.2.6 of the Staff's SER. SER at 3-76 through 3-79 (Ex. NYS00326A-F). The Staff's conclusions and bases for its conclusions for Entergy's metal fatigue TLAAs are documented in SER Section 4.3. SER at 4-18 through 4-41 (Ex. NYS00326A-F). Finally, the Staff's conclusions and bases for its conclusions on Entergy'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. SER at 4-41 through 4-46 (Ex. NYS00326AF) 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. Furthermore, details about the results of the IP71002 inspection performed for IP2 and IP3 regarding the Fatigue Monitoring Program are contained in IP71002 Report. IP71002 Report at 4 (Ex. NRC000107).

Q51 With respect to the Fatigue Monitoring Program, the metal fatigue TLAAs and environmentally-assisted fatigue analyses, do you agree with New York State and

Riverkeeper contention in NYS-26B/RK-TC-1B that Entergy's LRA does not include an adequate plan to monitor and manage the effects of aging due to metal fatigue on key reactor components in violation of 10 C.F.R. § 54.21(c)(1)(iii)?

A51 [GS, AH, OY, CN] No, we do not agree with New York State and Riverkeeper contention in NYS-26B/RK-TC-1B that Entergy's LRA does not include an adequate plan to monitor and manage the effects of aging due to metal fatigue on key reactor components in violation of 10 C.F.R. § 54.21(c)(1)(iii). There are six aspects to the Intervenors' contention. Each one will be discussed individually in summary form below, and in detail throughout the rest of our testimony. The six aspects are described in the Board's Order dated November 4, 2010.

As described in the Board's November 4, 2010, Order, NYS-26B/RK-TC-1B characterizes Entergy's re-analyses as inadequate under NRC regulations and the GALL Report because the re-analyses inappropriately limited the number of components subject to fatigue analyses. Order at 8. Entergy performed environmentally-assisted fatigue analyses for those components that were identified in NUREG/CR-6260, which was based on research performed by Idaho National Engineering Laboratory. Idaho National Engineering Laboratory selected these components to evaluate the effects of proposed interim environmental fatigue curves, and to give a representative overview of the impact of those curves on components that had higher CUFs and/or were important from a risk perspective. As provided in Commitment No. 43 and discussed later, Entergy will confirm prior to the period of extended operation whether there are additional locations for which it will need to address the effects of reactor water environment on metal fatigue. This approach, is consistent with the Commission's Order in Vermont Yankee, CLI-10-17, which states that the environmentally-assisted fatigue analyses need not be completed prior to the issuance of a renewed license in order to demonstrate an adequate AMP because these analyses are not TLAAs and they are not part of the CLB. Therefore, for metal fatigue, it is the Staff's opinion that Entergy did not inappropriately limit the number of components subject to environmentally-assisted fatigue analyses.

As described in the Board's November 4, 2010, Order, NYS-26B/RK-TC-1B characterizes Entergy's re-analyses as inadequate under NRC regulations and the GALL Report because Entergy has neither explained the methodology used to conduct its cumulative usage factor analyses nor included a detailed error analysis. Order at 8. As explained in our response to Q108, Entergy's re-analyses in WCAP-17199-P, *Environmental Fatigue Evaluation for Indian Point Unit 2*, (June 2010) (PROPRIETARY) (Ex. NYS000361) ("WCAP-17199") and WCAP-17200-P, *Environmental Fatigue Evaluation for Indian Point Unit 3*, (June 2010) (PROPRIETARY) (Ex. NYS000362) ("WCAP-17200") give the details that the Intervenors claim are lacking. Specifically, these reports describe the methodology used by Entergy to conduct the analyses for cumulative usage factor and environmentally-assisted fatigue. Furthermore, as discussed in our response to Q171, neither the Commission's regulations nor ASME Section III require a detailed error analysis to be performed for fatigue analyses. In addition, the Intervenors have not identified any requirements that specify the need for an error analysis to be performed for fatigue analyses, and the sample error analysis provided as an exhibit by the Intervenors contradicts their assertion that an error analysis is needed for a deterministic evaluation such as Entergy's fatigue evaluations.

As described in the Board's November 4, 2010, Order, NYS-26B/RK-TC-1B characterizes Entergy's re-analyses as inadequate under NRC regulations and the GALL Report because the re-analyses exclude a fatigue evaluation of important structures and fittings within the reactor pressure vessel. Order at 8. The research performed by Idaho National Engineering Laboratory, as documented in NUREG/CR-6260, did not identify the need to include structures and fittings within the reactor pressure vessel (i.e., reactor vessel internals that do not form a part of the reactor coolant pressure boundary) to address the effects of reactor water environment on metal fatigue. Instead, the report focused on reactor coolant pressure boundary components of operating nuclear power plants. The Intervenors also have not identified either specific structures and fittings within the reactor pressure vessel which should be considered, or events at IP2, IP3 or elsewhere in the industry in which these structures and fittings have either failed or have even been challenged due to environmentally-assisted fatigue.

As described in the Board's November 4, 2010, Order, NYS-26B/RK-TC-1B characterizes Entergy's re-analyses as inadequate under NRC regulations and the GALL Report because the re-analyses exclude evaluation of the potential failure of highly fatigued structures and fittings under certain types of "large thermal/pressure shock-type loads." Order at 8. The Intervenors' claim is incorrect because, as we discuss in further detail in our response to Q145, component evaluations in accordance with ASME Section III includes evaluation of Design Basis Accident (DBA) events, and fatigue crack initiation is not the relevant criterion for evaluation of DBA events such as "large thermal/pressure shock-type loads. Furthermore, the Intervenors' claim is based on the assumption that Entergy's Fatigue Monitoring Program is not capable of managing the CUF. If the CUF value is maintained to be less than or equal to 1.0, the likelihood of a small initiated fatigue crack is low. Therefore, the component has not degraded to the point that there are any structural effects due to fatigue cracking and the component continues to be acceptable under Entergy's CLB.

As described in the Board's November. 4, 2010, Order, NYS-26B/RK-TC-1B characterizes Entergy's re-analyses as inadequate under NRC regulations and the GALL Report because the re-analyses contain lower safety margins that create more risk since the new CUFs have been "reduced by more than an order of magnitude." Order at 8. The analyst's goal when performing an ASME Section III fatigue analysis is to ensure the component will continue to meet the component fatigue limit. The re-analyses accomplish this goal by using more precise and refined methods to calculate a CUF value. The Staff's opinion is that there is sufficient safety margin inherent in the ASME Section III CUF limit of 1.0 (remember this limit indicates a likelihood of initiating a fatigue crack, not failing the structure or component) and a reanalysis of the CUF that demonstrates compliance with the limit does not correspond to a reduction in safety margin. In fact, as detailed in our response to Q209, iterative calculations are routine for fatigue evaluations, and the level of iteration does not reflect a compromise in the safety margins for the component. In addition, the testimony provided by the Intervenors' expert witnesses does not discuss specifically how ASME Section III safety margins are degraded, nor do they explain how the consequences of lower safety margins that they assert create more risk from the new CUFs being "reduced by more than an order of magnitude."

As described in the Board's November 4, 2010, Order, NYS-26B/RK-TC-1B characterizes Entergy's re-analyses as inadequate under NRC regulations and the GALL Report because Entergy has not committed to repair or replace components when the CUF approaches unity (1.0). Order at 8. No provision of NRC regulations, the GALL Report, nor the ASME Code requires the preemptive repair or replacement of components that have calculated CUF values less than or equal to 1.0, or even for components where the calculated CUF is greater than 1.0; therefore, the Intervenors' request for a commitment to repair or replace components when the CUF approaches unity is not required and not necessary. Appendix L of ASME Section XI, *Operating Plant Fatigue Assessment*, (Ex. NRC000113) ("ASME Appendix L") allows for continued component service if the CUF is less than or equal to 1.0, and even for cases when the CUF is greater than 1.0 and acceptable inspection and flaw tolerance evaluation have been provided. ASME Appendix L at 422 through 427 (Ex. NRC000113).

Q52 Does the Staff agree with the Intervenors' claim in contention NYS-38/RK-TC-5 that Entergy does not have a program that will manage the effects of aging related to metal fatigue and environmentally-assisted fatigue analyses?

A52 [GS, 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. LRA at B-44 to B-46 (Ex. ENT000015A-B). 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 and finding of acceptability of Entergy's metal fatigue program, including all amendments made by Entergy to its aging management programs.

Q53 Summarize Entergy's Fatigue Monitoring Program and why the Staff concluded that it will adequately manage metal fatigue.

A53 [GS, AH, OY, CN] Based on a review of the LRA, responses to requests for additional information and audit items related to the metal fatigue TLAAs, the Fatigue

Monitoring Program, the results from the IP 71002 inspection, and the environmentally-assisted fatigue analyses for IP2 and IP3, the Staff concluded that the Fatigue Monitoring Program will adequately manage the metal fatigue TLAAs and the environmentally-assisted fatigue analyses that Entergy dispositioned in accordance with 10 C.F.R. 54.21(c)(1)(iii). The basis for the Staff position is summarized as follows:

- The scope of Entergy'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 and severity of fatigue transients, and are dispositioned in accordance with 10 C.F.R. 54.21(c)(1)(iii). The program also includes the sample of locations identified in NUREG/CR-6260. Entergy submitted these amendments to the scope of program in Letter NL-08-057 from Entergy Nuclear Operations, Inc. to NRC, *Indian Point Nuclear Generating Units 2 and 3, License Renewal Application Amendment 3*, (March 24, 2008) (Ex. NRC000109) ("NL-08-057"). In Letter NL-11-032 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) (Ex. NRC000110) ("NL-11-032"), Entergy provided Commitment No. 43 to review its design basis ASME Code Class 1 fatigue evaluations to confirm whether the NUREG/CR-6260 locations that were 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, to bound the other locations. NL-11-032, Attachment 1 at 26 and Attachment 2 at 17 (Ex. NRC000110). The Staff's review of Commitment No. 43 is documented in the Section 4.3.3.2 of SER Supp. 1. SER Supp. 1 at 4-1 through 4-3 (Ex. NYS000160).

- The Fatigue Monitoring Program monitors the numbers and severities of actual plant transients. The numbers and severities are evaluated against those used in the fatigue calculations. These actions ensure that the severity and number of cycles for each transient experienced by the plant remains within the analyzed severity and number of cycles. By ensuring that the severity and number of experienced cycles for each transient remains within those used in the fatigue evaluation for that transient, the program ensures that the component CUFs remain below the values calculated in the fatigue evaluations. Section 3.2.5 of the Staff's AMP Audit Report documents Entergy's response to Audit Item 39, in which Entergy describes that the program monitors the numbers and severities of 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 Updated Final Safety Analysis Report (UFSAR) and Table 4.1-8 of the IP3 UFSAR, by performing a review of site data to determine the transients that have occurred since the last review, and then updates the list of total transients to date. AMP Audit Report at 52 and 53 (Ex. NRC000108).
- As described by Entergy's RAI response in Entergy's Letter NL-08-084 to the 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) (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 number of transient cycles or the transient severity. NL-08-084, Attachment 1 at 4 (Ex. ENT000194). Entergy's program for IP2 includes 'alert cycles', which are defined as the number of transient cycles which are projected to accumulate in the next two operating periods. The number of 'alert cycles' is calculated by taking the number of transient cycles accumulated during the prior operating cycle and multiplying by 2; the total projected number of transient cycles is determined by adding the 'alert cycles' to the total accumulated number of transients to date. If this projected number of transient cycles remains below the number of transient cycles used in the fatigue evaluation, no corrective action is required. If this projected number of transient cycles exceeds the number of transient cycles used in the fatigue evaluation, a condition report is generated to ensure that corrective actions are taken. Conversely, Entergy's program for IP3 does not include the use of alert cycles and does not allow continued plant operation if the number of transient cycles used in the fatigue evaluation for any 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. AMP Audit Report at 53 (Ex. NRC000108).

- As described by Entergy's RAI response in NL-08-084, the Fatigue Monitoring Program periodically ensures that the numbers and severities of all transients used in the metal fatigue TLAAs and environmentally-assisted fatigue analyses are not exceeded, thereby, ensuring that the calculated CUF, CUF_{en}, and the fatigue limit of 1.0 are not exceeded. NL-08-084, Attachment 1 at 3 through 4 (Ex. ENT000194). In order for the CUF and CUF_{en} of an analyzed component to reach the calculated CUF or CUF_{en}, the assumptions defined in the fatigue analyses associated with each of the analyzed transient types (i.e., plant heat-ups and cool-downs), in particular the numbers and severities of all of the transients, must actually occur. Since it is unlikely for all of these assumptions to be met simultaneously, significant margin exists in the calculated CUF and CUF_{en} values.

- The Fatigue Monitoring Program includes corrective actions that are initiated if the monitoring of the plant transients indicates the potential for a condition outside those analyzed in the underlying fatigue evaluation. Corrective actions include performing a more rigorous analysis to remedy the condition, or repair or replacement of affected components, before the CUF or CUF_{en} exceeds 1.0. These corrective actions were described in Entergy's Letter NL-08-021 to the NRC, *Indian Point Nuclear Generating Units 2 and 3, License Renewal Application Amendment 2*, (January 22, 2008) at Attachment 1, 2 through 3 (Ex. NYS000351) ("NL-08-021") and 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 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's LRA does not include an adequate plan to monitor and manage the effects of aging due to metal fatigue on key reactor components, in violation of 10 C.F.R. § 54.21(c)(1)(iii).

Q54 In Tables 4.3-13 and 4.3-14 of Indian Point's LRA, the CUF_{en} values for some of the listed components are greater than 1.0. Why is it acceptable to "refine" predicted CUF_{en} values to obtain values less than 1.0?

A54 [GS, AH, OY, CN] Design fatigue calculations for components generally use conservative assumptions in the calculation of the CUF, typically to make performing the calculation more simple. These conservative assumptions can relate to the severity of the transients (where less severe transients are grouped with and treated as higher severity transients), the number of transients (where the number of transients expected to occur over the service life of the component is intentionally increased for use in the calculation), and the stresses on the component generated by the transient (where simplifying assumptions are made to provide conservative estimates of the component stresses). If the calculated CUF values with these various conservative assumptions are less than or equal to the limit of 1.0, then the analysis is considered acceptable because it has met the acceptance limit defined by the ASME Code and the analyst stops work on the analysis. Although the analyst could have reduced conservatism in the calculation to achieve a more accurate value of CUF, this isn't typical or necessary because the objective of acceptability was achieved by demonstrating that the CUF is less than 1.0.

When a value of CUF_{en} is calculated for that same component, that value may exceed the allowable value of 1.0 because the F_{en} value, which is multiplied by the CUF value to calculate the CUF_{en}, can be larger than 1.0. Since the limit for CUF_{en} is also 1.0, the analyst can perform a refined analysis for the component to reduce the CUF to an acceptable level by removing one or more of the conservative assumptions that were used in the original calculation. In other words, the analyst continues the calculation originally developed for the component with the addition of an extra multiplier. The iterative process that results from the addition of the F_{en}

multiplier often involves the use of fewer conservative assumptions than originally applied so that a more refined value of CUF_{en} can be calculated. Since these iterative calculations involve the use of fewer conservative assumptions and methods, the refined value of CUF_{en} will be lower than the original value. For example, actual plant transients that occur are typically less severe than the design transients, which are defined on a generic basis by the vendor for the component design. The use of the actual severity of the transients experienced by the plant would typically result in a CUF_{en} value that is lower than that from the original design calculation. In addition, transients may occur less frequently than specified by the original design, which would lead to lower calculated CUF and CUF_{en} values for the component, or the design calculations may have grouped less severe transients with more severe transients because it simplified the CUF calculation. In the iterative analyses, these transients may be ungrouped and individually analyzed as separate transients.

These refinements are considered acceptable because they more accurately reflect the actual operating conditions of the component compared to the conservative assumptions used during the design of the component. These refinements are also the same steps that would have been used by the original analyst to achieve an acceptable result if the original simplifications had led to an original design CUF value that exceeded 1.0.

A detailed discussion regarding the methods an analyst can use to make adjustments to fatigue calculations to reduce the CUF is documented in Section 4.3 of NUREG/CR-6260. NUREG/CR-6260 states that the adjustments typically fall into two broad categories, conservative assumptions made by the analyst or ASME Code changes that have been made since the edition of the ASME Code of record that was used to design the plant (“Code of Construction”). NUREG/CR-6260 at 4-5 (Ex.

NYS000355). Two examples that NUREG/CR-6260 discusses to reduce CUF values are (1) separating the enveloped load pair with the overall combined number of cycles into more detailed load pairs, each with its own set of cycles, and (2) using actual cycles that the plant has experienced to date if the numbers of cycles extrapolated are less than the numbers of design basis cycles. NUREG/CR-6260 at 4-5 and 4-6 (Ex. NYS000355).

Section 7 of NUREG/CR-6909 discusses the conservatisms present in the ASME Section III fatigue evaluations, which include (1) design transients considerably more severe than those experienced in service, (2) grouping of transients, and (3) simplified elastic–plastic analysis that leads to higher stresses. Furthermore, Section 7 of NUREG/CR-6909 states that the margin in the current ASME Section III fatigue evaluations is quite conservative. NUREG/CR-6909 at 71 (Ex. NYS000357).

Entergy uses the Fatigue Monitoring Program to ensure that the assumptions (e.g., transient severity, number of transients) made in the underlying CUF or CUF_{en} evaluations remain valid based on the actual operation of the plant.

Q55 Is the use of NUREG/CR-6583 to calculate F_{en} for carbon and low-alloy steel and NUREG/CR-5704 to calculate F_{en} for stainless steel outdated when compared to NUREG/CR-6909 for the respective materials?

A55 [GS, AH, OY, CN] No, the use of NUREG/CR-6583 to calculate the F_{en} factor for carbon and low-alloy steels and NUREG/CR-5704 to calculate the F_{en} factor for stainless steels is not outdated when compared to NUREG/CR-6909 for the respective materials. Further discussion of this is provided in our response to Q157. The Staff’s guidance in GALL Report Rev. 2 and SRP-LR Rev. 2 includes the option to use NUREG/CR-6583 and NUREG/CR-5704 with the original ASME Section III fatigue design curves for the components. Alternatively, applicants may also use

NUREG/CR-6909 for determining the environmental adjustment factors. SRP-LR Rev. 2 at 4.3-9 through 4.3-10 (Ex. NYS000161) and GALL Report Rev. 2 at X.M1-1 (Ex. NYS00147A-D). Therefore, the use of these documents to calculate F_{en} values is not outdated and these documents are still referenced for use and are acceptable for use in the current guidance documents for the review of LRAs.

Q56 Do you agree with Dr. Hopenfeld's assertions that Entergy's re-analyses have a wide margin of error and therefore those CUF_{en} values close to unity would exceed unity if these errors had been corrected?

A56 [GS, AH, OY, CN] No we do not agree with Dr. Hopenfeld's assertions that Entergy's re-analyses have a wide margin of error and therefore those CUF_{en} values close to unity would exceed unity if these errors had been corrected. The license renewal process presumes that applicants perform their calculations accurately and consistent with their CLB in accordance with their Quality Assurance (QA) Program (e.g., Appendix B to 10 C.F.R. Part 50). As described throughout our testimony, and in particular in our responses to Q53, Q54, Q126, Q208, and Q210, there is margin within the re-analyses and within the methods in which Entergy's Fatigue Monitoring Program is managing metal fatigue.

In order for the actual CUF of a component to approach the calculated CUF, at a minimum, both of the following conditions must be satisfied:

- the severity of each and every transient that occurs at IP2 and IP3 must be equal to the transient severity used in the fatigue calculations, and
- the actual number of cycles for each and every transient must approach the number of each and every transient used in the fatigue calculation.

Entergy is managing cumulative fatigue damage with its Fatigue Monitoring Program by (1) tracking actual plant transients, (2) evaluating actual transient severities against the transient severities used in the fatigue evaluation, and (3) ensuring that the numbers of transients experienced by the plant remain within the numbers of transients used in the fatigue evaluations.

Entergy's program for IP2 includes 'alert cycles', which are defined as the number of transient cycles which are projected to accumulate in the next two operating periods. The number of 'alert cycles' is calculated by taking the number of transient cycles accumulated during the prior operating cycle and multiplying by 2; the total projected number of transient cycles is then determined by adding the 'alert cycles' to the total accumulated number of transients to date. If this projected number of transient cycles remains below the number of transient cycles used in the fatigue evaluation, no corrective action is required. If this projected number of transient cycles exceeds the number of transient cycles used in the fatigue evaluation, a condition report is generated to ensure that corrective actions are taken. Conversely, Entergy's program for IP3 does include the use of alert cycles and does not allow continued plant operation if the number of transient cycles used in the fatigue evaluation for any 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. AMP Audit Report at 53 (Ex. NRC000108).

The Fatigue Monitoring Program for both IP2 and IP3, as described in Entergy's implementing procedures, requires corrective actions when a single transient type (e.g., heat-up transient or cool-down transient) approaches the respective number of transients used in the fatigue evaluation for that transient type. Since a design fatigue calculation or environmentally-assisted fatigue evaluation for a component typically include multiple transients (e.g., not just a single transient), this approach provides additional margin against the CUF or CUF_{en} exceeding its limit. Entergy further describes in response to Audit Item 39 by Letter NL-07-153 from Entergy to NRC, *Indian Point Nuclear Generating Units 2 and 3, License Renewal Application Amendment 1*, (December 18, 2007) (Ex. NRC000111) ("NL-07-153"), that the program monitors the transients listed in Tables 4.3-1 (IP2) and 4.3-2 (IP3) of the LRA and Table 4.1-8 of the UFSAR for each unit by performing a review of the site operational data to determine the transients that have occurred since the last review, and then updating the list of total transients to date. Furthermore, Entergy stated that this review is done by the cognizant engineer at IP2 and IP3. As described, Entergy's Fatigue Monitoring Program continually validates that the assumptions used in these fatigue evaluations remain valid, thus ensuring that the design limit of 1.0 is not exceeded. NL-07-153, Attachment 3 at 7 (Ex. NRC000111).

Therefore, Entergy will have taken corrective actions, in accordance with its Fatigue Monitoring Program if the monitoring of the plant transients indicates the potential for a condition outside those analyzed in the underlying fatigue evaluation and prior to the calculated CUF, CUF_{en} and the fatigue limit of 1.0 being approached. Thus, the consequences of any error that is asserted by Dr. Hopenfeld is not significant based on the margin inherent in Entergy's methods for management of metal fatigue, the environmentally-assisted fatigue analyses and ASME Section III fatigue analyses.

Q57 What two things did Entergy state that it would do in Commitment No. 43?

A57 [GS, AH, OY, CN] Entergy stated that it will review its ASME Code Class 1 fatigue evaluations to confirm that the NUREG/CR-6260 locations that were evaluated for the effects of the reactor water environment are the limiting locations for IP2 and IP3. If more limiting locations are identified, the most limiting location will be evaluated for the effects of the reactor water environment and added to the Fatigue Monitoring

Program. Further, Entergy will use the NUREG/CR-6909 methodology, which currently represents the best methodology for nickel alloys, in the evaluation of any nickel alloy limiting locations. NL-11-032, Attachment 1 at 26 and Attachment 2 at 17 (Ex. NRC000110).

Q58 Why is this commitment limited to ASME Code Class 1 fatigue evaluations?

A58 [GS, AH, OY, CN] Limiting this commitment to ASME Code Class 1 fatigue evaluations, as described in our response to Q24, is consistent with Commission policy in this area since these evaluations are the only CLB fatigue evaluations. Specifically, the Commission limited the scope of the license renewal review to the effects of age-related degradation related to the license renewal operating period, 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 ASME Code Class 1 fatigue evaluations that are part of the applicant's CLB are the subject of this commitment.

Q59 How does the Staff know that Entergy's review of its fatigue calculations will be performed correctly or if the fatigue calculations are performed correctly?

A59 [GS, AH, OY, CN] Entergy is required by Appendix B to 10 C.F.R. Part 50 to implement a QA 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 analysis or calculation.

This means that any fatigue analyses that may be performed in the future, as well as the evaluation that was completed as part of Commitment No. 43, are governed by Entergy's QA Program. An individual performing any evaluation must have specialized experience and be specifically trained so that a quality analysis or evaluation is produced. In addition, Entergy's QA Program is required by Appendix B to 10 C.F.R. Part 50 to provide training of personnel performing activities affecting quality, as necessary, to assure that suitable proficiency is achieved and maintained. Entergy's QA Program must ensure that, for all analyses, there are sufficient records, and that those records are maintained to document all activities affecting quality. 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 the originator. Furthermore, Entergy's QA Program is required to provide measures for verifying or checking the adequacy of design, such as by the performance of design reviews or independent design verifications.

Q60 What are the "NUREG/CR-6260 locations" and what is their significance?

A60 [GS, AH, OY, CN] 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 for an older vintage Westinghouse plant:

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

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

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 newer and older vintage plants designed by the four U.S. nuclear steam supply system (NSSS) vendors (i.e., Westinghouse, General Electric, Combustion Engineering and Babcock & Wilcox). 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. NUREG/CR-6260 at xxi (Ex. NYS000355). The components in this report were intended to provide a representative overview of the effects of the reactor water environment on component fatigue lives for critical components from facilities designed by each of the four U.S. NSSS vendors. NUREG/CR-6260 at iii (Ex. NYS000355).

Entergy has addressed the effects of reactor water on metal fatigue for these NUREG/CR-6260 locations for both IP2 and IP3.

Q61 Which components "have been evaluated for the effects of reactor water environment on fatigue usage for Indian Point?"

A61 [GS, AH, OY, CN] The following components were 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

LRA at 4.3-24 through 4.3-25 (Ex. ENT00015A-B).

Q62 What is the purpose of Commitment No. 43?

A62 [GS, AH, OY, CN] 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 plant configuration. GALL Report Rev. 1 at X M-1 (Ex. NYS00146A-C).

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. GALL Report Rev. 2 at X M-1 (Ex. NYS00147A-D).

Therefore, the purpose of Commitment No. 43 is for Entergy 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 in that the limiting CUF locations for IP2 and IP3 are included by considering the NUREG/CR-6260 locations. If, as a part of this commitment, Entergy identifies that additional locations are more limiting from a CUF perspective and should be managed, then it will also manage these additional locations with its Fatigue Monitoring Program.

Based on the Staff's experience related to the identification of locations to consider the effects of reactor water environment on metal fatigue for other licensees, 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.

Q63 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?

A63 [GS, AH, OY, CN] The Staff noted that none of the NUREG/CR-6260 locations for IP2 and IP3 were fabricated from nickel alloy. However, because the Staff did not previously identify in any of its guidance documents how to determine the environmental factor for nickel alloy components, and because of the possibility that other more limiting components that are fabricated from nickel alloy may be identified as a part of this commitment, the Staff specifically requested Entergy to identify the methodology it would use for determining the F_{en} factor for nickel alloy components, if necessary. RAI Letter at 11 through 13 (Ex. NYS000150). Entergy specified the use of NUREG/CR-6909 for nickel alloy components in this commitment. NL-11-032, Attachment 1 at 26 and Attachment 2 at 17 (Ex. NRC000110).

The 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 (DO) level. 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 and accurate values for the environmental factor will be used for nickel alloy locations. NL-11-032, Attachment 1 at 26 and Attachment 2 at 17 (Ex. NRC000110).

Q64 Does the NRC inspect how Entergy uses this NUREG/CR-6909 methodology?

A64 [GS, AH, OY, CN] The methodologies described in NUREG/CR-5704 and NUREG/CR-6583 are endorsed for use in GALL Report Rev. 1 and SRP-LR Rev. 1, and the methodologies described in NUREG/CR-5704, NUREG/CR-6583 and NUREG/CR-6909 are endorsed for use in GALL Report Rev. 2 and SRP-LR Rev. 2. GALL Report Rev. 1 at X.M-1 (Ex. NYS00146A-C), GALL Report Rev. 2 at X.M-1 (Ex. NYS00147A-D), SRP-LR Rev. 1 at 4.3-7 (Ex. NYS000195) and SRP-LR Rev. 2 at 4.3-9 through 4.3-10 (Ex. NYS000161). The calculations that apply the methodologies of these reports, as applicable, are available at Entergy's site for the Staff to review, including during inspections performed in accordance with IP71003. The NRC inspects renewed license holder completion of commitments using Inspection Procedure 71003. IP71003 at 1 (Ex. ENT000251). For the case of IP2, the Staff performed an on-site inspection regarding the completion of commitments in accordance with TI 2516/001 in 2013. In the case of IP3, which is in timely renewal, the Staff will perform an on-site inspection regarding the completion of commitments in accordance with IP71013 in the fall of 2015.

Q65 How does the NRC know that Entergy is using the NUREG/CR-6909 methodology correctly?

A65 [GS, AH, OY, CN] As discussed in our response to Q64, for IP2, the Staff performed an on-site inspection regarding the completion of commitments and Entergy's use of NUREG/CR-6909 in accordance with TI 2516/001 in 2013. The Staff will discuss the results of these inspections in its testimony below. In the case of IP3, which is in timely renewal, the Staff will perform an on-site inspection in accordance with IP71013 regarding the completion of commitments and Entergy's use of NUREG/CR-6909 in 2015. These inspections afford an opportunity for the Staff to verify Entergy's fulfillment of their license renewal commitments. 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, Entergy's analyses, like all other records, are available at Entergy's site for the Staff to review during routine inspections as part of the ongoing oversight process.

Q66 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?"

A66 [GS, AH, OY, CN] As described in our response to Q60, the locations in NUREG/CR-6260 were chosen to provide a representative sample of components that had higher CUFs and/or were important from a risk perspective from a representative older and newer vintage plant for each NSSS vendor. In the case of IP2 and IP3, the representative plant is the older vintage Westinghouse plant because the IP2 and IP3 piping systems were designed to the requirements of USAS B31.1. Since the locations evaluated in NUREG/CR-6260 represent a generic evaluation, the commitment will cause Entergy to consider the plant-specific configurations and CUF analyses of record at IP2 and IP3 to ensure that Entergy's evaluation covers the locations that are most 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 the older vintage Westinghouse plant in NUREG/CR-6260 are sufficient for IP2 and IP3. If Entergy's evaluation 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. Most typically, the fatigue susceptibility of a component is assessed by the value of CUF.

Q67 Do the documents referenced in the GALL AMP X.M1 (e.g., NUREG/CR-5704, NUREG/CR-6260, NUREG/CR-6583, and NUREG/CR-6909) form the basis of this AMP?

A67 [GS, AH, OY, CN] These documents do not form the entire basis for AMP X.M1 for Fatigue Monitoring, but are a part of its basis. The Fatigue Monitoring AMP described in the GALL Report in X.M1 manages fatigue cracking of metal components by counting transients that occur in the plant and monitoring their severity to verify that the existing fatigue usage calculations, and those that consider environmental effects when applicable, remain within the allowable fatigue limit. GALL Report Rev. 2 at X.M1 (Ex. NYS00147A-D)

In order to accomplish this, Entergy's program counts the number of cycles and monitors transient severity to ensure that the assumptions in the underlying fatigue analyses, including environmental effects where applicable, remain valid; therefore, the CUF and CUF_{en} values, which were calculated to be below the fatigue limit of 1.0, also remain valid.

The cited documents (e.g., NUREG/CR-6260, NUREG/CR-5704, NUREG/CR-6583 and NUREG/CR-6909) only provide the bases to account for the effects of the reactor water environment on metal fatigue.

Q68 What commitments has Entergy provided in relation to its Fatigue Monitoring Program?

A68 [GS, AH, OY, CN] Entergy provided the following commitments in relation to its Fatigue Monitoring Program:

Commitment No. 6 -

- Enhance the Fatigue Monitoring Program for IP2 to monitor steady state cycles and feedwater, cycles or perform an evaluation to determine monitoring is not required for these events. Review the number of allowed events and resolve discrepancies between reference documents and monitoring procedures.
- Enhance the Fatigue Monitoring Program for IP3 to include all the transients identified. Assure all fatigue analysis transients are included with the lowest limiting numbers of transients. Update the numbers of transients accumulated to date.

SER at A-6 (Ex. NYS00326A-F)

Commitment No. 33 -

- At least two years prior to entering the period of extended operation, for the locations identified in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3), under the Fatigue Monitoring Program, IP2 and IP3 will implement one or more of the following:

(1) Consistent with the Fatigue Monitoring Program, Detection of Aging Effects, update the fatigue usage calculations using refined the fatigue analyses to determine valid CUFs and CUF_{en}s that remain less than the fatigue limit of 1.0. This includes applying the appropriate F_{en} factors to valid

CUFs to calculate CUF_{en}s determined in accordance with one of the following.

- For locations in LRA Table 4.3-13 (IP2) and LRA Table 4.3-14 (IP3) with existing fatigue analyses that remain valid for the period of extended operation, use the existing CUF.
- Additional plant-specific locations with a valid CUF may be evaluated. In particular, the pressurizer lower shell will be reviewed to ensure the surge nozzle remains the limiting component.
- Representative CUF values from other plants, adjusted to or enveloping the IP plant-specific transients and other applicable loads, may be used if they are demonstrated to be applicable to IP.
- An analysis using an NRC-approved version of the ASME Code or NRC-approved alternative (e.g., NRC-approved code case) may be performed to determine a valid CUF.

(2) Consistent with the Fatigue Monitoring Program, Corrective Actions, repair or replace the affected locations prior to exceeding a CUF or CUF_{en} of 1.0.

SER at A-22 and A-33 (Ex. NYS00326A-F).

Commitment No. 43 -

- Entergy will review the governing IP ASME Code Class 1 fatigue evaluations to determine whether the NUREG/CR-6260 locations that were evaluated for the effects of the reactor coolant 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 coolant environment on fatigue usage. Entergy will use the NUREG/CR-6909 methodology in the evaluation of any limiting locations fabricated from nickel alloy material.

SER Supp. 1 at A-24 (Ex. NYS000160).

Commitment No. 44 -

- Entergy will include a written explanation and justification of any user intervention in future evaluations using the WESTEMS™ "Design CUF" module.

SER Supp. 1 at A-25 (Ex. NYS000160)

Commitment No. 45 -

- Entergy will not use the ASME Section III, Subarticle NB-3600, *Piping Design* ("NB-3600"), option of the WESTEMS™ program in future design calculations until the issues identified during the NRC review of the program have been resolved.

SER Supp. 1 at A-25 (Ex. NYS000160).

Commitment No. 49 –

- Recalculate each of the limiting CUFs provided in section 4.3 of the LRA for the reactor vessel internals to include the reactor coolant environment effects (F_{en}) as provided in the IPEC Fatigue Monitoring Program using NUREG/CR-5704 or NUREG/CR-6909. In accordance with the corrective actions specified in the Fatigue Monitoring Program, corrective actions include further CUF re-analysis, and/or repair or replacement of the affected components prior to the CUFen reaching 1.0.

NL-13-052, *Reply to Request for Additional Information Regarding the License Renewal Application Indian Point Nuclear Generating Unit Nos. 2 & 3, Docket Nos. 50-247 and 50-286, License Nos. DPR-26 and DPR-64,* (May 7, 2013) (ADAMS Accession No. ML13142A202) (Ex. NYS000501) ("NL-13-052").

Q69 In addition to Commitment No. 43, NYS-38/RK-TC-5 identifies issues that are related to Entergy's Commitment No. 44 as part of its basis. What is Commitment No. 44?

A69 [GS, AH, OY, CN] On March 28, 2011, Entergy submitted Commitment No. 44 in letter NL-11-032. NL-11-032 at 1 (Ex. NRC000110). Commitment No. 44 states that Entergy will include written explanation and justification of any user intervention in future evaluations using the WESTEMS™ "Design CUF" module.

Q70 Did the NRC request relevant information related to Commitment No. 44?

A70 [GS, AH, OY, CN] No, the Staff did not request the information related to Commitment No. 44 from Entergy. In letter NL-11-032, Entergy voluntarily provided Commitment No. 44 in Attachment 1 of the letter. Entergy 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. NL-11-032 at 1 (Ex. NRC000110).

Q71 What is WESTEMS™?

A71 [GS, AH, OY, CN] WESTEMS™ is Westinghouse proprietary computer software. The software is used by Westinghouse engineers to perform ASME Section III design stress and fatigue analyses. The inputs to this computer program include plant operating data such as temperature and pressure acquired 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.

Q72 Does Entergy mention the use of WESTEMS™ in its LRA?

A72 [GS, AH, OY, CN] No. The LRA as submitted on April 23, 2007, does not provide information regarding the use of WESTEMS™. Entergy's amendments to the LRA during the course of the Staff's review also did not provide information regarding the use of WESTEMS™. Entergy also did not request the review and approval of WESTEMS™ in the LRA.

Q73 If Entergy did not mention the use of WESTEMS™ in its LRA, why did Entergy offer Commitment No. 44?

A73 [GS, 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™. NL-11-032 at 1 (Ex. NRC000110).

