NRC000197
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,
JEFFREY POEHLER, AND GARY STEVENS ON NYS-25 AND NYS-38/RK-TC-5

INTRODUCTION

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

A1a. [AH] My name is Dr. Allen L. Hiser, Jr. I am employed as the Senior Technical Advisor for License Renewal Aging Management in the Division of License Renewal, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission ("NRC"), in Washington, D.C. I have been employed by the NRC for over 25 years. A statement of my professional qualifications has been previously submitted. Exhibit ("Ex.") NRCR00103.

A1b. [JP] My name is Jeffrey C. Poehler. I am a Senior Materials Engineer employed by the U.S. Nuclear Regulatory Commission (NRC) in the Office of Nuclear Reactor Regulation, Division of Engineering, Vessel and Internals Integrity Branch. I have been employed by the NRC for over seven years. My statement of qualifications is attached as Ex. NRC000226.

A1c. [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).

Q2. Please describe your current responsibilities.

A2a. [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 responsibilities include serving as a lead technical expert for aging management evaluation and assisting other NRC Staff as they implement their review of license renewal applications. 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.

A2b. [JP] My current responsibilities include technical reviews of licensing applications, topical reports, and other submittals related to the integrity of the reactor vessel internals of commercial nuclear power reactors (pressurized water reactors and boiling water reactors). I perform reviews related to power uprates, license renewal applications, relief requests, surveillance capsule schedule changes, and reviews of technical bases for proposed

rulemaking or generic guidance documents related to reactor vessel integrity. I led the technical review for four submittals related to plant-specific reactor vessel and internals inspection programs. I am the technical lead for resolution of generic issues related to closure of action items from MRP-227-A.

A2c. [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.

Q3. Please describe any specific experience not already highlighted in previous answers and relating to reactor vessel internals (“RVI”) and aging management programs (“AMPs”).

A3a. [AH] Before joining the NRC, I worked for a small company that did contract research for a variety of customers. Included in my experience was testing of irradiated stainless steel to characterize the post-irradiation properties of the materials, and characterization of the properties of cast austenitic stainless steel (CASS) before and after thermal aging treatments, which were used to simulate the material response after long-term exposure to reactor operating conditions. While at the NRC, I have had a variety of experiences related to RVI. During the review of the first license renewal applications, for the Calvert Cliffs and Oconee plants, I was one of the reviewers for the RVI. When the industry initiated its extensive activities to address aging management of RVI in the early 2000s, which ultimately led to the development of MRP-227-A, I was the NRC technical lead contact for the industry program, maintaining cognizance of the industry program and providing technical feedback on their plans and progress. Over the last several years, I have been a peer reviewer of RVI AMPs and inspection plans from several license renewal applications and plants with renewed licenses (submitted in compliance with license renewal commitments). Included in this was assistance and peer review of LR Interim Staff Guidance LR-ISG-2011-04: Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors. May 28, 2013 (ADAMS Accession No. ML12270A251), which incorporated aging management guidance into the license renewal guidance documents, the GALL Report and the SRP-LR. In addition, I have been involved in generic reviews of several documents and technical issues related to RVI, including Westinghouse Report, WCAP-17780-P, Rev. 0, “Reactor Internals Aging Management MRP-227-A Applicability for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs,” June 2013, and MRP-227-A Applicability Guidelines for

Combustion Engineering and Westinghouse Pressurized Water Reactor Designs, Enclosure to MRP Letter 2013-025, October 14, 2013, transmitted via email from K. Amberge to J. Holonich, November 15, 2013 (ADAMS Accession No. ML13322A454), and was one of the reviewers who developed Evaluation of WCAP-17780-P, "Reactor Internals Aging Management MRP-227-A Applicability for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs," and MRP-2013-025, "MRP-227-A Applicability Template Guideline," November 7, 2014 (ADAMS Accession No. ML14309A484). The technical issue reviews include screening evaluation of RVI fabricated from CASS, and aging management of lower support columns.

A3b. [JP] While employed at Baltimore Gas & Electric Company (BGE) and Constellation Energy, as a materials engineer at Calvert Cliffs Nuclear Power Plant, I developed a draft aging management program for Cast Austenitic Stainless Steel components, which included some highly irradiated internals components. Additionally, from 1999-2001, I represented BGE/Constellation on the Electric Power Research Institute (EPRI), Materials Reliability Program (MRP) Reactor Internals Issue Task Group (RI-ITG). While employed at Constellation Nuclear Services (CNS), a subsidiary of Constellation Energy, I prepared portions of the technical basis for the license renewal application for boiling water reactors Nine Mile Point, Units 1 and 2, including portions related to the reactor vessel internals. While employed at Constellation Energy, the licensee for Nine Mile Point, Units 1 and 2 in the corporate engineering organization for the nuclear fleet, I assisted in license renewal audits and inspections related to the NRC review of the Nine Mile Point Units 1 & 2 License Renewal Application. This support included responding to NRC Staff concerns related to aging management of the RVI. I also led a self-assessment of the Nine Mile Point, Units 1 & 2 Boiling Water Reactor Vessel and Internals Program, and updated an engineering report assessing the performance of hydrogen water chemistry and noble metal chemical addition for Nine Mile Point.

While employed at NRC, as a Senior Materials Engineer in the Office of Nuclear Reactor Regulation, Division of Component Integrity and Division of Engineering, I participated in resolving comments on the draft safety evaluation of MRP-227, Rev. 0. I am also the lead for resolution of issues and developing guidance related to Applicant/Licensee Action Items 1 and 7 from the NRC Staff's final safety evaluation of MRP-227, Rev. 0. I participated in numerous public meetings related to action items which led to development of guidance for resolution of A/LAI 1 that was acceptable to the NRC. I was the lead reviewer for three plant-specific RVI inspection plans (in addition to IP2 and IP3) for which safety assessments have been issued acknowledging closure of license renewal commitments.

Q4. Please explain your duties in connection with the Staff's review of the License Renewal Application ("LRA") submitted by Entergy Nuclear Operations, Inc. ("Entergy," "Applicant" or "Licensee") for the renewal of Indian Point's License Nos. 50-247-LR and 50-286-LR.

A4a. [AH] As the Senior Technical Advisor in the Division of License Renewal, I assist and guide the Staff in its review of information from applicants. For the review of the Indian Point LRA related to the RVI AMP and inspection plan, I assisted the lead technical reviewer of the RVI AMP and inspection plan by doing a peer review of his findings and reviewed his various products, such as the input for the safety evaluation report.

A4b. [JP] I was the lead technical reviewer for the AMP for reactor vessel internals, and the plant-specific reactor vessel internals (RVI) inspection plan. My review included Entergy's proposed AMP including changes to the program to align it with the MRP-227-A program, the inspection plan, resolution of eight (8) action items, and associated changes to the aging management review tables for the reactor vessel internals in Entergy's LRA. As the lead technical reviewer, I prepared requests for additional information and reviewed the applicant's

responses. I also prepared inputs to and authored several sections of the supplemental safety evaluation report ("SER") including Sections 3.0.3.3.9, "Reactor Vessel Internals," 3.0.3.3.10, "Reactor Vessel Internals Inspection Plan," and Sections 3.1.2.2.6, 3.1.2.29, 3.1.2.2.12, 3.1.2.2.15, and 3.1.2.2.17, which document the Staff's review of several further evaluations of specific aging effects applicable to RVI. Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3, Supplement 2 ("SSER"), Exhibit ("Ex.") NYS000507.

A4c. [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.

Q5. Why are you testifying here today?

A5. [AH][JP][GS] The purpose of our testimony is to present the Staff's views with respect to Entergy's proposed RVI AMP and Contention NYS-25, filed in this proceeding by the State of New York ("State" or "New York") and Joint Contention NYS-38 and Riverkeeper TC-5 (NYS-38/RK-TC-5). That contention generally challenges the adequacy of the Applicant's AMP(s) to manage the effects of aging on reactor vessel internals during the period of extended operation.

Q6. What did you review in order to prepare your testimony?

A6a. [JP][AH] I reviewed the Board's orders related to NYS-25 and NYS-38/RK-TC-5. I also reviewed

- State of New York's Motion for Leave to Supplement Previously-Admitted Contention NYS-25 (Feb. 13, 2015) (ADAMS Accession No. ML15044A508) (Non-Public);
- New York State February 2015 Supplement to Previously-Admitted Contention NYS-25 (Feb. 13, 2015) (ADAMS Accession No. ML15044A507) (Non-Public);
- Declaration of Richard T. Lahey, JR. (Feb. 13, 2015) (ADAMS Accession No. ML15044A505) (Non-Pubic);
- Declaration of Lisa S. Kwong (Feb. 13, 2015) (ADAMS Accession No. ML15044A506) (Non-Public);
- *Entergy Nuclear Operations, Inc.* (Indian Point Nuclear Generating Units 2 and 3), LBP-08-13, 68 NRC 43 (2008);
- State of New York's Motion for Leave to File Additional Bases for Previously-Admitted Contention NYS-25 in Response to Entergy's July 14, 2010 Proposed Aging Management Program for Reactor Pressure Vessels & Internal Components, dated (Sept. 15, 2010) (ADAMS Accession No. ML103050402);
- State of New York Initial Statement of Position Contention NYS-25 (Dec. 22, 2011) (Ex. NYS000293);
- Pre-filed Written Testimony of Richard T. Lahey, Jr. Regarding Contentions NYS-25 (Dec. 22, 2011) (Ex. NYS000294);
- Report of Dr. Richard T. Lahey, Jr. in Support of Contentions NYS-25 and NYS-26B/RK-TC-1B (Dec. 22, 2011) (Ex. NYS000296);

- Supplemental Report of Dr. Richard T. Lahey, Jr. in Support of Contentions NYS-25 and NYS-26B/RK-TC-1B (Dec. 21, 2011) (Ex. NYS000297);
- Declaration of Dr. Richard T. Lahey, Jr., In Support of the State of New York's Supplemental Contention 26-A (Dec. 22, 2011) (Ex. NYS000299);
- Declaration Of Dr. Richard T. Lahey, Jr. (Sept. 9, 2010) (NYS000300);
- Letter NL-10-063, from Fred Dacimo, Vice President, Entergy, to NRC, Amendment 9 to License Renewal Application (LRA) – Reactor Vessel Internals Program Indian Point Nuclear Generating Unit Nos. 2 & 3 Docket Nos. 50-247 and 50-286 License Nos. DPR-26 and DPR-64 (July 14, 2010) (ADAMS Accession No. ML102010102);
- Letter MRP 2009-004, from Christian Larsen, Nuclear Vice President & Chief Nuclear Officer, EPRI, to NRC, Report Transmittal; *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," (MRP-227-Rev. 0)*, EPRI Palo Alto, CA, 2008, 1016596 (Jan. 12, 2009) (ADAMS Accession No. ML090160204);
- EPRI 1016596, Final Report, *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0)* (Dec. 2008) (ADAMS Accession Nos. ML090160212 and ML090160206);
- Letter MRP 2011-036, from Scot Greenlee, to NRC, Transmittal: PWR Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A) (Jan. 9, 2012) (ADAMS Accession No. ML12017A193);
- EPRI 1022863, Final Report, *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)* (Dec. 2011) (ADAMS Accession Nos. ML12017A194 , ML12017A196, ML12017A197, ML12017A191, ML12017A192, ML12017A195, and ML12017A199);
- Letter NL-12-089, from Fred Dacimo, Vice President, Entergy, to NRC, Reply to Request

for Additional Information Regarding the License Renewal Application Indian Point Nuclear Generating Units 2 & 3 Docket Nos. 50-247 and 50-286 License Nos. DPR-26 and DPR-64 (June 14, 2012) (ADAMS Accession No. ML12184A037);

- Letter NL-12-037, from Fred Dacimo, Vice President, Entergy, to NRC, License Renewal Application - Revised Reactor Vessel Internals Program and Inspection Plan Compliant with MRP-227-A Indian Point Nuclear Generating Unit Nos. 2 and 3 Docket Nos. 50-247 and 50-286–License Nos. DPR-26 and DPR-64 (Feb. 17, 2012) (ADAMS Accession No. ML12060A312);
- NUREG-1930, Vol. 2, *Safety Evaluation Report Related to The License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3, Dockets No. 50-247 and 50-286* (Nov. 30, 2009) (ADAMS Accession No. ML093170671);
- Letter from Robert Nelson, Deputy Director, NRC, to Neil Wilmshurst, Vice President and Chief Nuclear Officer, EPRI, Revision 1 of the Final Safety Evaluation of Electric Power Research Institute (EPRI) Report, Materials Reliability Program (MRP) Report 1016596 (MRP-227), Revision 0, “PWR (PWR) Internals Inspection and Evaluation Guidelines” (TAC NO. ME0680) (Dec. 16, 2011) (ADAMS Accession No. ML11308A770);
- WCAP-17096-NP, Rev. 2, Reactor Internals Acceptance Criteria Methodology and Data Requirements (Dec. 2009) (ADAMS Accession No. ML101460157).
- EPRI 1013234, Technical Report, *Materials Reliability Program: Screening, Categorization and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Designs (MRP-191)* (Nov. 2006) (ADAMS Accession No. ML091910130);
- Letter NL-12-134, from Fred Dacimo, Vice President, Entergy, to NRC, 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 (Sept. 28, 2012) (ADAMS Accession No. ML12285A232);

- Memorandum from Joseph Golla, Project Manager, NRC, to Sheldon Stuchell, Acting Chief, NRC, Summary of November 28, 2012, Category II Public Meeting with the Electric Power Research Institute and Industry Representatives (Jan. 29, 2013) (ADAMS Accession No. ML13009A066);
- Memorandum from Joseph Golla, Project Manager, NRC, to Anthony Mendiola, Chief, NRC, Summary of January 22-23, 2013, Closed Meeting with the Electric Power Research Institute and Westinghouse (Feb. 21, 2013) (ADAMS Accession No. ML13042A048 and available in Document Package No. ML13043A062);
- Memorandum from Joseph Golla, Project Manager, NRC, to Anthony Mendiola, Chief, NRC, Summary of February 25, 2013, Telecom with the Electric Power Research Institute and Westinghouse Electric Company (Mar. 15, 2013) (ADAMS Accession No. ML13067A262);
- Memorandum from Joseph Golla, Project Manager, NRC, to Sheldon Stuchell, Senior Project Manager, NRC, Summary of May 21, 2013, Public Meeting Regarding Pressurized Water Reactor (PWR) Vessel Internals Inspections (June 24, 2013) (ADAMS Accession No. ML13164A126);
- NRC Presentation Slides, Status of MRP-227-A Action Items 1 and 7 (June 5, 2013) (ADAMS Accession No. ML13154A152);
- WCAP-17780-P, Rev. 0, Reactor Internals Aging Management MRP-227-A Applicability for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs (June 2013);

- Enclosure to MRP Letter 2013-025, MRP-227-A Applicability Guidelines for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs (Oct. 14, 2013) (ADAMS Accession No. ML13322A454);
- Evaluation of WCAP-17780-P, "Reactor Internals Aging Management MRP-227-A Applicability for Combustion Engineering [(CE)] and Westinghouse [Electric Company (Westinghouse)] Pressurized Water Reactor Designs," and MRP-2013-025, "MRP-227-A Applicability Template Guideline" (Nov. 7, 2014) (ADAMS Accession No. ML14309A484);
- Letter NL-14-010, from Fred Dacimo, Vice President, Entergy, to NRC, 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 (Jan. 16, 2014) (ADAMS Accession No. ML14027A413);
- Enclosure 1 to NL-14-010, Final Response to U.S. NRC RAI 6-A Items 1 and 2 on the RVI Program and RVI Inspection Plan for Indian Point Units 2 and 3 (Proprietary) (Jan. 9, 2014) (ADAMS Accession No. ML14027A416) (Proprietary);
- Enclosure 2 to NL-14-010, Final Response to U.S. NRC RAI 6-A Items 1 an 2 on the RVI Program and RVI Inspection Plan for Indian Point Units 2 and 3 (Non-Proprietary) (Jan. 9, 2014) (ADAMS Accession No. ML14027A415);
- Letter NL-12-166, From Fred Dacimo, Vice President, Entergy, to NRC, 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 (Nov. 20, 2012) (ADAMS Accession No. ML12340A154);
- Letter NL-11-101, From Fred Dacimo, Vice President, Entergy, to NRC, Clarification for Request for Additional Information (RAI) Aging Management Programs Indian Point

Nuclear Generating Unit Nos. 2 & 3 Docket Nos. 50-247and 50-286 License Nos. DPR-26 and DPR-64 (Aug. 22, 2011) (ADAMS Accession No. ML11243A085);

- EPRI 1002792, Final Report, *Materials Handbook for Nuclear Plant Pressure Boundary Applications* (Dec. 2002);
- Letter from Christopher Grimes, Chief, NRC, to Douglas Walters, NEI, License Renewal Issue No. 98-0030, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components" (May 19, 2000) (NRC ADAMS Accession No. ML003717179);
- Letter NL-12-140, from Fred Dacimo, Vice President, Entergy, to NRC, 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 (Oct. 17, 2012) (ADAMS Accession No. ML12300A391);
- Letter NL-13-052, from Fred Dacimo, Vice President, Entergy, to NRC, 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);
- NUREG/CR-6960, Crack Growth Rates and Fracture Toughness of Irradiated Austenitic Stainless Steels in BWR Environments (Mar. 31, 2008) (ADAMS Accession No. ML081130709);
- NRC Position on Aging Management of CASS Reactor Vessel Internal Components (June 11, 2014) (ADAMS Accession No. ML14163A112);
- NUREG/CR-7027, Degradation of LWR Core Internal Materials due to Neutron Irradiation (Dec. 31, 2010) (ADAMS Accession No. ML110100377);
- Memorandum from Joseph Golla, Project Manager, to Anthony Mendiola, Chief, Summary of January 22-23, 2013, Closed Meeting with the Electric Power Research