Q74 Did the Staff request additional information from other license renewal applicants related to the use of WESTEMS™?

A74 [GS, AH, OY, CN] Yes, the Staff requested additional information in November 2010, regarding the use of WESTEMS™ during the review of the LRA for the Salem Nuclear Generating Station.

Q75 Did Entergy respond to the concerns raised by the Staff for this other license renewal applicant?

A75 [GS, AH, OY, CN] Yes, Entergy responded by voluntarily providing Commitment No. 44. NL-11-032 at 1 (Ex. NRC000110).

Q76 Did the Staff make the information and issues related to WESTEMS™ widely available to the public?

A76 [GS, 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. 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.

Q77 Did the Staff make this operating experience related to the use of computer software to perform fatigue evaluations publicly available and widely distributed?

A77 [GS, 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 ASME Boiler and Pressure Vessel 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 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 ASME Section III. RIS-2011-14 at 1. (Ex. NRC000112)

Q78 Has the Staff identified any conclusions on the operating experience aspect of Entergy's Fatigue Monitoring program based on how Entergy has addressed the WESTEMS™ issues that were identified for another applicant?

A78 [GS, 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 Entergy'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. SRP-LR Rev. 2 at A.1-7 (Ex. NYS000161). Entergy 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.

Q79 In the Staff's opinion, did Commitment No. 44 provide any missing information in the LRA?

A79 [GS, AH, OY, CN] No, Commitment No. 44 does not provide any information that is missing in the LRA. Commitment No. 44 is not used to demonstrate the adequacy of Entergy's Fatigue Monitoring program.

Q80 What is the purpose of Commitment No. 44?

A80 [GS, 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 audit 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. 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 Entergy's program acceptable.

Q81 Has Entergy implemented Commitments Nos. 6, 33, 43, 44, 45 and 49 for IP2?

A81 [GS, AH, OY, CN] In letter NL-13-114, *Implementation of License Renewal Regulatory Commitments Indian Point Nuclear Generating Unit No. 2 Docket No. 50-247, License No. DPR-26*, (August 28, 2013) (ADAMS Accession No. ML13247A175) (Ex. NRC000184) ("NL-13-114"), Entergy confirmed that license renewal Commitment Nos. 6, 33, 43, 44, 45 and 49, which were required to be implemented prior to entry into the period of extended operation, are complete IP2.

Q82 Is there a similar confirmation for IP3?

A82 [GS, AH, OY, CN] No, there is not a similar confirmation for IP3. Entergy's current operating license for IP3 does not expire until December 12, 2015. Consistent with the implementation schedule for Commitment Nos. 6, 43, 44, 45 and 49, Entergy will implement these commitments prior to December 12, 2015. Entergy notified the Staff regarding its completion of Commitment No. 33 for IP3 in its letter NL-10-082, *License Renewal Application - Completion of Commitment #33 Regarding the*

Fatigue Monitoring Program Indian Point Nuclear Generating Unit Nos. 2 and 3, Docket Nos. 50-247 and 50-286, License Nos. DPR-26 and DPR-64, (August 10, 2010) (Ex. NYS000352) ("NL-10-082").

However, in letter NL-15-060, *Entergy Actions Required Prior to Entry into Period of Extended Operation Associated with License Renewal Application (LRA) for Indian Point Nuclear Generating Unit No. 3, Docket No. 50-286 License, No. DPR-64*, (May, 12, 2015) (Ex. NRC000188) ("NL-15-060"), Entergy acknowledged that the review and approval of the LRA by the Staff will not be completed prior to the expiration of the Unit 3 Operating License. Furthermore, Entergy provided the following two actions as regulatory commitments to ensure full implementation of its license renewal commitments prior to entering the Unit 3 period of extended operation:

Action 1

> By September 25, 2015, Entergy will submit a letter to the NRC confirming implementation of the Unit 3 license renewal commitments required to be implemented prior to entry into the PEO (i.e., December 12, 2015). Entergy will reference the most recent list of regulatory commitments which was provided in Entergy's reply to RAI 3.0.3-2 from Set 2014-01 for the Review of the Indian Point Nuclear Generating Unit Nos. 2 & 3, LRA.

Action 2

> By September 25, 2015, Entergy will submit a letter to the NRC confirming that the Unit 3 Updated Final Safety Analysis Report has been updated to implement the Unit 3 aging management programs into the licensing basis.

NL-15-060 at 1 through 4. (Ex. NRC000188)

Q83 How does the Staff have assurance that Entergy appropriately implemented these commitments for IP2?

A83 [GS, AH, OY, CN] As discussed in our response to Q64, for IP2, the Staff performed an inspection in accordance with TI 2516/001. One of the inspection objectives was to report the status of Entergy's implementation of all of their license renewal commitments, license conditions and selected aging management programs, as described in a plant's license renewal safety evaluation report. TI 2516/001 at 1 (Ex. NRC000151).

Q84 Were these inspections completed for IP2 for Commitment Nos. 6, 33, 43, 44, 45 and 49?

A84 [GS, AH, OY, CN] Yes, on-site inspections performed in accordance with TI 2516/001 were completed for IP2 in 2013. The Staff conducted three separate license renewal inspections that reviewed a total of 44 license renewal commitments for IP2. However, only the second and third on-site inspections were associated with the implementation of Commitment Nos. 6, 33, 43, 44 and 45. The first on-site inspection did not include commitments associated with metal fatigue or the Fatigue Monitoring Program and is not relevant for the purposes of the Staff's testimony.

The second license renewal inspection for IP2 was conducted from May 6-10 and May 20-23, 2013, and the results are documented in NRC Inspection Report 05000247/2013009 (Ex. NRC000181). The third license renewal inspection for IP2 was conducted from September 9-12, 2013, and the results are documented in NRC Inspection Report 05000247/2013010 (Ex. NRC000182).

The Staff's review, evaluation, and acceptance of Commitment No. 49 was not complete at the time the TI 2516/001 inspection for IP2 was performed. Thus, it was not appropriate to verify completion of Commitment No. 49 during the TI 2516/001 inspection for IP2. Additional discussion regarding Commitment No. 49 is provided in the Staff's response to Q90.

Q85 Please describe the Staff's findings and observations for Commitment Nos. 6, 33, 43, 44, and 45?

A85 [GS, AH, OY, CN] The inspection scope, findings and observations for Commitment No. 6 from the second TI 2516/001 inspection are documented in Inspection Report 05000247/2013009. The inspectors reviewed the commitment implementation plan and discussed this commitment with applicable plant staff and license renewal personnel. The inspectors noted that Entergy had awarded contracts to perform the calculations that will determine whether monitoring of steady state cycles and feedwater cycles is required. Entergy expected that the calculations would be completed prior to the period of extended operation. The inspectors also noted that Entergy planned to revise procedure 2-PT-2Y015, *Thermal Cycle Monitoring Program*, ("2-PT-2Y015") if the calculations demonstrate that a change in the numbers of allowable steady state cycles and feedwater cycles is identified. The inspectors stated that no findings were identified; however, the inspectors determined that additional inspection was merited, including review of the results of the calculations and any changes to 2-PT-2Y015. Inspection Report 05000247/2013009 at 5 (Ex. NRC000181).

The inspection scope, findings and observations for Commitment No. 6 from the third TI 2516/001 inspection are documented in Inspection Report 05000247/2013010. During the third license renewal inspection, the inspectors reviewed Westinghouse Calculation IPP-13-20, Revision 1, *Steady State Fluctuations and Feedwater Cycling Transient Disposition*, (August 14, 2013) ("IPP-13-20"), which reported the "Steady State Fluctuations and Feedwater Cycling

Transient Disposition," LTR-PAFM-13-87, Revision 2 ("LTR-PAFM-13-87"). IPP-13-20 reported that the steady state transient does not significantly affect the CUFs for the primary system, so that transient could be removed from the transient cycle counting program. However, it was also reported that the feedwater cycling transient must be tracked for the secondary side of the steam generators and feedwater piping, with a 25,000 cycle limit. The inspectors verified that procedure 2-PT-2Y015 tracked the feedwater cycling transient. The inspectors stated that no findings were identified. Inspection Report 05000247/2013010 at 2 (Ex. NRC000182).

The inspection scope, findings and observations for Commitment No. 33 from the second TI 2516/001 inspection are documented in Inspection Report 05000247/2013009. The inspectors reviewed the calculations and discussed this commitment with applicable plant staff. These calculations are documented as WCAP 17149-P, Rev. 1, *Evaluation of Pressurizer Insurge/Outsurge Transients for Indian Point Unit 2*, (July 2010) (PROPRIETARY) (Ex. NYS000363) ("WCAP-17149") and WCAP-17199. The inspectors reviewed these calculations and noted that both documents concluded that all CUF_{en}s were below the fatigue limit of 1.0 for all transients postulated for 60 years of operation. The inspectors noted that Entergy's action plan included revising the Thermal Cycle Monitoring Program procedure to reflect changes in the number of projected transients used in WCAP-17199. The inspectors stated that no findings were identified; however, the inspectors determined that additional inspection was merited regarding any changes to the Thermal Cycle Monitoring Program procedure to ensure that it appropriately reflects the input values used by WCAP-17199. Inspection Report 05000247/2013009 at 17 (Ex. NRC000181).

The inspection scope, findings and observations for Commitment No. 33 from the third TI 2516/001 inspection are documented in Inspection Report

05000247/2013010. During the third license renewal inspection, the inspectors reviewed 2-PT-2Y015. The inspectors noted that the procedure was updated to reflect the number of transients used in WCAP-17199. The procedure referenced WCAP-17199 and included a table in Attachment 1 that included the actual number of plant transients rather than the number of cycles used in the underlying fatigue calculations. The procedure also was revised to better reflect operational assumptions used and the bases for the revised calculations of WCAP-17199. The inspectors stated that no findings were identified. Inspection Report 05000247/2013010 at 5 (Ex. NRC000182).

The inspection scope, findings and observations for Commitment No. 43 from the second TI 2516/001 inspection are documented in Inspection Report 05000247/2013009. The inspectors noted that Entergy awarded contracts to perform the calculations that will support closure of this commitment. The inspectors also noted that Entergy and its vendor completed initial screening calculations to determine the locations that may require further analysis. The inspectors stated that no findings were identified; however, the inspectors determined that additional inspection was merited regarding review of the results of the calculations and any changes to the Thermal Cycle Monitoring Program procedure. Inspection Report 05000247/2013009 at 21 (Ex. NRC000181).

The inspection scope, findings and observations for Commitment No. 43 from the third TI 2516/001 inspection are documented in Inspection Report 05000247/2013010. During the third license renewal inspection, the inspectors reviewed calculation CN-PAFM-13-32, Revision 0, *Indian Point Unit 2 (IP2) and Unit 3 (IP3) Refined EAF Analyses and EAF Screening Evaluations*, (2013) (Ex. NYS000511) (“CN-PAFM-13-32”). This calculation documented the evaluation of locations previously screened by calculation CN-PAFM-12-35, Revision 1, *Indian*

Point Unit 2 and Unit 3 EAF Screening Evaluations, (November 2012) (Ex. NYS000510”) (“CN-PAFM-12-35”) that might be more limiting than the locations identified in NUREG/CR-6260. The inspectors noted the CUF_{en} result for the pressurizer nozzle was 0.999 at 60 years. Entergy was aware that accumulation of the actual number of transients at a rate greater than the number of transients assumed in the underlying fatigue calculation requires more refined analysis, application of non-destructive examination, or repair/replacement of the pressurizer nozzle. The inspectors stated that no findings were identified. Inspection Report 05000247/2013010 at 7 (Ex. NRC000182).

The inspection scope, findings and observations for Commitment Nos. 44 and 45 from the second TI 2516/001 inspection are documented in Inspection Report 05000247/2013009. The inspectors noted that Entergy revised procedure 2-PT-2Y015 to incorporate the two restrictions described in Commitment Nos. 44 and 45. The inspectors also noted that this procedure would be referenced whenever fatigue calculations are developed or modified in the future. The inspectors further noted that if an engineer were to inadvertently not use 2-PT-2Y015, procedures EN-LI-100, *Process Applicability Determination* (“EN-LI-100”), and EN-AD-101, *Procedure Process* (“EN-AD-101”), would be used. These procedures control changes, tests and experiments, and procedure changes, and also require review of all NRC commitments. EN-LI-100 and EN-AD-101 act as backup to the thermal cycling monitoring procedure. The inspectors stated that no findings were identified. Inspection Report 05000247/2013009 at 21 (Ex. NRC000181).

In summary, the NRC inspectors reviewed Entergy’s implementation of Commitments Nos. 6, 33, 43, 44 and No. 45 for IP2 over the course of two on-site license renewal inspections performed in accordance with TI 2516/001 and determined that no findings were identified at the conclusion of the license renewal inspections.

Q86 What are the Staff's plans for IP3 inspections regarding Commitment Nos. 6, 33, 43, 44 and 45?

A86 [GS, AH, OY, CN] IP3's current operating license does not expire until December 12, 2015. The Staff is planning to perform on-site inspections of Commitment Nos. 6, 33, 43, 44 and 45 for IP3 in the fall of 2015.

By letter dated March 4, 2015, the Staff issued its annual assessment letter for IP2 and IP3 and provided a plan for inspections through June 30, 2016, which includes TI 2516/001. The inspection plan stated that the expected start and end dates for the IP3 inspections are November 16, 2015 and November, 20, 2015, respectively. (ADAMS Accession No. ML15061A401) (Ex. NRC000185).

Furthermore, the Staff noted that IP 71013 was issued on September 25, 2013, and states that TI 2516/001 expires on December 31, 2013, at which point timely renewal inspection activities will be conducted in accordance with IP 71013. An inspection objective of IP71013 is to determine whether commitments made by the licensee to implement actions such as proposed license conditions, other regulatory commitments accepted by the Staff during the course of license renewal, selected AMPs, and TLAAs are implemented or completed. IP71013 at 1 (Ex. NRC000183).

Thus, the Staff will perform an inspection of Commitment Nos. 6, 33, 43, 44 and 45 during the Fall of 2015 in accordance with IP71013 to verify that Entergy has adequately completed the commitments made to implement actions such as proposed license conditions, other regulatory commitments accepted by the Staff during license renewal, AMPs and TLAAs for IP3.

Q87 Did you review any calculations supporting Entergy's completion of Commitment Nos. 6, 33, 43, 44 and 45?

A87 [GS, AH, OY, CN] No, the Staff did not review Entergy's calculations that support completion of Commitment Nos. 6, 33, 43, 44 and 45. In addition, these calculations were not reviewed by the Staff as part of the LRA review process to ascertain if the Fatigue Monitoring Program complied with GALL Report. The Staff views these calculations as part of implementing corrective actions under the AMP and completion of commitments made by Entergy to implement actions associated with its LRA.

As described in Section 2516-06 of IMC 2516, the fundamental purpose of license renewal inspections is to ensure that there is reasonable assurance that the effects of aging will be managed consistent with the CLB during the period of extended operation. In addition, one of the objectives of the license renewal inspections is to determine the status of compliance with 10 C.F.R. Part 54 and other areas relating to maintaining and operating the plant such that the continued operation beyond the current licensing term will not be inimical to the public health and safety. Revised IMC 2516 at 6 (Ex. NRC000180).

IMC 2516 states that post-approval site inspections for license renewal will be conducted in accordance with IP71003. The post-approval site inspections will be performed by a team, in part, to verify the license conditions added as part of the renewed operating license, regulatory commitments, selected AMPs, and TLAAs are adequately implemented and/or completed. Revised IMC 2516 at 7 (Ex. NRC000180).

In addition, IMC 2516 states that IP71013 was developed for the inspection of license renewal programs for applicants with timely renewal applications. In the case where the plant is approaching timely renewal, IP71003 is not applicable.

Furthermore, license renewal inspections will be performed at plants with timely renewal applications to assess the applicant's readiness to operate beyond the expiration date of the original operating license through the timely verification that the applicant has made sufficient progress in implementing its AMPs, TLAAs, commitments, and proposed license conditions. Revised IMC 2516 at 8 (Ex. NRC000180).

Guidance is provided to inspectors to contact the license renewal program office and request appropriate technical expertise during inspections performed in accordance with IP71003 and IP71013. Revised IMC 2516 at 7 and 8 (Ex. NRC000180).

[CN] I attended a portion of the second on-site license renewal inspection for IP2 performed in accordance with TI 2516/001 during May 20-23, 2013. During this inspection, I provided support to the Region I staff related to a neutron fluence TLAA, as well as evaluations associated with environmentally-assisted fatigue for which Entergy had dispositioned in accordance with 10 C.F.R. 54.21(c)(1)(iii) and had associated license renewal commitments. Specifically, it was noted that Entergy has revised procedure 2-PT-2Y015 to incorporate the two restrictions described in Commitment Nos. 44 and 45. Inspection Report 05000247/2013009 at 21 (Ex. NRC000181). For Commitment No. 43, it was noted that Entergy awarded contracts to perform the calculations that will support closure of this commitment. Inspection Report 05000247/2013009 at 21 (Ex. NRC000181).

Q88 Why didn't the Staff review any of the calculations supporting Entergy's completion of Commitment Nos. 6, 33, 43, 44 and 45 for IP2?

A88 [GS, AH, OY, CN] The Staff's opinion is that the completion of Commitment Nos. 6, 33, 43, 44 and 45 is not needed prior to a licensing decision. Per the requirements in 10 C.F.R. 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 aging management.

Consistent with the processes established in IMC 2516 and the associated TI 2516/001, IP71003, and IP71013, site inspections were performed by a team, in part, to verify the license conditions added as part of the renewed operating license, regulatory commitments, selected AMPs, and TLAAs are adequately implemented and/or completed.

Q89 Why did one Staff member witness attend a portion of the second TI 2516/001 license renewal inspection performed on May 20-23, 2013?

A89 [GS, AH, OY, CN] Consistent with Section 2516/01-10 of TI 2516/001, the NRC inspectors requested assistance and contacted the Division of License Renewal for technical support during the inspection of Entergy's license renewal commitments. TI 2516/001 at 10 (Ex. NRC000151).

Q90 Why did Entergy provide Commitment No. 49?

A90 [GS, AH, OY, CN] In letter NL-12-140, *Reply to Request for Additional Information Regarding the License Renewal Application, Indian Point Nuclear Generating Unit Nos. 2 and 3, Docket Nos. 50-247 and 50-286, License Nos. DPR-26 and DPR-64*, (October 17, 2012) (ADAMS Accession No. ML13142A202) (Ex. NYS000499) ("NL-12-140") Entergy responded to NRC RAI 15, which requested clarification on how

Entergy planned to address reactor vessel internal (RVI) locations with existing CUFs as described in the Fatigue Monitoring Program. In the response to RAI 15, Entergy indicated that it planned to use the RVI Program, rather than the Fatigue Monitoring Program, because the inspections provided in the RVI Program were sufficient to ensure that the effects of aging due to fatigue would be adequately managed. NL-12-140, Attachment 1, at 4 through 12 (Ex. NYS000499).

In NRC letter, *Request for Additional Information for the Review of the Indian Point Nuclear Generating Unit Nos. 2 and 3, License Renewal Application, SET 2013-01*, (February 6, 2013) (ADAMS Accession No. ML13015A175) (Ex. NRC000187) ("RVI RAI Letter"), the Staff requested additional justification to demonstrate that the inspection plan and inspection frequency provided in Entergy's RVI Program were sufficient to ensure that the effects of aging due to fatigue on those internals locations with existing CUFs would be adequately managed during the period of extended operation, including either consideration of the environmental effects from the reactor coolant environment or a flaw tolerance evaluation. RVI RAI Letter at 2 through 3 (Ex. NRC000187).

In letter NL-13-052, Entergy revised its response to RAI 15 and provided Commitment No. 49. NL-13-052, Attachment 1 at 8 through 9

Q91 What is purpose of Commitment No. 49?

A91 [GS, AH, OY, CN] Commitment No. 49 addresses Part 5 of Applicant/Licensee Action Item No. 8 of the Staff's final safety evaluation of MRP-227-A*, Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines,* which is used in the development of the RVI Program. In summary, Part 5 of Applicant/Licensee Action Item No. 8 indicates that use of MRP-227-A to manage fatigue cracking aging effects for RVI components with a TLAA for fatigue is subject to the following:

1. For those CUF analyses that are TLAAs, the applicant may use the RVI Program only if the RVI components in the CUF analyses are periodically inspected for fatigue-induced cracking during the period of extended operation.
2. The periodicity of the inspections of these components must be justified as adequate to resolve the TLAA.

Part 5 of Applicant/Licensee Action Item No. 8 further indicates the use of the Fatigue Monitoring Program to manage fatigue cracking aging effects for RVI components with a fatigue TLAA by including the effects of the reactor coolant system water environment in the existing fatigue CUF analyses. MRP-227-A, NRC Safety Evaluation at 26. (Ex. NRC000114A-F).

Q92 Why was Commitment No. 49 not addressed as part of the inspections performed in accordance with TI 2516/001 for IP2?

A92 [GS, AH, OY, CN] At the time of the inspections performed in accordance with TI 2516/001 for IP2, which were during the 2012 to 2013 timeframe, the Staff did not complete its review, evaluation and acceptance of Entergy's RVI Program and Commitment No. 49. Thus, it would not have been appropriate to verify the completion of Commitment No. 49 for IP2 at the time of the TI 2516/001 inspections.

Q93 How does the Staff know that these fatigue calculations associated with Commitment No. 49 were performed correctly for Indian Point Units 2 and 3?

A93 [GS, AH, OY, CN] As with all licensees, Entergy is required by Appendix B to 10 C.F.R. Part 50 to implement a QA 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 analysis or calculation. This means that any evaluations, including any fatigue analyses, that are performed in the future, including the evaluation completed in response to Commitment No. 49 for IP2, are governed by Entergy's QA Program. An individual performing any evaluation must have specialized experience and be specifically trained so that a quality analysis or evaluation is produced. In addition, Entergy's QA Program requires training of personnel performing activities affecting quality, as necessary, to assure that suitable proficiency is achieved and maintained. Entergy is also required by their QA Program to ensure that all analyses have sufficient records and there is adequate record maintenance and retention for these records to properly document all activities affecting quality. Furthermore, Entergy's QA Program requires that design analyses and calculations are sufficiently detailed such that a person technically qualified in the subject area can review and understand the analyses, and reproduce and verify the adequacy of the results without recourse to the originator. Finally, Entergy's QA Program provides measures for verifying or checking the adequacy of design, such as by the performance of design reviews and independent design verifications.

Q94 How will the Staff verify Entergy's completion of Commitment No. 49 for IP2?

A94 [GS, AH, OY, CN] The Staff expects the NRC Inspectors will verify the completion of Commitment No. 49 for IP2 as a part of the IP71013 inspection for IP3, which is planned for the Fall of 2015. The NRC Inspectors may engage the Staff for support during the inspection, if needed, as their procedures allow.

Q95 Will the competition of Commitment No. 49 be verified for IP3?

A95 [GS, AH, OY, CN] Yes, consistent with the processes established in IMC 2516 and the associated IP71013, site inspections will be performed by a team, in part, to verify the license conditions added as part of the renewed operating license, regulatory commitments, selected AMPs, and TLAAs are adequately implemented and/or completed, which will include Commitment No. 49.

Q96 Describe the on-site audits that the Staff performed for the IP license renewal review.

A96 [GS, AH, OY, CN] The Staff performed two audits at the Indian Point site. One audit was performed for the AMPs, AMRs and TLAAs. This on-site audit was conducted during the following four time periods: August 27 - 31, 2007, October 22 - 26, 2007, November 27 - 29, 2007, and February 19 - 21, 2008. The on-site audit for the AMPs, AMRs and TLAAs is the principal review of the program elements of an AMP, as it covers seven of the ten program elements. The second audit was performed for the scoping and screening methodology. This on-site audit was conducted during the week of October 8 - 12, 2007. The scoping and screening methodology audit reviews the remaining three program elements of all AMPs.

Q97 Provide an overview of the Staff's on-site audit for the AMPs, AMRs and TLAAs of IP2 and IP3.

A97 [GS, AH, OY, CN] As a part of the Staff's routine process for its safety review of LRAs, a project team audited and reviewed its assigned AMPs, AMRs, and TLAAs for the IP2 and IP3 LRA. As a basis for its audit and review activities, the Staff used the requirements of 10 C.F.R. Part 54 and the guidance provided in the SRP-LR and the GALL Report.

The scope of work for the audit is contained in an Audit Plan. For the work defined in the Audit Plan, the project team sought to verify that Entergy'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.

Since the LRA for IP2 and IP3 identified the Fatigue Monitoring AMP to be consistent with the GALL Report with exception and enhancement, the audit team sought to verify that this program contained the program elements of the referenced GALL AMP, that the exception was acceptable based on an adequate technical justification, and that the conditions at the plant were bounded by the conditions for which the GALL AMP was evaluated. Audit Plan at 1 (Ex. NRC000123).

The project team included Staff and engineers from Brookhaven National Laboratory (BNL), a technical contractor that supported NRC during the review. Appendix A of the Audit Plan lists the project team members, project team support, and Entergy personnel that participated in the audit and review. The project team performed its work at NRC Headquarters in Rockville, Maryland; at the BNL office in Long Island, New York; and at the IP2 and IP3 site in Buchanan, New York. During the course of the audits, the Staff asked numerous questions to Entergy personnel. The Staff's questions and Entergy's responses are documented in NL-07-153, and NL-08-057. NL-07-153 (Ex. NRC000111) and NL-08-057 (Ex. NRC000109), general. To the extent that the questions related to the consistency of Entergy's AMP with the GALL Report, they are documented in the AMP Audit Report. AMP Audit Report at 52 through 55 (Ex. NRC000108).

Q98 In the area of NYS's and RK's concern with metal fatigue and environmentally-assisted fatigue, what was the Staff's audit and review plan?

A98 [GS, AH, OY, CN] For the Indian Point Fatigue Monitoring Program, the Audit Plan provided guidance for review of AMPs that are consistent with the GALL Report, even if the AMP has exceptions and/or enhancements. Audit Plan at 19 through 23 (Ex. NRC000123).

The Audit Plan shows that, for the analyses associated with the effects of reactor water environment on metal fatigue, the resolution option identified in the LRA was either the analyses remain valid under 10 C.F.R. 54.21(c)(1)(i), or the aging effects will be managed under 10 C.F.R. 54.21(c)(1)(iii). Audit Plan at 11 through 13 (Ex. NRC000123).

The Audit Plan provided the Staff with additional details about how to plan and conduct the audit of the Metal Fatigue Analyses to determine if the TLAAs identified in the IP2 and IP3 LRA were within the NUREG-1800 TLAA category of “metal fatigue” and provided adequate information to meet the requirements of 10 C.F.R. 54.21(c)(1) and 10 C.F.R. 54.21(c)(2).

Q99 The Audit Plan lists reviews of the resolution option of the "analyses remain valid under 10 C.F.R. 54.21(c)(1)(i)" for analyses associated with the Reactor Vessel, Reactor Vessel Internals, Pressurizer, and numerous other metal parts and components. Are these reviews related to the Intervenors’ contention?

A99 [GS, AH, OY, CN] Yes, the Intervenors contend that Entergy’s LRA does not include an adequate plan to monitor and manage the effects of aging due to metal fatigue on key reactor components in violation of 10 C.F.R. § 54.21(c)(1)(iii). The Staff’s review of the resolution option of the "analyses remain valid under 10 C.F.R. 54.21(c)(1)(i)" for analyses associated with the Reactor Vessel, Reactor Vessel Internals, Pressurizer, and numerous other metal parts and components are related to the Intervenors’ contention. These metal parts and components have been analyzed for fatigue, which is an age-related degradation mechanism caused by cyclic stressing of a component by either mechanical or thermal stresses. In response to the Staff’s

Audit Item 12, Entergy amended several IP2 and IP3 dispositions from 10 C.F.R. 54.21(c)(1)(i), "the analyses will remain valid during the period of extended operation," to 10 C.F.R. 54.21(c)(1)(iii), "the aging effects will be managed by the Fatigue Monitoring Program." The details of the Staff's questions, Entergy's responses and revisions to the dispositions are documented in Entergy's letter dated March 24, 2008. NL-08-057, Attachment 4 at 5 (Ex. NRC000109). The Intervenors contend that Entergy's LRA does not include an adequate plan to monitor and manage the effects of aging due to metal fatigue on key reactor components in violation of 10 C.F.R. § 54.21(c)(1)(iii).

Q100 Did the Staff ask questions during the audit on metal fatigue?

A100 [GS, AH, OY, CN] Yes, the Staff identified several areas where additional information was needed during the Indian Point audit. These areas are described in Section 3.2.5 of the Staff's AMP Audit Report under Audit Item 39 (regarding the methods used to determine the number of effective transients on a component-by-component basis), Audit Item 40 (regarding when re-analyses or updates to the fatigue usage calculations are performed), Audit Item 41 (concerning transients, procedures, and differences between IP2 and IP3), Audit Item 42 (concerning operating experience and re-evaluation of usage factors), and Audit Item 164 (concerning counting feedwater cycles). AMP Audit Report at 52 through 55 (Ex. NRC000108).

Q101 What are the findings in the audit report as they relate to New York's and Riverkeeper's Contention?

A101 [GS, AH, OY, CN] The Intervenors contend that Entergy's LRA does not include an adequate plan to monitor and manage the effects of aging due to metal fatigue on key reactor components in violation of 10 C.F.R. § 54.21(c)(1)(iii). Section 3.2.5 of the Staff's AMP Audit Report documents the Staff's audit questions and Entergy's responses associated with the Fatigue Monitoring Program. AMP Audit Report at 52 through 55 (Ex. NRC000108). This section of the Staff's AMP Audit Report states that, based on the Staff's review of Entergy's onsite documents, review of Entergy's responses to the Staff's questions, and interviews with Entergy personnel, the Staff determined that Entergy's "scope of program," "preventive actions," "monitoring and trending," and "acceptance criteria" program elements were consistent with the GALL Report. AMP Audit Report at 55 (Ex. NRC000108).

Q102 That is only four of the ten program elements for an aging management program. Were the other program elements also reviewed?

A102 [GS, AH, OY, CN] Yes, the other program elements were also reviewed. The audit and review of the "corrective actions," "confirmation process" and "administrative controls" program elements was performed during the Scoping and Screening Methodology Audit. As discussed in the Scoping Audit Report, the audit team reviewed each individual AMP basis document to ensure consistency in the use of the quality assurance attributes for each program. The purpose of this review was to assure that the aging management activities were consistent with the Staff's guidance described in SRP-LR, Section A.2, *Quality Assurance for Aging Management Programs (Branch Technical Position IQMB-1)*. AMP Audit Report at 2 through 3 (Ex. NRC000108).

As documented in SER Section 3.0.4, the Staff concluded that the descriptions and applicability of the plant-specific AMPs and their associated quality attributes provided in Appendix A, Section A.2.1, and Appendix B, Section B.0.3 of the LRA, and the quality assurance elements "corrective actions," "confirmation process," and "administrative controls," as applied to Entergy's programs were consistent with the

Staff's position regarding QA for aging management and consistent with 10 C.F.R. 54.21(a)(3). SER at 3-214 through 3-216 (Ex. NYS00326A-F).

The Staff's review and conclusions of the "parameters monitored or inspected," "detection of aging effects" and "operating experience" program elements are documented in Section 3.0.3.2.6 of the Staff's SER in NUREG-1930. Based on LRA Amendment 2 dated January 22, 2008, and Entergy's clarification on the corrective actions for the program, the Staff found that the "detection of aging effects" program element for the Fatigue Monitoring Program was consistent with the "detection of aging effects" program element in GALL AMP X.M1 without exception. The Staff found that, when the enhancement to Entergy's "parameters monitored or inspected" program element is implemented, Entergy will be monitoring all plant transients that cause cyclic strain, consistent with the Staff's "parameters monitored or inspected" program element in GALL AMP X.M1. The Staff confirmed that the "operating experience" program element satisfies the recommendations in the GALL Report and the guidance in SRP-LR Section A.1.2.3.10. SER at 3-76 through 3-79 (Ex. NYS00326A-F).

Thus, the Staff's review did consider all ten program elements of the aging management program and found all of them to be acceptable.

Q103 Other than the audit questions identified above, did the Staff ask Entergy any other questions related to metal fatigue?

A103 [GS, AH, OY, CN] Yes, the Staff asked Entergy other questions related to metal fatigue, beyond the audit questions identified above. In addition to the audit questions that were related to the consistency of Entergy's AMP with the GALL Report, which are documented in the AMP Audit Report, the Staff asked numerous audit questions related to LRA Section 4, which are documented in Entergy's Letter

NL-08-057 dated March 24, 2008. The Staff identified areas where additional information was needed and asked forty-six audit questions related to metal fatigue. The details of the Staff's questions and Entergy's responses are documented in Attachment 4 to Entergy's Letter NL-08-057. NL-08-057, Attachment 4 at 1 through 16 (Ex. NRC000109).

In addition to these audit questions, the Staff asked several RAIs in NRC Letter, *Request for Additional Information for the Review of the Indian Point Nuclear Generating Unit Numbers 2 and 3, License Renewal Application Time-Limited Aging Analyses, Bolted Connections, and Boraflex*, (April 18, 2008) (Ex. NRC000116) ("RAI Letter 2"), and in the RAI Letter dated February 10, 2011. RAI Letter 2, general (Ex. NRC000116), and RAI Letter at 11 through 13 (Ex. NYS000150).

Q104 Is the Staff' review in the area of metal fatigue complete?

A104 [GS, AH, OY, CN] Yes, the Staff's review in the area of metal fatigue is complete and is documented in the Staff's SER and SER Supp. 1. The Staff did not perform any review related to metal fatigue after the issuance of SER Supp. 1. Thus, SER Supp. 2 does not have any information related to the Staff's review related to metal fatigue. In addition, the Staff's audit reports provide additional information that supports the Staff's review documented in the Staff's SER and SER Supp. 1.

Q105 Does Entergy need to provide the results of Commitment Nos. 6, 33, 43, 44, 45 and 49 to demonstrate that the aging effects for metal fatigue and EAF will be managed?

A105 [GS, AH, OY, CN] No, the completion of these commitments 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 ASME Code design limit of 1.0, during both the initial 40-year operating license term and the period of extended operation.

Q106 Does the Staff's review documented in the SER and SER Supp. 1, as supported by the information in the Staff's Audit Reports and the associated audit questions, address the concerns raised by the Intervenors?

A106 [GS, AH, OY, CN] Yes, the Staff's review documented in the SER and SER Supp. 1, as supported by the Staff's audit reports and the associated audit questions, addresses the concerns raised by the Intervenors, which contend that Entergy's license renewal application does not include an adequate plan to monitor and manage the effects of aging due to metal fatigue on key reactor components in violation of 10 C.F.R. § 54.21(c)(1)(iii).

The Staff found that Entergy's Fatigue Monitoring Program, which is the AMP for fatigue monitoring and the environmentally-assisted fatigue analyses, is sufficient, based on a review of the LRA, the Staff's audits, Entergy's responses to requests for additional information, and audit questions related to the metal fatigue TLAAs. The basis for the Staff concluding this is as follows.

The scope of Entergy'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 and severity of fatigue transients, and are dispositioned in accordance with 10 C.F.R. 54.21(c)(1)(iii). The program tracks and monitors the number of actual plant transients and evaluates their severity against the severity of the transients used in the fatigue calculations. The program also includes periodic updates to the numbers of plant transients at least once each operating cycle, which determines if the numbers or severities of transients may be exceeded before the next update, and ensures corrective actions will be taken prior to exceeding the analyzed number or severity of each and every transient. This aspect of the program ensures that the assumptions in the analyses regarding these quantities remain bounding and applicable for the analyses.

Thus, the program also ensures that the calculated CUF and CUF_{en} values do not exceed the fatigue limit of 1.0. Condition reports are generated to ensure that corrective actions are taken prior to exceeding the analyzed number of each and every transient. Additional details of Entergy's Fatigue Monitoring Program can be found in the Staff's audit report, SER and SER Supp. 1. AMP Audit Report at 52 through 55 (Ex. NRC000108). SER at 3-76 through 3-79 (Ex. NYS00326A-F). SER Supp. 1 at 4-1 through 4-3 (Ex. NYS000160). In addition, Entergy's responses to the Staff's audit questions in NL-08-057 and NL-07-153 document additional details of the Fatigue Monitoring Program. NL-08-057, Attachment 4 (Ex. NRC000109) and NL-07-153, Attachment 3 (Ex. NRC000111), general.

It should be noted that, during the TI 2516/001 inspection for IP2, the NRC inspector noted that 2-PT-2Y015 was updated to reflect the number of transients used in WCAP-17199, and the procedure also included a table with the number of actual plant transients rather than the number of transients used in the underlying fatigue calculations. The NRC inspector also noted that the procedure was revised to better reflect operational assumptions used and the bases for the revised calculations contained in WCAP-17199. Inspection Report 05000247/2013010 at 5 (Ex. NRC000182). The observations of the NRC inspector support our testimony that the Fatigue Monitoring Program is managing cumulative fatigue damage by (1) tracking actual plant transients, (2) evaluating actual transient severity against the transient severity used in the fatigue evaluations, and (3) ensuring that the number of transients experienced by the plant remain within the number of transients used in the fatigue evaluations.