Institute and Westinghouse (Feb. 21, 2013) (ADAMS Accession No. ML13042A048);

- Memorandum from Joseph Golla, Project Manager, to Anthony Mendiola, Chief, Summary of February 25, 2013, Telecom with the Electric Power Research Institute and Westinghouse Electric Company (Mar. 15, 2013) (ADAMS Accession No. ML13067A262);
- Letter NL-13-122, from Fred Dacimo, Vice President, Entergy, to NRC, 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 (Sept. 27, 2013 (ADAMS Accession No. ML13277A007);
- Memorandum from Stacey Rosenburg, Chief, to Yoira Diaz-Sanabria, Chief, Indian Point Nuclear Generating, Units 2 and 3 - Audit Report, April 24-25, 2013, Related to Amendment 9 to License Renewal Application - Reactor Vessel Internals Program and the Reactor Vessel Internals Inspection Plan, Docket Nos. 50-247 and 50-286 (TAC Nos. MD5407 and MD5408) (Oct. 17, 2013) (ADAMS Accession No. ML13280A905);
- Letter NL-11-107, from Fred Dacimo, Vice President, Entergy, to NRC, License Renewal Application - Completion of Commitment #30 Regarding the Reactor Vessel Internals Inspection Plan Indian Point Nuclear Generating Unit Nos. 2 and 3 Docket Nos. 50-247 and 50-286 License Nos. DPR-26 and DPR-64 (Sept. 28, 2011) (ADAMS Accession No. ML11280A121);
- Letter NL-14-013, from Fred Dacimo, Vice President, Entergy, to NRC, Additional Information Regarding the License Renewal Application - Action Item 7 from MRP-227-A Indian Point Nuclear Generating Unit Nos. 2 & 3 Docket Nos. 50-247 and 50-286 License Nos. DPR-26 and DPR-64 (Jan. 28, 2014) (ADAMS Accession No. ML14038A150);

- Letter NL-14-067, from Fred Dacimo, Vice President, Entergy, to NRC, 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 (June 9, 2014) (ADAMS Accession No. ML14176A159);
- Letter NL-14-093, from Fred Dacimo, Vice President, Entergy, to NRC, 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 (Aug. 5, 2014) (ADAMS Accession No. ML14225A777);
- Letter from John Lubinski, Director, NRC, to Jean Smith, EPRI, License Renewal Interim Staff Guidance LR-ISG-2011-04: Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors (May 28, 2013) (ADAMS Accession No. ML12270A251);
- EPRI 1012081, Topical Report, Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175) (Dec. 31, 2005) (ADAMS Accession No. ML061880278) (Non-proprietary);
- Letter from Fred Dacimo, Vice President, Entergy, to NRC, Entergy Nuclear Operations Inc. Indian Point Nuclear Generating Unit Nos. 2 & 3 Docket Nos. 50-247 and 50-286 Amendment 3 to License Renewal Application (LRA) (Mar. 24, 2008) (ADAMS Accession No. ML081070255);
- PWR Owners Group Presentation Slides, Industry and NRC Coordination Meeting Materials Programs Technical Exchange (June 5, 2014) (ADAMS Accession No. ML14163A520);
- EPRI 1016593, Final Report, Materials Reliability Program: Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals (MRP-232)

(Dec. 31, 2008) (ADAMS Accession No. ML092230745);

- Revised Pre-Filed Written Testimony of Dr. Richard T. Lahey, Jr., Regarding Contention NYS-25 (June 9, 2015) (Ex. NYS000482);
- Regulatory Information Conference Presentation Slides, “Recent Materials Inspections of PWR Reactor Internals” (Mar. 2015) (ADAMS Accession No. ML15104A041);
- EPRI 1021001NP/BWRVIP-100NP, Rev.1, Final Report, BWR Vessel and Internals Project Updated Assessment of the Fracture Toughness of Irradiated Stainless Steel for BWR Core Shrouds (Oct. 2010) (ADAMS Accession No. ML12044A189);
- PWROG-14048-P, Rev. 0, Functionality Analysis: Lower Support Columns (Dec. 2014) (ADAMS Accession No. ML15077A114) (Non-Public);
- EPRI 1013396/BWRVIP-100-A, Final Report, BWR Vessel and Internals Project Updated Assessment of the Fracture Toughness of Irradiated Stainless Steel for BWR Core Shrouds (Aug. 2006) (ADAMS Accession No. ML062570229) (Non-Public).
- Indian Point Unit 2 Updated Final Safety Evaluation Report, Rev. 25, at Chapter 3 (2014);
- Indian Point Unit 2 Updated Final Safety Evaluation Report, Rev. 25, at Chapter 14 (2014);
- Indian Point Unit 2 Updated Final Safety Evaluation Report, Rev. 25, at Chapter 16 (2014);
- Indian Point Unit 3 Updated Final Safety Evaluation Report, Rev. 4, at Chapter 3 (2011);
- Indian Point Unit 3 Updated Final Safety Evaluation Report, Rev. 4, at Chapter 14 (2011);
- Indian Point Unit 3 Updated Final Safety Evaluation Report, Rev. 4, at Chapter 16 (2011);

- MRP Letter 2014-09, Biennial Report of Recent MRP-227-A Reactor Internals Inspection Results (Project 694) (May 12, 2014) (ADAMS Accession Nos. ML14135A383, ML14135A384, ML14135A385) ;
- Babcock & Wilcox Report S-1473-002, Rev. 0, Examination of Clevis Bolts Removed from D. C. Cook Nuclear Plant (ADAMS Accession No. ML14253A318);
- NUREG/CR-6909, Rev. 1, Draft Report for Comment, Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials (Mar. 2014) (ADAMS Accession No. ML14087A068);
- NUREG/CR-7153, Vol. 1, Expanded Materials Degradation Assessment (EMDA): Executive Summary of EMDA Process and Results (Oct. 2014) (ADAMS Accession No. ML14279A321);
- NUREG/CR-7153, Vol. 2, Expanded Materials Degradation Assessment (EMDA): Aging of Core Internals and Piping Systems (Oct. 2014) (ADAMS Accession No. ML14279A331);
- NUREG/CR-7153, Vol. 3, Expanded Materials Degradation Assessment (EMDA): Aging of Reactor Pressure Vessels (Oct. 2014) (ADAMS Accession No. ML14279A349);
- NUREG/CR-7153, Vol. 4, Expanded Materials Degradation Assessment (EMDA): Aging of Concrete and Civil Structures (Oct. 2014) (ADAMS Accession No. ML14279A430);
- NUREG/CR-7153, Vol. 4, Expanded Materials Degradation Assessment (EMDA): Aging of Cable and Cable Systems (Oct. 2014) (ML14279A461); and
- NUREG/CR-7184, Crack Growth Rate and Fracture Toughness Tests on Irradiated Cast Stainless Steels (Dec. 2014) (ADAMS Accession No. ML14356A136).

A6b. [GS] I reviewed Dr. Lahey's Lahey's Prefiled Testimony (Ex. NYS000374); Dr. Lahey's Revised Prefiled Testimony (Ex. NYS000482); WCAP-17199-P (Ex. NYS000361);

WCAP-17200-P (Ex. NYS000362); the 1980 Spent Fuel Report (Ex. NYS000348); the Staff's Supplemental Safety Evaluation Report (NUREG-1930, Supplement 1, August 2011 (Ex. NYS000160), the NRC presentation for the WESTEMS Audit (Ex. NRC000119); NUREG-2101, *Safety Evaluation Report Related to the License Renewal of Salem Nuclear Generating Station*, June 2011 (Ex. ENT000195); and RIS-2011-14 (Ex. NRC000112).

Q7. Based on your review of Entergy's proposed RVI AMP, what is your expert opinion regarding the adequacy of the program?

A7. [JP][AH] Entergy's currently proposed RVI AMP for Indian Point Units 2 (IP2) and 3 (IP3) and inspection plan provides a fully developed and effective condition monitoring program that provides for timely inspections of the reactor vessel internals components that are most likely to have aging degradation, and thus is capable of managing the aging effects that may be experienced by RVI components through the period of extended operation. As documented in the SSER, the Staff concluded that Entergy's AMP is consistent with the approved program as documented in License Renewal Interim Staff Guidance LR-ISG-2011-04, "Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors," and satisfies the Commission's regulatory requirements including 10 C.F.R. § 54.29(a).

The use of condition monitoring by inspections, such as in the RVI AMP, is entirely consistent with the requirements for license renewal at 10 C.F.R. § 54.21(c)(3), which states that 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 [current licensing basis] for the period of extended operation."

Q8. Based on your review, what is your expert opinion regarding New York State's assertions in Contention 25 ("NYS-25") and New York State and Riverkeeper Joint Contention 38 ("NYS-38/RK-TC-5")?