In addition, the NRC inspectors noted that, with respect to the pressurizer nozzle, Entergy was aware that the accumulation of transients at a rate greater than the rate assumed in the fatigue calculation requires more refined analysis, application of a non-destructive examination, or repair or replacement of the nozzle. Inspection Report 05000247/2013010 at 7 (Ex. NRC000182). Again, the observations of the NRC inspector support our testimony that the Fatigue Monitoring Program is managing cumulative fatigue damage by (1) tracking actual plant transients, (2) evaluating actual transient severity against the transient severity used in the fatigue evaluations, and (3) ensuring that the number of transients experienced by the plant remain within the number of transients used in the fatigue evaluations.

Staff's Opinion regarding Contentions

NYS-26B/RK-TC-1B and NYS-38/RK-TC-5

Q107 You stated previously that Contention NYS-26B/RK-TC-1B has six elements. Do you have an opinion on whether the six elements of this contention have merit?

A107 [GS, AH, OY, CN] No, our opinion is that the six elements of this contention do not have merit. Our review and our findings provide reasonable assurance that Entergy's program will adequately manage metal fatigue and the effects of reactor water environment on metal fatigue at both IP2 and IP3 during the period of extended operation.

Q108 What is your basis for concluding that the six element of Contention NYS-26B/RK-TC-1B s of this contention are without merit?

A108 [GS, AH, OY, CN] Our basis for concluding that each individual element of the contention is without merit is provided below for each element of the contention.

As described in the Board's November 4, 2010, Order, NYS-26B/RK-TC-1B characterizes Entergy's re-analyses as inadequate under NRC regulations and the GALL Report because (1) the re-analyses inappropriately limited the number of components subject to environmentally-assisted fatigue analyses. Order at 8. On the contrary, Entergy performed environmentally-assisted fatigue analyses for those components that were identified in NUREG/CR-6260. Based on its research, Idaho National Engineering Laboratory selected a representative overview of components that had higher CUFs and/or were important from a risk perspective. Besides the analyses that Entergy completed, the Staff also concluded that it would be prudent to have Entergy verify that the locations currently analyzed for the effects of reactor water environment will bound all other fatigue locations, thus providing additional assurance that Entergy has fully addressed the breadth of components with existing fatigue analyses in its CLB. Based on the analyses completed and this verification by Entergy, we conclude that Entergy did not inappropriately limit the number of components subject to environmentally-assisted fatigue analyses.

As described in the Board's November 4, 2010, Order, NYS-26B/RK-TC-1B characterizes Entergy's re-analyses as inadequate under NRC regulations and the GALL Report because (2) the re-analyses neither explain the methodology used to conduct their cumulative usage factor analyses nor include a detailed error analysis. Order at 8. Two of the exhibits provided by the New York State, Entergy's re-analyses in WCAP-17199 and WCAP-17200, give the details that the Intervenors claim are lacking, specifically the methodology used to conduct Entergy's CUF and CUF_{en} analyses. (Ex. NYS000361 and Ex. NYS000362, respectively).

For the re-analyses for IP2, the methodology used to calculate the CUF and CUF_{en} values is described in WCAP-17199. The following is a partial summary of the information that Dr. Hopenfeld states is not explained:

- [redacted]
- [redacted]
- [redacted]

WCAP-17199 as noted (Ex. NYS000361).

For the re-analyses for IP3, the methodology used to calculate the CUF and CUF_{en} values are described in WCAP-17200. The following is a partial summary of the information that Dr. Hopenfeld states is not explained:

[redacted] WCAP-17200 as noted (Ex. NYS000362).

Neither the Commission's regulations, nor ASME Section III, require detailed error analysis to be performed for fatigue analyses. In addition, the Intervenors have not identified any requirements that specify the need for an error analysis to be performed. As discussed in detail in our response to Q51, the conservatism that is inherent in the fatigue calculation methodology dictated by the ASME Code is sufficient for deterministic evaluations such as those used to calculate CUF and CUF_{en} values.

As described in the Board's Order, NYS-26B/RK-TC-1B characterizes Entergy's re-analyses as inadequate under NRC regulations and the GALL Report because (3) the re-analyses exclude a fatigue evaluation of important structures and fittings within the reactor pressure vessel. Order at 8. The research performed by Idaho National Engineering Laboratory as documented in NUREG/CR-6260 did not identify the need to include structures and fittings within the reactor pressure vessel to address the effects of reactor water environment on metal fatigue because internal components of the reactor vessel are not important to the structural integrity of the reactor coolant pressure boundary. Instead, the report focused on reactor coolant pressure boundary components of operating nuclear power plants. The Intervenors also have not identified either any regulatory requirements mandating consideration of environmental effects on metal fatigue (i.e., F_{en} factor) of structures and fittings within the reactor pressure vessel, any specific structures and fittings within the reactor pressure vessel which should be considered, nor have they identified any plant operating experience in which structures and fittings within the reactor pressure vessel have failed or been challenged due to environmentally-assisted fatigue.

As described in the Board's November 4, 2010, Order, NYS-26B/RK-TC-1B characterizes Entergy's re-analyses as inadequate under NRC regulations and the GALL Report because (4) the re-analyses exclude from evaluation "the potential failure of highly fatigued structures and fittings under" certain types of "large thermal/pressure shock-type loads." Order at 8. The Intervenors' claim is faulted in that "highly fatigued" structures, components, and "fittings," those that may have a high CUF or a CUF_{en} value but does not exceed 1.0, are unlikely to have initiated a fatigue crack that would be necessary for the "highly fatigued" structure, component, or "fitting" to be challenged by design basis loads. In addition, fatigue crack initiation is not a relevant failure criterion for DBA events such as large thermal/pressure shock-type loads. As detailed in our response to Q145, gross component failure and deformation are the important parameters to safety for DBA events where safe shutdown of the plant is the main objective (rather than complete component integrity for continued operation).

The Intervenors' claim is also incorrect in its assertion that the Fatigue Monitoring Program is not capable of managing the CUF and CUF_{en} within the fatigue limit of 1.0, or that having a CUF or CUF_{en} value within the fatigue limit of 1.0 fails to prevent fatigue cracking of the component. Based on the information provided by Entergy and reviewed by the Staff, the Staff has reasonable assurance that Entergy's Fatigue Monitoring Program will manage CUF and CUF_{en} values within the fatigue limit of 1.0, and that corrective actions will be implemented prior to exceeding the fatigue limit. Further, maintaining the CUF and CUF_{en} to values less than or equal to the fatigue limit of 1.0 provides reasonable assurance that the likelihood of a small initiated fatigue crack in the component is very low. Thus, assurance that the CUF and CUF_{en} values remain within the fatigue limit has the effect of providing assurance that the component does not have fatigue cracks that would challenge any of the CLB analyses for the component. Therefore, maintaining the CUF and CUF_{en} values within the fatigue limit using the Fatigue Monitoring Program renders this element of the contention without merit.

As described in the Board's November 4, 2010, Order, NYS-26B/RK-TC-1B characterizes Entergy's re-analyses as inadequate under NRC regulations and the GALL Report because (5) the re-analyses contain lower safety margins that create more risk because the new CUFs have been "reduced by more than an order of magnitude." Order at 8. As discussed in our response to Q210, the Staff believes that this contention is without merit because reanalyzed CUF or CUF_{en} values do not have "lower safety margins." Rather, refined CUF and CUF_{en} values only indicate the level of effort required to show acceptability for the component. It is a requirement of the ASME Code methods used to calculate the CUF and CUF_{en} values that required safety factors are maintained in the analysis. Furthermore, when performing the fatigue calculations, the analyst's goal is to demonstrate acceptability of the component; it is not the analyst's intent to show margin with respect to the fatigue limit. As a result, the analyst makes only those simplifying assumptions that are conservative and necessary to demonstrate that the results of the calculation indicate that the CUF or CUF_{en} meets the fatigue limit. Re-analyses may require fewer simplifying assumptions to demonstrate that the CUF or CUF_{en} value remains acceptable. The Staff's opinion is that there is safety margin inherent to the ASME Code CUF methodology and the fatigue limit of 1.0, and that a reanalysis of the CUF and CUF_{en} values using valid assumptions does not correspond to a reduction in safety margins. Entergy's implementation of its Fatigue Monitoring Program provides, in part, assurance that the assumptions used in the re-analyses will be validated during the period of extended operation, and invalidation of these assumptions requires corrective actions by Entergy. The testimony provided by the Intervenors' expert witnesses does not discuss any consequences of their assertion that lower safety margins create more risk from the refined CUF and CUF_{en} values being reduced by more than an order of magnitude.

As described in the Board's November 4, 2010, Order, NYS-26B/RK-TC-1B characterizes Entergy's re-analyses as inadequate under NRC regulations and the GALL Report because (6) Entergy has not committed to repair or replace components when the CUF approaches unity (1.0). Order at 8. This element of the contention is faulty because there are no regulatory or technical requirements that would dictate repair or replacement of components that still meet all applicable regulatory and technical requirements. Specifically, ASME Section III (which is referenced in 10 C.F.R. 50.55a) does not require the preemptive repair or replacement of components that have CUF values less than, equal to, or greater than 1.0, and Appendix L of ASME Section XI states that a component is acceptable for continued service if the CUF is less than or equal to 1.0, or, in the event that the CUF is greater than 1.0, the component is acceptable with inspection provisions if an acceptable flaw tolerance evaluation is performed. ASME Section III at 81 (Ex. NYS0000349) and ASME Appendix L at 422 through 427 (Ex. NRC000113). It is incumbent that an applicant implements corrective actions to ensure that the CUF values for its components do not exceed the fatigue limit, and, as described in the GALL Report Rev. 1 and GALL Report Rev. 2, such corrective actions can include more rigorous analysis, repair or replacement. GALL Report Rev. 1 at X M-2 (Ex. NYS00146A-C) and GALL Report Rev.2 at X M-2 (Ex. NYS00147A-D). Therefore, the Intervenors request for a commitment to repair or replace components when the CUF approaches unity is not required and is inconsistent with the Staff's guidance for appropriate corrective actions.

Q109 You stated previously that Joint Contention NYS-38/RK-TC-5 has 4 bases and this testimony addresses Basis (1) and Basis (2). With respect to the intervenors' basis (1) for Joint Contention NYS-38/RK-TC-5, has Entergy deferred defining the methods used for determining the most limiting locations for metal fatigue calculations and selection of those locations as related to Commitment No. 43?

A109 [GS, AH, OY, CN] It is the Staff's opinion that Entergy has not deferred defining the methods used for determining the most limiting locations for metal fatigue calculations because Commitment No. 43 has been completed by Entergy for IP2. For IP3, Commitment No. 43 will be completed by Entergy before its entrance of the extended period of operation in December 2015.

Q110 With respect to the intervenors' basis (2) for Joint Contention NYS-38/RK-TC-5, has Entergy not specified the criteria it will use and assumptions upon which it will rely for modifying the WESTEMS™ computer model for CUF_{en} calculations as related to Commitment No. 44?

A110 [GS, AH, OY, CN] It is the Staff's opinion that Entergy has specified the criteria it will use and assumptions upon which it will rely for modifying the WESTEMS™ computer model for CUF_{en} calculations because Commitment No. 44 has been completed for

IP2. For IP3, Commitment No. 44 will be completed by Entergy before its entrance of the extended period of operation in December 2015.

<u>Staff Testimony in Response to Dr. Joram Hopenfeld for NYS-26B/RK-TC-1B</u>

Q111 Have you read Prefiled Written Testimony of Dr. Joram Hopenfeld Regarding NYS-26-B/RK-TC-1B-Metal Fatigue, (December 22, 2011) ("Hopenfeld") (Ex. RIV000034) and, Report of Dr. Joram Hopenfeld in Support of Contention Riverkeeper TC-1B-Metal Fatigue, (December 22, 2011) ("Hopenfeld Report") (Ex. RIV000035)?

A111 [GS, AH, OY, CN] Yes, we have Dr. Hopenfeld's pre-filed written testimony regarding NYS-26-B/RK-TC-1B- dated December 22, 2011.

Q112 What is your opinion of Dr. Hopenfeld's statement regarding, "What is metal fatigue?"

A112 [GS, AH, OY, CN] The information provided in Dr. Hopenfeld's statement regarding metal fatigue is partially acceptable and it requires supplementation. Hopenfeld at 4 (Ex. RIV000034). The Staff does not agree with Dr. Hopenfeld that, "[T]he [ASME] Code requires that the CUF at any given location be maintained below one." For component design, Paragraph NB-3222.4 of ASME Section III requires that the CUF be less than or equal to one. ASME Section III at 81 (Ex. NYS0000349). After a component is placed into service, ASME Section XI recognizes that the CUF may in fact exceed one due to unanticipated loads or loads that are more severe than what was assumed during the design. Thus, ASME Appendix L provides provisions that may be used when the calculated fatigue usage exceeds the fatigue usage limit defined in the original Construction Code of the facility. If the CUF is projected to exceed one, corrective actions including repair, replacement, or inspection may be implemented. Article L-3000 of ASME Appendix L provides provisions for performing flaw tolerance assessment of a postulated flaw in situations when the CUF exceeds one. Therefore, Dr. Hopenfeld's statement that CUF must be <u>maintained</u> below one is incorrect, and it should be replaced in its entirety, to provide correctness, context and completeness, with the description from Section 4.3.1 of the SRP-LR Rev. 2, which states, “A CUF below a value of one provides assurance that no crack has been formed. A CUF above a value of one allows for the possibility that a crack may form, and that if left untreated, the crack could propagate exponentially under fatigue loading and eventually lead to coolant leakage in reactor pressure boundary components…” SRP-LR Rev. 2 at 4.3-1 (Ex. NYS000161). Thus, even if the CUF value for a component exceeds 1.0, the component is not subject to imminent failure or rupture, and may be evaluated for acceptability in accordance with the ASME Code, in contrast to the implications throughout Dr. Hopenfeld’s testimony.

Q113 Is Dr. Hopenfeld’s description of the “safety implications of metal fatigue” accurate?

A113 [GS, AH, OY, CN] No, Dr. Hopenfeld’s attempt to address the safety implication of metal fatigue in his testimony is not accurate. Hopenfeld at 5 (Ex. RIV000034). He testified that, “Fatigue may also create small cracks that propagate and cause a given component to malfunction or break up and form loose parts which can interfere with the safe operation of the plant. Such failures may have serious consequences to public health and safety.” Hopenfeld at 5 (Ex. RIV000034). Dr. Hopenfeld’s testimony on this point is unclear. Dr. Hopenfeld appears to mistake the aspect of metal fatigue that is within the scope of this contention (that is “crack initiation”) and the aspect of metal fatigue that would cause a component to break up and create loose parts (that is “crack propagation or crack growth”). The aspect of metal fatigue that is within the scope of the contention is calculations that are used to demonstrate that there is a low likelihood of fatigue crack initiation in the component when the calculated CUF or CUF_{en} is below the fatigue usage limit of 1.0. The only “safety implication of metal fatigue” in the scope of the initiation would be the low likelihood of the presence of a small, structurally insignificant, fatigue flaw, rather than “failure” of the component, resulting in "loose parts." Fatigue crack initiation is controlled by maintaining calculated CUF and CUF_{en} values to be less than or equal to 1.0 in accordance with the appropriate requirements of the ASME Code – both ASME Section III during the design and construction phase and ASME Section XI during the operating and maintenance phase of the component's life. For instances where the CUF or CUF_{en} values are calculated to exceed one, or in instances where actual fatigue cracking may be found by inspection, ASME Section XI also provides provisions for evaluation to remedy the situation, including possible repair or replacement of the component.

Fatigue crack growth that could, potentially, lead to structural failure is governed by ASME Section XI, which contains the rules for inservice inspection of nuclear power plant components. ASME Section XI makes use of crack growth rates, which are not related to CUF. The Staff would also like to point out the incorrect characterization of the term "failure" by Dr. Hopenfeld for a case in which the CUF limit is exceeded. The "failure" discussed in his sentence, "Such failures may have serious consequences to public health and safety…" is a result of fatigue crack growth leading to complete, through-wall cracks in components, which is different than metal fatigue crack initiation as indexed using CUF or CUF_{en}. Thus, the example provided by Dr. Hopenfeld is not relevant to the contention and Dr. Hopenfeld appears to be confused between the mechanisms of crack growth and crack initiation in his testimony.

Q114 What is your opinion of Dr. Hopenfeld's view of "how component susceptibility to metal fatigue is predicted?"

A114 [GS, AH, OY, CN] The Staff does not address with Dr. Hopenfeld's testimony related to "how component susceptibility to metal fatigue is predicted". Hopenfeld at 5 (Ex.

RIV000034). The Staff does not agree with Dr. Hopenfeld's statements that, "Crack growth rate for a given stress intensity can be predicted using an equation that includes empirical constants that were derived from laboratory tests in air under controlled conditions. However, this equation can predict crack growth reliably only as long as the equation is used under the conditions that were used to calibrate the empirical constants. In order to account for crack propagation in the actual reactor environment….." Hopenfeld at 5 (Ex. RIV000034). The phrases, "crack growth rate", "crack growth", and "crack propagation" in his response refer to a known flaw, or a postulated flaw under the circumstances when CUF or CUF_{en} is calculated to be greater than one. However, Dr. Hopenfeld said, in particular, "In order to account for crack propagation in the actual reactor environment, the individual usage factor in air is multiplied by a corresponding environmental adjustment factor, 'F_{en}.' " Hopenfeld at 5 (Ex. RIV000034). The environmental multiplication factor, F_{en}, is used in evaluating crack initiation, and is not relevant to the evaluation of crack propagation. Dr. Hopenfeld appears to have again confused the two mechanisms of fatigue crack initiation (which is within the scope of the contention) and fatigue crack growth (which is not within the scope of the contention).

Q115 What is your opinion of Dr. Hopenfeld's view of the validity of the results of Entergy's refined analyses?

A115 [GS, AH, OY, CN] The Staff does not agree with Dr. Hopenfeld's statement that, "the methodology employed to calculate Entergy's new CUF_{en} values is highly suspect, and that the validity of the results is questionable." Hopenfeld at 6 (Ex. RIV000034).

The Staff's SER was issued in November 2009 (Ex. NYS00326A-F), which was prior to the date of the submittal of the CUF_{en} re-analyses in WCAP-17199 and WCAP-17200, so those two reports were not reviewed by the Staff as part of the

LRA review process to ascertain if the AMP complied with the GALL Report. The Staff reviews these re-analyses as part of the implementation phase of corrective actions under the AMP. The Commission determined that if the NRC concludes that an aging management program is consistent with the GALL Report, then it accepts the applicant's commitment to implement that aging management program, finding the commitment itself to be an adequate demonstration of reasonable assurance under section 54.29(a). *NextEra Energy Seabrook, LLC* (Seabrook Station, Unit 1), CLI-12-5, 75 NRC 301, 304 (2012) (citing *Entergy Nuclear Vermont Yankee, LLC* (Vermont Yankee Nuclear Power Station), CLI-10-17, 72 NRC 1, 36 (2010); *AmerGen Energy Co., LLC* (Oyster Creek Nuclear Generating Station), CLI-08-23, 68 NRC 461, 467-68 (2008)). With respect to Entergy's re-analyses for IP2 and IP3, the re-analyses show appropriate implementation of the AMP.

Nevertheless, the Staff reviewed the reports for the purposes of this testimony. In the re-analyses, Entergy indicated that the F_{en} was calculated in accordance with NUREG/CR-5704 for the stainless steels components and NUREG/CR-6583 for the carbon and low-alloy steel components. The Staff noted that GALL Report Rev. 1, states that, "Formulae for calculating the environmental life correction factors are contained in NUREG/CR-6583 for carbon and low-alloy steels and in NUREG/CR-5704 for austenitic stainless steels." GALL Report Rev. 1 at X.M-2 (Ex. NYS00146A-C). Thus, the methodologies that were used in Entergy's re-evaluations for IP2 and IP3 to calculate F_{en} are consistent with the Staff's guidance, and the Staff has no reason to question the validity of the CUF_{en} values from the refined analyses.

Q116 What bases does Dr. Hopenfeld provide to support his view on the lack of validity of the results of Entergy's refined analyses?

A116 [GS, AH, OY, CN] Dr. Hopenfeld identified four reasons to support his view. Hopenfeld at 6 and 7 (Ex. RIV000034). However, as our testimony in the responses to the following questions indicate, Dr. Hopenfeld did not provide sufficient bases for his assertions.

[redacted] Hopenfeld at 6 and 7 (Ex. RIV000034).

Q117 Focusing on the first item in Dr. Hopenfeld's testimony - what is your opinion of Dr. Hopenfeld's statement that laboratory data must be adjusted to account for the actual reactor environment in order to determine appropriate F_{en} factors in the calculation of CUF_{en}?

A117 [GS, AH, OY, CN] In his testimony, Dr. Hopenfeld provided his explanation of, "why the use of laboratory data must be adjusted in order to determine appropriate F_{en} factors in the calculation of CUF_{en}." Hopenfeld at 7 (Ex. RIV000034). The Staff does not agree with Dr. Hopenfeld's statement regarding the NRC-sponsored research performed by Argonne National Laboratory (ANL):

> *Due [to] the significant differences that exist between the laboratory and reactor environment, there are numerous uncertainties in applying the F_{en} equations to actual reactor components. In, NUREG/CR-6909, Effect of LWR Coolant*

Environment on Fatigue Life of Reactor Materials, ANL identifies numerous such uncertainties, which include material composition, component, size and geometry, surface finish, loading history, strain rate, mean stress, water chemistry, dissolved oxygen levels, temperature, and flow rate. Such uncertainties can have a significant affect upon fatigue life and ignoring them will result in underestimated CUF_{en} calculations.

Hopenfeld at 7 (Ex. RIV000034) (footnote omitted) (emphasis added).

NUREG/CR-6909 actually states the following:

The variables that can affect fatigue life in air and LWR environments can be broadly classified into three groups: (a) Material (Composition, metallurgy, processing, size and geometry, surface finish, and surface preparation), (b) Loading (strain rate, sequence, mean stress, and biaxial effects), and (c) Environment (water chemistry, temperature, and flow rate).

NUREG/CR-6909 at 72 (Ex. NYS000357) (emphasis added).

Dr. Hopenfeld has incorrectly interpreted or characterized this report, where these "variables" that can affect fatigue life become "uncertainties" that presumably would undermine the acceptability of the resultant CUF_{en} calculations.

Reading further from NUREG/CR-6909, the authors stated that the existing fatigue database covers an adequate range of material parameters (composition, metallurgy and processing), the strain rate loading parameter, and the water chemistry and temperature environmental parameters. NUREG/CR-6909 72 (Ex. NYS000357). In particular, the authors said "therefore, the variability and uncertainty in fatigue life due to these parameters have been incorporated into the model." NUREG/CR-6909 72 (Ex. NYS000357). Thus, contrary to Dr. Hopenfeld's statement that, "the user must consider all of the relevant uncertainties, and the results must be adjusted to account for the varying parameters," the authors of NUREG/CR-6909 indicated that material, loading, and environmental parameters have already been adequately addressed. Furthermore, the authors of this report further discussed how the existing fatigue data are conservative with respect to the effect of surface preparation, fabrication procedures, and biaxial effects. NUREG/CR-6909 at 72 through 73 (Ex. NYS000357). Therefore, the Staff does not believe it is necessary or reasonable to expect Entergy to account for the uncertainties cited by Dr. Hopenfeld since they are already accounted for in the evaluation of the existing fatigue data.

Further, the Staff does not agree with Dr. Hopenfeld's statement in his pre-filed testimony that: "In NUREG/CR-6909, *Effect of LWR Coolant Environment on Fatigue Life of Reactor Materials*, ANL specifies that appropriate bounding F_{en} values of 12 for stainless steel and 17 for carbon and low alloy steel to account for the numerous uncertainties in using the F_{en} equations." Hopenfeld at 7 (Ex. RIV000034).

NUREG/CR-6909 states that, "Under certain environment and loading conditions, fatigue lives in water relative to those in air can be a factor of ~12 lower for austenitic stainless steels, ~ 3 lower for Ni-Cr-Fe alloys, and ~17 lower for carbon and low-alloy steels." NUREG/CR-6909 at iii (Ex. NYS000357). This statement indicates that these values represent extreme values for the environmental factor. The words, "appropriate" and "bounding" cited in Dr. Hopenfeld's statement do not appear on page iii of the NUREG/CR-6909, nor does NUREG/CR-6909 anywhere recommend that F_{en} values of 12 for stainless steel and 17 for carbon and low alloy steel should be used. Instead, the Staff noted that the values of F_{en} should be based on plant-specific information, which really is the whole purpose of the report, to provide an acceptable methodology to determine environmental effects without resorting to overly conservative and bounding values, as Dr. Hopenfeld apparently wants Entergy to implement. The Staff strongly disagrees with Dr. Hopenfeld's assertion that use of these highly conservative values is necessary or appropriate.

Q118 What is your opinion of Dr. Hopenfeld's view and response to, "How did you reach the conclusion that Entergy's refined fatigue evaluations fail to properly adjust laboratory data to account for the actual reactor environment" in his testimony?

A118 [GS, AH, OY, CN] The Staff does not agree with Dr. Hopenfeld's statement that:

> [redacted]

Hopenfeld at 8 (Ex. RIV000034).

NUREG/CR-6909 states that the existing fatigue database covers an adequate range of material parameters, loading parameters and environmental parameters; therefore, the variability and uncertainty in fatigue life due to these parameters have been incorporated into the model. NUREG/CR-6909 at 72 through 73 (Ex. NYS000357). Dr. Hopenfeld's claim that there is a need to adjust the laboratory data for uncertainties is in direct contradiction to what is stated in NUREG/CR-6909.

Dr. Hopenfeld's claim that Entergy should apply a bounding F_{en} factor is in direct contradiction to the purpose of NUREG/CR-6909, which is intended to allow for the computation of an appropriate F_{en} based on the plant-specific parameters of DO, temperature, sulfur content, and strain-rate. NUREG/CR-6909 at Appendix A, general (Ex. NYS000357).

Q119 What is your opinion of Dr. Hopenfeld's view in his testimony on "how Entergy's failure to properly adjust laboratory data to the actual reactor environment affects the results of the refined EAF analysis"?

A119 [GS, AH, OY, CN] The Staff does not agree with Dr. Hopenfeld's statement that:

> [redacted] *The use of the bounding F_{en} values recommended by ANL,* [redacted] *would increase the CUF_{en} values beyond unity for eight of the components analyzed, as I calculated in my expert report.*

Hopenfeld at 8 (Ex. RIV000034) (emphasis added).

We again strongly disagree with Dr. Hopenfeld's testimony and the cited inadequacies in Entergy's analyses. Dr. Hopenfeld provides no substantiation that the values calculated and used by Entergy are not realistic, other than misquoting and misinterpreting NUREG/CR-6909 to indicate that the bounding values identified in the ANL report are "realistic" rather than what they are – among the highest values identified by ANL in the report.

Q120 Do you agree with Dr. Hopenfeld's testimony regarding the state of knowledge of the DO in operating reactors?

A120 [GS, AH, OY, CN] No, the Staff does not agree with Dr. Hopenfeld's statement that the DO, in many cases, is unknown in a reactor plant. Hopenfeld at 9 (Ex. RIV000034).

DO is an important input to the calculation of F_{en} and should be based on available plant-specific data. As documented in the Staff's SER in Section 3.0.3.2.17, (SER-F at 3-143 through 3-145 (Ex. NYS00326A)), and based on its audit and review of Entergy's Water Chemistry Control - Primary and Secondary Program, the Staff concluded that the program elements for which Entergy claimed consistency with the GALL Report are in fact consistent with the GALL AMP XI.M2, including the enhancement to the program. LRA Section B.1.41 states that the program relies on monitoring and control of reactor water chemistry based on the EPRI guidelines in Technical Report 1002884 (previously TR-105714), *Pressurized Water Reactor Primary Water Chemistry Guidelines – Volume 1, Revision 5* (September 2003) (Ex. NRC000115) ("TR-1002884 Rev. 5"), (PROPRIETARY). LRA at B-137 (Ex. ENT00015A-B). [redacted] TR-1002884 Rev. 5 at 3-1 through 3-23 (Ex. NRC000115). Since Entergy's Water Chemistry Control – Primary and Secondary Program monitors [redacted], plant-specific data for this parameter is known.

Dr. Hopenfeld provides no basis or supporting data for his statement other than his own opinion.

Q121 Do you agree with Dr. Hopenfeld that Entergy should have used bounding DO values in the calculation of the F_{en} factors?

A121 [GS, AH, OY, CN] No, the Staff does not agree with Dr. Hopenfeld's statement that [redacted]

[redacted] Hopenfeld at 10 (Ex. RIV000034).

As described in their Letter NL-08-084, Entergy provides its treatment of DO levels for the stainless steel and low alloy steel components listed in LRA Tables 4.3-13 and 4.3-14 and states that reactor coolant system (RCS) DO is controlled at less than 50 parts-per-billion (ppb). NL-08-084, Attachment 1 at 2 through 3 (Ex. ENT000194). In addition, for the refined fatigue evaluations, Westinghouse provided an explanation regarding the treatment of DO levels in the refined CUF analyses of carbon steel components addressed for IP2. WCAP-17199 at 5-24 (Ex. NYS000361). Westinghouse also provided an explanation regarding the treatment of DO levels in the refined CUF analyses of stainless steel components addressed for IP2. WCAP-17199 at 5-2 through 5-3 (Ex. NYS000361). Furthermore, Westinghouse provided an explanation regarding the treatment of DO levels in the refined CUF analyses of carbon steel components addressed for IP3. WCAP-17200 at 5-24 (Ex. NYS000362). Westinghouse also provided an explanation regarding the treatment of DO levels in the refined CUF analyses of stainless steel components addressed for IP3. WCAP-17200 at 5-2 through 5-3 (Ex. NYS000362). In both reports, [redacted] Also, in both reports, Westinghouse explains their treatment of DO by stating:

1) [redacted]

2) [redacted]

WCAP-17199 at 5-24 (Ex. NYS000361) and WCAP-17200 at 5-24 (Ex. NYS000362).

LRA Section B.1.41 states that Entergy's Water Chemistry Control - Primary and Secondary Program, which the Staff found acceptable in Section 3.0.3.2.17 of its SER, relies on monitoring and control of reactor water chemistry based on the EPRI guidelines in TR-1002884 Rev. 5 (previously TR-105714). SER at 3-143 through 3-145 (Ex. NYS00326A-F). Section 3 of TR-1002884 Rev. 5 [redacted] TR-1002884 Rev. 5 at 3-1 through 3-23 (Ex. NRC000115). Specifically, Table 3-3 of 1002884 Rev. 5 [redacted] TR-1002884 Rev. 5 at 3-15 (Ex. NRC000115). It further states that [redacted] which includes the following:

[redacted]

TR-1002884 Rev. 5 at 3-13 (Ex. NRC000115).

Thus, Entergy's Water Chemistry Control - Primary and Secondary Program, which the Staff found to be acceptable, maintains DO at IP2 and IP3 consistent with the values used by Entergy for DO in WCAP-17199 and WCAP-17200, negating the need to use the bounding values that Dr. Hopenfeld suggests.

Q122 Do you agree with Dr. Hopenfeld that Entergy and/or Westinghouse has not provided any explanation regarding the treatment of DO levels in the refined EAF analyses?

A122 [GS, AH, OY, CN] No, the Staff does not agree with Dr. Hopenfeld.

Specifically, Westinghouse provided an explanation regarding their treatment of DO levels in the refined CUF analyses of carbon steel components and stainless steel components at IP2. WCAP-17199 at 5-24 and 5-3 (Ex. NYS000361). Similarly, Westinghouse provided an explanation regarding the treatment of DO levels in the refined CUF analyses of carbon steel components and stainless steel components at IP3. WCAP-17200 at 5-24 and 5-3 (Ex. NYS000362). In addition, NL-08-084 indicated that DO levels of less than 50 ppb (0.050 ppm) were used for those components described in LRA Tables 4.3-13 and 4.3-14. NL-08-084, Attachment 1 at 2 through 3 (Ex. ENT000194).

Westinghouse states that it used [redacted] WCAP-17199 and WCAP-17200 at 1-2 (Ex. NYS000361 and NYS000362, respectively). The equation for the F_{en} factor for carbon steel provided in NUREG/CR-6583 is $F_{en} = \exp(0.585 - 0.00124T - 0.101S^* T^* O^* \varepsilon^*)$, where T is test temperature, S* is transformed sulfur content, T* is transformed temperature, O* is transformed DO and ε* is transformed strain-rate. NUREG/CR-6583 at 68 and 69 (Ex. NYS000356). In addition, NUREG/CR-6583 states that a value of 25°C is used for T if the environmental adjustment factor is defined relative to room temperature air. NUREG/CR-6583 at 69 (Ex. NYS000356).

Consistent with Westinghouse's appropriate use of a DO level [redacted] and the guidance provided in NUREG/CR-6583, the equation with temperature greater than 150°C is:

[redacted]

When the temperature is less than 150°C, the transformed temperature T* has a value of 0.0, so the F_{en} is:

[redacted]

Based on the above description of the guidance and equation from [redacted] was appropriately calculated for carbon steel components for all temperature conditions.

For stainless steel, Westinghouse stated that, [redacted] WCAP-17199 and WCAP-17200 at 5-3 (Ex. NYS000361 and NYS000362, respectively).

Therefore, Westinghouse clearly identified its treatment of DO for both carbon steel and stainless steel components in WCAP-17199 and WCAP-17200. In addition, Entergy provided its treatment of DO in NL-08-084. NL-08-084, Attachment 1 at 1 through 2 (Ex. ENT000194), for the low-alloy steel components listed in LRA Tables 4.3-13 and 4.3-14.

Further, Entergy's determination of F_{en} factors is consistent with Staff guidance in the GALL Report Rev. 1 and GALL Report Rev. 2, both of which recommend the use of NUREG/CR-6583 to determine the F_{en} factor for carbon and low alloy steel components and NUREG/CR-5704 to determine the F_{en} factor for stainless steel components. GALL Report Rev. 1 and GALL Report Rev. 2 at X.M-1 (Ex. NYS00146A-C and NYS00147A-D, respectively).

Therefore, it is not clear to the Staff as to what additional information Dr. Hopenfeld thinks is missing, given the information provided by Entergy's reports, in which Westinghouse's treatment of DO in the refined calculations is included and fully explained in WCAP-17199 and WCAP-17200, and Entergy's treatment of DO is described in NL-08-084.

Q123 Do you agree with Dr. Hopenfeld's statement on how the refined EAF analyses do not properly consider DO in the calculation of F_{en} values?

A123 [GS, AH, OY, CN] No, the Staff does not agree with Dr. Hopenfeld's statement that the refined EAF analyses do not properly consider DO in the calculation of F_{en} values. Hopenfeld at 11 (Ex. RIV000034).

As described in our response to Q122, Westinghouse's and Entergy's treatment of DO are clearly documented in WCAP-17199, WCAP-17200, and NL-08-084. In addition, as mentioned in our response to Q121, one of Entergy's aging management programs is the Water Chemistry Control - Primary and Secondary Program, which, as described in LRA Section B.1.41, relies on monitoring and control of reactor water chemistry based on the EPRI guidelines in TR-1002884 Rev. 5 (previously TR-105714). LRA at B-137 (Ex. ENT00015A-B). Section 3 of TR-1002884 Rev. 5 provides power operation chemistry control recommendations, which provides specific guidance for the control of DO. TR-1002884 Rev. 5 at 3-1 through 3-23 (Ex. NRC000115).

Specifically, Table 3-3 of TR-1002884 Rev. 5 states that [redacted]

[redacted] TR-1002884 Rev. 5 at 3-15 (Ex. NRC000115). In addition, Dr. Hopenfeld describes the DO levels as uncertainties; however, since Entergy uses its Water Chemistry Control - Primary and Secondary Program to monitor and manage DO, the DO level is managed within a specified range rather than treated as an uncertainty. The figure that Dr. Hopenfeld references on page 4-22 of MRP-47, *Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application, Revision 1* (September 2005) (Ex. NYS000350) ("MRP-47") is a graphical representation of F_{en} values as a function of DO for constant values of temperatures and strain rates. Contrary to Dr. Hopenfeld's implication that Entergy should use F_{en} values as high as 130 at high DO levels, Entergy has plant-specific data on its DO levels, which are well below the levels that lead to a F_{en} value of 130, and were in turn used in the calculation of its F_{en} values. Furthermore, measurement of DO in the plant on a periodic basis is consistent with how it is measured during laboratory testing of specimens, as documented in Section 2 of NUREG/CR-5704 and Section 2 of NUREG/CR-6583. NUREG/CR-5704 at 16 (Ex. NYS000354) and NUREG/CR-6583 at 5 (Ex. NYS000356). In both situations, some variation in DO is both measured and expected.