A8. [JP][AH] In our opinion, the issues raised by New York in NYS-25 do not identify any issues that would undermine the Staff's conclusion that the RVI AMP is capable of adequately managing any aging effects, including potential synergistic effects resulting from multiple aging mechanisms through the period of extended operations. New York's assertions do not appreciate full scope of and program activities for the RVI AMP. For example, the RVI AMP (1) uses inspections that will detect cracking regardless of whether one or multiple aging effects are occurring, (2) selects primary components that are subject to an increased likelihood of crack initiation and growth and will be inspected first in order to assess the likelihood of crack initiation elsewhere in the RVI, (3) uses conservative acceptance criteria for most Primary and Expansion components that are inspected, including no detectable cracking for cracking-related mechanisms, and (4) requires issues identified from inspections, operating experience at IP2 and IP3, and other relevant operating experience within the United States and outside the United States to be dispositioned in its corrective action program

<u>DESCRIPTION OF WESTINGHOUSE REACTOR VESSEL INTERNALS</u>

Q9. With respect to the RVI components, what components are within the scope of license renewal?

A9. [JP][AH] For IP2 and IP3, the entire RVI are generally within the scope of license renewal, except for those portions that do not meet 10 C.F.R. § 54.21 (a)(1)(i) and (a)(1)(ii). As will be discussed subsequently, there are several components that have been identified by New York and its witness as needing to be managed within the RVI AMP, however, those

components generally do not require aging management because they are not passive long-lived components or are covered by separate programs that have not been challenged by New York or its witness.

Q10. What components that are contained within the reactor pressure vessel are not considered RVI components?

A10. [JP][AH] Non-passive components such as control rods and control rod drive mechanisms, are not subject to aging management under the RVI AMP and are not considered RVI components. Fuel assemblies are not subject to aging management because they are not long-lived, and nuclear instrumentation is not subject to aging management because it is not passive. The RVI also does not include any pressure boundary components, such as the reactor pressure vessel, or the penetrations for the control rod drive mechanisms in the reactor pressure vessel closure head; these components are addressed in other programs.

Q11. Can you provide a general overview of the types of components within the scope of license renewal and part of the RVI?

A11. [JP][AH] The lower core support structure, the upper core support structure, and the incore instrumentation support structure are the three major parts of the reactor vessel internals that are within the scope of license renewal. The components within these parts generally have the functions of structural support, flow distribution, or shielding. There are other components covered within the scope of license renewal in the reactor vessel internals program including some but not all of the components identified by Dr. Lahey in his testimony. Specifically, the following components are within the scope of license renewal and any aging effects are managed by the RVI AMP: core barrel (and its welds); baffle-former assembly including baffle plates, former plates, baffle-former bolts, barrel-former bolts, and baffle-edge

bolts; lower core plate and support structures; clevis bolts; fuel alignment pins; thermal shield; the lower support columns and the upper support column assemblies; the control rod guide tubes, plates, and welds; the baffle-to-baffle bolts, the barrel-to-former bolts, and baffle-to-former bolts. Certain components including the control rods and the control rod drive mechanism penetrations identified by Dr. Lahey are either not subject to aging management under license renewal or are addressed by other AMPs not challenged by New York. When Entergy amended their LRA to include the RVI aging management guidance provided in MRP-227, Rev. 0[1], it updated LRA Tables 3.1.2-2-IP2 and 3.1.2.-2-IP3 to provide a complete list of the IP2 and IP3 RVI components that are in scope and subject to aging management review (NL-10-063 at 18-81, Exhibit NYS000313). Entergy initially used MRP-227, Rev. 0, to develop its RVI AMP because it was the best industry guidance available at the time. In addition, Table 5-1 at 29-36 of NL-12-037, Exhibit NYS000496, provides a listing of all the RVI components subject to aging management, with a cross index between the tables in NL-10-063 and MRP-227-A, since IP2 and IP3 component names differ from the generic MRP-227-A component names.

Q12. Why are the control rods not addressed by the RVI AMP?

A12. [JP][AH] The control rods are within the scope of license renewal, but are not subject to aging management under license renewal, because they are not long-lived components. The control rods screen out from aging management review in accordance with 10 CFR 54.21(a)(1)(ii).

[1] MRP-227, Rev. 0 was the version of topical report MRP-227 submitted to the NRC Staff for review and approval. The NRC Staff-approved version is designated MRP-227-A and incorporates changes due to conditions imposed by the NRC Staff in its final safety evaluation related to MRP-227. Rev.0.

Q13. Why are the control rod drive mechanism penetrations not addressed by the RVI Program?

A13. [JP][AH] The control rod drive mechanism penetrations in the reactor pressure vessel closure head, are in the scope of license renewal and are subject to aging management, but they are addressed by the Reactor Vessel Head Penetration Inspection Program.

Q14. For the components covered by the RVI AMP, are there any other programs that help to manage the aging effects of the RVI components?

A14. [JP][AH] Yes. All the RVI are also covered by the "Water Chemistry Control – Primary and Secondary" AMP, which is credited for managing the aging effect (loss of material). This program maintains the reactor coolant system water chemistry such that loss of material due to pitting, crevice corrosion, and general corrosion is minimized.

Q15. How does the water chemistry program minimize loss of material due to corrosion and stress corrosion cracking?

A15. [JP][AH] The water chemistry program minimizes loss of material due to corrosion by minimizing and controlling the concentration of dissolved oxygen and impurities such as chloride, fluoride, and sulfate ions in the reactor coolant, and by controlling the pH. If concentrations of any of these chemical species or parameters are outside specified control limits, corrective actions are initiated to bring these species and parameters back within the limits. The water chemistry program also minimizes the potential for stress corrosion cracking (SCC) of the RVI by controlling the concentrations of these same parameters.

Q16. Are any of the RVI components covered by TLAA or other type of analysis?

A16. [JP][AH] Certain RVI components are also covered by a time-limited aging analysis (TLAA) for fatigue. A TLAA may be credited for managing aging in lieu of or in addition to an AMP. A TLAA is an analysis that meets the following six criteria as defined in 10 CFR 54.3:

(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 current licensing basis (CLB).

A TLAA may be resolved for license renewal by demonstrating in accordance with To C.F.R. § 54.21(c)(1), that either:

(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.

Mechanical design analyses were performed for the RVI during design of the plant considering normal operation and accident conditions. These analyses demonstrate the components are able to perform their intended function during the period of extended operation.

Q17. Which RVI components are covered by TLAA?

A17. [JP][AH] Components covered by the fatigue TLAAs for IP2 are the upper support plate assembly, upper support plate flange, upper core plate, mid core barrel, upper core barrel, core barrel nozzle, core barrel flange, lower radial key plate, lower radial key 45° plane, lower core plate, lower core support plate, and lower support columns. Components covered by the fatigue TLAA for IP3 are the upper support plate assembly, upper core plate, core barrel to lower support plate junction, thermal shield, lower core plate, instrumentation columns, and lower support columns.

Q18. For the RVI components covered by a TLAA, did the analysis show that the components would perform their intended function during the period of extended operation?

A18. [JP][AH] Yes, each of the TLAAs for the RVI components was demonstrated to meet the criteria identified in 10 C.F.R. § 54.21(c)(1). In particular, the cumulative usage factor (CUF) calculations for the RVI components to cover the license renewal operating period are demonstrated in the license renewal application to be less than 1.0 in all cases, indicating that the RVI components will not exhibit fatigue crack initiation.

In addition, Entergy committed to reevaluate the fatigue CUFs for the RVI components in Commitment 49 to account for the effects of the reactor water environment. If these environmentally assisted CUF values (CUF_{en} values) exceed 1.0 for any component, then additional corrective actions would be taken. Commitment 49 has been completed for IP2. Letter from Entergy to U.S. NRC Regarding Indian Point Nuclear Generating Unit 2 Implementation of License Renewal Commitments (ML13247A175) Ex. NRC000184.

DESCRIPTION OF THE INDIAN POINT REACTOR VESSEL INTERNALS

Q19. Could you briefly describe the functions performed by the reactor vessel internals?

A19. [AH] [JP] The components of the RVI support the core, direct flow of reactor coolant, or provide alignment and guidance for components such as control rods or instrumentation.

IP2 and IP3 are Westinghouse-design PWRs, and the following discussion is for Westinghouse-design RVI. Figure 1 below is an overview of typical Westinghouse-design internals and is a fair representation of the reactor vessel and internal components for IP2 and IP3.

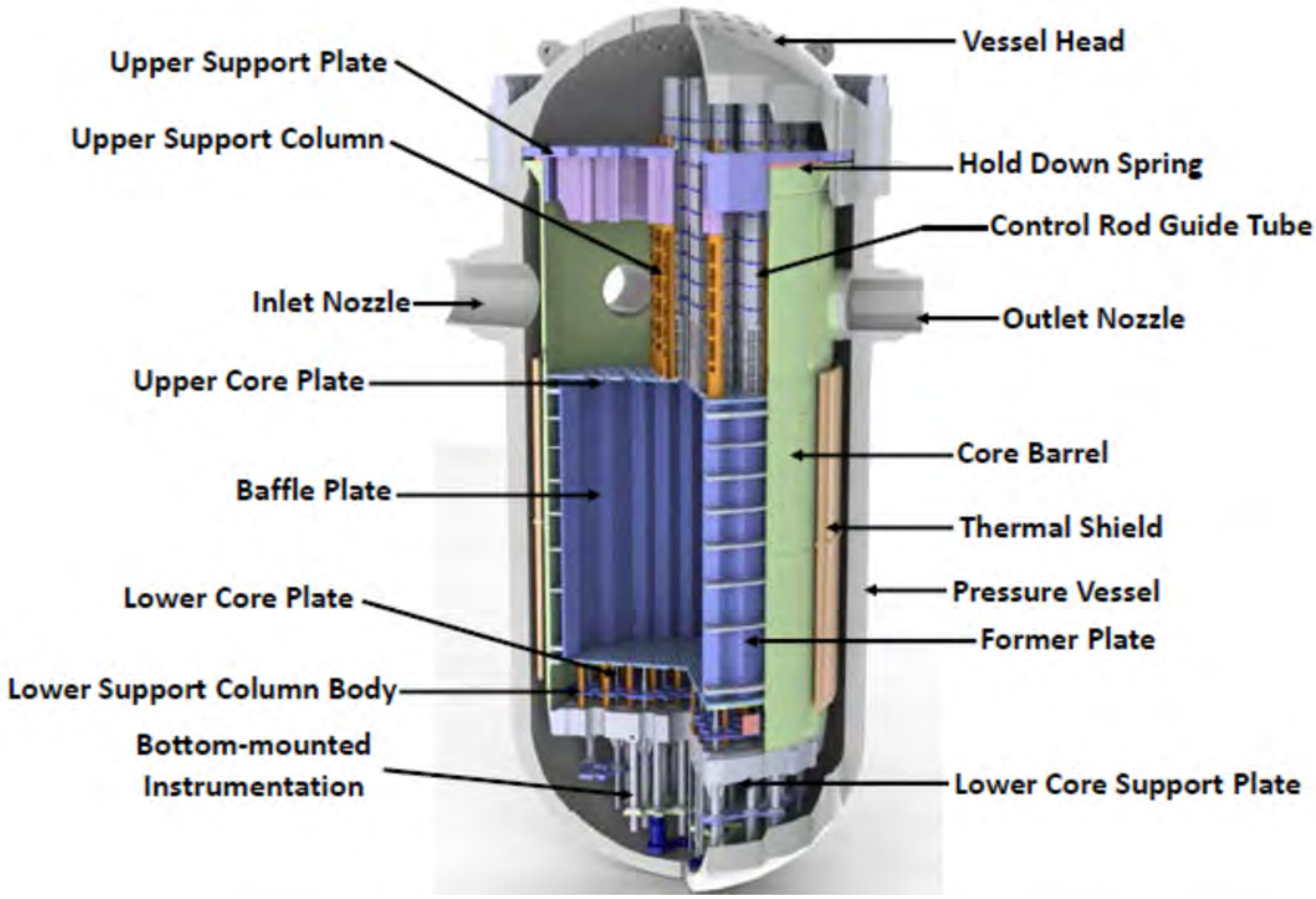

Figure 1 – Overview of Typical Westinghouse-design Reactor Vessel Internals (reproduced from update Westinghouse Figures Presented at Public Meeting on MRP-227, Rev. 1, March 31, 2015, ADAMS Accession No. ML15091A136 Exhibit NRC000211)

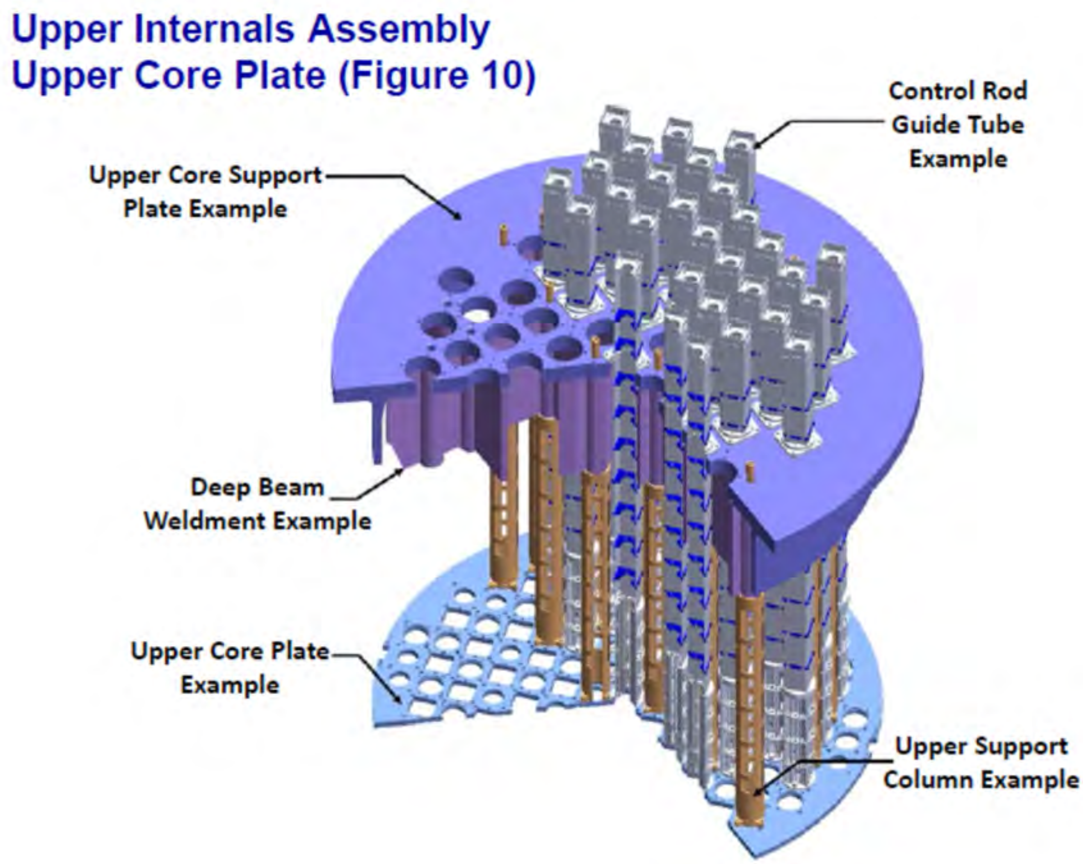

Figure 2 – Example of Upper Internals Assembly. The design is slightly different from IP2 and IP3 because the upper support plate is not a top hat design. (reproduced from updated Westinghouse Figures Presented at Public Meeting on MRP-227, Rev. 1, March 31, 2015, ADAMS Accession No. ML15091A136 Exhibit NRC000211)

Q20. What are the major assemblies that make up the RVI?

A20. [JP][AH] As shown in Figure 1, the major assemblies of Westinghouse-design RVI consist of the upper internals assembly (also illustrated in Figure 2) that is removed during each refueling operation to obtain access to the reactor core, and a lower internals assembly that can be removed following a complete core off-load. In Figure 1, the lower internals assembly includes the core barrel, baffles and former plates, lower core plate, lower core support plate, and lower support columns. The lower internals assembly is essentially the full height of the

reactor pressure vessel, as it is supported from a ledge just below the reactor pressure vessel closure flange. The upper internals assembly sits within approximately the upper half of the lower internals assembly, and includes the upper core plate, upper support plate, and control rod guide tube assemblies.

Q21. Which components comprise the upper internals assembly?

A21. [JP][AH] The major sub-assemblies that constitute the upper internals assembly are the: (1) upper core plate (UCP); (2) upper support column assemblies; (3) control rod guide tube assemblies; and (4) upper support plate (USP). A typical upper internals assembly is shown in Figure 2.

Q22. Describe the functions of the upper internals assembly.

A22. [JP][AH] The primary function of the upper internals assembly is core support. The upper assembly also provides guidance to the control rods and instrumentation for temperature measurement.

Q23. Describe how the upper internals assembly performs its primary function of core support.

A23. [JP][AH] During reactor operation, the upper internals assembly is preloaded against the fuel assembly springs and the internals hold down spring by the reactor vessel head pressing down on the outside edge of the USP, as illustrated in Figure 1. The USP acts as the divider between the upper plenum and the reactor vessel head and as a relatively stiff base for the rest of the upper internals. The upper support columns and the control rod guide tubes are attached to the USP (see Figure 2). The upper core plate (UCP), in turn, is attached to the upper support columns. The USP design at IPEC is designated as a top hat design, in which the

UCP is perforated to permit coolant to pass from the core below into the upper plenum between the USP and the UCP. The coolant then exits through the outlet nozzles in the core barrel. The UCP positions and laterally supports the core by fuel alignment pins extending below the plate. The UCP contacts and preloads the fuel assembly springs and thus maintains contact of the fuel assemblies with the lower core plate (LCP) during reactor operation. The upper support columns vertically position the UCP and are designed to take the uplifting hydraulic flow loads and fuel spring loads on the UCP. The control rod guide tubes are bolted to the USP and pinned at the UCP so they can be easily removed if replacement is needed.

Q24. Describe how the upper internals performs the function of guidance for the control rods.