Finally, it should be noted that monitoring of DO is not continuous. Rather, DO samples are taken at a frequency consistent with Entergy's implementing procedures, which may be during steady state or transient conditions. The data available from the Water Chemistry Control - Primary and Secondary Program provides a reasonable source of information for DO level, which is consistent with Entergy's treatment of DO. In addition, it is not likely that DO levels will change significantly between samples without a source of supply of oxygen that would cause the DO level to change. It should also be noted that there have been no identified instances of failures due to fatigue cracking in a nuclear power plant as a result of environmentally-assisted fatigue, which evidences the robustness and conservatism in ASME Section III fatigue evaluations, the calculation of F_{en} factors, and the associated calculations of CUF and CUF_{en}. Additional assurance that metal fatigue will be adequately managed is provided by Entergy's Fatigue Monitoring Program, as described throughout our testimony.

Q124 What is the Staff's opinion about Dr. Hopenfeld's testimony regarding Entergy's heat transfer coefficient for the calculation of CUF_{en}?

A124 [GS, AH, OY, CN] Dr. Hopenfeld's pre-filed testimony references WCAP-14950, *Mitigation and Evaluation of Pressurizer Insurge/Outsurge Transients*, (February 1998) (Ex. RIV000050) ("WCAP-14950") for the pressurizer surge line. Hopenfeld at 14 and 15 (Ex. RIV000034). He also references CENC-1110, *Analytical Report for Indian Point Reactor Vessel Unit No. 2*, (April 22, 1968) (Ex. RIV000052A-D) ("CENC-1110") and CENC-1122, *Analytical Report for Indian Point Reactor Vessel Unit No. 3*, (June 1969) (Ex. RIV000053) ("CENC-1122") for the IP2 reactor inlet and outlet nozzles and the IP3 reactor inlet and outlet nozzles, respectively. Hopenfeld at 15 (Ex. RIV000034).

In all cases, for the IP2 and IP3 pressurizer surge lines and the IP2 and IP3 reactor inlet and outlet nozzles, Dr. Hopenfeld questions Entergy's analyses that are part of its CLB. Specifically, the heat transfer coefficient Dr. Hopenfeld is questioning is used to calculate the stress values, which are then used as an input to existing CUF calculations that are part of Entergy's CLB. Since these CUF calculations are part of Entergy's CLB they are not part of the review for license renewal.

The Staff noted that WCAP-17199 and WCAP-17200 do not discuss the specific heat transfer coefficients that were used in the CUF_{en} calculations. The Staff also noted that, through Entergy's and its vendors' QA Programs in accordance with Appendix B to 10 C.F.R. Part 50, the review and verification processes would ensure that reasonable and appropriate inputs, such as the heat transfer coefficient, are used in these calculations.

Q125 Do you agree with Dr. Hopenfeld statements regarding [redacted] Hopenfeld at 17 (Ex. RIV000034).

With regards to "past transients," Entergy provided the number of accrued cycles for IP2 as of May 24, 2005, in LRA Table 4.3-1, and those for IP3 as of March 31, 2006, in LRA Table 4.3-2. The information in these tables was clarified and supplemented in NL-08-057, in response to the Staff's audit questions. Specifically, Audit Questions #3, 5, 6, 7, 10, 14, 134 and 138, related to the cycle counts and projections that Entergy provided in LRA Tables 4.3-1 and 4.3-2. NL-08-057, Attachment 4 at 1 through 16 (Ex. NRC000109) and LRA at 4.3-4 through 4.3-7 (Ex. ENT00015A-B). Dr. Hopenfeld does not provide any data to support his assertions that Entergy's records are in any way incomplete or insufficient.

As described in LRA Section B.1.12, Entergy's Fatigue Monitoring Program includes an enhancement to Entergy's current IP3 Fatigue Monitoring Program to revise appropriate procedures to include all the transients identified, to assure all fatigue analysis transients are included with the lowest limiting numbers, and to update the number of design transients accumulated to date. LRA at B-45 (Ex. ENT00015A-B). This enhancement was clarified in response to Audit Item 4 in NL-08-057, in which Entergy stated:

> *[t]he IP2 and IP3 Class 1 systems were designed for similar cyclic duty during original design and construction. Both units track these design cycles, which are included in the FSAR, to ensure that the original design requirements are not exceeded during plant operation. In addition to the original design cycles, IP2 has added a number of additional duty cycles to its fatigue monitoring program to address enhancements developed during the design of newer vintage plants but which were not included as part of the original plant design basis. IP3 is reviewing its fatigue monitoring program to determine if additional transients should be added to its monitoring program to improve its effectiveness.*

NL-08-057, Attachment 4 at 1 through 2 (Ex. NRC000109).

Furthermore, in response to Audit Item 39 in NL-07-153, Entergy stated that "[t]he basis for the IP2 design cycles is described in WCAP-12191, Revision 3, *Transient and Fatigue Cycle Monitoring Program Transient History Evaluation Final Report for Indian Point 2 Addendum 1*, (September 2003) (Ex. NYS00369A-B) ("WCAP-12191"). Entergy further clarified that "[a]s described in the enhancement to the Fatigue Monitoring Program, IP3 will complete a review of existing fatigue analyses of record and enhance the Fatigue Monitoring Program to include additional transient cycles similar to what has been done for IP2." NL-07-153, Attachment 3 at 7 (Ex. NRC000111).

Consistent with Section 5.5.5 of the IP3 Plant Technical Specifications (Ex. NRCR00118), Entergy is required to have an existing program that tracks the cyclic and transient occurrences listed in Section 4.1.5 of the IP3 UFSAR to ensure that components are maintained within the design limits. Consistent with the Staff's guidance in Section 3.0.1 of SRP-LR Rev. 1 and SRP-LR Rev. 2, an applicant may choose an existing program that does not currently meet all of the program elements defined in the GALL Report AMP. In such a situation, the applicant may enhance the existing program to satisfy the GALL Report AMP element prior to the period of extended operation. GALL Report Rev. 1 (Ex. NYS0014A-C) and GALL Report Rev. 2 (Ex. NYS00147A-D) at 3.0-3. Therefore, for IP3, Entergy provided an enhancement to the "parameters monitored or inspected" program element of its existing Fatigue Monitoring Program, which will complete a review of existing fatigue analyses of record and assure all fatigue analysis transients are included with the lowest limiting numbers and have an updated number of design transients accumulated to date. Therefore, upon entering the period of extended operation, the data for past transients for IP2 and IP3 will be incorporated into each of their respective existing Fatigue Monitoring Programs.

With regard to Dr. Hopenfeld's testimony related to "future transients," Entergy provided its justification for using a linear extrapolation to determine the number of cycles to the end of 60-years of operation in its response to the Staff's Audit Item 3 documented in NL-08-057. Specifically, Entergy stated that:

> *[o]perating data shows that the rate of occurrence of transients is decreasing. Continued reduction of transient rate is economically desirable and thus will continue to be pursued. As operating experience is accrued and lessons learned are implemented, the reduction in the rate of transient occurrence is expected to continue. Many transients are projected using a linear rate that is much higher than actually experienced in recent years. The reactor trips used the more recent timeframe to determine the projection rate, but the results are still realistic.*

NL-08-057, Attachment 4 at 1 (Ex. NRC000109).

Sixty-year cycle projections are meant to provide an estimate of what is expected to occur at the end of 60 years of plant operation. The exact methodology used for cycle projection is critical if the methodology is being relied upon to validate an existing analysis for 60-years (i.e., demonstrate an analysis is acceptable in accordance with 10 C.F.R. 54.21(c)(1)(i)) or substantiate a new analysis is acceptable for 60 years (i.e., demonstrate an analysis is acceptable in accordance with 10 C.F.R. 54.21(c)(1)(ii)). However, when the assumption on the number of cycles is being verified through monitoring (i.e., demonstrating an analysis is acceptable in accordance with 10 C.F.R. 54.21(c)(1)(iii)), the amount of margin between the number of accrued cycles and the number of assumed cycles is always known and the cycle projections are not critical to continued operation. Therefore, Dr. Hopenfeld's assertion that Entergy has not justified a straight line extrapolation to determine the remaining number of transients lacks merit because Entergy is using its Fatigue Monitoring Program to verify the extrapolated number of transients and take corrective action in the event those extrapolations are found to be non-conservative. This aging management approach for the CUF_{en} calculations assures that the fatigue limit of 1.0 will not be exceeded. Entergy's procedures for IP2 and IP3 specifically prohibit the actual number of transient cycles from exceeding the number of transient cycles assumed in the fatigue calculations without corrective action, as described in Entergy's response to Audit Item 40 in NL-07-153. NL-07-153, Attachment 3 at 7 through 8 (Ex. NRC000111).

In summary, the "past transients" for IP2 and IP3, referred to by Dr. Hopenfeld, will be incorporated in Entergy's Fatigue Monitoring Program prior to entering the period of extended operation, and Entergy's use of this program does not rely on the prediction of "future" transient cycles to justify continued operation. Such projections only give an indication that components continue to meet the fatigue limit of 1.0

during the period of extended operation consistent with their fatigue analyses. Should Entergy find that the actual number of occurrences for any transient exceeds the projected number assumed in the fatigue calculations, then they would implement corrective actions consistent with their program to ensure that the CUF and CUF_{en} calculations remain valid.

Q126 Do you agree with the points of Dr. Hopenfeld testimony in his summarized conclusions about Entergy's refined EAF analyses?

A126 [GS, AH, OY, CN] No, the Staff does not agree with the points that Dr. Hopenfeld makes in his summarized conclusion about Entergy's refined EAF analyses. Hopenfeld at 19 (RIV000034).

Repeating some of points he made previously in his testimony, Dr. Hopenfeld states, in part, that many assumptions are not explained, numerous potential uncertainties are largely not presented, and several parts of the analyses are far from adequately addressed. Hopenfeld at 19 (Ex. RIV000034).

As we described in our responses to Q120 through Q123, Entergy and Westinghouse clearly described the treatment of DO in the calculation of the F_{en} factors in both the refined EAF analyses presented in WCAP-17199 and WCAP-17200, and the CUF_{en} values for the components presented in LRA Tables 4.3-13 and 4.3-14. As we previously indicated in our response to Q122 and Q123, Westinghouse's and Entergy's justifications for their treatment of DO levels were presented in WCAP-17199 and WCAP-17200, and NL-08-084.

Dr. Hopenfeld also states in part that, without an error analysis, the claimed high degree of accuracy of the results remains questionable at best. Hopenfeld at 19 (Ex. RIV000034). Dr. Hopenfeld then states that an error analysis is required, but provides no basis either from the NRC regulations or the ASME Code for his assertion. The Staff notes that neither the NRC regulations nor the ASME Code require an error analysis, or even hint at any situation or conditions for which such an analysis might be necessary for deterministic calculations, as detailed in our response to Q171.

The regulations at 10 C.F.R. 50.55a incorporate the ASME Code requirements. Dr. Hopenfeld does not discuss this in his pre-filed testimony. In particular, 10 C.F.R. 50.55a(c) requires, in part, that components of the reactor coolant pressure boundary must meet the requirements for Class 1 components in ASME Section III, with limited exceptions specified in 10 C.F.R. 50.55a(c)(2) through 10 C.F.R. 50.55a(c)(4). These regulatory provisions require CUF calculations; to meet this requirement, [redacted], as described in WCAP-17199 and WCAP-17200. WCAP-17199 and WCAP-17200 at 5-20 (Ex. NYS000361 and NYS000362, respectively). For determination of the CUF_{en} values, the F_{en} factors were calculated using the guidance recommended in the GALL Report. Consistent with the guidance in the GALL Report, Westinghouse used the methods [redacted] to compute F_{en}. The CUF_{en} is equal to the CUF multiplied by the F_{en} factor.

Therefore, the refined EAF calculations prepared by Westinghouse [redacted] consistent with the regulations at 10 C.F.R. 50.55a(c). In addition, the refined EAF calculations prepared by Westinghouse [redacted] which is consistent with the guidance of the GALL Report and the fSRP-LR. WCAP-17199 and WCAP-17200 at 1-2 (Ex. NYS000361 and NYS000362, respectively).

Dr. Hopenfeld describes his concerns with the margin of error given that some of the CUF_{en} values are, in his words, “a hair below unity.” Hopenfeld at 19 (Ex.

RIV000034). However, in order for the actual CUF_{en} of a component to approach its calculated CUF_{en}, be it a "hair below unity" or significantly below unity, the severity of each and every one of the transients that occur at IP2 and IP3 must be equal to the severity of each and every one of the transients evaluated in the fatigue calculations, and the actual number of cycles for each and every transient assumed in the fatigue analysis must approach the number of occurrences assumed for each and every transient evaluated in the fatigue calculations. Typically, actual transient severity is significantly less than the transient severity used in the fatigue calculations; however, since this may not always be the case, Entergy uses its Fatigue Monitoring Program to (1) track actual plant transient severities, (2) evaluate these actual transient severities against transient severities used in the fatigue calculations, and (3) ensure that the actual number of transients experienced by the IP units remain within the number of transients evaluated in the fatigue calculations.

In summary, Entergy's Fatigue Monitoring Program provides reasonable assurance that the assumptions made in the fatigue calculations remain valid. Entergy's enhanced Fatigue Monitoring Program was determined to be consistent with the recommendations of AMP X.M1 of GALL Report Rev. 1, as documented in Section 3.0.3.2.6 of the Staff's SER. SER at 3-76 through 3-79 (Ex. NYS00326A-F). Additional information provided in Section 3.2.5 of the AMP Audit Report supports the Staff's conclusions in the SER. AMP Audit Report at 52 through 55 (Ex. NRC000108). Therefore, the CUF and CUF_{en} values from actual transients that occur at the plant will be monitored to remain less than the CUF and CUF_{en} values determined in the fatigue calculations, or corrective action is necessary.

Entergy's calculated CUF and CUF_{en} values provide reasonable and appropriate estimates for the period of extended operation. Additional margin between the actual accumulated CUF and CUF_{en} and the calculated CUF and CUF_{en} is provided by

Entergy's procedures for its Fatigue Monitoring Program for IP2, which include alert limits as an additional measure to ensure that the number of analyzed transients will not be exceeded without initiation of corrective actions. Entergy's procedures for its Fatigue Monitoring Program for IP3 does not allow plant operation if the analyzed number of occurrences of any transient is exceeded unless an appropriate engineering evaluation under the corrective action program has determined that operation of the plant is acceptable. The Staff noted this aspect of the program in Section 3.2.5 of Staff's AMP Audit Report. AMP Audit Report at 53 (Ex. NRC000108). Furthermore, there is additional margin between the calculated CUF and CUF_{en} values and actual CUF and CUF_{en} values because the Fatigue Monitoring Program for both IP2 and IP3, as described by Entergy's implementing procedures, states that they will take corrective actions when a single transient type (e.g., heat-up transient or cool-down transient) approaches its respective action limit, even if the remaining transients included in the analysis are below their action limits. AMP Audit Report at 53 (Ex. NRC000108). Thus, Entergy requires corrective actions in accordance with its Fatigue Monitoring Program prior to the calculated CUF or CUF_{en} values reaching the fatigue limit of 1.0.

Therefore, based on these factors and the method in which Entergy is managing metal fatigue with its Fatigue Monitoring Program, there is margin between the actual accumulated fatigue usage when compared to the calculated cumulative usage factor, including environmental effects of reactor water where applicable.

Q127 Does the Staff agree with Dr. Hopenfeld's opinions that Entergy does not have an adequate program for managing the effects of metal fatigue during the proposed period of extended operation?

A127 [GS, AH, OY, CN] No, the Staff does not agree with Dr. Hopenfeld's opinion that Entergy does not have an adequate program for managing the effects of metal fatigue during the proposed period of extended operation.

First, Dr. Hopenfeld states that Entergy's new calculations do not demonstrate that the CUF_{en} for the components evaluated will not exceed unity during the proposed extended licensing terms. Hopenfeld at 20 (Ex. RIV000034).

As described earlier in our testimony, Entergy is managing metal fatigue with its Fatigue Monitoring Program that (1) tracks actual plant transients, (2) evaluates these actual transients against transient definitions used in the fatigue evaluations to ensure the actual severity is not greater than the severity used in the fatigue evaluations, and (3) ensures that the numbers of transients experienced by the plant remain within the analyzed numbers of transients in the fatigue evaluations. This ensures that the accumulated CUF and CUF_{en} during the period of extended operation will not exceed the fatigue limit of 1.0, or that corrective action is taken before the CUF or CUF_{en} reach the fatigue limit of 1.0, at any time during plant operation. There is also additional margin between calculated CUF and CUF_{en} and actual CUF and CUF_{en}, as previously described in our response to Q126, associated with Entergy taking corrective actions in accordance with its Fatigue Monitoring Program when a single transient type (e.g., heat-up transient or cool-down transient) approaches its respective cycle limit defined in the fatigue analyses.

Second, Dr. Hopenfeld states that in order for Entergy to have an effective AMP to monitor for metal fatigue, it must expand the scope of the fatigue analysis beyond simply representative components, to identify other components whose CUF_{en} may be greater than 1.0. Hopenfeld at 20 and 21 (Ex. RIV000034). As discussed in our response to Q53, Entergy proposed Commitment No. 43, which addresses this concern. The Staff verified the completion of Commitment No. 43 for IP2 during inspections performed in accordance with TI 2516/001. Inspection Report 05000247/2013010 at 6 through 7 (Ex. NRC000182). In addition, verification of IP3's completion of Commitment No. 43 will be performed in accordance with IP73013 in the fall of 2015.

Finally, Dr. Hopenfeld states, in part, that Entergy does not have an adequate AMP for metal fatigue because Entergy has failed to define specific criteria concerning component inspection, monitoring, repair, and replacement. Hopenfeld at 21 and 22 (Ex. RIV000034).

Entergy's Fatigue Monitoring Program is a preventative and mitigative program; therefore, specific criteria concerning component inspection, repair and replacement are not applicable. The ASME Code does not require the preemptive repair or replacement of components that have CUF values that have not exceeded 1.0. In addition, ASME Appendix L indicates that a component is acceptable for continued service if the CUF is greater than 1.0 provided that an inspection of the component reveals no indications and an acceptable flaw tolerance evaluation has been performed. ASME Appendix L at 423 through 427 (Ex. NRC000113). Monitoring of the components is performed by ensuring that the CUF and CUF_{en} values accrued by the components do not exceed the fatigue limit of 1.0 or that corrective action is taken before the CUF and CUF_{en} values exceed the fatigue limit of 1.0. Therefore, Dr. Hopenfeld's assertion that Entergy has failed to define specific criteria concerning component inspection, repair, and replacement is not correct because the "corrective actions" program element of Entergy's Fatigue Monitoring Program includes repair and replacement of a component if it cannot be qualified for service by maintaining the CUF and CUF_{en} to less than 1.0.

<u>Staff Testimony in Response to Dr. Joram Hopenfeld for NYS-38/RK-TC-5</u>

Q128 Have you read the pre-filed written Testimony of Dr. Joram Hopenfeld (Ex. RIV000102 (ADAMS Accession No. ML12171A559) ("Hopenfeld June") dated June 19, 2012?

A128 [GS, AH, OY, CN] Yes, the Staff has read the Testimony of Dr. Joram Hopenfeld, Jr. dated June 19, 2012.

Q129 What is your opinion on Dr. Hopenfeld's view that Entergy is required to identify and investigate additional reactor locations?

A129 [GS, 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.*

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.

Q130 Dr. Hopenfeld refers to MRP-47, Rev. 1 in his pre-filed testimony. Hopenfeld June at 8 (Ex. RIV000102). What is MRP-47, Rev. 1?

A130 [GS, 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.

Q131 Is Dr. Hopenfeld correct that Entergy is required to identify additional locations based on the guidance provided in MRP-47, Rev. 1?

A131 [GS, 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. 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).

Q132 Do you agree with Dr. Hopenfeld's view that the components analyzed for fatigue will likely exceed unity?

A132 [GS, 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. 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 4.314 in Letter NL-10-082 are less than 1.0. 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.

Q133 What is your opinion on Dr. Hopenfeld's view that it is not appropriate for Staff to accept Entergy's vague commitment?

A133 [GS, 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 Staff to accept Entergy's vague commitment to determine at some point in the future what additional locations must be analyzed." 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. SRP-LR Rev. 1 and Rev. 2 at 3.0-3 (Ex. NYS000195 and Ex. NYS000161, respectively). Furthermore, the Commission determined that if the NRC concludes that an aging management program is consistent with the GALL Report, then it accepts the applicant's commitment to implement that aging management program, finding the commitment itself to be an adequate demonstration of reasonable assurance under section 54.29(a). *NextEra Energy Seabrook, LLC* (Seabrook Station, Unit 1), CLI-12-5, 75 NRC 301, 304 (2012) (citing *Entergy Nuclear Vermont Yankee, LLC* (Vermont Yankee Nuclear Power Station), CLI-10-17, 72 NRC 1, 36 (2010); *AmerGen Energy Co., LLC* (Oyster Creek Nuclear Generating Station), CLI-08-23, 68 NRC 461, 467-68 (2008)).

Q134 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?

A134 [GS, AH, OY, CN] No, the Staff does not agree with Dr. Hopenfeld's view. 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. NL-11-032, Attachment 1 at 26 and Attachment 2 at 17 (Ex. NRC000110). Thus, the LRA 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.

It should be noted that the licensee confirmed that that license renewal Commitment No. 43 for IP2, which was required to be implemented prior to entry into the period of extended operation, is complete. NL-13-114 at 1 (NRC000184). Consistent with the implementation schedule for Commitment No.43 for IP3, Entergy will implement the commitment prior to December 12, 2015.

Q135 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?

A135 [GS, 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. 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.

Q136 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?

A136 [GS, 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." 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 period, which may be additional analyses performed during a power uprate license amendment or due to industry experience at other pressurized water reactor plants.

Q137 What is your opinion on Dr. Hopenfeld's view that thermal striping during stratification should be generally considered?

A137 [GS, 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." 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 of the component. 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.

Q138 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?

A138 [GS, 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. 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 CLB for IP2 and IP3. However, by fulfilling Commitment No. 43, the licensee has evaluated additional plant-specific locations that may be more limiting. 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 CLB Class 1 fatigue analyses.

Q139 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"?

A139 [GS, 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." 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." Hopenfeld June at 15-16 (Ex. RIV000102).

Q140 Is Dr. Hopenfeld's use of the term "user intervention" in the context of Commitment No. 44, WESTEMS™ and the Staff's concern accurate?

A140 [GS, 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" as 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.

Q141 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?

A141 [GS, 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." 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 mean 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 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 and WCAP-17200 and performed to fulfill 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.

Q142 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?

A142 [GS, 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." 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 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.

Staff Testimony in Response to Supplemental Report from

Dr. Joram Hopenfeld for NYS-26B/RK-TC-1B and NYS-38/RK-TC-5

Q143 Have you read Supplemental Report of Dr. Joram Hopenfeld In Support of Contention NYS-26/RK-TC-1B and Amended Contention NYS-38/RK-TC-5, dated June 8, 2015 ("Hopenfeld Supplemental Report") (Ex. RIV000144)?

A143 [GS, AH, OY, CN] Yes, we have read the cited Supplemental Report of Dr. Hopenfeld..

Q144 Dr. Hopenfeld states that, given how close Entergy's latest CUF_{en} results are to 1.0, an uncertainty analysis should have been conducted. Hopenfeld Supplemental Report at 4 (Ex. RIV000144). Do you agree?

A144 [GS, AH, OY, CN] No, the Staff does not agree with Dr. Hopenfeld because he does not cite the basis, an NRC regulatory requirement, nor an ASME Code provision for his assertion.

As discussed in our response to Q171, an uncertainty analysis involves the investigation of the variation, or uncertainties, that are possible with inputs used in a calculation. However, such analyses are only performed in probabilistic calculations that estimate the probabilities that certain outcomes will occur. The CUF_{en} calculations performed by Entergy are not probabilistic calculations, so an uncertainty analysis is not necessary. Rather, the CUF_{en} calculations performed by Entergy in accordance with ASME Section III are deterministic. The characteristics of a deterministic calculation are to use conservative and bounding constant input values and required design factors to produce results that are conservative compared to what is actually expected. There is no requirement, either in ASME

Section III or in any NRC regulations, to perform uncertainty analyses for these type of deterministic fatigue calculations.

Q145 Dr. Hopenfeld states that Westinghouse/Entergy failed to conduct a safety assessment to show that IP2 and IP3 can operate safely during normal operations and DBAs, despite the fact that many of the refined CUF_{en} values are very close to 1 without any uncertainty allowance. Hopenfeld Supplemental Report at 5 (Ex. RIV000144). Do you agree?

A145 [GS, AH, OY, CN] No, the Staff does not agree with Dr. Hopenfeld's assertion because it inappropriately mixes the requirements for normal operations, DBA events, and CUF_{en} analyses. Furthermore, he does not mention or credit the plant design requirements that address his concern.

First, to clarify, CUF or CUF_{en} analyses are not required for safety assessments of DBA events. This is because CUF and CUF_{en}, which are indicators of possible fatigue crack initiation, are not a significant contributor to safety during DBA events. DBA events are classified by ASME Section III as Emergency/Faulted (or Service Level C/D) events. Dr. W. E. Cooper, *Background of the Factors of Safety Used in Divisions 1 of Sections III and XI of the ASME Rules for Nuclear Vessels*, (October 1984) (Ex. NRC000174) ("ASME Background Document") at 5. ASME Section III, Paragraph NCA-2142.4, *Design, Service, and Test Limits*, provides definitions for Service Level C and D events and identifies that gross component failure and deformation are the important contributors to safety where safe shutdown of the plant is the main objective (rather than complete component integrity for continued operation). ASME Boiler & Pressure Vessel Code (ASME Code) Section III, *Rules for Construction of Nuclear Power Plant Components, Subsection NCA, General Requirements for Division 1 and 2* (Ex. NRC000178) ("Subsection NCA") at 7. As a result, CUF and CUF_{en} analyses, which are focused on component integrity for continued operation, are not required for Service Levels C and D. As indicated in Figure NB-3222-1 (for Service Levels A and B), Figure NB-3224-1 and Paragraph NB-3224.5 (for Service Level C), and Paragraph NB-3225 (for Service Level D), only Service Levels A and B for normal, upset, and test conditions require fatigue analysis. ASME Section III at 78, 83 and 82, and 82, respectively (Ex. NYS000349) and ASME Background Document at 5 through 9 (Ex. NRC000174).

Second, ASME Section III requires fatigue evaluation for all Class 1 components. Demonstration during plant design that the CUF is less than 1.0 fulfills that requirement and demonstrates that the likelihood of fatigue crack initiation is low. Satisfying ASME Section III requirements is a key input to the plant's overall safety assessment.

Q146 Dr. Hopenfeld states that, based on a review of Entergy's latest fatigue evaluations, the conclusion remains that Entergy has failed to demonstrate that the CUFs of components at Indian Point will not exceed unity and/or succumb to metal fatigue during the proposed periods of extended operation, or that it otherwise has an adequate program for managing the effects of metal fatigue at Indian Point during the proposed periods of extended operations for IP2 and IP3. Hopenfeld Supplemental Report at 5 (Ex. RIV000144). Do you agree?

A146 [GS, AH, OY, CN] No, the Staff does not agree with Dr. Hopenfeld's assertions for the following three reasons, as previously discussed by the Staff and as summarized below for the following three items:

1. Fatigue analyses are performed by trained/qualified individuals and the adequacy of the analysis results are verified without recourse to the originator.

2. The adequacy of Entergy's Fatigue Monitoring Program was addressed in the SER and the associated IP71002 inspection report.
3. Entergy's implementation of the IP2 Fatigue Monitoring Program for the license renewal period was addressed in inspection reports performed in accordance with TI 2516/001. The implementation of the IP3 Fatigue Monitoring Program for the license renewal period will be assessed by the Staff using IP71013 in the fall of 2015.

First, as discussed in our response to Q59, through Entergy's and its vendors' QA Programs that were developed in accordance with Appendix B to 10 C.F.R. Part 50, the review and verification processes ensure that reasonable and appropriate inputs are used for analyses and calculations. In addition, Entergy is required by Appendix B to 10 C.F.R. Part 50 to implement a QA Program that ensures that all 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.

In addition, as given in our response to Q59, an individual performing fatigue analyses must have specialized experience and be specifically trained. Entergy is required by Appendix B to 10 C.F.R. Part 50 to implement a QA 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. Entergy is also required by their QA Program to provide for training of personnel performing activities affecting quality, as necessary, to assure that suitable proficiency is achieved and maintained, and to ensure that there are sufficient records and these records are maintained to document activities affecting quality. Entergy's QA Program provides measures for verifying or checking the adequacy of design, such as by the performance of design reviews or independent design verifications. 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.

Second, the ten program elements of Entergy's Fatigue Monitoring Program were reviewed by the Staff during a combination of the on-site Aging Management Program audit, the Scoping and Screening audit, in-office reviews, and the IP71002 inspection. The following summarizes the Staff's review of the ten program elements of Entergy's Fatigue Monitoring Program.

As discussed in our response to Q101, Section 3.2.5 of the Staff's AMP Audit Report documents the Staff's audit questions and Entergy responses associated with the Fatigue Monitoring Program. AMP Audit Report at 52 through 55 (Ex. NRC000108). This section states that, based on the Staff's review of Entergy's onsite documents, review of Entergy's responses to the Staff's questions, and interviews with Entergy's personnel, the Staff determined that Entergy's "scope of program," "preventive actions," "monitoring and trending," and "acceptance criteria" program elements are consistent with the GALL Report. AMP Audit Report at 55 (Ex. NRC000108).

As documented in SER Section 3.0.4, the Staff concluded that the descriptions and applicability of the plant-specific AMPs and their associated quality attributes provided in Appendix A, Section A.2.1, and Appendix B, Section B.0.3, of the LRA, as well as the "corrective actions," "confirmation process," and "administrative controls" quality assurance elements applied to Entergy's programs, were determined to be consistent with the Staff's position regarding QA for aging management and were consistent with 10 C.F.R. 54.21(a)(3). SER at 3-214 through 3-216 (Ex. NYS00326A-F).

As discussed in our response to Q102, the Staff's review and conclusions of the "parameters monitored or inspected," "detection of aging effects" and "operating experience" program elements are documented in Section 3.0.3.2.6 of the Staff's SER. Based on LRA Amendment 2 dated January 22, 2008, and Entergy's clarification on the corrective actions for the program, the Staff found that the "detection of aging effects" program element for the Fatigue Monitoring Program was consistent with the "detection of aging effects" program element in GALL AMP X.M1 without exception. The Staff found that when the enhancement to Entergy's "parameters monitored or inspected" program element is implemented, Entergy will be monitoring all plant transients that cause cyclic strain, consistent with the Staff's "parameters monitored or inspected" program element in GALL AMP X.M1. The Staff confirmed that the "operating experience" program element satisfies the recommendations in the GALL Report and the guidance in SRP-LR Section A.1.2.3.10. SER at 3-76 through 3-79 (Ex. NYS00326A-F).

As discussed in our response to Q21, the IP71002 Report documented that 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 each 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. IP71002 Report at 4 (Ex. NRC000107).

Third, as previously discussed in our response to Q19 for IP2, the NRC issued TI 2516/001, and the Staff conducted three separate license renewal inspections in accordance with TI 2516/001 that have reviewed a total of 44 license renewal commitments for IP2.

As discussed in our response to Q85, the inspectors stated that no findings were identified associated with the completion of Commitment No. 6, which is an enhancement to Entergy's Fatigue Monitoring Program. Inspection Report 05000247/2013010 at 2 (Ex. NRC000182). The implementation of Commitment No. 6 and the enhancement to the "parameters monitored or inspected" program element of the IP3 Fatigue Monitoring Program will be inspected during the fall of 2015 in accordance with IP71013 to verify that Entergy has adequately completed this enhancement and Commitment No. 6.

Thus, Entergy has demonstrated an adequate aging management program in accordance with NRC guidance in the GALL Report that manages fatigue analyses that were performed by qualified and trained personnel.

Q147 Dr. Hopenfeld states that, throughout the course of the Indian Point license renewal proceeding, Entergy has taken the erroneous position that the ANL data can be applied to IP2 and IP3 directly for accurate metal fatigue predictions without accounting for the known differences between the laboratory and plant environments. Hopenfeld Supplemental Report at 7 (Ex. RIV000144). Do you agree?

A147 [GS, AH, OY, CN] No. As discussed in our response to Q117, the Staff does not agree with Dr. Hopenfeld's statement that laboratory data must be adjusted to account for differences between the laboratory and the actual reactor environment.

The F_{en} methods developed by ANL under NRC funding were specifically created to account for the differences between laboratory test specimens and nuclear components. This accounting is done via two means: (i) variable parameters in the F_{en} equations, and (ii) adjustment factors applied to mean air fatigue curves.

To account for the effects of key environmental variables that affect fatigue life, the F_{en} equations contain expressions for temperature, oxygen, strain rate, and (in the case of ferritic materials) metal sulfur content. These parameters can have significant effects on fatigue life, so capturing their variation between laboratory and plant conditions is important. These expressions were developed based on variation of each parameter during laboratory testing to isolate their individual impact on fatigue life. This allows for evaluation of the actual component conditions for each variable during CUF_{en} calculations so that differences from the laboratory test conditions can be properly considered in the calculations for the plant components.

Other parameters that contribute to fatigue life are accounted for in the development of the air fatigue design curve by the use of adjustment factors. Section 7 of NUREG/CR-6909 describes these factors and how they were developed. NUREG/CR-6909 at 71 (Ex. NYS000357). The factors account for material variability and data scatter, size effects, surface finish effects, and load sequence effects. These factors were specifically developed to account for the observed variation in test results and the differences in key parameters between laboratory test specimens and actual components.

Collectively, the use of the variable parameters and the adjustment factors in the CUF_{en} method accounts for the differences between laboratory test specimens and nuclear components, and the F_{en} methods were developed specifically with that intent.

Q148 Dr. Hopenfeld states that ANL recommended using 0.4 ppm for the DO value during transients for carbon and low alloy steels and 0.05 ppm for the DO value for stainless steel during all transients. Hopenfeld Supplemental Report at 9 (Ex. RIV000144). Do you agree?

A148 [GS, AH, OY, CN] No, the Staff does not agree with Dr. Hopenfeld's statement. Dr. Hopenfeld's statement comes from page A.5 of NUREG/CR-6909, which states, "*For carbon and low-alloy steels, the dissolved oxygen content, DO, associated with a stress cycle is the highest oxygen level in the transient, and for austenitic stainless steels, it is the lowest oxygen level in the transient. A value of 0.4 ppm for carbon and low-alloy steels and 0.05 ppm for austenitic stainless steels can be used for the DO content to perform a conservative evaluation.*" NUREG/CR-6909 at A.5 (Ex. NYS000357). The recommendation in the NUREG is to use the highest oxygen level during a transient for carbon and low-alloy steels, and the lowest oxygen level during a transient for austenitic stainless steels, and not the fixed conservative values recommended by Dr. Hopenfeld. The approach recommended in the NUREG will yield conservative (higher) values of F_{en} for each material, relative to the actual transient encountered at the plant. A simplification of this recommendation that the analyst may choose to use as a very conservative assumption and to avoid the effort necessary to collect DO data for a particular location, is to use a conservative (bounding) value for the DO level. The quoted sentence in NUREG/CR-6909 provides a DO level for each material type that can be used to perform a conservative evaluation; however, the use of the simplified assumption or the values provided is not necessary.

Q149 Dr. Hopenfeld provides an assessment of Entergy's consideration of DO at IP2 and IP3. Hopenfeld Supplemental Report at 9 (Ex. RIV000144). Do you agree with his assessment?

A149 [GS, AH, OY, CN] No, the Staff does not agree with Dr. Hopenfeld's assessment. Dr. Hopenfeld asserts that Westinghouse made an *"…incorrect assumption that a DO level of 0.005 ppm was appropriate."* Hopenfeld Supplemental Report at 9 (Ex. RIV000144). The entirety of the assumption made by Westinghouse states, "*For this screening evaluation, a value of 0.005 ppm is used for the dissolved oxygen (DO) content; which is typical of the PWR environment. Note that for carbon and low alloy steels, this assumption could greatly impact the maximum F_{en}. For PWRs, DO is generally well below 0.05 ppm, except during heatup/cooldown operations. However, elevated DO content usually only occurs when water temperature is low. During these periods of operations, fluid temperatures are in the range where T* = 0 in the second term in the applicable F_{en} equation exponent reduces to zero anyway. Therefore, this assumption is acceptable."* CN-PAFM-12-35 at 28 (Ex. NYS000510). Westinghouse's assumption is consistent with the methods described in NUREG/CR-6909.