A24. [JP][AH] The control rod guide tubes (Figure 3), are bolted to the USP and pinned at the UCP. They, guide the control rods in and out of the fuel assemblies to control power generation. Guide tube cards (Figure 4) are located within each control rod guide tube.

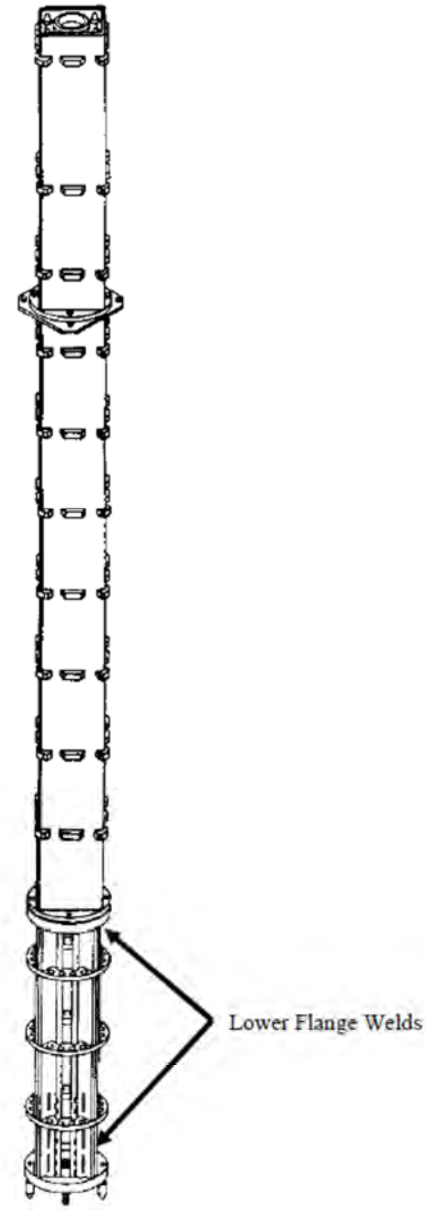

Figure 4-21
Typical Westinghouse control rod guide tube assembly

Figure 3 – Typical Westinghouse-design RVI Control Rod Guide Tube Assembly, from MRP-227-A Figure 4-21 (Exhibit NRC000114B at 4-60 ADAMS Accession No. ML12095A218)

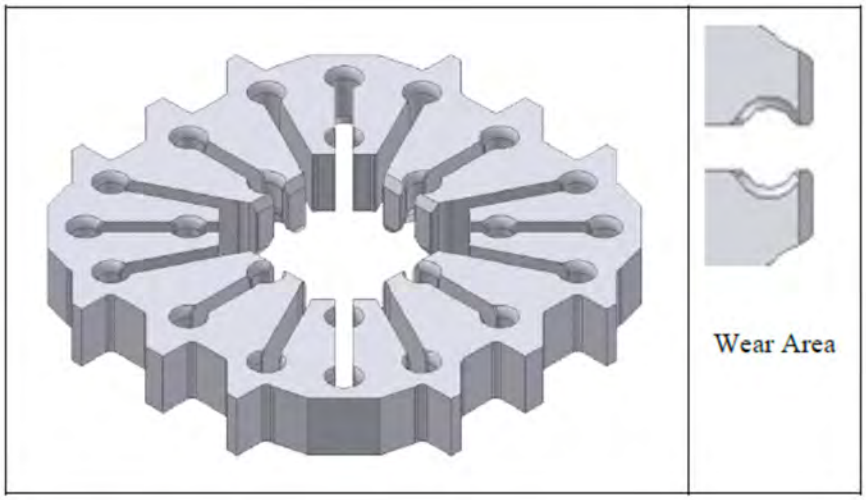

Figure 4-20
Typical Westinghouse control rod guide card (17x17 fuel assembly)

Figure 4 – Control Rod Guide Card from Westinghouse-design RVI, from MRP-227-A Figure 4-20 (Exhibit NRC000114B at 4-59 ADAMS Accession No. ML12095A218)

Q25. What are the major subassemblies of the lower internals assembly?

A25. [JP][AH] The major subassemblies of the lower internals assembly are the core barrel assembly, the baffle-former assembly (a subassembly of the core barrel assembly) and the lower core support assembly.

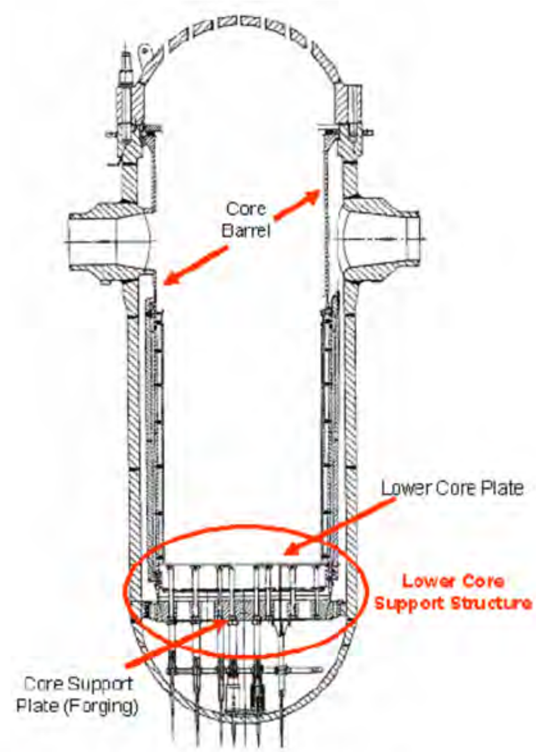

Figure 4-32
Schematic indicating location of Westinghouse lower core support structure. Additional details shown in Figure 4-33

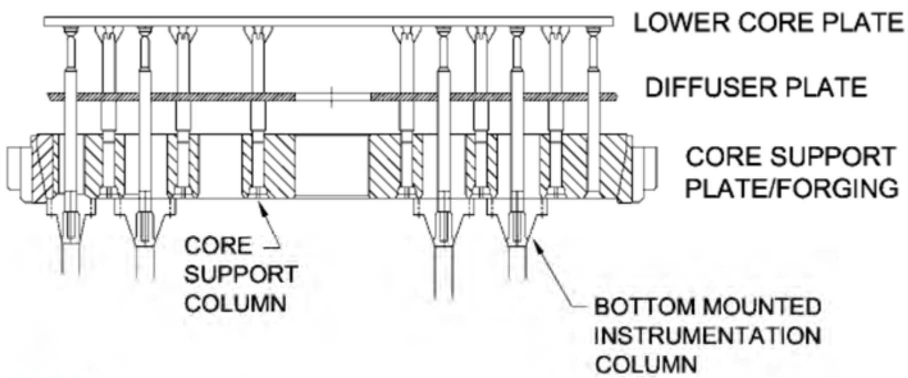

Figure 4-33
Westinghouse lower core support structure and bottom mounted instrumentation columns. Core support column bolts fasten the core support columns to the lower core plate

Figure 5 – Typical Lower Core Support Structure from Westinghouse-design RVI. Top figure shows location of lower support structure within the RVI. Bottom figure shows detail of lower support structure. From MRP-227-A Figures 4-32 and 4-33, (Exhibit NRC000114B at 4-70 ADAMS Accession No. ML12095A218)