PWR water chemistry controls limit DO levels to very low levels (5 ppb (0.005 ppm) or less) other than during low-temperature periods after plant outages during plant ascension before water chemistry is under full control. The DO levels later in plant startup, when the temperatures are above 300 degrees F (150 degrees C), are well below 0.05 ppm, which is the DO threshold value in the F_{en} expressions in NUREG/CR-6583 where the value of the transformed oxygen, O*, changes. For carbon and low-alloy steels, the O* expression in NUREG/CR-6583 is zero when DO levels are below 0.05 ppm, which means the effects of DO are insignificant on F_{en} during these DO conditions. CN-PAFM-12-35 at 26 (Ex. NYS000510). Similarly, the transformed temperature, T*, expression in NUREG/CR-6583 is zero when the temperature is below 302 degrees F (150 degrees C), which means the effects of temperature are insignificant on F_{en} during these temperature conditions. CN-PAFM-

12-35 at 26 (Ex. NYS000510). Water chemistry controls during power ascension are implemented before the temperature reaches 302 degrees F, so the T* value of zero negates the need to consider the time period when the DO level is above 0.05 ppm because the product of T* and O* in the F_{en} expression is zero.

Note that values of O* and T* equal to zero do not mean that environmental effects are neglected. The constants in the F_{en} expressions (0.898 for low-alloy steel and 0.554 for carbon steel) will still produce F_{en} factors that have an approximate value of two, even when the T* and O* terms are zero. This is reflected in the F_{en} values shown for IP2 and IP3 in Tables 5-1 and 5-2 of the Westinghouse screening evaluation. CN-PAFM-12-35 at 40-43 (Ex. NYS000510).

Q150 Dr. Hopenfeld provides a discussion of Entergy's incorrect DO "theory." Hopenfeld Supplemental Report at 10 (Ex. RIV000144). Do you agree with his discussion?

A150 [GS, AH, OY, CN] No, the Staff does agree with Dr. Hopenfeld's discussion. Entergy's application of DO in the F_{en} expressions is consistent with the methods described in NUREG/CR-6909. Further specifics on this topic are provided in our responses to Q151 and Q152.

Q151 Dr. Hopenfeld states that Entergy has never provided or acknowledged the existence of plant measurements of DO during transients at key locations. A review of Entergy's documentation of chemistry control at IP2 and IP3 provides no description of how water chemistry is controlled so that T* O* will always remain zero. Hopenfeld Supplemental Report at 10 (Ex. RIV000144). Is it necessary to use plant measurements of DO during transients at key locations to determine F_{en} factors for carbon steel, low-alloy steel and stainless steel components?

A151 [GS, AH, OY, CN] No, the Staff does not think that it is necessary to use plant measurements of DO during transients at key locations to determine F_{en} factors for carbon steel, low-alloy steel and stainless steel components. As discussed in our responses to Q120 and Q123, water chemistry is tightly controlled by one of Entergy's aging management programs, the Water Chemistry Control - Primary and Secondary Program. This program, whic is described in LRA Section B.1.41, relies on monitoring and control of reactor water chemistry based on the EPRI water chemistry guidelines in TR-1002884 Rev. 5. LRA at B-137 (Ex. ENT00015A-B). Section 3 of TR-1002884 Rev. 5 summarizes power operation chemistry control recommendations that provide specific guidance for the control of DO. TR1002884 Rev. 5 at 3-1 through 3-23 (Ex. NRC000115). It is valid to use bounding DO values based on the chemistry control for the plant for calculating F_{en} as suggested by NUREG/CR-6909 and discussed in our response to Q148. Usually when this is done, the values used are selected to bound the DO levels maintained during actual plant operation. As an example, if water chemistry control parameters restrict plant DO to levels less than 0.010 ppm, it would be appropriate to use 0.010 ppm in the F_{en} calculation for ferritic materials where higher DO values lead to higher F_{en} values. It would also be appropriate to use 0.010 ppm in the F_{en} calculation for austenitic materials, where lower DO values lead to higher F_{en} values, because normal operating DO levels of 0 – 0.010 ppm are well below the O* threshold value of 0.050 ppm in the austenitic stainless steel F_{en} expressions. The approach used by Entergy for the Indian Point F_{en} calculations is consistent with the methods described in NUREG/CR-6909.

Q152 Is it appropriate to use available plant measurements of DO that are obtained from Entergy's Water Chemistry Program for determining F_{en} factors for carbon steel, low-alloy steel and stainless steel components, even if these measurements may be taken during steady state operation?

A152 [GS, AH, OY, CN] Yes, it is appropriate. The available plant measurements of DO are the best indicator of DO levels in the plant, so they are appropriate for use in the F_{en} calculations. In doing so, the Staff considers it to be reasonable to take into account the normal variation in the measurements to yield a conservative Fen calculation. In addition, the guidance in MRP-47 uses a time-averaged value of DO from plant records as a best-estimate of DO for use in the Fen relationships. The Staff agrees with such an approach for use when actual plant DO measurements are used in the Fen expressions. Entergy presented such an approach for using measured DO values for Vermont Yankee using a mean plus one standard deviation value during ASLB Hearings in 2008. Entergy Nuclear Vermont Yankee, LLC (Vermont Yankee Nuclear Power Station), LBP-08-25, 68 NRC 763, 807-809 (2008) (addressing a claim supported by Dr. Hopenfeld, and opposed by, inter alia, Gary Stevens, that Entergy's CUF_{en} calculations do not adequately account for the DO chemistry of the reactor water), (rev'd in part on other grounds CLI-10-17, 72 NRC 1 (2010)). The Board stated in part: "The use of actual DO data from the feedwater system, as well as the use of industry guidance DO values in other systems, was reasonable and appropriate." Id. at 809.

Q153 Dr. Hopenfeld states that any analysis of the effects of the LWR environment on fatigue must consider the synergistic effects of radiation, stress corrosion cracking and thermal embrittlement and that a first step towards this end would be to incorporate the effects of radiation into the F_{en} equation. Hopenfeld Supplemental Report at 15 (Ex. RIV000144). Do you agree?

A153 [GS, AH, OY, CN] NRC agrees with a portion of this argument regarding the effects of radiation on fatigue. The effects of stress corrosion cracking and thermal embrittlement are addressed separately from fatigue. The NRC and consensus standards such as the ASME Code currently treat these effects separately, and it is intended that the separate evaluation approach for these mechanisms is conservative. The NRC continues to find such an approach acceptable and conservative, and is not aware of any specific data to indicate otherwise. Dr. Hopenfeld does not offer any specific research data or evidence to support his contention that treating these mechanisms separately is inadequate, nor does he provide any synergistic models, methods, or evaluations to support his contention. Furthermore, WCAP-14577, Rev. 1-A, *License Renewal Evaluation: Aging Management for Reactor Internals*, (March 2001) (NYS000341) ("WCAP-14577"), states that, "Irradiation embrittlement, by itself, does not result in the initiation of cracks in reactor internals components. Rather, irradiation embrittlement decreases the resistance to crack propagation. Therefore, a crack produced by some other initiation and subcritical growth mechanism would need to be present before irradiation embrittlement became potentially significant." WCAP-14577 at 3-2 of (Ex. NYS000341). Because Entergy's Fatigue Monitoring Program ensures CUF and CUF_{en} calculations remain valid and that the fatigue limit of 1.0 is not exceeded, there is no reason to assume the presence of cracks caused by fatigue that could lead to potentially significant irradiation effects on the structural integrity of the components.

With respect to the effects of radiation on fatigue, the F_{en} methods are considered appropriate for application to materials exposed to significant levels of irradiation, including austenitic stainless steel RVI components, when mandated by regulation or required by the CLB, as discussed in Section 1.3.2 of the draft of

NUREG/CR-6909, Rev. 1, *Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials*, (March 2014) (Ex. NYS00490A-B) ("Draft NUREG/CR-6909 Rev. 1"). Draft NUREG/CR-6909 Rev. 1 at 9 (Ex. NYS00490A-B). Generally speaking, radiation increases the mechanical properties of materials (i.e., yield and ultimate tensile strengths) which, in turn, improves fatigue life (increases the number of applied cycles before the onset of fatigue crack initiation). A 1997 publication by H. Kanasaki, et al., *Fatigue and Stress Corrosion Cracking Behaviors of Irradiated Stainless Steels in PWR Primary Water*, Proceedings of the 5th International Conference on Nuclear Engineering (ICONE5-2372), (May 26-30, 1997) (Ex. NRC000177) ("ICONE5"), which is cited as Reference #110 in Draft NUREG/CR-6909 Rev. 1, concludes that the fatigue life of irradiated stainless steel was longer than that of un-irradiated stainless steel in the range considered by this research and that this increase in fatigue strength was considered due to an increase of tensile strength after irradiation. ICONE5 at 3 (Ex. NRC000177) and Draft NUREG/CR-6909 Rev. 1 at 186 (Ex. NYS00490A-B). For this reason, the NRC and ANL made the conclusion that it is still appropriate to apply the F_{en} method to irradiated components.

Q154 If radiation increases the mechanical properties of materials, which, in turn, improves fatigue life, why doesn't the Staff further improve the F_{en} formulations?

A154 [GS, AH, OY, CN] The total amount of available research test data on irradiation effects on fatigue is insufficient to perform a detailed, comprehensive, statistical evaluation of data in the same manner that was done for unirradiated data to develop the F_{en} expressions. The Staff believes application of the current F_{en} expressions and fatigue curves is conservative for application to irradiated components.

Q155 Dr. Hopenfeld indicates high cycle tube fatigue is an aging phenomenon and must be incorporated into Entergy's Fatigue Monitoring Program. However, Entergy/Westinghouse did not identify tube locations that would be vulnerable to such flow induced vibration failures. Hopenfeld Supplemental Report at 27 (Ex. RIV000144). Do you agree?

A155 [GS, AH, OY, CN] No, the Staff does not agree with Dr. Hopenfeld that high cycle fatigue and flow induced vibrations for steam generator tubes must be incorporated into Entergy's Fatigue Monitoring Program. Age-related degradation of the steam generator tubes is separately managed by Entergy's Steam Generator Integrity Program; thus, it is not necessary to manage age-related degradation of the steam generator tubes in Entergy's Fatigue Monitoring Program. The Staff's review and conclusions regarding the Steam Generator Integrity Program is documented in the Staff's SER Section 3.0.3.2.14. SER at 3-115 through 3-118 (Ex. NYS00326A-F). Therefore, based on its review of Entergy's existing Steam Generator Tube Integrity Program, the Staff concluded that the program elements for which Entergy claimed consistency with the GALL Report are in fact consistent with the GALL AMP XI.M19, "Steam Generator Tube Integrity," including the enhancement to the program.

Q156 Does Entergy manage steam generator tubes for age-related degradation? If so, how does Entergy manage steam generator tubes?

A156 [GS, AH, OY, CN] Entergy's Steam Generator Integrity Program was determined to be consistent with GALL Report Rev. 1, which states that the scope of the program is specific to steam generator tubes, plugs, sleeves and tube supports, and that the inspection activities in the program detect flaws in tubing, plugs, sleeves, and degradation of tube supports needed to maintain tube integrity. Furthermore, the program states that nondestructive examination techniques that are used to inspect all tubing materials and sleeves to identify tubes with degradation that may need to be removed from service or repaired in accordance with plant technical specifications. GALL Report Rev. 1 at XI.M-68 through XI.M-69 (Ex. NYS00146A-C).

Inspection of steam generator tubes is also required by Entergy's Technical Specifications for both IP2 and IP3. The following is a summary of those requirements for IP2 and IP3.

Consistent with Section 5.5.7 of the IP2 Technical Specifications, Entergy is required to establish and implement a Steam Generator Program to ensure that steam generator tube integrity is maintained. IP2 Technical Specifications at 5.5-6 (Ex. NRC000169). Section 5.5.7 of the IP2 Technical Specifications also states, in part, that periodic steam generator tube inspections and the methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along steam generator tubes. IP2 Technical Specifications at 5.5-7 (Ex. NRC000169). In addition, Section 3.4.17 of the IP2 Technical Specifications establishes a limiting condition for operation (LCO), which is a minimum requirement for ensuring safe operation of the unit, that steam generator tube integrity shall be maintained and all steam generator tubes satisfying the tube plugging criteria shall be plugged in accordance with the Steam Generator Program. IP2 Technical Specifications at 3.4.17-1 (Ex. NRC000169).

Consistent with Section 5.5.8 of the IP3 Technical Specifications, Entergy is required to establish and implement a Steam Generator Program to ensure that steam generator tube integrity is maintained. IP3 Technical Specifications at 5.0-13 through 5.0-15 (Ex. NRCR00118). Section 5.5.8 of IP3 Technical Specifications also states, in part, that periodic steam generator tube inspections shall be performed and the methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along steam generator tubes. IP3 Technical Specifications at 5.0-14 (Ex. NRCR00118). In addition, Section 3.4.17 of the IP3 Technical Specifications establishes a limiting condition for operation, which is a minimum requirement for ensuring safe operation of the unit, that steam generator tube integrity shall be maintained and all steam generator tubes satisfying the tube plugging criteria shall be plugged in accordance with the Steam Generator Program. IP3 Technical Specifications at 3.4.17-1 (Ex. NRCR00118). Thus, while Dr. Hopenfeld is correct that steam generator tubes should be managed for age-related degradation for the period of extended operation, he is incorrect in his assertion that Entergy's Fatigue Monitoring Program is the appropriate aging management program for the steam generator tubes. In his assertion, Dr. Hopenfeld failed to acknowledge that Entergy has a separate aging management program for steam generators, and also has existing requirements in the IP2 and IP3 plant Technical Specifications to establish and implement a program that requires inspections of steam generators tubes.

Q157 Dr. Hopenfeld states that ANL has obtained additional data and modified the definitions of T* and O* in ANL-LWRS-47, *Report on Assessment of Environmentally-Assisted Fatigue for LWR Extended Service Conditions*, (September 2011) (Ex. RIV000150) ("ANL-LWRS-47"). Hopenfeld Supplemental Report at 11 (Ex. RIV000144). What is your opinion of using the T* and O* equations in ANL-LWRS-47?

A157 [GS, AH, OY, CN] The expressions for T* and O* in ANL-LWRS-47 are very similar to the expressions in Draft NUREG/CR-6909 Rev. 1. However, ANL-LWRS-47 was completed in 2011, and the NRC's updated research activities that led to Draft

NUREG/CR-6909 Rev. 1 were still underway at that time. As a result, the expressions in Draft NUREG/CR-6909 Rev. 1 are more recent, and the Staff considers them to supersede the expressions in ANL-LWRS-47.

In his testimony, Dr. Hopenfeld's characterization of the F_{en} expressions in Figure 1 of RIV000144 is misleading. In that figure, Dr. Hopenfeld shows that T* = 0 for temperatures less than 150°C using the definitions in NUREG/CR-6909. Hopenfeld Supplemental Report at 11 (Ex. RIV000144). While that is correct, Entergy's argument for the T* O* term being equal to zero is based on the value of O*, not T* (refer to our response to Q149). Therefore, the value of T* is inconsequential.

However, the earlier expressions for F_{en} in NUREG/CR-5704 and NUREG/CR-6583 remain acceptable for use because those expressions are conservative with respect to the updated expressions. Although the NRC does not recommend the use of the expressions in ANL-LWRS-47 in our guidance, they are very similar to the NRC's updated guidance in Draft NUREG/CR-6909 Rev. 1, as discussed in the remainder of this response, so their use will not lead to significantly different CUF_{en} results. In fact, Dr. Hopenfeld does not offer any specific evidence to support his argument favoring the use of the expressions in ANL-LWRS-47. The expressions in ANL-LWRS-47 are not significantly different from the more-recent expressions in Draft NUREG/CR-6909 Rev. 1. In some cases, the expressions between these two documents are identical with the exception of some of the ranges of applicability. For example, for carbon and low-alloy steels, ANL-LWRS-47 defines T* as follows in Equation (11):

$T^* = 0.395$ (for $T \leq 150°C$)

$T^* = (T - 75)/190$ (for $150 < T \leq 325°C$)

$T^* = 1.316$ (for $T \geq 325°C$)

ANL-LWRS-47 at 14 (Ex. RIV000150).

Similarly, Draft NUREG/CR-6909 Rev. 1 defines T* as follows in Equation (43):

$T^* = 0.395$ (for $T < 150°C$)

$T^* = (T - 75)/190$ (for $150 \leq T \leq 325°C$)

Draft NUREG/CR-6909 Rev. 1 at 99 (Ex. NYS00490A-B).

These expressions are mathematically identical, with the exception that Draft NUREG/CR-6909 Rev. 1 does not provide a definition for temperatures higher than 325°C (617°F), whereas ANL-LWRS-47 defines T* as a constant at temperatures above 325°C. The reason for this limitation is described in Draft NUREG/CR-6909 Rev. 1, where it is noted that there is an insignificant amount of fatigue data available at temperatures above 290°C. Draft NUREG/CR-6909 Rev. 1 at 89 (Ex. NYS00490A-B). Consequently, the maximum temperature limit was set at 325°C as a reasonable bound to cover all anticipated LWR operating conditions.

Similarities are also evident when comparisons of the T* and O* expressions are made for austenitic stainless steels and nickel alloys.

Thus, there is nothing in Dr. Hopenfeld's testimony that supports the conclusion that the use of the expressions in ANL-LWRS-47 will lead to significantly different values for CUF_{en}.

Q158 Dr. Hopenfeld states that the methodology employed by Entergy/Westinghouse also introduces another major uncertainty by apparently relating the CUF_{en} to the CUF of record, i.e. $CUF_{en} = F_{en}$ x CUF (of record), as has erroneously been done in all of Entergy's fatigue evaluations to date. Hopenfeld Supplemental Report at 19 (Ex. RIV000144). Do you agree?

A158 [GS, AH, OY, CN] No, the Staff does not agree with Dr. Hopenfeld's statement. In his testimony, Dr. Hopenfeld implies that the effects of the LWR environment were not considered in the design of the Indian Point reactor vessels, which is false. In fact, it is a requirement of ASME Section III, to which the Indian Point vessels were designed, to address the environment to which the component will be exposed. ASME Section III, Paragraph NB-3121, *Corrosion*, requires that, "*Material subject to thinning by corrosion, erosion, mechanical abrasion, or other environmental effects shall have provision made for these effects during the design or specified life of the component by a suitable increase in or addition to the thickness of the base metal over that determined by the design equations."* ASME Section III at 36 (Ex. NYS000349). For other, less prominent aging mechanisms, ASME Section XI requires inspections to look for degradation and, if found, to evaluate or repair it. Dr. Hopenfeld does not provide any supporting inspection data or other evidence to support his claims that the IP units are experiencing any type of measured degradation from other effects, such as swelling, pitting, or cavitation, which would obviate the governing CUF calculations. Finally, using the CUF of record, with possible adjustments necessitated by considerations from actual plant operation, is consistent with ASME Section XI guidance, e.g., L-2000, *Fatigue Usage Evaluation*, of ASME Appendix L. ASME Appendix L at 422 (Ex. NRC000113). ASME Appendix L provides procedures that may also be used when the calculated fatigue usage exceeds the fatigue usage limit defined in the original Construction Code of the facility. ASME Section XI at 140 (Ex. NRC000195).

Q159 Dr. Hopenfeld states that the geometry of many components at IP2 and IP3 have been changed over the past 40 years due to flow-accelerated corrosion and, therefore, the analyses of record are no longer valid due to large and inappropriate reduction in the CLB CUFs. Hopenfeld Supplemental report at 20 (Ex. RIV000144). Do you agree?

A159 [GS, AH, OY, CN] No, the Staff not agree with Dr. Hopenfeld's statement. As stated in our response to Q158, corrosion was addressed during design of the Indian Point vessels, using the CUF analysis of record is consistent with ASME Code procedures, and ASME Section XI requires inspections to look for degradation and, if found, to evaluate or repair it. Dr. Hopenfeld does not provide any supporting inspection data or other evidence to support his claims that the Indian Point units are experiencing any type of measured degradation from other effects, such as swelling, pitting, or cavitation, which would obviate the governing CUF calculations.

Q160 Dr. Hopenfeld states that relying on CUFs of record without allowing for surface changes during service completely ignores the overwhelming importance of surface topography on fatigue life. Hopenfeld Supplemental Report at 21 (Ex. RIV000144). Do you agree?

A160 [GS, AH, OY, CN] No, the Staff not agree with Dr. Hopenfeld's statement. As discussed in our response to Q147, one of the adjustment factors included in the development of the fatigue design curve is to address surface finish effects. That factor is intended to address differences in the surface finish between laboratory test specimens and actual components. Dr. Hopenfeld does not offer any inspection evidence to support his claim that the surfaces of any components in the Indian Point reactors have undergone changes that lead to overwhelming effects on fatigue life, nor is the Staff aware of any field experience that supports this assertion.

Q161 Dr. Hopenfeld states that discontinuities due to wall thinning are expected to occur in components that are located on the balance of plant side (i.e., steam generator secondary side). Hopenfeld Supplemental Report at 21 (Ex. RIV000144). In addition, Dr. Hopenfeld discusses his concerns regarding degradation of the steam generator tubes and steam generator secondary side. Hopenfeld Supplemental Report at 25 (Ex. RIV000144). Do you agree?

A161 [GS, AH, OY, CN] Yes, the Staff agrees with Dr. Hopenfeld that age-related degradation of the steam generators, whether it's the primary side or the secondary side, is a concern for both the period of the original license or the period of extended operation. Dr. Hopenfeld appears to have only focused his attention on CUF analyses and the fact that Entergy's Fatigue Monitoring Program does not manage steam generator components. Consequently, Dr. Hopenfeld has overlooked the fact that Entergy is managing age-related degradation of the primary and secondary side of the steam generators.

The following is a summary of the two activities Entergy performs in order to manage degradation of the steam generators:

(1) LRA Section B.1.18, "Inservice Inspection"

10 C.F.R. 50.55a(g)(4) provides the inservice inspection standards requirements for operating plants and states that, "throughout the service life of a boiling or pressurized water-cooled nuclear power facility, components (including supports) that are classified as ASME Code Class 1, Class 2, and Class 3 must meet the requirements, except design and access provisions and preservice examination requirements, set forth in ASME Section XI editions and addenda of the American Society of Mechanical Engineers Boiler & Pressure Vessel Code Boiler and Pressure Vessel Code.

In addition, Entergy's existing Inservice Inspection Program, which manages age-related degradation of Class 1, 2, and 3 pressure-retaining components, their attachments, and supports, was reviewed by the Staff. The Staff's conclusions regarding the Inservice Inspection Program are documented in Section 3.0.3.3.4 of the Staff's SER. SER at 3-173 through 3-189 (Ex. NYS00326A-F). ASME Section

XI, Subarticle IWB-2500, *Examination and Pressure Test Requirements*, states that components shall be examined and tested as specified in Table IWB-2500-1. ASME Section XI at 80 (Ex. NRC000195). Table IWB-2500-1 provides the following examination requirements specific to the primary side of the steam generators:

- Examination Category B-B, "Pressure Retaining Welds In Vessels Other Than Reactor Vessels" – Steam Generators (Primary Side) - Item Nos. B2.20, B2.31 B.32 and B2.40. ASME Section XI at 83 (Ex. NRC000195)
- Examination Category B-D, "Full Penetration Welded Nozzles in Vessels" - Steam Generators (Primary Side) – Item No. B3.130. ASME Section XI at 84 (Ex. NRC000195)
- Examination Category B-F, "Pressure Retaining Dissimilar Metal Welds In Vessel Nozzles" - Steam Generator – Item Nos. B5.70, B5.80 and B5.90. ASME Section XI at 86 (Ex. NRC000195)
- Examination Category B-G-1, "Pressure Retaining Bolting, Greater Than 2 in. (50 mm) in Diameter" - Steam Generators – Item Nos. B6.90, B6.100 and B6.110. ASME Section XI at 87 (Ex. NRC000195)
- Examination Category B-G-2, "Pressure Retaining Bolting, 2 In. (50 mm) and Less in Diameter" - Steam Generators – Item No. B7.30. ASME Section XI at 89 (Ex. NRC000195)
- Examination Category B-Q, "Steam Generator Tubing" – Item No. B16.10 (Straight Tube Design) and Item No. B16.20 (U-Tube Design). ASME Section XI at 97 (Ex. NRC000195)

In addition, ASME Section XI, Subarticle IWC-2500, *Examination and Pressure Test Requirements*, states that components shall be examined and pressure tested as specified in Table IWC-2500-1. ASME Section XI at 145 (Ex. NRC000195).

Table IWC-2500-1 provides the following examination requirements for the secondary side of the steam generators:

- Examination Category C-A, “Pressure Retaining Welds in Pressure Vessels.” ASME Section XI at 146 (Ex. NRC000195)
- Examination Category C-B, “Pressure Retaining Nozzle Welds In Vessels.” ASME Section XI at 147 (Ex. NRC000195)

(2) LRA Section B.1.37, "Steam Generator Integrity”

Age-related degradation of the steam generator tubes is also managed by Entergy’s existing Steam Generator Integrity Program. The Staff’s review and conclusions regarding the Steam Generator Integrity Program is documented in Section 3.0.3.2.14 of the Staff’s SER. SER at 3-115 through 3-118 (Ex. NYS00326A-F). In summary, based on its review of Entergy’s existing Steam Generator Tube Integrity Program, the Staff concluded that the program elements for which Entergy claimed consistency with the GALL Report are in fact consistent with the GALL AMP XI.M19, "Steam Generator Tube Integrity," including the enhancement to the program. Entergy’s Steam Generator Integrity Program is specific to steam generator tubes, plugs, sleeves and tube supports, and the inspection activities in the program detect flaws in tubing, plugs, sleeves, and degradation of tube supports needed to maintain tube integrity. The inspection activities include nondestructive examination techniques that are used to inspect all tubing materials and sleeves to identify tubes with degradation that may need to be removed from service or repaired in accordance with plant Technical Specifications.

Inspection of steam generator tubes is required by Entergy’s Technical Specifications for both IP2 and IP3, as discussed in our response to Q156.

In addition, as documented in Section 3.0.3.2.17 of the Staff's SER, and based on the Staff's audit and review of Entergy's existing Water Chemistry Control - Primary and Secondary Program, the Staff concluded that the program elements for which Entergy claimed consistency with the GALL Report are in fact consistent with the GALL AMP XI.M2, including the enhancements to the program.

LRA Section B.1.41 states that the Water Chemistry Control – Primary and Secondary Program is an existing program that manages aging effects caused by corrosion and cracking mechanisms. The program relies on monitoring and control of reactor water chemistry based on the EPRI guidelines in Technical Report 1002884 Rev. 5 and Technical Report 1008224 (previously TR-102134), *Pressurized Water Reactor Secondary Water Chemistry Guidelines* – *Revision 6*, (December 2004) (ADAMS Accession No. ML050840514) (Ex. NRC000170) ("TR-1008224 Rev. 6"), (PROPRIETARY). LRA at B-137 (Ex. ENT00015A-B).

TR-1002884 Rev. 5, states, [redacted]

TR-1002884 Rev. 5 at 1-1 (Ex. NRC000115).

TR-1008224 Rev. 6 states, in part, [redacted]

[redacted] TR-1008224 Rev. 6 at 1-1 (Ex. NRC000170).

Therefore, Entergy has requirements and aging management programs developed to manage the primary and secondary sides of the steam generators during the period of extended operation, so it is not necessary for Entergy to perform CUF_{en} calculations for the steam generator tubes or the secondary side of the steam generators as Dr. Hopenfeld suggests. Dr. Hopenfeld has not raised any specific issues or concerns with Entergy's Inservice Inspection Program, Steam Generator Tube Integrity Program, Water Chemistry Program or the IP2 and IP3 Technical Specifications.

Q162 Dr. Hopenfeld states that, since Entergy assumed that CUF = CUF of record, any strain rate related errors in the CUF of record will be reflected in the CUF_{en}. Dr. Hopenfeld further states that using the CUF of record without a correction for non-conservatism with respect to the strain rate would also introduce non-conservatism in the CUFen values. Hopenfeld Supplemental Report at 23 (Ex. RIV000144). Do you agree?

A162 [GS, AH, OY, CN] No, the Staff does not agree with Dr. Hopenfeld's statement. First, the Staff is not aware of any "errors" in the CUF of record associated with strain rates. We assume what Dr. Hopenfeld is referring to are the competing effects from assumptions made with respect to transient severities that can affect strain rate. For example, shorter duration temperature changes will lead to higher thermal stress (which causes higher CUF), but the strain rate associated with shorter duration temperature changes is higher (because the rate of change in stress is faster) so the F_{en} is lowered under this assumption (because F_{en} decreases as strain rate increases). Therefore, there are competing effects from assumptions used to maximize the CUF (e.g., maximize the stress) compared to the assumptions made to maximize F_{en} (e.g., minimize the strain rate).

It is well-recognized and well-documented by component design experts in the nuclear industry that the CUF of record has been shown to be consistently conservative with respect to actual plant operation. For example, NUREG/CR-6260, which documented key stress analyses performed on critical components for the fleet of U.S. nuclear reactors and provided significant insight into the NRC's early investigations of environmental fatigue, recognized this: "*Two studies based on fatigue monitoring of BWR feedwater nozzles in other plants showed that the monitored CUF was a factor of 30 to 50 less than the design basis CUF.*" NUREG/CR-6260 at xxiii (Ex. NYS000355). This observed conservatism in the CUF of record was a result of a common assumption made in those analyses to assume more rapid, or "step," changes in temperature compared to what is actually experienced in operating plants, thereby maximizing the thermal stress in components, which in turn maximized the CUF. Step changes in temperature were often assumed as instantaneous, or "zero ramp time" transients.

The industry performed studies on these competing factors. The EPRI report, *Materials Reliability Program: Evaluation of Controlling Transient Ramp Times Using Piping Methodologies When Considering Environmental Fatigue (F_{en}) Effects (MRP-218)* (September 2007) (Ex. NRC000171) ("MRP-218") documents results from such investigations. The evaluations documented in this report revealed, "*For all cases of ferritic material (i.e., carbon and low alloy steel) on Side B of the joint with low DO levels, the traditional expectation of a step (i.e., zero ramp time) being limiting was obtained.*" MRP-218 at 6-1 (Ex. NRC000171).

The Staff's evaluations of the competing effects of stress vs. strain rate are consistent with EPRI's findings, and support the conclusion that use of the CUF of record is generally conservative and appropriate for use when performing CUF_{en} calculations. Dr. Hopenfeld's testimony does not offer any specific examples or evidence to the contrary.

Q163 Dr. Hopenfeld discusses unexplained changes to the CUF values for Indian Point. Hopenfeld Supplemental Report at 24 (Ex. RIV000144). Do you agree with his discussion?

A163 [GS, AH, OY, CN] No, the Staff does not agree with Dr. Hopenfeld's statements. In his testimony, Dr. Hopenfeld states that, "*…it is impossible to accept the more than two order of magnitude reduction in the CUFs…*" Hopenfeld Supplemental Report at 24 (Ex. RIV000144). However, as discussed in our response to Q162, it is well-recognized and well-documented by stress experts in the nuclear industry that the CUF of record has been shown to be consistently conservative with respect to actual plant operation.

In addition to the example from NUREG/CR-6260 cited in our responses to Q162 and Q210, two other examples are provided that document similar findings with regard to these reductions in CUF that date back more than 20 years. First, T. Sakai, et al., *Implementation of Automated, On-Line Fatigue Monitoring in a Boiling Water Reactor*, 1993 Pressure Vessel and Piping Conference, PVP-Vol. 252, Plant Systems/Components Aging Management, Denver, CO, (July 25-29, 1993) (not in ADAMS but Gary has this) (Ex. NRC000175) ("1993 PVP Paper"), concludes, "*This prediction* [CUF = 0.0074] *is substantially lower than the value estimated in the original design basis stress report for 40 years (0.387 for the thermal sleeve location). The improvements in the rates and magnitudes of the temperature* *changes, as shown in Figure 9, led to this drastically reduced fatigue usage prediction."* 1993 PVP Paper at 7 (Ex. NRC000175). Second, , *Fatigue Evaluation of a BWR Feedwater Nozzle Using an Online Fatigue Monitoring System*, 2002 EPRI Second International Conference on Fatigue (MRP-84), (July 29 – August 1, 2002) (Ex. NRC000173) ("MRP-84"), concludes, "*The fatigue usage computed by* [EPRI FatiguePro Version 2 software (*FPro*)] *for actual transients during a period of 7.75 years is significantly lower than the fatigue usage calculated for the equivalent design basis transients and lower than the allowable value of 1.0 specified in the ASME Code. Two reasons that contribute to this reduction are: (i) the sequence of occurrence of the transients is not the "worst case" sequence typically postulated in traditional design basis evaluation, and (ii) the observed transients, although very similar in shape to the design ones, are in general significantly less severe than the design basis definitions specified in the D*[esign] *S*[pecification]*."* MRP-84 at 28-19 (Ex. NRC000173). There are many other examples of similar results available in the open technical literature.

In light of the well-established industry experience with CUF evaluations, Entergy's reductions in CUF values are not surprising; in fact, they are consistent with typical industry experience accumulated over the past 30 years. Dr. Hopenfeld offers no specific evaluations or data to support his difficulties in accepting Entergy's calculation results.

Q164 Dr. Hopenfeld discusses his concerns regarding degradation of the reactor head penetrations and outlet/inlet nozzle safe ends because of the susceptibility of Alloy 600/82/182 to PWSCC. Hopenfeld Supplemental Report at 26 (Ex. RIV000144). Do you agree?

A164 [GS, AH, OY, CN] Yes, Dr. Hopenfeld is correct that there are concerns regarding degradation of the reactor head penetrations and outlet and inlet nozzle safe ends because of the susceptibility of Alloy 600/82/182 to primary water stress corrosion cracking (PWSCC). However, as discussed in our response to Q165, those concerns are addressed by Commission-mandated requirements in 10 CFR 50.55a.

Q165 Is the susceptibility of Alloy 600/82/182 to primary water stress corrosion cracking an issue addressed in license renewal? Do you agree?

A165 [GS, AH, OY, CN] Yes, the susceptibility of Alloy 600/82/182 to PWSCC is addressed in license renewal and for all plant operating periods because there are Commission-mandated requirements to address PWSCC for all nuclear power plants. Dr. Hopenfeld failed to acknowledge that there are existing Commission requirements in 10 C.F.R. 50.55a that mandate augmented inservice requirements for reactor vessel head inspections and reactor coolant pressure boundary visual inspections. Specifically, 10 C.F.R. 50.55a(g)(6)(ii)(D) states, in part, that all licensees of pressurized water reactors must augment their inservice inspection program with ASME Code Case N-729-1, *Alternative Examination Requirements for PWR Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds, Section XI, Division 1*. In addition, 10 C.F.R. 50.55a(g)(6)(ii)(E) states, in part, all licensees of pressurized water reactors must augment their inservice inspection program by implementing ASME Code Case N-722-1, *Additional Examinations for PWR Pressure Retaining Welds in Class 1 Components Fabricated with Alloy 600/82/182 Materials, Section XI, Division 1*, unless pressure retaining welds fabricated with Alloy 600/82/182 materials that have been mitigated by weld overlay or stress improvement. All licensees, including Entergy, are required to perform actions associated with these Commission-mandated requirements, regardless of whether a renewed license has been granted.

Therefore, Dr. Hopenfeld's assertions regarding the susceptibility of Alloy 600/82/182 to PWSCC are addressed by Entergy.

Q166 Do you have any other comments or opinions on Dr. Hopenfeld's testimony and report?

A166 [GS, AH, OY, CN] No.

Staff Testimony in Response to Dr. Richard T. Lahey, Jr. for NYS-26B/RK-TC-1B

Q167 Have you read the "Pre-Filed Written Testimony of Richard T. Lahey, Jr. Regarding Consolidated Contention NYS-26B/RK-TC-1B," dated December 27, 2011 (Ex. NYSR00344) ("Lahey") and the "Revised Pre-Filed Written Testimony of Dr. Richard T. Lahey, Jr., in Support of Consolidated Contention NYS-26B/RK-TC-1B," dated June 9, 2015 (Ex. NYS000530) ("Revised Lahey")?

A167 [GS, AH, OY, CN] Yes, we have read the cited Pre-Filed Written Testimony of Richard T. Lahey, Jr..

Q168 What is your opinion of Dr. Lahey's pre-filed testimony?

A168 [GS, AH, OY, CN] Significant portions of Dr. Lahey's testimony are not relevant to Consolidated Contention NYS-26B/TC-1B.

First, the portions of Dr. Lahey's Revised Testimony that starts on page 22 and end on page 29 under the titles, "Embrittlement" and "The Consequences of Embrittlement" are not applicable to Consolidated Contention NYS-26B/TC-1B. Revised Lahey at 22 through 29 (Ex. NYS000530). These portions of his testimony relate to neutron embrittlement of the reactor vessel internals and are not associated with the contention that Entergy's LRA does not provide a sufficient demonstration that Entergy will manage the effects of aging due to metal fatigue on key reactor components, in violation of 10 C.F.R. § 54.21(c)(1)(iii).