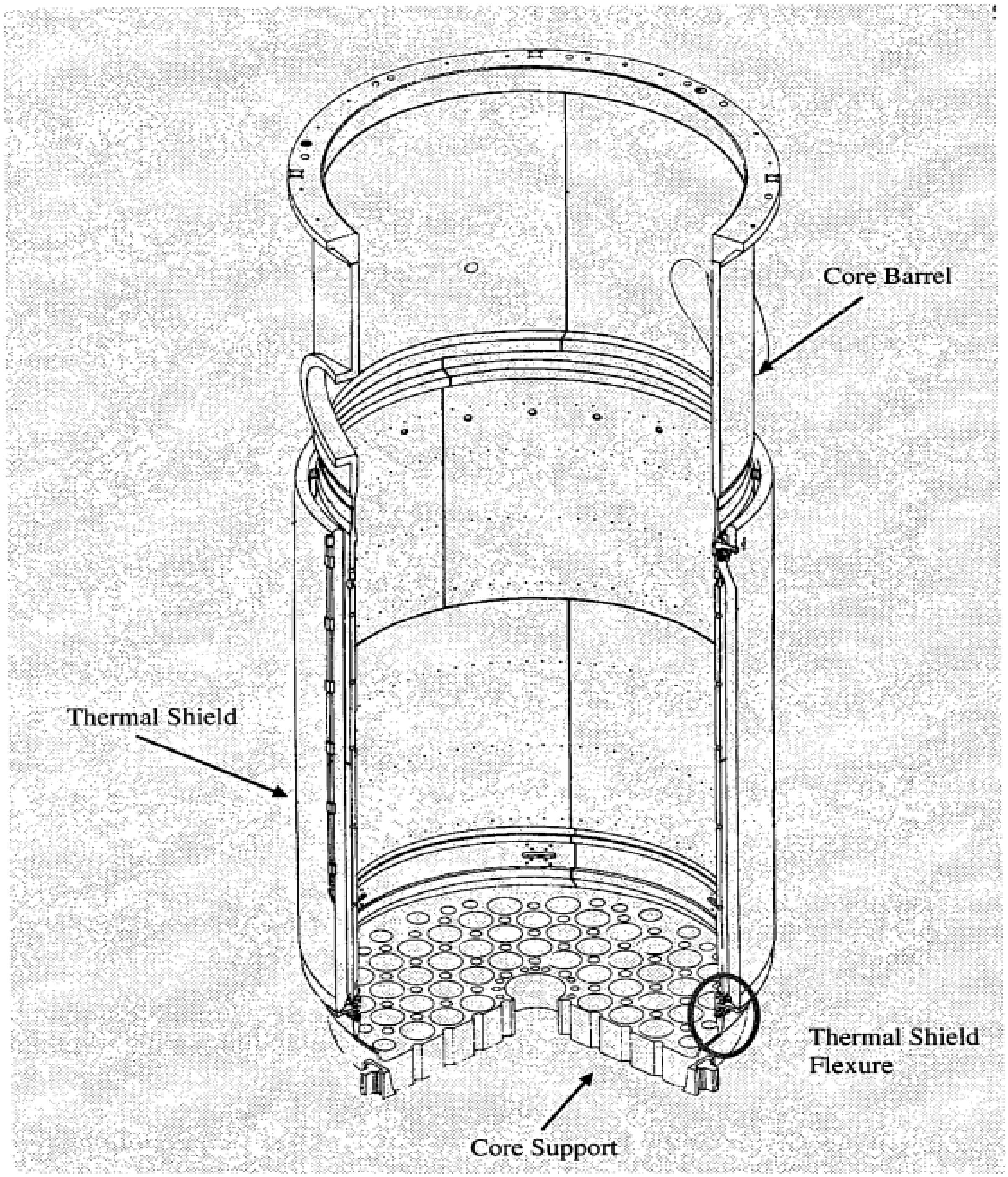

Figure 6 – Demonstrative illustration of Core Barrel from Westinghouse-design RVI

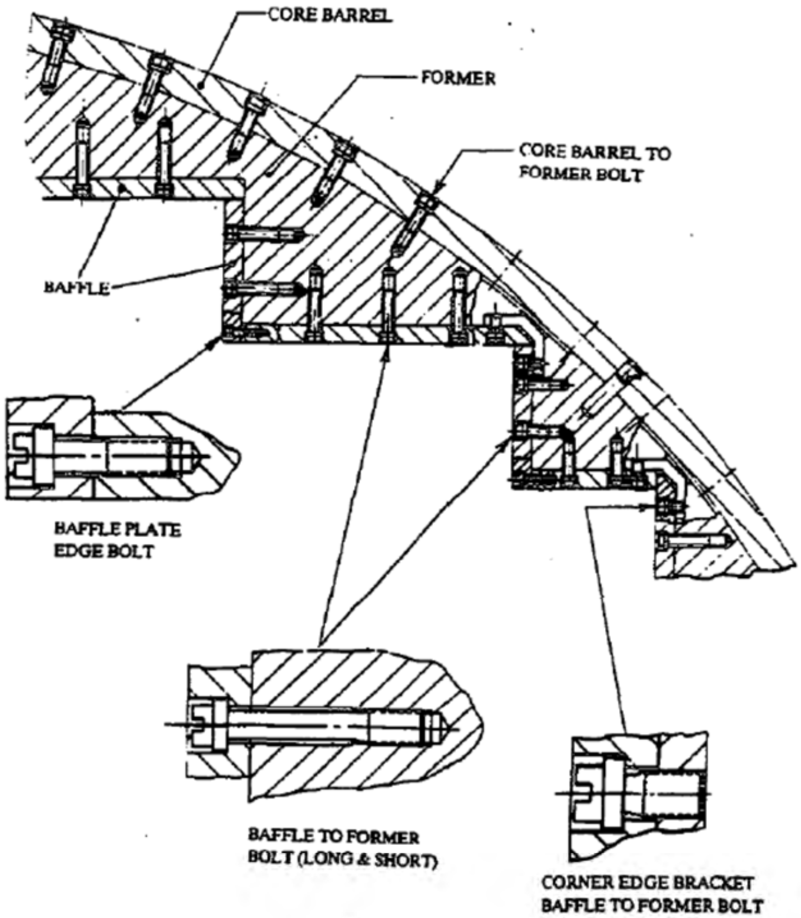

Figure 4-23
Bolt locations in typical Westinghouse baffle-former-barrel structure. In CE plants with bolted shrouds, the core shroud bolts are equivalent to baffle-former bolts and barrel-shroud bolts are equivalent to barrel-former bolts

Figure 7 – Plan view of a cross-section of a portion of the baffle-former assembly in Westinghouse-design RVI, showing bolt locations, from MRP-227-A Figure 4-23 (Exhibit NRC000114B at 4-62, ADAMS Accession No. ML12095A218)

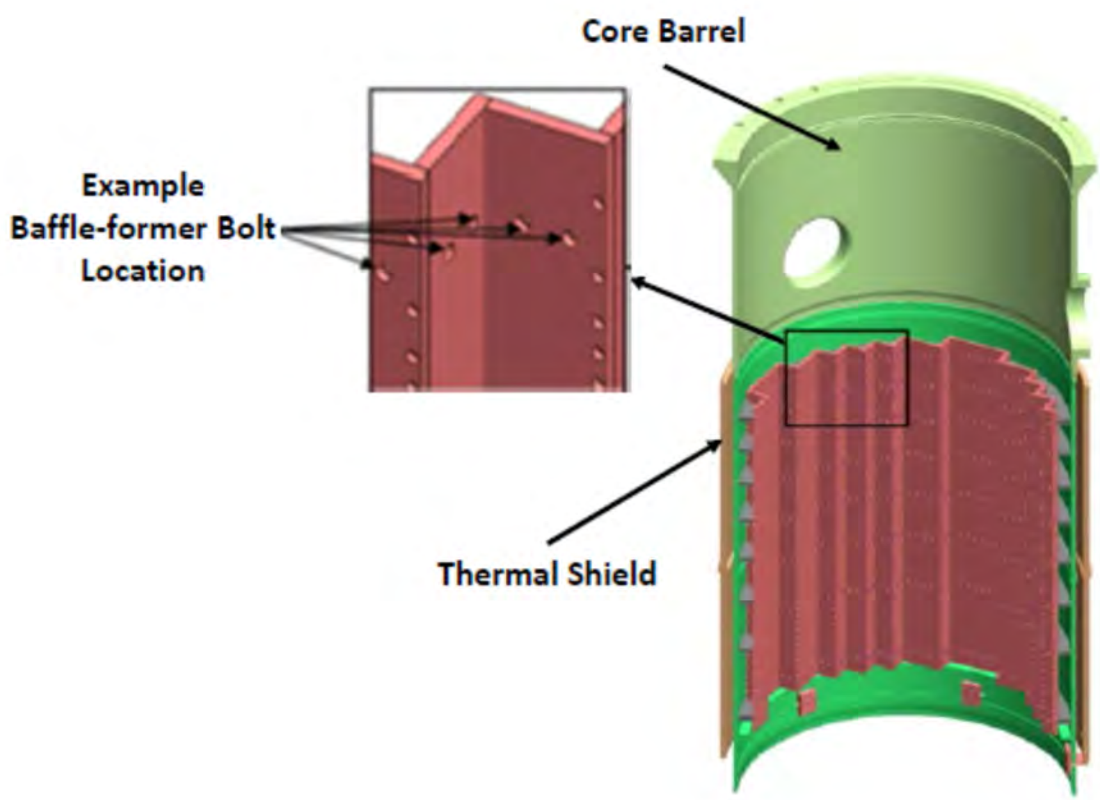

Figure 8 – Three-dimensional view of a portion of the baffle-former assembly in Westinghouse-design RVI, showing baffle-former bolt locations (reproduced from updated Westinghouse Figures Presented at Public Meeting on MRP-227, Rev. 1, March 31, 2015, ADAMS Accession No. ML15091A136, Exhibit NRC000211)

Q26. What are the functions of the lower internals assembly?

A26. [JP][AH] The major functions of the lower internals assembly are core support and positioning, direction of flow through the core, and shielding.

Q27. How does the lower internals assembly perform its function of core support?

A27. [JP][AH] The fuel assemblies are supported inside the lower internals assembly on top of the lower core plate (LCP). The functions of the LCP are to position and support the

core and provide a metered control of reactor coolant flow into each fuel assembly. As illustrated in Figure 5, the LCP is elevated above the lower support casting by support columns and bolted to a ring support attached to the inside diameter of the core barrel. The support columns transmit vertical fuel assembly loads from the LCP to the much thicker lower support casting. The function of the lower support casting is to provide support for the core. As shown in Figure 6, the lower support casting is welded to and supported by the core barrel, which transmits vertical loads to the vessel through the core barrel flange. A large number of components are attached to the core barrel, including the baffle/former assembly, the core barrel outlet nozzles, the thermal shields, the alignment pins that engage the UCP, the lower support casting, and the LCP, some of which are shown in Figure 1. The lower radial support/clevis assemblies restrain large transverse motions of the core barrel but at the same time allow unrestricted radial and axial thermal expansion.