Second, the portions of Dr. Lahey's Revised Testimony that start on page 29 and end on page 40 under the titles, "GALL, Revision 1"; "Standard Review Plan, Revision 1"; "GALL, Revision 2"; "MRP-227, Revision 0"; and "MRP-227-A" are not applicable to Consolidated Contention NYS- 26B/TC-1B because they discuss the Staff's guidance on the aging management program for reactor vessel internals of pressurized water reactors, embrittlement of reactor vessel internals and inspections of reactor vessel internals. Revised Lahey at 29 through 40 (Ex. NYS000530). These portions of his testimony are unrelated to the contention that Entergy's LRA does not provide a sufficient demonstration that Entergy will manage the effects of aging due to metal fatigue on key reactor components, in violation of 10 C.F.R. § 54.21(c)(1)(iii).

Third, the portion of Dr. Lahey's Revised Testimony that starts on page 32 and ends on page 33 under the title, "Entergy's Opposition to NYS Contention 25" is not applicable to Consolidated Contention NYS-26B/TC-1B. Revised Lahey at 32 and 33 (Ex. NYS000530). This portion is related to NYS Contention 25.

Fourth, the portions of Dr. Lahey's Revised Testimony that start on page 40 and end on page 50 under the titles, "Entergy's License Renewal Application"; "The 2007 LRA and the IP3 Reactor Pressure Vessel"; "The 2007 LRA and RPV Internals"; and "Entergy's NL-10-063 Communication" are not applicable to Consolidated Contention NYS-26B/TC-1B. Revised Lahey at 40 through 50 (Ex. NYS000530). These portions discuss embrittlement of reactor vessel internals. These topics are not applicable to the contention that Entergy's LRA does not provide a sufficient demonstration that Entergy will manage the effects of aging due to metal fatigue on key reactor components, in violation of 10 C.F.R. § 54.21(c)(1)(iii).

Finally, the portions of Dr. Lahey's Revised Testimony that starts on page 50 and ends on page 62 under the titles, "Entergy's NL-11-107 Communication"; and "Entergy's Amended and Revised RVI Plan, and USNRC Staff's November 2014 SSER2" are not applicable to Consolidated Contention NYS-26B/TC-1B because they discuss embrittlement and inspection of reactor vessel internals with Entergy's aging management program for reactor vessel internals. Revised Lahey at 50 through 62 (Ex. NYS000530). This subject is unrelated to the contention that

Entergy's LRA does not provide a sufficient demonstration that Entergy will manage the effects of aging due to metal fatigue on key reactor components, in violation of 10 C.F.R. § 54.21(c)(1)(iii).

Q169 Are there portions of Dr. Lahey's Pre-Filed or Revised Testimony that are related to Consolidated Contention NYS-26B/TC-1B?

A169 [GS, AH, OY, CN] Yes, there are portions of Dr. Lahey's Pre-Filed or Revised Testimony that are related to Consolidated Contention NYS-26B/TC-1B.

Q170 Do you agree with Dr. Lahey's Pre-Filed or Revised Testimony that is related to Consolidated Contention NYS-26B/TC-1B?

A170 [GS, AH, OY, CN] No, the Staff does not agree with Dr. Lahey's Pre-Filed or Revised Testimony that is related to Consolidated Contention NYS-26B/TC-1B. Dr. Lahey states, in part, that Entergy has not presented an error analysis for use of WESTEMS™ to quantify the modeling uncertainties and the effect of code user interactions in WESTEMS™ and that Entergy has not done a systematic fatigue evaluation of limiting RPV internals, considering the synergistic effect of radiation-induced embrittlement and stress corrosion cracking on fatigue-induced failures, and the possible failures due to accident-induced pressure and/or thermal shock loads. Lahey at 38 through 40 (Ex. NYSR10344). Dr. Lahey also raised similar arguments in his revised Testimony. Revised Lahey at 15 through 16 (Ex. NYS000530)

The Staff does not agree with Dr. Lahey. An error analysis is not required for CUF analyses performed in accordance with ASME Section III. We have further specific testimony regarding this point in our response to Q171. As a part of their CUF_{en} analyses, the assumptions and methods used by Westinghouse to calculate the F_{en} factors are described in WCAP-17199 and WCAP-17200 and by Entergy in

NL-08-084, which follow the acceptable guidance in NUREG/CR-5704 and NUREG/CR-6583. The guidance in the GALL Report and SRP-LR specify that NUREG/CR-5704 and NUREG/CR-6583 are acceptable methods to the Staff for addressing the effects of reactor water environment. GALL Report Rev. 1 at X M-1 (Ex. NYS00146A-C) and SRP-LR Rev. 1 at 4.3-5 and 4.3-7 (Ex. NYS000195). Neither the guidance in NUREG/CR-5704 or NUREG/CR-6583 require an error analysis for determining the F_{en} factor.

The NRC regulations at 10 C.F.R. 50.55a incorporate by reference the ASME Code requirements. In particular, 10 C.F.R. 50.55a(c) requires, in part, that components of the reactor coolant pressure boundary must meet the requirements for Class 1 components in ASME Section III, with limited exceptions specified in 10 C.F.R. 50.55a(c)(2) through 10 C.F.R. 50.55a(c)(4). ASME Section III does not require an error analysis for CUF calculations.

A CUF_{en} calculation is performed in two parts. The first part is to calculate the CUF, in which Westinghouse [redacted] as described in WCAP-17199 and WCAP-17200. WCAP-17199 and WCAP-17200 at 5-20 (Ex. NYS000361 and Ex. NYS000362, respectively). The second part is to calculate the F_{en} factor by using the guidance recommended in the GALL Report, in which Westinghouse and Entergy [redacted] respectively. The CUF_{en} is equal to the CUF multiplied by the F_{en} factor. Therefore, Entergy has followed the NRC regulations in 10 C.F.R. 50.55a(c) to perform its CUF calculations [redacted]

However, in order for the fatigue usage of a component to approach the calculated cumulative usage factor, the severity of the transients that occur at IP2 and IP3 must be equal to the transient severity assumed in the underlying fatigue calculations, and the actual number of cycles for each and every transient assumed in the calculation must approach their respective numbers of cycles used in the fatigue analysis (hereafter referred to as cyclic limits). Further, Entergy is managing cumulative fatigue damage with its Fatigue Monitoring Program, which (1) tracks actual plant transients, (2) evaluates these against design transient definitions, and (3) ensures that the number of cycles experienced by the plant remain within the analyzed number of cycles in the fatigue evaluations. Furthermore, for additional margin between the calculated cumulative usage factor and actual accumulated fatigue usage, the Fatigue Monitoring Program for both IP2 and IP3, as described by its implementing procedures, will prompt Entergy to take corrective actions when a single transient type (e.g., heat-up transient or cool-down transient) approaches its respective cycle or action limit, and typically a design fatigue calculation or environmentally-assisted fatigue evaluation for a component does not involve only a single transient type. Thus, prior to the calculated cumulative usage factor and fatigue limit of 1.0 being approached, Entergy will be required to take corrective actions in accordance with its Fatigue Monitoring Program when any single transient type reaches its action or cycle limit. This action or cycle limit for IP2 and IP3 was described earlier in our testimony. Based on the methods in which Entergy's Fatigue Monitoring Program manages metal fatigue, there is margin between the actual accumulated fatigue usage when compared to (1) the calculated cumulative usage factor, including environmental effects of reactor water where applicable, and (2) the fatigue limit of 1.0.

In addition, Dr. Lahey cites the need for a "systematic fatigue evaluation of limiting RPV internals, considering the synergistic effect of radiation-induced embrittlement and stress corrosion cracking on fatigue-induced failures, and the possible failures due to accident-induced pressure and/or thermal shock loads." Lahey at 44 (Ex. NYSR10344). This concern does not directly relate to this contention because it speaks to a broader concern with the aging management of RPV internals, which is addressed by Entergy's RVI Program and industry report MRP-227-A, *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines*, (December 2011) ("MRP-227-A") (Ex. NRC000114A-F). This is also related to a separate contention for the Indian Point LRA.

Q171 Dr. Lahey is also concerned that no error analysis was completed by Entergy for its fatigue calculations. Is there merit to Dr. Lahey's concerns?

A171 GS, AH, OY, CN] No, there is not merit to Dr. Lahey's concern. First, to clarify semantics, the Staff believes Dr. Lahey's testimony relates to an "uncertainty analysis" rather than an "error analysis." These two analyses are different and they are differentiated as follows. An error analysis involves the investigation of errors, or mistakes, made in an analysis. That is addressed by Entergy's QA Program that was developed under Appendix B to 10 C.F.R. Part 50. Entergy's QA Program provides measures for verifying or checking the adequacy of design, such as by the performance of design reviews or independent design verifications, and 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 independently verify the adequacy of the results. The QA Program does not address the possible variation, or uncertainty, of inputs used in a calculation as Dr. Lahey discusses in his testimony.

An uncertainty analysis involves the investigation of the variation, or uncertainty, which is possible with inputs used in a calculation. However, such analyses are only performed in probabilistic calculations that estimate the probabilities that certain outcomes will occur. The fatigue calculations performed by Entergy (i.e., CUF or CUF_{en}) are not probabilistic calculations, so an uncertainty analysis is not necessary. Fatigue calculations performed in accordance with ASME Section III, such as those performed by Entergy for IP2 and IP3 (i.e., CUF or CUF_{en}) are deterministic, rather than probabilistic. The characteristics of a deterministic calculation are to use conservative, bounding, and constant input values and required design factors to produce results that are conservative compared to what is actually expected. For example, lower-bound material strength values and wall thicknesses that account for corrosion are used in the calculation. There is no requirement, either in the ASME Code or in any NRC regulations, to perform uncertainty analyses for these type of deterministic fatigue calculations.

So, in summary, Entergy's fatigue calculations were deterministic in nature using conservative, bounding input values and required design factors, and uncertainty (not error) analysis is neither required nor appropriate.

Q172 Dr. Lahey stated that an error analysis was previously performed in support of a proposal to add more spent fuel into the spent fuel pool at IP2 (NYS000348), and that a similar error analysis should be performed for IP2 and IP3 fatigue analyses. Lahey at 40 (Ex. NYSR10344). Revised Lahey at 71 (Ex. NYS000530). Do you agree?

A172 [GS, AH, OY, CN] No, Staff does not agree with Dr. Lahey's statement. Dr. Lahey cited a calculation for spent fuel at IP2, *Final Design Report for Reracking the Indian Point Unit No. 2 Spent Fuel Pool, Docket No. 50-247* (May 1980) (Ex. NYS000348) ("Spent Fuel Report"). We do not agree with Dr. Lahey's concerns in this regard.

As previously stated, an uncertainty analysis is not required for deterministic evaluations. The uncertainty analysis Dr. Lahey refers to is a probabilistic nuclear analysis using a Monte Carlo code (KENO IV) for the proposed spent fuel pool rerack configuration to demonstrate the multiplication constant (keff) of the system is less than criticality criterion. Spent Fuel Report at 26 (Ex. NYS000348). In that same report, deterministic structural and seismic analyses did not include an uncertainty analysis, and these analyses are similar to those analyses that calculate CUF and CUF_{en}. Spent Fuel Report at 9 through 16 (Ex. NYS000348). The report cited by Dr. Lahey is therefore consistent with our discussion in the response to Q171 that an uncertainty analysis is not performed for deterministic analyses. Therefore, the Spent Fuel Report does not support Dr. Lahey's contention that an error analysis should be performed for the IP2 and IP3 CUF_{en} analyses.

Q173 What are Dr. Lahey's concerns about the refined CUF_{en} reanalysis with respect to the use of WESTEMS™, and do you agree with them?

A173 [GS, AH, OY, CN] Dr. Lahey cites several concerns in his pre-filed testimony. One concern is that engineering judgment or user intervention could have affected the results of the refined CUF_{en} re-analyses. Furthermore, he is also concerned about the analytical framework employed by the WESTEMS™ computer code. Lahey at 40 through 41 (Ex. NYSR10344). Revised Lahey at 71 through 72 (Ex. NYS000530). We do not agree with Dr. Lahey's concerns in this regard.

To support the first concern, Dr. Lahey states the following, citing the Staff's Supplemental SER:

when USNRC Staff issued the Supplemental Safety Evaluation Report, Staff instructed Entergy and Westinghouse, on a going forward basis, to document and disclose the use of engineering judgment and user intervention when *conducting future fatigue analysis using the WESTEMS code . . . Also, USNRC Staff instructed Entergy not to use WESTEMS when conducting analyses under the ASME Standard know as NB-3600.*

Lahey at 41 (Ex. NYSR10344).

This is not an accurate restatement of the SER, Supp. 1. The SER, Supp. 1, does not indicate that the Staff "instructed" Entergy to take any actions regarding the use of WESTEMS™, either to document the use of engineering judgment and user intervention, or regarding the use of WESTEMS™ to conduct NB-3600 fatigue analyses. SER Supp. 1 at 4-2 and 4-3 (Ex. NYS000160). As stated in the SER, Supp. 1, Entergy provided the two commitments regarding use of WESTEMS™ based on documented concerns by the Staff in a separate review related to the AP1000 design certification application. SER Supp. 1 at 4-2 and 4-3 (Ex. NYS000160).

At a public meeting on March 11, 2011, the Staff made a presentation related to an audit performed on Salem Nuclear Generating Station's use of WESTEMS™ fatigue software (March, 2011) (Ex. NRC000119) ("WESTEMS Audit Presentation") during the license renewal process. In that presentation, the Staff indicated that options were currently being considered on how to generically communicate the concerns and results of the audit.

Subsequent to this public meeting, the Staff issued RIS-2011-14. The Staff published a notice of opportunity for public comment on this RIS in the Federal Register (76 FR60939) on September 30, 2011. With respect to Entergy's two commitments, the first commitment relates to documenting the use of engineering judgment and user intervention when conducting future fatigue analysis using the WESTEMS™ code. Entergy committed to document any future use of the WESTEMS™ code, which the Staff noted is in accordance with Appendix B to 10

C.F.R. Part 50. As described in RIS-2011-14, the Staff's review of WESTEMS™ for another applicant (but the same vendor) did not identify issues with the engineering judgment and user intervention exercised for that particular applicant's fatigue evaluations, and thus did not question the accuracy or validity of that particular applicant's fatigue evaluations. RIS-2011-14 at 3 (Ex. NRC000112). The commitment by Entergy for Indian Point confirms that Entergy would provide sufficient documentation of the engineering judgment and user intervention such that these items would be readily retrievable for Staff inspection.

Regarding the second commitment on use of WESTEMS™ to conduct NB-3600 fatigue analyses, this commitment provides assurance that Entergy will not use the WESTEMS™ NB-3600 module for any calculations until the issues identified by the NRC have been resolved.

It should be clarified that the provisions of NB-3600 address the design stress and fatigue analyses for piping, not reactor vessel or RVI components. Therefore, the WESTEMS™ NB-3600 module was not used by Entergy in their fatigue calculations for reactor vessel or RVI components. Instead, Westinghouse used another module of WESTEMS™ based on the methodology of NB-3200. Therefore, Dr. Lahey's testimony regarding the NB-3600 module of the WESTEMS™ computer code is not relevant with respect to Entergy's CUF_{en} calculations. Furthermore, the Staff noted that the WESTEMS™ NB-3600 module was not used in the re-analyses described in WCAP-17199 and WCAP-17200 because [redacted] WCAP-17199 and WCAP-17200 at 5-20 (Ex. NYS000361 and Ex. NYS000362, respectively).

As indicated previously, the Staff performed an audit on Salem Nuclear Generating Station's use of WESTEMS™ fatigue software during the Salem license renewal process. The Staff's review of Salem Nuclear Generating Station's use of

WESTEMS™ NB-3200 module is documented in Section 3.0.3.2.18 of NUREG-2101, *Safety Evaluation Report Related to the License Renewal of Salem Nuclear Generating Station*, (June 2011) (Ex. ENT000195) ("NUREG-2101"). A benchmarking evaluation was performed for the pressurizer surge nozzle and the 1.5-inch boron injection tank (BIT) line locations, as the representative locations, monitored with WESTEMS™ by Salem Nuclear Generating Station. NUREG-2101 at 3-164 (Ex. ENT000195). The Staff determined, based on its review and audit, that Salem Nuclear Generating Station's application of WESTEMS™ provided results that are consistent with a traditional NB-3200 analysis for the Salem Unit 2 pressurizer surge nozzle safe end to pipe weld and the Unit 2 safety injection BIT nozzle to cold leg weld. NUREG-2101 at 3-168 and 3-169 (Ex. ENT000195). No further concerns were identified with the method in which the WESTEMS™ NB-3200 module performs fatigue calculations in accordance with NB-3200.

Dr. Lahey has not supported his concerns, regarding the thermal-hydraulic models and framework, in that he does not identify any instance where the models and framework employed by WESTEMS™ for IP2 and IP3 have invalidated Entergy's environmentally-assisted fatigue calculations.

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Q174 Page 17 of Revised Lahey (Ex. NYS000530) at line 11 states, "the Department of Energy (DOE) and USNRC, in conjunction with various national laboratories, have recently embarked on an ambitious R&D program to understand and resolve issues related to these interacting and synergistic effects [NUREG/CR-7153, Vol. 2, "Expanded Materials Degradation Assessment (EMDA), Aging of Core Internals and Piping Systems" (October 2014) ("EMDA"), at 1-5 (Ex. NYS00484A-B)]." Describe the contents and purpose of NUREG/CR-7153.

A174 [GS, AH, OY, CN] The overall purpose of NUREG/CR-7153 ("EMDA") was to develop the technical bases for subsequent license renewal, or operation beyond 60 years. While applications for subsequent license renewal may not be prepared for several years, both NRC and DOE have an interest in acting proactively to identify issues that may affect the ability of plants to operate for up to 80 years. First, the analytical timeframe is extended to 80 years, encompassing the subsequent license renewal-operating period. Second, the materials and systems addressed in the EMDA are generally extended to all of those which fall within the scope of aging management review for license renewal. Thus, in addition to piping and core internals, the EMDA also includes the reactor pressure vessel (RPV), electrical cables, and concrete structures. EMDA at 1 through 3. (Ex. NYS00484A-B).

Q175 Explain what timeframe is addressed with respect to nuclear reactors operation in NUREG/CR-7153.

A175 [GS, AH, OY, CN] As discussed in the EMDA, applications for subsequent license renewal may not be prepared for several years, but both NRC and DOE have an interest in acting proactively to identify issues that may limit the ability of plants to operate for up to 80 years. In addition, in preparation for likely future submittals, the NRC is obligated to provide guidance to applicants on the expected contents of subsequent license renewal applications and to develop the technical bases for making safety determinations in their license reviews. Through the Light Water Reactor Sustainability (LWRS) Program, the DOE undertakes research to understand the fundamentals of component aging, thereby supporting industry in sustaining the domestic fleet as an economic and strategic resource. To take advantage of their common interests, NRC and DOE have entered into a Memorandum of Understanding (MOU) to cooperate on research activities related to long-term operations. One activity initiated under the MOU is the EMDA. EMDA at 1. (Ex. NYS00484A-B).

The EMDA expands the analytical timeframe to 80 years to encompass a potential second 20-year license-renewal operating-period, beyond the initial 40-year licensing term and a first 20-year license renewal. The EMDA provides assessments for a second 20-year license renewal operating period, and provides analyses of the key degradation modes for current and expected future service, key degradation modes expected for extended service, and suggested research needs to provide technical information for operation up to 80 years. EMDA at iii. (Ex. NYS00484A-B).

Q176 What is your opinion with respect to the applicability of the results in NUREG/CR-7153 to the IP2 and IP3 license renewal application?

A176 [GS, AH, OY, CN] The results in the EMDA are not applicable for the IP2 and IP3 LRA, since the LRA is associated with continued operation from 40 years to 60 years. The results of the EMDA may potentially be applicable to IP2 and IP3 in the future, if Entergy decides to pursue an LRA for continued operation from 60 years to 80 years.

Q177 Page 17 of Revised Lahey (Ex. NYS000530) at line 17 states, "In addition, the federal government has also embarked on a fairly large research program, known as the Light Water Reactor Sustainability Program, which includes research into whether the different materials and LWR components can continue to perform their intended function during the extended operation of a nuclear reactor. [DOE, Light Water Sustainability Program, Material Aging and Degradation Technical Program Plan (August 2014) ("DOE Report") (Ex. NYS000485)]." Explain what timeframe is addressed in the Light Water Sustainability Program.

A177 [GS, AH, OY, CN] The purpose of the research performed under the LWRS program is to develop the scientific basis for understanding and predicting long-term environmental degradation behavior of materials in nuclear reactors. The work is intended to provide data and methods to assess performance of systems, structures, and components (SSCs) essential to safe and sustained reactor operations. The LWRS program is designed to support the long-term operation (LTO) of existing domestic nuclear power generation. The report, which was published in August 2014, focused on extending reactor service to beyond 60 years of operation because extended operation will increase the demands on materials and components. DOE Report at page iii (Ex. NYS000485).

Q178 What is your opinion with respect to the applicability of the results in the Light Water Sustainability Program to the IP2 and IP3 LRA?

A178 [GS, AH, OY, CN] The IP2 and IP3 LRA is for the first 20-year license renewal period (i.e., continued operation from 40 years to 60 years). Thus, the results of the LWRS program are not applicable to the IP2 and IP3 LRA because the LRA does not extend reactor service beyond 60 years. Since the Light Water Reactor Sustainability Program is not applicable to the IP2 and IP3 LRA, this reference cited by Dr. Lahey is not relevant and does not support his assertion that Entergy has not demonstrated that it has a program that will manage the effects of aging of critical components.

Q179 Page 18 of Revised Lahey (Ex. NYS000530) at line 3 states, "the effects of embrittlement, especially loss of fracture toughness, make existing cracks in the affected materials and components less resistant to growth." Dr. Lahey also cites of the Staff presentation to the ACRS, "Technical Brief on Regulatory Guidance for

Evaluating the Effects of Light Water Reactor coolant Environments in Fatigue Analyses of Metal Components" (December 2, 2014) ("ACRS Brief"), at 56-58 (Ex. NYS000486). Please describe the purpose of the ACRS meeting that was held at NRC Headquarters on December 2, 2014.

A179 [GS, AH, OY, CN] The purpose of the ACRS meeting that was held at NRC Headquarters on December 2, 2014 was to respond to a request, from the ACRS Subcommittee on Metallurgy and Reactor Fuels, for a technical brief on Draft Regulatory Guide DG-1309 and Draft NUREG/CR-6909 Rev. 1 from the ACRS Subcommittee on Metallurgy and Reactor Fuels. ACRS review of draft regulatory guides is commonly solicited prior to or during their release for public comment.

Q180 Please describe the content and the status of DG-1309, "Guidelines for Evaluating the Effects of Light-Water Reactor Coolant Environments in Fatigue Analyses of Metal Components."

A180 [GS, AH, OY, CN] DG-1309 is a proposed revision to RG 1.207. DG-1309 contains methods and procedures that the Staff considers acceptable for use in determining the acceptable fatigue lives of components evaluated by a CUF calculation in accordance with the fatigue design rules in ASME Section III to account for the effects of light-water reactor coolant environments.

DG-1309 was published for 60 days of public comment in November 2014. The public comment period closed in January 2015. The Staff is working to address all public comments received and finalize the document for publication in concert with the final publication of Draft NUREG/CR-6909 Rev. 1. Final publication of DG-1309, which will then become RG 1.207, Rev. 1, is currently anticipated in early 2016.

Q181 Please describe the content and the status of Draft NUREG/CR-6909 Rev. 1, Rev. 1, "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials."

A181 [GS, AH, OY, CN] Draft NUREG/CR-6909 Rev. 1 is currently in draft form and is a revision to NUREG/CR-6909, which was published in February 2007. Draft NUREG/CR-6909 Rev. 1 was published for public comment in April 2014. The Staff is still working to address all of the public comments received and finalize the report for publication. The document contains the technical basis for the guidance provided in DG-1309, which is the proposed Revision 1 of RG 1.207.

Draft NUREG/CR-6909 Rev. 1 contains the results of recent NRC research activities, and documents the complete technical bases for the guidance provided in the proposed Revision 1 of RG 1.207. Draft NUREG/CR-6909 Rev. 1 provides updates and improvements to the environmental fatigue factor (F_{en}) approach based on an extensive update to the fatigue cyclic strain data from testing and results available since the report was first published in 2007.

Q182 Please describe the outcome(s) of the ACRS meeting that was held at NRC Headquarters on December 2, 2014.

A182 [GS, AH, OY, CN] As indicated in the presentation material provided to the ACRS, the purpose of the December 2, 2014 meeting was to brief the ACRS subcommittee on the background, research activities, and content of DG-1309 and Draft NUREG/CR-6909 Rev. 1. ACRS Brief at 2 (Ex. NYS000486). As described in the meeting transcripts, the subcommittee's intent was to gather information, analyze relevant issues and facts, and formulate a proposed position and action as appropriation for deliberation by the full committee. ACRS Transcripts at 4 (Ex. NRC000173). At the conclusion of the meeting, the subcommittee identified that the brief was a preliminary review in parallel with the documents' publications for public comment, that they would address the issue further after receipt of public comments, and that there would not be any full committee action at that time. ACRS Transcripts at 106 (Ex. NRC000173).

Q183 Can you summarize the information on pages 56-58 of the presentation (Ex. NYS000486) made to the ACRS subcommittee?

A183 [GS, AH, OY, CN] Slides 56-58 of the ACRS presentation contain a summary of information regarding the latest technical reasoning behind the mechanism of environmentally-assisted cracking. Leading researchers have postulated two mechanisms to explain enhanced fatigue cracking that occurs in water environments. The first mechanism is identified as film rupture/slip dissolution, whereby incremental strain breaks the oxide layer that forms on the material, and continued crack extension occurs by dissolution/oxidation of the freshly exposed surface. The second mechanism is identified as hydrogen-induced cracking, whereby corrosion of the material causes hydrogen production that diffuses into inclusions in the material, which then act as new crack initiation sites. Either mechanism, or some combination of both mechanisms, are possible explanations of observed cracking behavior in laboratory testing. In addition, the effects of dynamic strain aging are discussed, which is a condition where material solutes segregate to dislocations that result in strong elastic interactions between the solute and dislocation stress-strain field. It is believed by the ANL and NRC researchers that dynamic strain aging may also be a factor in the enhanced fatigue cracking that occurs when materials are exposed to water environments.

The discussion on slides 56-58 of the ACRS presentation do not indicate additional effects on fatigue cracking that have not been previously considered. Rather, they are offering a more informed and detailed scientific explanation for the causes of enhanced cracking with the intent of enhancing the direction of future research and modeling endeavors.

Q184 Can you explain whether pages 56-58 of the presentation (Ex. NYS000486) made to the ACRS subcommittee discusses the effects of embrittlement or loss of fracture toughness?

A184 [GS, AH, OY, CN] Neither the effects of embrittlement, nor loss of fracture toughness, are discussed in my ACRS presentation. The effects of radiation, which leads to material embrittlement, is discussed in our response to Q193. Loss of fracture toughness is another matter that is not related to fatigue crack initiation and CUF or CUF_{en} evaluations, so it is not relevant to this fatigue contention.

Q185 What is your opinion with respect to Dr. Lahey's claim that the presentation (Ex. NYS000486) made to the ACRS subcommittee on pages 56-58 support his statement, "the effects of embrittlement, especially loss of fracture toughness, make existing cracks in the affected materials and components less resistant to growth"?

A185 [GS, AH, OY, CN] Dr. Lahey provides no specific, supporting evidence or data to support this portion of his testimony with respect to the impact of embrittlement and loss of fracture on calculations of CUF or CUFen. Whereas NRC agrees that these effects are important to the growth of existing cracks that could initiate due to fatigue, stress corrosion cracking or other mechanisms, and that these effects should be evaluated accordingly for either postulated cracks or cracks found in service, CUF and CUFen calculations are intended to demonstrate the absence of fatigue cracks, so discussion of the effects of loss of fracture toughness on CUF and CUFen is not relevant.

Q186 What is your opinion on this portion of the testimony with respect to Dr. Lahey's claim that Entergy has not demonstrated that it has a program that will manage the effects of aging of critical components?

A186 [GS, AH, OY, CN] In summary, once again, this reference cited by Dr. Lahey is not relevant and does not support his assertion that Entergy has not demonstrated that it has a program that will manage the effects of aging of critical components.

Q187 Page 18 of Dr. Lahey's Revised Testimony at line 19 states, "a combined effect of thermal aging and irradiation embrittlement could reduce the fracture resistance even further to a level neither of these degradation mechanisms can impart alone" [Chen, et al., "Crack Growth Rate and Fracture Toughness Tests on Irradiated Cast Stainless Steels," NUREG/CR-7184 (Revised December 2014) ("NUREG/CR-7184"), at xv (Ex. NYS00488A-B)]. What is your opinion with respect to the quoted statement from page xv of NUREG/CR-7184?

A187 [GS, AH, OY, CN] The complete sentence from NUREG/CR-7184 is, "*It is suspected that a combined effect of thermal aging and irradiation embrittlement could reduce the fracture resistance even further to a level neither of these degradation mechanisms can impart alone."* NUREG/CR-7184 at xv (Ex. NYS00488A-B). Dr. Lahey truncated the sentence and left out the words, "*It is suspected*." With these additional words, the context of the statement does not agree with Dr. Lahey's assertion.

Q188 Does NUREG/CR-7184 provide a scientific basis to support a definite conclusion that, "a combined effect of thermal aging and irradiation embrittlement could reduce the fracture resistance even further to a level neither of these degradation mechanisms can impart alone"?

A188 [GS, AH, OY, CN] No, the report does not provide a scientific basis to support a definitive conclusion. Although, NUREG/CR-7184 suspected that the combined effects of thermal aging and irradiation embrittlement could further reduce the fracture resistance, the test results presented in the report do not support that concern. As a matter of fact, the report identifies that a test program has been initiated to investigate the joint effects of thermal aging and irradiation damage on the cracking susceptibility and fracture resistance of CASS. NUREG/CR-7184 at xv (Ex. NYS00488A-B).

Q189 Does Dr. Lahey provide a scientific basis to support a definite conclusion that, "a combined effect of thermal aging and irradiation embrittlement could reduce the fracture resistance even further to a level neither of these degradation mechanisms can impart alone"?

A189 [GS, AH, OY, CN] No, Dr. Lahey did not provide a scientific basis to support such a conclusion.

Q190 What is your opinion on this portion of the testimony with respect to Dr. Lahey's claim that Entergy has not demonstrated that it has a program that will manage the effects of aging of critical components?

A190 [GS, AH, OY, CN] Yes, Dr. Lahey provided reference to the EMDA by USNRC, DOE and associated contractors that discusses "embrittlement" and "high or low cycle fatigue". However, as discussed in our responses to some of the previous questions, the EMDA is intended to extend operation from 60 years to 80 years. The applicability of that document and its results do not pertain to the LRA for IP2 and IP3, which is associated with continued operation from 40 years to 60 years.

Q191 Page 19 of Dr. Lahey's Revised Testimony at line 14 stated that, "Multiple recent reports and studies from USNRC, DOE, and associated contractors recognize the lack of understanding of the interrelationship between embrittlement, high or low cycle fatigue, and shock loads for highly fatigued and/or embrittled components made of CASS, non-cast stainless steels, or other alloys." Does Dr. Lahey provide any references by "USNRC, DOE and associated contractors" that discuss "embrittlement," or "high or low cycle fatigue"?

A191 [GS, AH, OY, CN] Yes, Dr. Lahey provided reference to NUREG/CR-7153 by USNRC, DOE and associated contractors that discusses "embrittlement" and "high or low cycle fatigue". However, as discussed in our responses to some of the previous questions, NUREG/CR-7153 is intended to extend operation from 60 years to 80 years. The applicability of that document and its results do not pertain to the LRA for IP2 and IP3, which is associated with continued operation from 40 years to 60 years.

Q192 Does Dr. Lahey provide any references by "USNRC, DOE and associated contractors" that discuss "shock loads"?

A192 [GS, AH, OY, CN] No, Dr. Lahey did not provide any reference by "USNRC, DOE and associated contractors" that discussed "shock loads." Once again, this reference cited by Dr. Lahey is not relevant and does not support his assertion that that Entergy has not demonstrated that it has a program that will manage the effects of aging of critical components.

Q193 Page 19 of Dr. Lahey's Revised Testimony at line 23 states, "Draft NUREG/CR-6909 Rev. 1 (March 2014 15 (draft) (Ex. NYS000490A-B)), at 11 ("it is not possible to quantify the impact of irradiation on the prediction of fatigue lives in PWR primary water environments compared to those in air.")" Can you provide the context of the sentence cited on page 11 of Draft NUREG/CR-6909 Rev. 1 and explain how it relates to the F_{en} methods described in that document?

A193 [GS, AH, OY, CN] Dr. Lahey truncated the sentence and interprets the portion of the sentence that he quotes out of context. The complete sentence on page 11 states, "*Although some small-scale laboratory fatigue ε–N test data indicate that neutron irradiation decreases the fatigue lives of austenitic SSs, particularly at high strain amplitudes, it is not possible to quantify the impact of irradiation on the prediction of fatigue lives based on the limited data currently available*." As we discuss in our response to Q154, the total amount of research test data available is insufficient to perform a detailed, comprehensive, statistical evaluation of data in the same manner that was done for unirradiated data to develop the F_{en} expressions. The Staff believes application of the current F_{en} expressions and fatigue curves is generally conservative for application to irradiated components. Therefore, in Draft NUREG/CR-6909 Rev. 1, the NRC noted that additional fatigue data on reactor structural materials irradiated under light water reactor operating conditions are needed to determine whether there are measurable effects of neutron irradiation on the fatigue lives of these materials and, if so, to better define how those impacts may be quantified.

Q194 Please describe the process and the public involvement during the development of Draft NUREG/CR-6909 Rev. 1.

A194 [GS, AH, OY, CN] As a part of developing Draft NUREG/CR-6909 Rev. 1, the Staff elected to publish a draft of the report for 60 days of public comment via the Federal Register process commonly used for other NRC documents. During the 60-day comment period, any member of the public may submit written comments to the

NRC on the draft report. The Staff is then required to respond to all comments received during the 60-day period as a part of finalizing the document for internal NRC technical and administrative reviews and final publication.

Q195 Is it common for the Staff to solicit public comment for NUREG-type document?

A195 [GS, AH, OY, CN] No, it is not common for the Staff to solicit public comment for NUREG-type document. The practice of soliciting public comments is not common for NUREG-type documents published by the NRC. It does happen from time to time, and is allowed by NRC procedures, but it is not the usual practice. In addition, the NRC practices an openness policy that allows any member of the public to comment on any NRC document at any time, even if the document is published.

Q196 Why was it necessary for the NRC to solicit public comments on Draft NUREG/CR-6909 Rev. 1?

A196 [GS, AH, OY, CN] For Draft NUREG/CR-6909 Rev. 1, the Staff felt that soliciting public comments was important in view of the significant stakeholder feedback provided to the Staff in the many industry and public forums that have taken place since 2007, primarily associated with ASME Code proceedings. As a part of those forums, several stakeholders requested an opportunity to review a draft of the NRC's research activities before they were published. In order to be responsive to those requests, the Staff decided to publish a draft of the report for public comment. In addition, the Staff was interested in receiving comments from interested external stakeholders to ensure the completeness and the quality of the document. The Staff was also interested in further stakeholder feedback on a few technical aspects of the report, as requested in the 2014 Federal Register Notice that noticed the report for public comments, Federal Register, Vol. 79, No. 74, (April 17, 2014) ("FRN 2014-08792") (Ex. NRC000186). FRN 2014-08792 at 21812 (Ex. NRC000186).

Q197 Can you describe the general technical quality of the comments provided by interested external stakeholders?

A197 [GS, AH, OY, CN] More than 200 separate comments were received on Draft NUREG/CR-6909 Rev. 1 from ten individual stakeholders. ACRS Brief at 51 and 52 (Ex. NYS000486). Most of these individuals represented research organizations from around the world that are, or were previously, involved in environmentally-assisted fatigue research. As such, the majority of the comments were of very high technical caliber and quality. For this reason, the Staff continues to work on responses to those comments more than one year after the public comment period closed in June 2014.

One of the NRC's goals during the review of Draft NUREG/CR-6909 Rev. 1 was to receive the most thorough and independent technical review possible on the results of the Staff's research activities. Based on the number of comments received and the affiliation of the commenters, the Staff believes that this was accomplished.

Q198 Please indicate whether the intervener organizations for the Indian Point license renewal contentions submitted any comments objecting to the completeness and the quality of Draft NUREG/CR-6909 Rev. 1 during the public comment period.

A198 [GS, AH, OY, CN] Neither, the State of New York, Riverkeeper, nor any of the witnesses for these organizations submitted public comments for either Draft NUREG/CR-6909 Rev. 1 or DG-1309..

Q199 Page 20 of Dr. Lahey's Revised Testimony at line 16 states, "A recent paper presented at an MPA Seminar in Stuttgart, Germany confirms that, at present, the USNRC staff does not have a clear solution to the challenges posed by synergistic age-related degradation mechanisms [Stevens, Gary L., et al., "Observations and Recommendations for Further Research Regarding Environmentally-Assisted Fatigue Evaluation Methods," 40th MPA- Seminar, Materials Testing Institute, University of Stuttgart, Stuttgart, Germany (October 6-7, 2014) ("MPA Paper") (Ex. NYS000491)]." Please describe the contents of the conference paper, "Observations and Recommendations for Further Research Regarding Environmentally-Assisted Fatigue Evaluation Methods."

A199 [GS, AH, OY, CN] The MPA seminar paper, *Observations and Recommendations for Further Research Regarding Environmentally-Assisted Fatigue Evaluation Methods*, was written by two staff members and an ANL contractor for presentation at the 40th MPA-Seminar sponsored by the Materials Testing Institute at the University of Stuttgart in Stuttgart, Germany on October 6-7, 2014.

The paper briefly summarizes results of the NRC's research activities that led to the drafts of DG-1309 and Draft NUREG/CR-6909 Rev. 1, and describes several observations made by the Staff and its contractors during the course of the performance of those activities.

The paper also provides recommendations for further research efforts that the Staff and its contractors identified where further research could yield reduced conservatism in EAF evaluations, including more refined, material-specific fatigue curves, fatigue curves for ferritic materials based on material tensile strength, component testing (rather than small-scale specimen testing), ASME Code CUF calculation methods, and the effect of neutron irradiation on fatigue crack initiation in austenitic stainless steels.

Publication of technical conference papers that document the status of on-going research activities performed by NRC researchers is common, and is another form of public interaction that the Staff uses to solicit interested stakeholder feedback on their research activities.

Q200 Please describe and identify any new information in the conference paper that is not already captured in Draft NUREG/CR-6909 Rev. 1.

A200 [GS, AH, OY, CN] There is no new research information in the paper compared to what is contained in Draft NUREG/CR-6909 Rev. 1. The recommendations made in the paper, however, are not contained in Draft NUREG/CR-6909 Rev. 1 because the paper contains the results of recent NRC research activities related to environmentally-assisted fatigue and the application of specific methods. It is not appropriate for a NUREG document to make recommendations for further research activities. Therefore, the paper was developed as a way for the NRC research staff and/or NRC contractors to document their observations and recommendations to interested stakeholders in an appropriate manner.

Q201 Do you agree with Dr. Lahey's assertion that, "the USNRC staff does not have a clear solution to the challenges posed by synergistic age-related degradation mechanisms"?

A201 [GS, AH, OY, CN] No, it is the Staff's opinion that the current approach to managing age-related degradation mechanisms is consistent with current knowledge and understanding. In areas where information is lacking, or indicating that further investigation or knowledge is needed, NRC is funding research activities to provide further understanding or confirmation of industry findings. As always, the Staff continue to monitor industry-wide field experience for indications of new challenges so that the Staff can re-direct its focus when warranted. Entergy's aging management programs are consistent with the Staff's current understanding of age-related degradation, which is documented in the GALL Report. The Staff's review that evaluated Entergy's aging management programs is documented in SER, SER Supp. 1 and SER Supp. 2. SER, SER Supp. 1 and SER Supp. 2 in general (Ex. NYS00326A-F, NYS000160, and NYS000507, respectively).

Q202 Does Dr. Lahey provide a clear proposal for addressing the challenges posed by synergistic age-related degradation mechanisms?

A202 [GS, AH, OY, CN] No, we have rebutted Dr. Lahey's testimony and his testimony does not provide specific evidence to demonstrate that Entergy's aging management programs are deficient.

Q203 Page 66 of Dr. Lahey's Revised Testimony at line 9 states, "For that component, Westinghouse was able bring the CUF_{en} just below unity by performing successive analyses using modified design transient conditions and 60 year projected cycles (rather than CLB cycles), and applying new, specially-developed heat-up and cool-down spray transients." Is using the 60-year projected number of cycles in a fatigue calculation prohibited?

A203 [GS, AH, OY, CN] No, the ASME Code does not prohibit the use of the 60-year number of projected cycles in fatigue calculations. Paragraph NB-3222.4 of ASME Section III requires consideration of the specified service loadings for the component involving cyclic application of loads and thermal conditions when performing an analysis for cyclic operation. ASME Section III at 81 (Ex. NYS0000349). Paragraph L-2210(c) of Appendix L in ASME Section XI allows the use of actual plant operating data when performing operating plant fatigue assessments. ASME Appendix L at 422 (Ex. NRC000113).

Dr. Lahey has not identified any restrictions on the use of 60-year projected cycles when performing fatigue calculations in accordance with the ASME Code.

Q204 Can you explain how Indian Point's fatigue monitoring program will be capable of managing the aging effects of metal fatigue for those components that used 60-year projected numbers of cycles?

A204 [GS, AH, OY, CN] Sixty-year cycle projections provide an estimate of the loadings that are anticipated after 60 years of plant operation. These projections represent the best estimate of the loadings expected over the 60-year operating period based on trending of past plant performance, and they are used in the fatigue calculations to assure that the fatigue limit of 1.0 is maintained throughout the period of extended operation. An objective of the Fatigue Monitoring Program is to continually validate the assumptions used in the fatigue analyses, including the number of cycles for each transient does not exceed that assumed in the fatigue analyses.

With the Fatigue Monitoring Program limiting each transient to its assumed number of cycles used in the fatigue analyses, both of the following conditions must be satisfied for the CUF or CUF_{en} to exceed the fatigue limit of 1.0:

- the severity of each and every transient experienced at the plants must be equal to or greater than the transient severities assumed in the fatigue calculations, and
- the actual number of cycles for each and every transient experienced at the plants must equal or exceed the number of cycles used in the fatigue calculations.

Entergy is managing cumulative fatigue damage with its Fatigue Monitoring Program by (1) tracking the actual severities and numbers of plant transients, (2) evaluating the severities of the actual transients to the severity of the transients used in the fatigue calculations, and (3) ensuring that the numbers of cycles of each plant transient remains within the numbers of cycles of the transient used in the fatigue calculations. It should be noted that if the actual severity and the actual number of cycles for each and every transient used in the fatigue calculation were equal to the assumptions, the accumulated fatigue usage would only be equal to that determined by the calculation and not necessarily the fatigue limit of 1.0.

Entergy's program for IP2 includes 'alert cycles', which are defined as the number of transient cycles which are projected to accumulate in the next two operating periods. The number of 'alert cycles' is calculated by taking the number of transient cycles accumulated during the prior operating cycle and multiplying by 2; the total projected number of transient cycles is then determined by adding the 'alert cycles' to the total accumulated number of transients to date. If this projected number of transient cycles remains below the number of transient cycles used in the fatigue evaluation, no corrective action is required. If this projected number of transient cycles exceeds the number of transient cycles used in the fatigue evaluation, a condition report is generated to ensure that corrective actions are taken. Conversely, Entergy's program for IP3 does include the use of alert cycles and does not allow continued plant operation if the number of transient cycles used in the fatigue evaluation for any transient is exceeded unless an appropriate engineering evaluation, developed under the corrective action program has determined that plant operation is acceptable. NL-08-057, Attachment 5 at 7 through 8 (Ex. NRC000109). . NL-08-057, Attachment 5 at 7 through 8 (Ex. NRC000109).

The Fatigue Monitoring Program for both IP2 and IP3, as described in Entergy's implementing procedures, require corrective actions when a single transient type (e.g., heat-up transient or cool-down transient) approaches its respective cycle limit. Since the fatigue calculations for each component typically include multiple transients (e.g., not just a single transient), this approach provides additional margin against exceeding any fatigue limits.

Therefore, the Fatigue Monitoring Program allows for taking corrective actions well before the CUF or CUF_{en} values approach the fatigue limit of 1.0.

Q205 Describe "specially-developed heat-up and cool-down spray transients."

A205 [GS, AH, OY, CN] Entergy [redacted] Westinghouse Report CN-PAFM-13-40, *Indian Point Unit 2 Pressurizer Spray Model Transfer Function Data Base Development and Environmental Fatigue Evaluations*, (2013) (PROPRIETARY) (Ex. NYS000512) ("CN-PAFM-13-40") at 15. Entergy explained [redacted] CN-PAFM-13-40 at 40 (Ex. NYS000512).

In addition, as described in their response to the Staff's audit questions in NL-08-057, Entergy explained the details of its modified operating procedures associated with the specially-developed heat-up and cool-down spray transients. Entergy stated that both IP2 and IP3 instituted two main operating changes consistent with the generic Westinghouse program to address surge line thermal cycling. The first change included continuous, reduced pressurizer spray flow, which minimizes the temperature differential between the reactor coolant system, the pressurizer, and the surge line. A reduced temperature differential minimizes the possible thermal stresses that can occur during an insurge transient. The second change included changing the plant startup operating procedures to eliminate drawing and then collapsing a pressurizer air bubble so that reactor coolant pumps are run to sweep air out of the reactor coolant system and reactor pressure vessel. The collapsing of the pressurizer air bubble early in the startup procedure resulted in a significant reduction in the number of insurge transients. NL-08-057, Attachment 5 at 7 through 8 (Ex. NRC000109).

Q206 Is the usage of those “specially-developed heat-up and cool-down spray transients” prohibited in fatigue calculations?

A206 [GS, AH, OY, CN] No, ASME Section III does not prohibit the use of specially-developed heat-up and cool-down spray transients in fatigue calculations. It should be noted that these “specially-developed heat-up and cool-down spray transients” were developed because they are more representative transients for heat-ups and cool-downs based on plant data. Entergy further clarified that it was assumed that the plant data from 1991 to 2012 was representative, regarding spray operations, for the entire plant life. Specifically, the plant computer data prior to April 14, 1997, was representative of heatup/cooldown operations prior to the implementation of modified operational procedures (MOPs and the plant computer data for the period after April 14, 1997, is representative of plant spray operations through 60 years of operation. CN-PAFM-13-40at 14 through 15 (Ex. NYS000512).

ASME Section III requires consideration of the specified service loadings for the component involving cyclic application of loads and thermal conditions when performing an analysis for cyclic operation in accordance with ASME Section III, Paragraph NB-3222.4.

Dr. Lahey has not identified restrictions on the use of transients based on plant data when performing a fatigue analysis in accordance with ASME Section III. When the plant was originally designed over 40 years ago, the design transients were defined conservatively, and the goal of the fatigue analysis was to meet the ASME Section III design limit of 1.0. Use of transients based on plant data to achieve a more accurate value of CUF or CUF_{en} can be performed to reduce the conservatism in the calculation.

Q207 In your opinion, can you explain how Indian Point's fatigue monitoring program will be capable of managing the aging effects of metal fatigue for those components that used specially-developed heat-up and cool-down spray transients" in fatigue calculation?

A207 [GS, AH, OY, CN] As previously discussed, in order for the CUF or CUF_{en} to exceed the fatigue limit of 1.0, both of the following conditions must be satisfied:

- the severity of each and every transient experienced at the plants must be equal to or greater than the transient severities assumed in the fatigue calculations, and
- the actual number of cycles for each and every transient experienced at the plants must equal or exceed the number of cycles used in the fatigue calculations.

Entergy is managing cumulative fatigue damage with its Fatigue Monitoring Program by (1) tracking the actual severities and numbers of plant transients, (2) evaluating the severities of the actual transients to the severity of the transients used in the fatigue calculations, and (3) ensuring that the numbers of each and every actual transient remains within the numbers of cycles of each and every transient used in the fatigue calculations. It should be noted that if the actual severity and the actual number of cycles for each and every transient used in the fatigue calculation were equal to the assumptions, the accumulated fatigue usage would only be equal to that determined by the calculation and not necessarily the fatigue limit of 1.0.

The Fatigue Monitoring Program for both IP2 and IP3, as described in Entergy's implementing procedures, require corrective actions when a single transient type (e.g., heat-up transient or cool-down transient) approaches its respective cycle limit. Since the fatigue calculations for each component typically include multiple transients (e.g., not just a single transient), this approach provides additional margin against exceeding any fatigue limits.

Therefore, the Fatigue Monitoring Program allows for taking corrective actions well before the CUF or CUF_{en} values approach the fatigue limit of 1.0.

Q208 Page 67 of Dr. Lahey's Revised Testimony at line 22 states, "it clearly shows the iterative process used by Westinghouse in which safety margin is removed in its environmentally-assisted fatigue (EAF) calculations in an effort to reduce the output or result below CUF_{en} = 1.0." Does the ASME Code define what safety margin is in a fatigue calculation?

A208 [GS, AH, OY, CN] The ASME Code does not explicitly define what safety margin is in a fatigue calculation. However, the document, *Criteria of the ASME Boiler and Pressure Vessel Code for Design by Analysis in Sections III and VIII, Division 2*, (1969) (Ex. ENT000188) ("ASME Section III Criteria Document") contains ASME's technical bases for the design requirements in ASME Section III, including fatigue analysis. That document discusses "safety factors" used to develop the design fatigue curves of two on stress or a factor of twenty on cycles. ASME Section III

Criteria Document at 20 (Ex. ENT000188). However, the document also notes, "*These factors were intended to cover such effects as environment, size effect, and scatter of data, and thus it is not to be expected that a vessel will actually operate safely for twenty times its specified life.*" ASME Section III Criteria Document at 20 (Ex. ENT000188). There are other design factors applied on stresses that are used in fatigue calculations so, even ignoring any conservative assumptions applied by analysts in the calculations, it is difficult to explicitly quantify an exact overall safety margin present in fatigue calculations. Generally, ASME Section III uses a factor of three against failure. K. R. Rao, *Companion Guide to the ASME Boiler & Pressure Vessel Code, Criteria and Commentary on Select Aspects of the Boiler & Pressure Vessel and Piping Codes, Third Edition, Volume 1* (2009) (Ex. ENT000191) ("ASME Companion Guide") at 159-160 and ASME Background Document at 4 and 6 (Ex. NRC000174).

Q209 Does the ASME Code prohibit the iterative process performed by Westinghouse?

A209 [GS, AH, OY, CN] No, the ASME Code does not prohibit the iterative process performed by Westinghouse. In fact, iteration during design has been quite common in ASME Code Class 1 fatigue calculations for more than 50 years. This is because the typical objective of the analyst is to demonstrate acceptability (as opposed to demonstrating margin). A secondary cause may have been because schedules and budgets were limited, which would limit the amount of time an analyst could spend performing the calculation. Therefore, especially in the early days of the industry in the 1960s before computers were prevalent or affordable, analysts would make many simplifying assumptions to shorten the problem and make it easier to solve. If the assumptions were too conservative, the analyst would modify one or more of the assumptions to be more realistic and less conservative,, the analyst would remove one or more of the assumptions and repeat the analysis. This iteration would continue until acceptable results were achieved. In the case of a fatigue calculation, acceptable results meant the CUF was calculated to be less than or equal 1.0. Experienced analysts tend to make fewer iterations because of their past trials and familiarity with performing fatigue calculations. That practice continues today, although perhaps to a much lesser extent with the availability of high-powered computers. The nature of these conservatisms is reflected in our response to Q56. For example, such iteration is described in NUREG/CR-6260, where it was noted, "*Since the licensees' design basis analyses were based on the ASME Code of record, it was uneconomical for the licensee to attempt to reduce the CUF to lower and lower values by removing conservative assumptions once the Code requirements were met. Given more funding and time, further calculations could have been performed to reduce the existing stress values by using more realistic loadings or more detailed analysis models. These reduced stresses would result in lower CUFs*." NUREG/CR-6260 at xxi. (Ex. NYS000355).

In our experience, it would be unusual for an experienced analyst to perform a fatigue calculation without some amount of iteration.

Q210 Can you identify other safety margins that exist in a fatigue calculation?

A210 [GS, AH, OY, CN] As described our response to Q208, there are two factors used to develop the design fatigue curves, and there are other design factors applied on stresses that are used in fatigue calculations. Beyond that, there may be other several additional margins in fatigue calculations based on conservative assumptions applied by analysts. Other factors of conservatism that are very common in fatigue calculations are transient severity and grouping of transients, as reflected in our response to Q54. Given the variability in assumptions made by different analysts, it is difficult to explicitly quantify an exact overall safety margin present in fatigue calculations. The NRC's opinion is that fatigue calculations tend to be very conservative, as evidenced by the lack of observed thermal fatigue failures for components where a design CUF calculation was made. A detailed discussion of this and the methods an analyst can use to make adjustments to fatigue calculations that might reduce the CUF is documented in Section 4.3 of NUREG/CR-6260. NUREG/CR-6260 states the changes fall into two broad categories, conservative assumptions made by the analyst or ASME Code changes that have been made since the edition of the ASME Code of record for the plant. NUREG/CR-6260 at 4-5 (Ex. NYS000355). Two examples that NUREG/CR-6260 discusses to reduce CUF values are (1) by separating the enveloped load pair with the overall combined number of cycles into more detailed load pairs, each with its own set of cycles, which can sometimes be significantly reduced, and (2) by using actual cycles that the plant has experienced to date if the numbers of cycles extrapolated are less than the numbers of design basis cycles. NUREG/CR-6260 at 4-5 and 4-6 (Ex. NYS000355).

Q211 Page 68 of Dr. Lahey's Revised Testimony at line 7 states, "For example, Westinghouse has reported that the CUF_{en} for the RHR/Accumulator nozzles and associated piping for IP3 is 0.9961, which is extremely close to unity [CN-PAFM-09-77 (2010), at 61 tbl. 5-36 (Ex. NYS000366)]. Line 20 of Page 68 states, "Even assuming this CUF_{en} calculation is accurate, it does not account for the possibility that a highly fatigued component, which does not yet have signs of significant surface cracking, may be exposed to an unexpected seismic event or shock load that could cause it to fail." Does Dr. Lahey provide any quantitative description (i.e., pressure or temperature profile) of the “shock load” or “unexpected seismic event” he is referring to?

A211 [GS, AH, OY, CN] No, Dr. Lahey does not provide any quantitative description (i.e., pressure or temperature profile) of the “shock load” or “unexpected seismic event” to which he is referring. As long as the CUF is less than the fatigue limit of 1.0, and therefore no cracks are expected to initiate, the structural performance of a “highly fatigued component” is unaffected, in particular for design basis loads. Other than his opinion, Dr. Lahey does not provide any quantitative description as a basis to support his opinion.

Q212 Does Dr. Lahey provide any quantitative comparison as a basis to conclude that the “shock load” is more severe than the design transients used in fatigue calculation?

A212 [GS, AH, OY, CN] No, other than his opinion, Dr. Lahey does not provide any quantitative comparison as a basis to support his opinion.

Q213 Does Dr. Lahey provide calculations to quantitatively support his statement that an unexpected seismic event or shock load would increase the CUF_{en} value beyond 1.0?

A213 [GS, AH, OY, CN] No, other than his opinion, Dr. Lahey does not provide any quantitative comparison as a basis to support his opinion. Furthermore, as described in our response to Q145, seismic and other accident loads are considered as part of the design calculations, and Dr. Lahey’s argument is irrelevant because the important failure mode for these severe loads is gross structural overload and deformation, not fatigue crack initiation.

Q214 Does Dr. Lahey provide any scientific basis or operating experience to support his statement that an unexpected seismic event or shock load would a highly fatigued component fail?

A214 [GS, AH, OY, CN] No, other than his opinion, Dr. Lahey does not provide any scientific basis or operating experience to support his opinion. Furthermore, his testimony indicates he is not familiar with ASME Code calculations and design because he argues that fatigue cracks exist in components where it has been demonstrated that they do not, and he suggests evaluations that are not required by NRC regulations or the ASME Code.

Q215 The Revised Statement of Position on Consolidated Contention NYS-26B/RK-TC-1B ("Revised SOP") (Ex. NYS000529) on pages 27 and 28 ascribes discussion of in-core fatigue failures of irradiated baffle-to-former bolts have been observed in operating PWRs to Dr. Lahey. What are your thoughts on the documents cited in the Revised SOP and their statements in support of the Revised SOP?

A215 [GS, AH, OY, CN] The Revised SOP cites several supporting documents, including NYS-000324, NYS000353, NYS000316, and NYS000315. In most cases the contents of the cited document have either been taken without the context provided or needed context has not been provided.

Q216 Does NYS-000324 support the statement that the baffle-to-former bolt failures are due to fatigue?

A216 [GS, AH, OY, CN] Not entirely. NYS000324 states that "Possible causes of baffle/former bolt cracking are irradiation-assisted SCC (IASCC), irradiation embrittlement, stress relaxation, and fatigue, or some combination of these."

NYS000324 at 2-30. Based on this document it is clear that the failures are not solely the result of “fatigue failures” as the Revised SOP asserts.

Q217 Are there other documents that discuss the cause of the baffle-to-former bolt failures?

A217 [GS, AH, OY, CN] Yes, MRP-51, *Hot Cell Testing of Baffle/Former Bolts Removed from Two Lead PWR Plants*, (November 2001) (Ex. NRC000176) (“MRP-51), provides information from evaluation of failed bolts removed from Point Beach Unit 2. MRP-51 at v.

Q218 How did MRP-51 characterize the failures?

A218 [GS, AH, OY, CN] MRP-51 states “The fractures appear to be caused by irradiation assisted stress corrosion cracking (IASCC).” MRP-51 at 4-1.

Q219 Was MRP-51 issued prior to or after NYS-000324?

A219 [GS, AH, OY, CN] MRP-51 is dated November 2001 and NYS-000324 was issued one year earlier, October 2000.

Q220 How does NYS000316 relate to fatigue failures of RVI components?

A220 [GS, AH, OY, CN] NYS000316 does not explicitly mention RVI components. One bullet in this document discusses flaws found in Babcock & Wilcox supplied components where the usage factor was significantly less than 1.0, but the document provides no context on what components, what plants, and most importantly the cause of the defects. For example, a flaw caused by stress corrosion cracking may have a fatigue usage factor that is very small, because fatigue and SCC different. Without the underlying context this statement has no value. NYS000316 at 2.

Q221 On page 55 of Revised Lahey, Dr. Lahey states that “Yes, similar to NL-10-063 (Ex. NYS000313), the applicant acknowledges, in NL-12-037, that other PWRs have experienced material degradation and failure of multiple RVI components …”. What does NL-12-037 say about industry operating experience with failure of PWR RVI components?

A221 [GS, AH, OY, CN] NL-12-037 has a different perspective on the PWR industry operating experience. Quoting from Dr. Lahey’s reference, the “Operating Experience” portion of NL-12-037, states that “[r]elatively few incidents of PWR internals aging degradation have been reported in operating U.S. commercial PWR plants.” NYS000496, Attachment 1 at 98.

Q222 How does NYS000353 describe fatigue failure of in-core components, or bolting in general?

A222 [GS, AH, OY, CN] NYS000353 does not address fatigue failures of in-core components such as baffle-to-former bolting. As the title of this document implies, this report addresses Class 1 piping and associated pressure boundary components. NYS000353 addresses aging management of non-RVI bolting, but it does not fatigue failures of bolting.

Q223 How does NYS000315 address fatigue failures of RVI components?

A223 [GS, AH, OY, CN] NYS000315 quotes from the LRA for Wolf Creek. The following passage is directed to baffle-to-former and barrel-to-former bolts:

> “That is, the high predicted usage factor, the additional aging effects requiring mitigation, and the fact that some of these are synergistic (e.g., fatigue and

> the other cracking mechanisms) dictate that management of the fatigue usage factor in these bolts will be insufficient by itself, and that an aging management program must be constructed for the bolts which either adequately address all of these effects, or which will ensure their safety function despite these effects."

NYS000315 at 2.

We agree with this statement, that fatigue usage factors alone are not sufficient to manage all forms of cracking in all cases, such as the cited case where IASCC is the predominant mechanism governing degradation of these bolts. That is exactly the reason aging management of the baffle-to-former and barrel-to-former bolts addressed by the RVI Program uses volumetric examination of the bolts to identify cracks without reliance on a fatigue usage calculation.

Q224 What are your thoughts on this statement from page 28 of Revised SOP: "Moreover, unlike postulated nuclear reactor accidents, the fatigue failures of in-core bolts are actual events that have happened and will likely happen again for sufficiently stressed materials."

A224 [GS, AH, OY, CN] From consideration of the source documents used to support this statement and the debunking that the documents support the notion of fatigue failures of in-core bolts, the statement in the Revised SOP is not accurate in stating that "fatigue failures of in-core bolts are actual events."

Staff Testimony in Response to Dr. Richard T. Lahey, Jr for NYS-38/RK-TC-5

Q225 Have you read the pre-filed written declaration of Dr. Richard T. Lahey, Jr. (Ex. NYS000374) (ADAMS Accession No. ML12171A513) ("Lahey June") dated June 18, 2012 and "Revised Pre-Filed Written Testimony of Dr. Richard T. Lahey, Jr., in Support of Joint Contention NYS-38/RK-TC-5," dated June 9, 2015 (Ex. NYS000562) ("Revised Lahey June")?

A225 [GS, AH, OY, CN] Yes, the Staff has read the declaration of Dr. Richard T. Lahey, Jr. dated June 18, 2012.

Q226 Has Dr. Lahey identified any errors in Entergy's license renewal application?

A226 [GS, AH, OY, CN] No.

Q227 Has he identified any omissions of required information in Entergy's LRA?

A227 [GS, AH, OY, CN] No.

Q228 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?

A228 [GS, AH, OY, CN] Dr. Lahey has identified two concerns as it relates to metal fatigue and Contention NYS-38/RK-TC-5. Lahey June at 12 (Ex. NYS000374).

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. 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. Lahey June at 12 (Ex. NYS000374).

Q229 How does Dr. Lahey describe "user intervention" in his testimony?

A229 [GS, 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 CUF_{en} values using codes such as WESTEMS™." Lahey June at 24 (Ex. NYS000374).

Q230 Is Dr. Lahey's characterization of "user intervention" consistent with the concerns identified with the use of WESTEMS™?

A230 [GS, 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.

Q231 Can you clarify what is meant by "user intervention" as it relates to WESTEMS™ and the Staff's concerns?

A231 [GS, 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. RIS-2011-14 at 2 (Ex. NRC000112).

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.

Q232 What was the purpose of the Staff issuing RIS 2011-14?

A232 [GS, 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.

Q233 Regarding WESTEMS™, Dr. Lahey states in his pre-filed testimony on June 18, 2012 (Ex. NYS000374) [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?

A233 [GS, 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 as discussed in Q59.

Further, the assumptions to be used in future analyses (that may be needed by Entergy 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 discussed in Q53.

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. As discussed in Q59, 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 and WCAP-17200, the EAF analyses that were performed 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.

Q234 Dr. Lahey states on Page 26 of in his pre-filed testimony dated June 18, 2012 (Ex. NYS000374) 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?

A234 [GS, 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 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.

Q235 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?

A235 [GS, AH, OY, CN] Dr. Lahey believes that an error analysis must be done. Lahey June at 27 (Ex. NYS000374). First, to clarify semantics, the Staffs believe Dr. Lahey's testimony relates to an "uncertainty analysis" rather than an "error analysis." These two analyses are different and they are differentiated as follows. An error analysis involves the investigation of errors, or mistakes, made in an analysis. An uncertainty analysis involves the investigation of the variation, or uncertainty, which is possible with inputs used in a calculation. However, such analyses are only performed in probabilistic calculations that estimate the probabilities that certain outcomes will occur. The fatigue calculations performed by Entergy (i.e., CUF or CUF_{en}) are not probabilistic calculations, so an uncertainty analysis is not necessary. Fatigue calculations performed in accordance with ASME Section IIII, such as those done by Entergy (i.e., CUF or CUF_{en}) are deterministic, rather than probabilistic. The characteristics of a deterministic calculation are to use conservative, bounding input values and required safety factors to produce results that are conservative compared to what is actually expected. For example, lower-bound material strength values are used in the calculation. There is no requirement, either in the ASME Code or in any NRC regulations, to perform uncertainty analyses for these type of deterministic fatigue calculations.

So, in summary, Entergy's fatigue calculations were deterministic in nature using conservative, bounding input values and required safety factors, and uncertainty (not error) analysis is neither required nor appropriate.

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. NL-07-153, Attachment 3 at 7 through 8 (Ex. NRC000111). Thus, the Staff believes an error analysis is not needed.

Q236 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?

A236 [GS, 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.

Q237 What is the Staff's opinion related to Dr. Lahey's concerns?

A237 [GS, AH, OY, CN] It is the Staff's opinion that Entergy has determined the most limiting locations for metal fatigue calculations because Commitment No. 43 has been completed by Entergy for IP2. For IP3, Commitment No. 43 will be completed by Entergy before its entrance of the extended period of operation in December 2015.

Q238 In its revised pre-filed testimony dated Jun 9, 2015, Dr. Lahey is also concerned that no error analysis was completed by Entergy for its fatigue calculations. Is there merit to Dr. Lahey's concerns?

A238 [GS, AH, OY, CN] No. Dr. Lahey states in part that Entergy has not completed an error analysis for its fatigue calculations. Revised Lahey June at 74 through 76 (Ex. NYS000562). As discussed in Q171 and Q235, in summary, Entergy's fatigue calculations were deterministic in nature using conservative, bounding, and constant input values and required design factors to produce results that are conservative compared to what is actually expected. Furthermore, 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.

Q239 On pages 74 through 77 of Revised Lahey June (Ex. NYS000562), Dr. Lahey stated that an error analysis was previously performed in support of a proposal to add more spent fuel into the spent fuel pool at Indian Point Unit 2 (NYS000348), and that a similar error analysis should be performed for IP2 and IP3 fatigue analyses. Do you agree?

A239 [GS, AH, OY, CN] No, Staff does not agree with Dr. Lahey's statement. Dr. Lahey cited a calculation for spent fuel at IP2, *Final Design Report for Reracking the Indian Point Unit No. 2 Spent Fuel Pool, Docket No. 50-247* (May 1980) (Ex. NYS000348) ("Spent Fuel Report"). We do not agree with Dr. Lahey's concerns in this regard.

As previously stated, an uncertainty analysis is not required for deterministic evaluations. The uncertainty analysis Dr. Lahey refers to is a probabilistic nuclear analysis using a Monte Carlo code (KENO IV) for the proposed spent fuel pool rerack configuration to demonstrate the multiplication constant (keff) of the system is less than criticality criterion. Spent Fuel Report at 26 (Ex. NYS000348). In that same report, deterministic structural and seismic analyses did not include an uncertainty analysis, and these analyses are similar to those analyses that calculate CUF and CUF_{en}. Spent Fuel Report at 9 through 16 (Ex. NYS000348). The report cited by Dr. Lahey is therefore consistent with our discussion in the response to Q171 that an uncertainty analysis is not performed for deterministic analyses. Therefore, the Spent Fuel Report does not support Dr. Lahey's contention that an error analysis should be performed for the IP2 and IP3 CUF_{en} analyses.

Q240 Page 20 of Revised Lahey June (Ex. NYS000562) at line 3 states, "the Department of Energy (DOE) and USNRC, in conjunction with various national laboratories, have recently embarked on an ambitious R&D program to understand and resolve issues related to these interacting and synergistic effects [NUREG/CR-7153, Vol. 2, "Expanded Materials Degradation Assessment (EMDA), Aging of Core Internals and Piping Systems" (October 2014), at 1-5 (Ex. NYS004800004A-B)]." Describe the contents and purpose of NUREG/CR-7153.

A240 [GS, AH, OY, CN] The overall purpose of NUREG/CR-7153 ("EMDA") was to develop the technical bases for subsequent license renewal, or operation beyond 60 years. While applications for subsequent license renewal may not be prepared for several years, both NRC and DOE have an interest in acting proactively to identify issues that may affect the ability of plants to operate for up to 80 years. First, the analytical timeframe is extended to 80 years, encompassing the subsequent license renewal-operating period. Second, the materials and systems addressed in the EMDA are generally extended to all of those which fall within the scope of aging management review for license renewal. Thus, in addition to piping and core internals, the EMDA also includes the reactor pressure vessel (RPV), electrical cables, and concrete structures. EMDA at 1 through 3. (Ex. NYS00484A-B).

Q241 Explain what timeframe is addressed with respect to nuclear reactors operation in NUREG/CR-7153.

A241 [GS, AH, OY, CN] As discussed in the EMDA, applications for subsequent license renewal may not be prepared for several years, but both NRC and DOE have an interest in acting proactively to identify issues that may limit the ability of plants to operate for up to 80 years. In addition, in preparation for likely future submittals, the NRC is obligated to provide guidance to applicants on the expected contents of subsequent license renewal applications and to develop the technical bases for making safety determinations in their license reviews. Through the Light Water Reactor Sustainability (LWRS) Program, the DOE undertakes research to understand the fundamentals of component aging, thereby supporting industry in sustaining the domestic fleet as an economic and strategic resource. To take advantage of their common interests, NRC and DOE have entered into a Memorandum of Understanding (MOU) to cooperate on research activities related to long-term operations. One activity initiated under the MOU is the EMDA. EMDA at 1. (Ex. NYS00484A-B).

The EMDA expands the analytical timeframe to 80 years to encompass a potential second 20-year license-renewal operating-period, beyond the initial 40-year licensing term and a first 20-year license renewal. The EMDA provides assessments for a second 20-year license renewal operating period, and provides analyses of the key degradation modes for current and expected future service, key degradation modes expected for extended service, and suggested research needs to provide technical information for operation up to 80 years. EMDA at iii. (Ex. NYS00484A-B).

Q242 What is your opinion with respect to the applicability of the results in NUREG/CR-7153 to the IP2 and IP3 LRA?

A242 [GS, AH, OY, CN] The results in the EMDA are not applicable for the IP2 and IP3 LRA, since the LRA is associated with continued operation from 40 years to 60 years. The results of the EMDA may potentially be applicable to IP2 and IP3 in the future, if Entergy decides to pursue an LRA for continued operation from 60 years to 80 years.

Q243 Page 20 of Revised Lahey June (Ex. NYS000562) at line 8 states, “In addition, the federal government has also embarked on a fairly large research program, known as the Light Water Reactor Sustainability Program, which includes research into whether the different materials and LWR components can continue to perform their intended function during the extended operation of a nuclear reactor. [DOE, Light Water Sustainability Program, Material Aging and Degradation Technical Program Plan (August 2014) (“DOE Report”) (Ex. NYS000485)].” Explain what timeframe is addressed in the Light Water Sustainability Program.

A243 [GS, AH, OY, CN] The purpose of the research performed under the LWRS program is to develop the scientific basis for understanding and predicting long-term environmental degradation behavior of materials in nuclear reactors. The work is intended to provide data and methods to assess performance of systems, structures, and components (SSCs) essential to safe and sustained reactor operations. The LWRS program is designed to support the long-term operation (LTO) of existing domestic nuclear power generation. The report, which was published in August 2014, focused on extending reactor service to beyond 60 years of operation because extended operation will increase the demands on materials and components. DOE Report at page iii (Ex. NYS000485).

Q244 What is your opinion with respect to the applicability of the results in the Light Water Sustainability Program to the IP2 and IP3 LRA?

A244 [GS, AH, OY, CN] The IP2 and IP3 LRA is for the first 20-year license renewal period (i.e., continued operation from 40 years to 60 years). Thus, the results of the LWRS program are not applicable to the IP2 and IP3 LRA because the LRA does not extend reactor service beyond 60 years. Since the Light Water Reactor Sustainability Program is not applicable to the IP2 and IP3 LRA, this reference cited by Dr. Lahey is not relevant and does not support his assertion that Entergy has not demonstrated that it has a program that will manage the effects of aging of critical components.

Q245 Page 20 of NYS000562 at line 17 states, "the effects of embrittlement, especially loss of fracture toughness, make existing cracks in the affected materials and components less resistant to growth." Dr. Lahey also cites of the NRC presentation to the Advisory Committee on Reactor Safeguards (ACRS), "Technical Brief on Regulatory Guidance for Evaluating the Effects of Light Water Reactor coolant Environments in Fatigue Analyses of Metal Components" (December 2, 2014), at 56-58 (Ex. NYS000486). Please describe the purpose of the ACRS meeting that was held at NRC Headquarters on December 2, 2014.

A245 [GS, AH, OY, CN] The purpose of the ACRS meeting that was held at NRC Headquarters on December 2, 2014 was to respond to a request, from the ACRS Subcommittee on Metallurgy and Reactor Fuels, for a technical brief on Draft Regulatory Guide DG-1309 and Draft NUREG/CR-6909 Rev. 1 from the ACRS Subcommittee on Metallurgy and Reactor Fuels. ACRS review of draft regulatory guides is commonly solicited prior to or during their release for public comment.

Q246 Please describe the content and the status of DG-1309, "Guidelines for Evaluating the Effects of Light-Water Reactor Coolant Environments in Fatigue Analyses of Metal Components."

A246 [GS, AH, OY, CN] DG-1309 is a proposed revision to RG 1.207. DG-1309 contains methods and procedures that the Staff considers acceptable for use in determining the acceptable fatigue lives of components evaluated by a CUF calculation in accordance with the fatigue design rules in ASME Section III to account for the effects of light-water reactor coolant environments.

DG-1309 was published for 60 days of public comment in November 2014. The public comment period closed in January 2015. The Staff is working to address all public comments received and finalize the document for publication in concert with the final publication of Draft NUREG/CR-6909 Rev. 1. Final publication of DG-1309, which will then become RG 1.207, Rev. 1, is currently anticipated in early 2016.

Q247 Please describe the content and the status of NUREG/CR-6909, Rev. 1, "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials."

A247 [GS, AH, OY, CN] Draft NUREG/CR-6909 Rev. 1 is currently in draft form and is a revision to NUREG/CR-6909, which was published in February 2007. Draft NUREG/CR-6909 Rev. 1 was published for public comment in April 2014. The Staff is still working to address all of the public comments received and finalize the report for publication. The document contains the technical basis for the guidance provided in DG-1309, which is the proposed Revision 1 of RG 1.207.

Draft NUREG/CR-6909 Rev. 1 contains the results of recent NRC research activities, and documents the complete technical bases for the guidance provided in the proposed Revision 1 of RG 1.207. Draft NUREG/CR-6909 Rev. 1 provides updates and improvements to the environmental fatigue factor (F_{en}) approach based on an extensive update to the fatigue cyclic strain data from testing and results available since the report was first published in 2007.

Q248 Please describe the outcome(s) of the ACRS meeting that was held at NRC Headquarters on December 2, 2014.

A248 [GS, AH, OY, CN] As indicated in the presentation material provided to the ACRS, the purpose of the December 2, 2014 meeting was to brief the ACRS subcommittee on the background, research activities, and content of DG-1309 and Draft NUREG/CR-6909 Rev. 1. ACRS Brief at 2 (Ex. NYS000486). As described in the meeting transcripts, the subcommittee's intent was to gather information, analyze relevant issues and facts, and formulate a proposed position and action as appropriation for deliberation by the full committee. ACRS Transcripts at 4 (Ex. NRC000173). At the conclusion of the meeting, the subcommittee identified that the brief was a preliminary review in parallel with the documents' publications for public comment, that they would address the issue further after receipt of public comments, and that there would not be any full committee action at that time. ACRS Transcripts at 106 (Ex. NRC000173).

Q249 Can you summarize the information on pages 56-58 of the presentation (Ex. NYS000486) made to the ACRS subcommittee?

A249 [GS, AH, OY, CN] Slides 56-58 of the ACRS presentation contain a summary of information regarding the latest technical reasoning behind the mechanism of environmentally-assisted cracking. Leading researchers have postulated two mechanisms to explain enhanced fatigue cracking that occurs in water environments. The first mechanism is identified as film rupture/slip dissolution, whereby incremental strain breaks the oxide layer that forms on the material, and continued crack extension occurs by dissolution/oxidation of the freshly exposed surface. The second mechanism is identified as hydrogen-induced cracking, whereby corrosion of the material causes hydrogen production that diffuses into inclusions in the material, which then act as new crack initiation sites. Either mechanism, or some combination of both mechanisms, are possible explanations of observed cracking behavior in laboratory testing. In addition, the effects of dynamic strain aging are discussed, which is a condition where material solutes segregate to dislocations that result in strong elastic interactions between the solute and dislocation stress-strain field. It is believed by the ANL and NRC researchers that dynamic strain aging may also be a factor in the enhanced fatigue cracking that occurs when materials are exposed to water environments.

The discussion on slides 56-58 of the ACRS presentation do not indicate additional effects on fatigue cracking that have not been previously considered. Rather, they are offering a more informed and detailed scientific explanation for the causes of enhanced cracking with the intent of enhancing the direction of future research and modeling endeavors.

Q250 Can you explain whether pages 56-58 of the presentation (Ex. NYS000486) made to the ACRS subcommittee discusses the effects of embrittlement or loss of fracture toughness?

A250 [GS, AH, OY, CN] Neither the effects of embrittlement, nor loss of fracture toughness, are discussed in my ACRS presentation. The effects of radiation, which leads to material embrittlement, is discussed in our response to Q193. Loss of fracture toughness is another matter that is not related to fatigue crack initiation and CUF or CUF_{en} evaluations, so it is not relevant to this fatigue contention.

Q251 What is your opinion with respect to Dr. Lahey's claim that the presentation (Ex. NYS000486) made to the ACRS subcommittee on pages 56-58 support his statement, "the effects of embrittlement, especially loss of fracture toughness, make existing cracks in the affected materials and components less resistant to growth"?

A251 [GS, AH, OY, CN] Dr. Lahey provides no specific, supporting evidence or data to support this portion of his testimony with respect to the impact of embrittlement and loss of fracture on calculations of CUF or CUF_{en}. Whereas NRC agrees that these effects are important to the growth of existing cracks that could initiate due to fatigue, stress corrosion cracking or other mechanisms, and that these effects should be evaluated accordingly for either postulated cracks or cracks found in service, CUF and CUF_{en} calculations are intended to demonstrate the absence of fatigue cracks, so discussion of the effects of loss of fracture toughness on CUF and CUF_{en} is not relevant.

Q252 What is your opinion on this portion of the testimony with respect to Dr. Lahey's claim that Entergy has not demonstrated that it has a program that will manage the effects of aging of critical components?

A252 [GS, AH, OY, CN] In summary, once again, this reference cited by Dr. Lahey is not relevant and does not support his assertion that Entergy has not demonstrated that it has a program that will manage the effects of aging of critical components

Q253 Page 21 of Revised Lahey June (Ex. NYS000562) at line 10 states, "a combined effect of thermal aging and irradiation embrittlement could reduce the fracture resistance even further to a level neither of these degradation mechanisms can impart alone" [Chen, et al., "Crack Growth Rate and Fracture Toughness Tests on Irradiated Cast Stainless Steels," NUREG/CR-7184 (Revised December 2014), at xv (Ex. NYS00488A-B)]. What is your opinion with respect to the quoted statement from page xv of NUREG/CR-7184?

A253 [GS, AH, OY, CN] On page xv of NUREG/CR-7184, the complete sentence is, "*It is suspected that a combined effect of thermal aging and irradiation embrittlement* *could reduce the fracture resistance even further to a level neither of these degradation mechanisms can impart alone.*" Dr. Lahey truncated the sentence and left out the words, "*It is suspected*." With these additional words, the context of the statement does not agree with Dr. Lahey's assertion.

Q254 Does NUREG/CR-7184 provide a scientific basis to support a definite conclusion that, "a combined effect of thermal aging and irradiation embrittlement could reduce the fracture resistance even further to a level neither of these degradation mechanisms can impart alone"?

A254 [GS, AH, OY, CN] No, the report does not provide a scientific basis to support a definitive conclusion. Although NUREG/CR-7184 suspected that the combined effects of thermal aging and irradiation embrittlement could further reduce the fracture resistance, the test results presented in the report do not support that concern. As a matter of fact, the report identifies that a test program has been initiated to investigate the joint effects of thermal aging and irradiation damage on the cracking susceptibility and fracture resistance of CASS. NUREG/CR-7184 at xv (Ex. NYS00488A-B).

Q255 Does Dr. Lahey provide a scientific basis to support a definite conclusion that, "a combined effect of thermal aging and irradiation embrittlement could reduce the fracture resistance even further to a level neither of these degradation mechanisms can impart alone"?

A255 [GS, AH, OY, CN] No, Dr. Lahey did not provide a scientific basis to support such a conclusion.

Q256 What is your opinion on this portion of the testimony with respect to Dr. Lahey's claim that Entergy has not demonstrated that it has a program that will manage the effects of aging of critical components?

A256 [GS, AH, OY, CN] Since neither Dr. Lahey or NUREG/CR-7184 provided a scientific basis to support a conclusion that combined effects of thermal again and irradiation embrittlement may be more severe than the effects of evaluating these mechanisms separately, once again, this reference cited by Dr. Lahey is not relevant and does not support his assertion that that Entergy has not demonstrated that it has a program that will adequately manage the effects of aging of critical components.

Q257 Page 22 of Revised Lahey June (Ex. NYS000562) at line 6 stated that, "Multiple recent reports and studies from USNRC, DOE, and associated contractors recognize the lack of understanding of the interrelationship between embrittlement, high or low cycle fatigue, and shock loads for highly fatigued and/or embrittled components made of CASS, non-cast stainless steels, or other alloys." Does Dr. Lahey provide any references by "USNRC, DOE and associated contractors" that discuss "embrittlement," or "high or low cycle fatigue"?

A257 [GS, AH, OY, CN] Yes, Dr. Lahey provided reference to the EMDA by USNRC, DOE and associated contractors that discusses "embrittlement" and "high or low cycle fatigue". However, as discussed in our responses to some of the previous questions, the EMDA is intended to extend operation from 60 years to 80 years. The applicability of that document and its results do not pertain to the LRA for IP2 and IP3, which is associated with continued operation from 40 years to 60 years.

Q258 Does Dr. Lahey provide any references by "USNRC, DOE and associated contractors" that discuss "shock loads"?

A258 [GS, AH, OY, CN] No, Dr. Lahey did not provide any reference by "USNRC, DOE and associated contractors" that discussed "shock loads." Once again, this reference cited by Dr. Lahey is not relevant and does not support his assertion that that Entergy has not demonstrated that it has a program that will manage the effects of aging of critical components.

Q259 Page 22 of Revised Lahey June (Ex. NYS000562) at line 6 states, "Draft NUREG/CR-6909 Rev. 1 (March 2014 15 (draft) (Ex. NYS000490)), at 11 ("it is not possible to quantify the impact of irradiation on the prediction of fatigue lives in PWR primary water environments compared to those in air.")" Can you provide the context of the sentence cited on page 11 of the draft of NUREG/CR-6909, Rev. 1 and explain how it relates to the F_{en} methods described in that document?

A259 [GS, AH, OY, CN] Dr. Lahey truncated the sentence and interprets the portion of the sentence that he quotes out of context. The complete sentence on page 11 states, "*Although some small-scale laboratory fatigue ε–N test data indicate that neutron irradiation decreases the fatigue lives of austenitic SSs, particularly at high strain amplitudes, it is not possible to quantify the impact of irradiation on the prediction of fatigue lives based on the limited data currently available*." As we discuss in our response to Q154, the total amount of research test data available is insufficient to perform a detailed, comprehensive, statistical evaluation of data in the same manner that was done for unirradiated data to develop the F_{en} expressions. The Staff believes application of the current F_{en} expressions and fatigue curves is generally conservative for application to irradiated components. Therefore, in Draft NUREG/CR-6909 Rev. 1, the NRC noted that additional fatigue data on reactor structural materials irradiated under light water reactor operating conditions are needed to determine whether there are measurable effects of neutron irradiation on the fatigue lives of these materials and, if so, to better define how those impacts may be quantified.

Q260 Please describe the process and the public involvement during the development of NUREG/CR-6909, Rev. 1, "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials."

A260 [GS, AH, OY, CN] As a part of the revision of NUREG/CR-6909, Rev. 1, the Staff elected to publish a draft of the report for 60 days of public comment via the Federal Register process commonly used for other NRC documents. During the 60-day comment period, any member of the world-wide public may submit written comments to the NRC on the draft report. The Staff is then required to respond to all comments received during the 60-day period as a part of finalizing the document for internal NRC technical and administrative reviews and final publication.

Q261 Is it common for the Staff to solicit public comment for NUREG-type document?

A261 [GS, AH, OY, CN] No, it is not common for the Staff to solicit public comment for NUREG-type document. The practice of soliciting public comments is not common for NUREG-type documents published by the NRC. It does happen from time to time, and is allowed by NRC procedures, but it is not the usual practice. In addition, the NRC practices an openness policy that allows any member of the public to comment on any NRC document at any time, even if the document is published.

Q262 Why was it necessary for the NRC to solicit public comments on the draft of NUREG/CR-6909, Rev. 1?

A262 [GS, AH, OY, CN] For Draft NUREG/CR-6909 Rev. 1, the Staff felt that soliciting public comments was important in view of the significant stakeholder feedback provided to the Staff in the many industry and public forums that have taken place since 2007, primarily associated with ASME Code proceedings. As a part of those forums, several stakeholders requested an opportunity to review a draft of the NRC's research activities before they were published. In order to be responsive to those requests, the Staff decided to publish a draft of the report for public comment. In addition, the Staff was interested in receiving comments from interested external stakeholders to ensure the completeness and the quality of the document. The Staff was also interested in further stakeholder feedback on a few technical aspects of the report, as requested in the 2014 Federal Register Notice that noticed the report for public comments, Federal Register, Vol. 79, No. 74, (April 17, 2014) ("FRN 2014-08792") (Ex. NRC000186). FRN 2014-08792 at 21812 (Ex. NRC000186).

Q263 Can you describe the general technical quality of the comments provided by interested external stakeholders?

A263 [GS, AH, OY, CN] More than 200 separate comments were received on Draft NUREG/CR-6909 Rev. 1 from ten individual stakeholders. ACRS Brief at 51 and 52 (Ex. NYS000486). Most of these individuals represented research organizations from around the world that are, or were previously, involved in environmentally-assisted fatigue research. As such, the majority of the comments were of very high technical caliber and quality. For this reason, the Staff continues to work on responses to those comments more than one year after the public comment period closed in June 2014.

One of the NRC's goals during the review of Draft NUREG/CR-6909 Rev. 1 was to receive the most thorough and independent technical review possible on the results of the Staff's research activities. Based on the number of comments received and the affiliation of the commenters, the Staff believes that this was accomplished.

Q264 Please indicate whether the intervener organizations for the Indian Point license renewal contentions submitted any comments objecting to the completeness and the quality of the draft of NUREG/CR-6909, Rev. 1 during the public comment period.

A264 [GS, AH, OY, CN] The State of New York, Riverkeeper, nor any of the witnesses for these organizations submitted public comments for either Draft NUREG/CR-6909 Rev. 1 or DG-1309.

Q265 Page 23 of Revised Lahey June (Ex. NYS000562) at line 9 states, “A recent paper presented at an MPA Seminar in Stuttgart, Germany confirms that, at present, the USNRC staff does not have a clear solution to the challenges posed by synergistic age-related degradation mechanisms [Stevens, Gary L., et al., “Observations and Recommendations for Further Research Regarding Environmentally-Assisted Fatigue Evaluation Methods,” 40th MPA- Seminar, Materials Testing Institute, University of Stuttgart, Stuttgart, Germany (October 6-7, 2014) (Ex. NYS000491)].” Please describe the contents of the conference paper, “Observations and Recommendations for Further Research Regarding Environmentally-Assisted Fatigue Evaluation Methods.”

A265 [GS, AH, OY, CN] The MPA seminar paper, *Observations and Recommendations for Further Research Regarding Environmentally-Assisted Fatigue Evaluation Methods*, was written by two staff members and an ANL contractor for presentation at the 40th MPA-Seminar sponsored by the Materials Testing Institute at the University of Stuttgart in Stuttgart, Germany on October 6-7, 2014.

The paper briefly summarizes results of the NRC’s research activities that led to the drafts of DG-1309 and Draft NUREG/CR-6909 Rev. 1, and describes several observations made by the Staff and its contractors during the course of the performance of those activities.

The paper also provides recommendations for further research efforts that the Staff and its contractors identified where further research could yield reduced conservatism in EAF evaluations, including more refined, material-specific fatigue curves, fatigue curves for ferritic materials based on material tensile strength, component testing (rather than small-scale specimen testing), ASME Code CUF calculation methods, and the effect of neutron irradiation on fatigue crack initiation in austenitic stainless steels.

Publication of technical conference papers that document the status of on-going research activities performed by NRC researchers is common, and is another form of public interaction that the Staff uses to solicit interested stakeholder feedback on their research activities.

Q266 Please describe and identify any new information in the conference paper that is not already captured in the draft of NUREG/CR-6909, Rev. 1.

A266 [GS, AH, OY, CN] There is no new research information in the paper compared to what is contained in Draft NUREG/CR-6909 Rev. 1. The recommendations made in the paper, however, are not contained in Draft NUREG/CR-6909 Rev. 1 because the paper contains the results of recent NRC research activities related to environmentally-assisted fatigue and the application of specific methods. It is not appropriate for a NUREG document to make recommendations for further research activities. Therefore, the paper was developed as a way for the NRC research staff and/or NRC contractors to document their observations and recommendations to interested stakeholders in an appropriate manner.

Q267 Do you agree with Dr. Lahey's assertion that, "the USNRC staff does not have a clear solution to the challenges posed by synergistic age-related degradation mechanisms"?

A267 [GS, AH, OY, CN] No, the Staff does not agree with Dr. Lahey's assertion. It is the Staff's opinion that the current approach to managing age-related degradation mechanisms is consistent with current knowledge and understanding. In areas where information is lacking, or indicating that further investigation or knowledge is needed, NRC is funding research activities to provide further understanding or confirmation of industry findings. As always, the Staff continue to monitor industry-wide field experience for indications of new challenges so that the Staff can re-direct its focus when warranted. Entergy's aging management programs are consistent with the Staff's current understanding of age-related degradation, which is documented in the GALL Report. The Staff's review that evaluated Entergy's aging management programs is documented in SER, SER Supp. 1 and SER Supp. 2. SER, SER Supp. 1 and SER Supp. 2 in general (Ex. NYS00326A-F, NYS000160, and NYS000507, respectively).

Q268 Does Dr. Lahey provide a clear proposal for addressing the challenges posed by synergistic age-related degradation mechanisms?

A268 [GS, AH, OY, CN] No, Dr. Lahey does not provide a clear proposal for addressing the challenges posed by synergistic age-related degradation mechanisms. We have rebutted Dr. Lahey's testimony and his testimony does not provide specific evidence to demonstrate that Entergy's aging management programs are deficient.

Q269 Page 71 of Revised Lahey June (Ex. NYS000562) at line 4 states, "For that component, Westinghouse was able bring the CUF_{en} just below unity by performing successive analyses using modified design transient conditions and 60 year projected cycles (rather than CLB cycles), and applying new, specially-developed heat-up and cool-down spray transients." Is using the 60-year projected number of cycles in a fatigue calculation prohibited?

A269 [GS, AH, OY, CN] No, the ASME Code does not prohibit the use of the 60-year number of projected cycles in fatigue calculations. Paragraph NB-3222.4 of ASME Section III requires consideration of the specified service loadings for the component involving cyclic application of loads and thermal conditions when performing an analysis for cyclic operation. ASME Section III at 81 (Ex. NYS0000349). Paragraph L-2210(c) of ASME Appendix L allows the use of actual plant operating data when performing operating plant fatigue assessments. ASME Appendix L at 422 (Ex. NRC000113).

Dr. Lahey has not identified any restrictions on the use of 60-year projected cycles when performing fatigue calculations in accordance with the ASME Code.

Q270 Can you explain how Indian Point's fatigue monitoring program will be capable of managing the aging effects of metal fatigue for those components that used 60-year projected numbers of cycles?

A270 [GS, AH, OY, CN] Sixty-year cycle projections provide an estimate of the loadings that are anticipated after 60 years of plant operation. These projections represent the best estimate of the loadings expected over the 60-year operating period based on trending of past plant performance, and they are used in the fatigue calculations to assure that the fatigue limit of 1.0 is maintained throughout the period of extended operation. An objective of the Fatigue Monitoring Program is to continually validate the assumptions used in the fatigue analyses, including the number of cycles for each transient does not exceed that assumed in the fatigue analyses.

With the Fatigue Monitoring Program limiting each transient to its assumed number of cycles used in the fatigue analyses, both of the following conditions must be satisfied for the CUF or CUF_{en} to exceed the fatigue limit of 1.0:

- the severity of each and every transient experienced at the plants must be equal to or greater than the transient severities assumed in the fatigue calculations, and
- the actual number of cycles for each and every transient experienced at the plants must equal or exceed the number of cycles used in the fatigue calculations.

Entergy is managing cumulative fatigue damage with its Fatigue Monitoring Program by (1) tracking the actual severities and numbers of plant transients, (2) evaluating the severities of the actual transients to the severity of the transients used in the fatigue calculations, and (3) ensuring that the numbers of cycles of each plant transient remains within the numbers of cycles of the transient used in the fatigue calculations. It should be noted that if the actual severity and the actual number of cycles for each and every transient used in the fatigue calculation were equal to the assumptions, the accumulated fatigue usage would only be equal to that determined by the calculation and not necessarily the fatigue limit of 1.0.

Entergy's program for IP2 includes 'alert cycles', which are defined as the number of transient cycles which are projected to accumulate in the next two operating periods. The number of 'alert cycles' is calculated by taking the number of transient cycles accumulated during the prior operating cycle and multiplying by 2; the total projected number of transient cycles is then determined by adding the 'alert cycles' to the total accumulated number of transients to date. If this projected number of transient cycles remains below the number of transient cycles used in the fatigue evaluation, no corrective action is required. If this projected number of transient cycles exceeds the number of transient cycles used in the fatigue evaluation, a condition report is generated to ensure that corrective actions are taken. Conversely, Entergy's program for IP3 does include the use of alert cycles and does not allow continued plant operation if the number of transient cycles used in the fatigue evaluation for any transient is exceeded unless an appropriate engineering evaluation, developed under the corrective action program has determined that plant operation is acceptable. NL-08-057, Attachment 5 at 7 through 8 (Ex. NRC000109). . NL-08-057, Attachment 5 at 7 through 8 (Ex. NRC000109).

The Fatigue Monitoring Program for both IP2 and IP3, as described in Entergy's implementing procedures, require corrective actions when a single transient type (e.g., heat-up transient or cool-down transient) approaches its respective cycle limit. Since the fatigue calculations for each component typically include multiple transients (e.g., not just a single transient), this approach provides additional margin against exceeding any fatigue limits.

Therefore, the Fatigue Monitoring Program allows for taking corrective actions well before the CUF or CUF_{en} values approach the fatigue limit of 1.0.

Q271 Describe "specially-developed heat-up and cool-down spray transients."

A271 [GS, AH, OY, CN] Entergy discusses that [redacted] Westinghouse Report CN-PAFM-13-40, *Indian Point Unit 2 Pressurizer Spray Model Transfer Function Data Base Development and Environmental Fatigue Evaluations*, (2013) (PROPRIETARY) (Ex. NYS000512) ("CN-PAFM-13-40") at 15. Entergy explained [redacted]

[redacted] CN-PAFM-13-40 at 40 (Ex. NYS000512).

In addition, as described in their response to the Staff's audit questions in NL-08-057, Entergy explained the details of its modified operating procedures associated with the specially-developed heat-up and cool-down spray transients. Entergy stated that both IP2 and IP3 instituted two main operating changes consistent with the generic Westinghouse program to address surge line thermal cycling. The first change included continuous, reduced pressurizer spray flow, which minimizes the temperature differential between the reactor coolant system, the pressurizer, and the surge line. A reduced temperature differential minimizes the possible thermal stresses that can occur during an insurge transient. The second change included changing the plant startup operating procedures to eliminate drawing and then collapsing a pressurizer air bubble so that reactor coolant pumps are run to sweep air out of the reactor coolant system and reactor pressure vessel. The collapsing of the pressurizer air bubble early in the startup procedure resulted in a significant reduction in the number of insurge transients. NL-08-057, Attachment 5 at 7 through 8 (Ex. NRC000109).

Q272 Is the usage of those "specially-developed heat-up and cool-down spray transients" prohibited in fatigue calculations?

A272 [GS, AH, OY, CN] No, ASME Section III does not prohibit the use of specially-developed heat-up and cool-down spray transients in fatigue calculations. It should be noted that these "specially-developed heat-up and cool-down spray transients" were developed because they are more representative transients for heat-ups and cool-downs based on plant data. Entergy further clarified that it was assumed that the plant data from 1991 to 2012 was representative, regarding spray operations, for the entire plant life. Specifically, the plant computer data prior to April 14, 1997, was representative of heatup/cooldown operations prior to the implementation of modified operational procedures MOPs and the plant computer data for the period after April 14, 1997, is representative of plant spray operations through 60 years of operation. CN-PAFM-13-40at 14 through 15 (Ex. NYS000512).

ASME Section III requires consideration of the specified service loadings for the component involving cyclic application of loads and thermal conditions when performing an analysis for cyclic operation in accordance with ASME Section III, Paragraph NB-3222.4.

Dr. Lahey has not identified restrictions on the use of transients based on plant data when performing a fatigue analysis in accordance with ASME Section III. When the plant was originally designed over 40 years ago, the design transients were defined conservatively, and the goal of the fatigue analysis was to meet the ASME Section III design limit of 1.0. Use of transients based on plant data to achieve a more accurate value of CUF or CUF_{en} can be performed to reduce the conservatism in the calculation.

Q273 In your opinion, can you explain how Indian Point's fatigue monitoring program will be capable of managing the aging effects of metal fatigue for those components that used specially-developed heat-up and cool-down spray transients" in fatigue calculation?

A273 [GS, AH, OY, CN] As previously discussed, in order for the CUF or CUF_{en} to exceed the fatigue limit of 1.0, both of the following conditions must be satisfied:

- the severity of each and every transient experienced at the plants must be equal to or greater than the transient severities assumed in the fatigue calculations, and
- the actual number of cycles for each and every transient experienced at the plants must equal or exceed the number of cycles used in the fatigue calculations.

Entergy is managing cumulative fatigue damage with its Fatigue Monitoring Program by (1) tracking the actual severities and numbers of plant transients, (2) evaluating the severities of the actual transients to the severity of the transients used in the fatigue calculations, and (3) ensuring that the numbers of each and every actual transient remains within the numbers of cycles of each and every transient used in the fatigue calculations. It should be noted that if the actual severity and the actual number of cycles for each and every transient used in the fatigue calculation were equal to the assumptions, the accumulated fatigue usage would only be equal to that determined by the calculation and not necessarily the fatigue limit of 1.0.

The Fatigue Monitoring Program for both IP2 and IP3, as described in Entergy's implementing procedures, require corrective actions when a single transient type (e.g., heat-up transient or cool-down transient) approaches its respective cycle limit. Since the fatigue calculations for each component typically include multiple transients (e.g., not just a single transient), this approach provides additional margin against exceeding any fatigue limits.

Therefore, the Fatigue Monitoring Program allows for taking corrective actions well before the CUF or CUF_{en} values approach the fatigue limit of 1.0.

Q274 Page 72 of Revised Lahey June (Ex. NYS000562) at line 18 states, "it clearly shows the iterative process used by Westinghouse in which safety margin is removed in its environmentally-assisted fatigue (EAF) calculations in an effort to reduce the output or result below CUF_{en} = 1.0." Does the ASME Code define what safety margin is in a fatigue calculation?

A274 [GS, AH, OY, CN] The ASME Code does not explicitly define what safety margin is in a fatigue calculation. However, the document, *Criteria of the ASME Boiler and Pressure Vessel Code for Design by Analysis in Sections III and VIII, Division 2*, (1969) (Ex. ENT000188) ("ASME Section III Criteria Document") contains ASME's technical bases for the design requirements in ASME Section III, including fatigue analysis. That document discusses "safety factors" used to develop the design fatigue curves of two on stress or a factor of twenty on cycles. ASME Section III Criteria Document at 20 (Ex. ENT000188). However, the document also notes, "*These factors were intended to cover such effects as environment, size effect, and scatter of data, and thus it is not to be expected that a vessel will actually operate safely for twenty times its specified life.*" ASME Section III Criteria Document at 20 (Ex. ENT000188). There are other design factors applied on stresses that are used in fatigue calculations so, even ignoring any conservative assumptions applied by analysts in the calculations, it is difficult to explicitly quantify an exact overall safety margin present in fatigue calculations. Generally, ASME Section III uses a factor of three against failure. K. R. Rao, *Companion Guide to the ASME Boiler & Pressure Vessel Code, Criteria and Commentary on Select Aspects of the Boiler & Pressure Vessel and Piping Codes, Third Edition, Volume 1* (2009) (Ex. ENT000191) ("ASME Companion Guide") at 159-160 and ASME Background Document at 4 and 6 (Ex. NRC000174).

Q275 Does the ASME Code prohibit the iterative process performed by Westinghouse?

A275 [GS, AH, OY, CN] No, the ASME Code does not prohibit the iterative process performed by Westinghouse. In fact, iteration during design has been quite common in ASME Code Class 1 fatigue calculations for more than 50 years. This is because the typical objective of the analyst is to demonstrate acceptability (as opposed to demonstrating margin). A secondary cause may have been because schedules and budgets were limited, which would limit the amount of time an analyst could spend performing the calculation. Therefore, especially in the early days of the industry in the 1960s before computers were prevalent or affordable, analysts would make many simplifying assumptions to shorten the problem and make it easier to solve. If the assumptions were too conservative, the analyst would modify one or more of the assumptions to be more realistic and less conservative,. This iteration would continue until acceptable results were achieved. In the case of a fatigue calculation, acceptable results meant the CUF was calculated to be less than or equal 1.0. Experienced analysts tend to make fewer iterations because of their past trials and familiarity with performing fatigue calculations. That practice continues today, although perhaps to a much lesser extent with the availability of high-powered computers. The nature of these conservatisms is reflected in our response to Q56. For example, such iteration is described in NUREG/CR-6260, where it was noted, "*Since the licensees' design basis analyses were based on the ASME Code of record, it was uneconomical for the licensee to attempt to reduce the CUF to lower and lower values by removing conservative assumptions once the Code requirements were met. Given more funding and time, further calculations could have been performed to reduce the existing stress values by using more realistic loadings or more detailed analysis models. These reduced stresses would result in lower CUFs*." NUREG/CR-6260 at xxi. (Ex. NYS000355). In our experience, it would be unusual for an experienced analyst to perform a fatigue calculation without some amount of iteration.

Q276 Can you identify other safety margins that exist in a fatigue calculation?

A276 [GS, AH, OY, CN] As described our response to Q208, there are two factors used to develop the design fatigue curves, and there are other design factors applied on stresses that are used in fatigue calculations. Beyond that, there may be other several additional margins in fatigue calculations based on conservative assumptions applied by analysts. Other factors of conservatism that are very common in fatigue calculations are transient severity and grouping of transients, as reflected in our response to Q54. Given the variability in assumptions made by different analysts, it is difficult to explicitly quantify an exact overall safety margin present in fatigue calculations. The NRC's opinion is that fatigue calculations tend to be very conservative, as evidenced by the lack of observed thermal fatigue failures for components where a design CUF calculation was made. A detailed discussion of this and the methods an analyst can use to make adjustments to fatigue calculations that might reduce the CUF is documented in Section 4.3 of NUREG/CR-6260. NUREG/CR-6260 states the changes fall into two broad categories, conservative assumptions made by the analyst or ASME Code changes that have been made since the edition of the ASME Code of record for the plant. NUREG/CR-6260 at 4-5 (Ex. NYS000355). Two examples that NUREG/CR-6260 discusses to reduce CUF values are (1) by separating the enveloped load pair with the overall combined number of cycles into more detailed load pairs, each with its own set of cycles, which can sometimes be significantly reduced, and (2) by using actual cycles that the plant has experienced to date if the numbers of cycles extrapolated are less than the numbers of design basis cycles. NUREG/CR-6260 at 4-5 and 4-6 (Ex. NYS000355).

Q277 Page 73 of Revised Lahey June (Ex. NYS000562) at line 7 states, “For example, Westinghouse has reported that the CUF_{en} for the RHR/Accumulator nozzles and associated piping for IP3 is 0.9961, which is extremely close to unity [CN-PAFM-09-77 (2010), at 61 tbl. 5-36 (Ex. NYS000366)]. Line 20 of Page 68 states, “Even assuming this CUF_{en} calculation is accurate, it does not account for the possibility that a highly fatigued component, which does not yet have signs of significant surface cracking, may be exposed to an unexpected seismic event or shock load that could cause it to fail.” Does Dr. Lahey provide any quantitative description (i.e., pressure or temperature profile) of the “shock load” or “unexpected seismic event” he is referring to?

A277 [GS, AH, OY, CN] No, Dr. Lahey does not provide any quantitative description (i.e., pressure or temperature profile) of the “shock load” or “unexpected seismic event” to which he is referring. As long as the CUF is less than the fatigue limit of 1.0, and therefore no cracks are expected to initiate, the structural performance of a “highly fatigued component” is unaffected, in particular for design basis loads. Other than his opinion, Dr. Lahey does not provide any quantitative description as a basis to support his opinion.

Q278 Does Dr. Lahey provide any quantitative comparison as a basis to conclude that the “shock load” is more severe than the design transients used in fatigue calculation?

A278 [GS, AH, OY, CN] No, other than his opinion, Dr. Lahey does not provide any quantitative comparison as a basis to support his opinion.

Q279 Does Dr. Lahey provide calculations to quantitatively support his statement that an unexpected seismic event or shock load would increase the CUF_{en} value beyond 1.0?

A279 [GS, AH, OY, CN] No, other than his opinion, Dr. Lahey does not provide any quantitative comparison as a basis to support his opinion. Furthermore, as described in our response to Q145, seismic and other accident loads are considered as part of the design calculations, and Dr. Lahey's argument is irrelevant because the important failure mode for these severe loads is gross structural overload and deformation, not fatigue crack initiation.

Q280 Does Dr. Lahey provide any scientific basis or operating experience to support his statement that an unexpected seismic event or shock load would a highly fatigued component fail?

A280 [GS, AH, OY, CN] No, other than his opinion, Dr. Lahey does not provide any scientific basis or operating experience to support his opinion. Furthermore, his testimony indicates he is not familiar with ASME Code calculations and design because he argues that fatigue cracks exist in components where it has been demonstrated that they do not, and he suggests evaluations that are not required by NRC regulations or the ASME Code.

Q281 Do you have any other comments or opinions on Dr. Lahey's testimony and report?

A281 [GS, AH, OY, CN] No.

Q282 Based on your review of the Entergy's metal fatigue TLAAs, environmentally-assisted fatigue analyses and Fatigue Monitoring Program, contention NYS-26/RK-TC-1B and NYS-38/RK-TC-5, and the exhibits and testimony of Dr. Hopenfeld and

Dr. Lahey, what is Staff's opinion regarding the adequacy of Entergy's Fatigue Monitoring Program?

A282 [GS, AH, OY, CN] Based on assumptions of transient severity and transient cycles, Entergy performed fatigue calculations and determined CUF and CUF_{en} values less than 1.0. This means that there is assurance that a fatigue crack not formed or initiated in the material provided that the assumptions in these fatigue evaluations are not invalidated. It should be noted there are inherent conservatisms present in CUF and CUFen calculations that provide additional assurance that a fatigue crack not formed or initiated. NUREG/CR-6260 at 4-5 and 4-9 (Ex. NYS000355) and NUREG/CR-6909 at 71 (Ex. NYS000357). However, instead of solely relying on these fatigue evaluations as means to demonstrate that metal fatigue and environmentally-assisted fatigue is valid for the period of extended operation, Entergy is relying on its Fatigue Monitoring Program for IP2 and IP3 to periodically validate that the assumptions in these fatigue evaluations have not been exceeded.

As the Staff has described previously, in order for the actual CUF of a component to approach the calculated CUF, at a minimum, both of the following conditions must be satisfied:

- the severity of each and every transient that occurs at IP2 and IP3 must be equal to the transient severity used in the fatigue calculations, and
- the actual number of cycles for each and every transient must approach the number of each and every transient used in the fatigue calculation.

Furthermore, Entergy's Fatigue Monitoring Program, requires corrective actions when a single transient type (e.g., heat-up transient or cool-down transient) approaches the respective number of transient cycles used in the fatigue evaluation for that transient type. Since a design fatigue calculation or environmentally-assisted fatigue evaluation for a component typically include multiple transients (e.g., not just a single transient), this approach provides additional margin against the CUF or CUF_{en} exceeding its limit its calculated value and the ASME design limit of 1.0.

These factors, support the Staff's opinion that Entergy's Fatigue Monitoring Program is adequate to manage metal fatigue will be adequately managed for the period of extended operation in accordance with 10 C.F.R. § 54.21(c)(1)(iii).

Q283 Does this conclude your testimony?

A283 [GS, 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.

I, Allen L. Hiser, Jr., 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 CFR 2.304(d)/

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August 9, 2015

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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AFFIDAVIT OF ON YEE

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 CFR 2.304(d)/

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August 10, 2015

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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(Indian Point Nuclear Generating	)	
Units 2 and 3)	)	

AFFIDAVIT OF GARY L. STEVENS

I, Gary L. Stevens, 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 CFR 2.304(d)/

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August 10, 2015

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
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AFFIDAVIT OF CHING HANG NG

I, Ching Hang 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 CFR 2.304(d)/

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August 10, 2015