Q28. How does the lower internals assembly perform its function to direct flow?

A28. [JP][AH] The baffle and former assembly consists of vertical plates called baffles and horizontal support plates called formers, as illustrated in Figures 7 and 8. The baffle plates are bolted to the formers by the baffle/former bolts, and the formers are attached to the core barrel inside diameter by the barrel/former bolts. Baffle plates are secured to each other at selected corners by edge bolts. In addition, at IP2 and IP3, corner brackets are installed behind and bolted to the baffle plates. The baffle/former assembly forms the interface between the core and the core barrel. The baffles provide a barrier between the core and the former region so that a high concentration of flow in the core region can be maintained. A secondary benefit is to reduce the neutron flux on the vessel.

Q29. How does the lower internals assembly perform its function of neutron shielding?

A29. [JP][AH] All the RVI components located radially outboard from the core provide some shielding to the reactor pressure vessel (RPV) by placing a metallic material between the core and the RPV. Additional neutron shielding of the reactor vessel is provided in the active core region by thermal shields attached to the outside of the core barrel.

MATERIALS OF CONSTRUCTION FOR REACTOR VESSEL INTERNALS

Q30. What are the primary materials of construction of the RVI components covered by MRP-227-A, for Westinghouse-design RVI such as those of IP2 and IP3?

A30. [JP][AH] The RVI components are constructed mostly of stainless steel. Most of the RVI are fabricated from Type 304 stainless steel, and weld metal of similar composition known as Type 308 stainless steel. Replacement split pins in IP3 are made from Type 316 stainless steel.

A few RVI components were constructed of cast austenitic stainless steel (CASS), such as the lower support columns (sometimes interchangeably referred to as the "lower core support columns"). A few RVI components are constructed from a nickel-based alloy known as Alloy X-750, such as the guide tube support pins (split pins) in the IP2 RVI. These materials are generally susceptible to the same aging mechanisms and effects as the stainless steel that makes up the vast majority of the RVI. In addition, the CASS material is susceptible to loss of fracture toughness due to thermal aging embrittlement, a mechanism which does not apply to the other stainless steel components used in the RVI.

Q31. In your previous answer, you identified several different types of stainless steel. Please explain the types of stainless steel and the differences between each identified type.

A31. [JP][AH] The majority of the RVI components are constructed of Type 304 stainless steel. Type 304 is categorized as an austenitic stainless steel. Stainless steels, in general, are alloys, or mixtures of several metallic elements including iron, chromium, nickel, and molybdenum. Type 304 stainless steel contains roughly 18 weight percent chromium, 8 weight percent nickel, and 72 weight percent iron, and one-half or less percent molybdenum and other metals. Type 308 and Type 316 are similar to Type 304 stainless steel with different weight percentages of chromium, nickel, and other metals.

Each of these identified types is an austenitic stainless steel. Austenitic stainless steels describe alloys of steel with a face-centered crystal structure. They possess high ductility, good fracture toughness (a parameter related to the amount of energy required to break or fracture the material), and high corrosion resistance. The chromium and nickel elements in the material allow it to form a thin, protective oxide film that acts as a barrier to additional corrosion. Additionally, austenitic stainless steels do not experience a ductile-to-brittle transition, wherein the fracture behavior of the transitions from brittle fracture at low temperature to a ductile fracture at higher temperatures. Stainless steels exhibit ductile behavior at all temperatures.

Q32. Why was Type 304 stainless steel chosen to construct the majority of RVI components?

A32. [JP][AH] Type 304 stainless steel was chosen because of its high corrosion resistance, ductility and toughness. In a PWR, all the wetted components in the reactor coolant system are fabricated from corrosion resistant materials such as stainless steel or nickel-based alloys, or are clad on the wetted surfaces with a corrosion-resistant material. For example, the

low-alloy steel RPV is clad on its interior surfaces with a thin layer of stainless steel. Use of corrosion resistant materials, in combination with strict water chemistry control, provides superior performance against general corrosion of the components.

Q33. How do austenitic stainless steels differ from low-alloy and carbon steels with respect to brittle fracture?

A33. [JP][AH] Unlike carbon and low-alloy steels, austenitic stainless steels do not experience a ductile-to-brittle transition. A ductile-to-brittle transition refers to a change in the fracture behavior of a material as a function of temperature, wherein the fracture behavior of the material "transitions" from brittle fracture at low temperatures to a ductile fracture at higher temperatures. A transition temperature is often defined in terms of the temperature corresponding to a specified impact energy level, such as 30 ft-lbs. Figure 9 depicts impact energy versus temperature for a material with a ductile-to-brittle transition, as measured by a Charpy V-notch test. The Charpy V-notch test measures the energy required to break a small notched bar, and is commonly used as an indirect measure of fracture toughness.

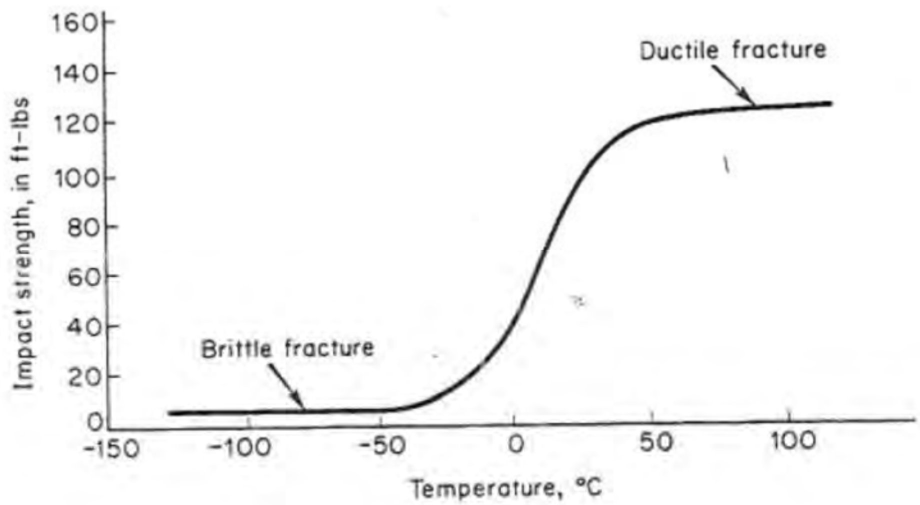

Figure 9 – Schematic of Charpy V-Notch Impact Test Results that Show a Ductile-to-Brittle Transition (from Physical Metallurgy Principles, Reed-Hill, Robert E., Abbaschian, R., 1992, PWS-KENT Publishing Company, Boston, MA (Exhibit NRC000212)

Q34. Why don't austenitic stainless steels experience a ductile-to-brittle transition?

A34. [JP][AH] Austenitic stainless steels do not experience a ductile-to-brittle transition because they have a different type of crystal structure from carbon and low-alloy steels, which exhibit a ductile-to-brittle transition. Austenitic stainless steels have a face-centered cubic crystal structure, while carbon and low-alloy steels (classified as ferritic steels), have a body-centered cubic structure. The difference in crystal structure results in differences in the way that the materials experience plastic (permanent) deformation. In body-centered cubic materials such as carbon steel, the way that the material undergoes plastic deformation is highly temperature dependent, while in face-centered cubic materials such as austenitic stainless steel, it is not. Therefore, face-centered cubic alloys will bend or stretch rather than break regardless of the temperature, while body-centered cubic materials will bend or stretch at higher temperatures, but break or fracture at lower temperatures without significant plastic deformation.

TYPICAL AGING EFFECTS EXPERIENCED BY REACTOR VESSEL INTERNALS

Q35. What are the types of aging mechanisms that impact RVI components?

A35. [AH][JP] The RVI components can experience (1) stress corrosion cracking, (2) irradiation-assisted stress corrosion cracking, (3) fatigue, (4) wear, (5) thermal embrittlement, (6) irradiation embrittlement, (7) void swelling, (8) irradiation-induced creep, and (9) irradiation-enhanced stress relaxation. Some components experience one or more these aging mechanisms.

Q36. What is stress corrosion cracking?

A36. [AH][JP] Stress Corrosion Cracking (SCC) is localized cracking of a material due to a combination of tensile stress, environment, and metallurgical properties. The actual mechanism that causes SCC involves a complex interaction of environmental and metallurgical factors. Cracks develop in a susceptible material under the combined effects of a chemical environment and a tensile stress that is below the yield strength of the material. Components in compression do not experience SCC.

Q37. What is irradiation-assisted stress corrosion cracking?

A37. [JP][AH] Irradiation-assisted stress corrosion cracking (IASCC) is a form of SCC that occurs only in highly-irradiated components. IASCC operates similarly to SCC but cracks may initiate and grow more quickly once initiated because of radiation exposure.

Q38. What is fatigue?

A38. [JP][AH] Fatigue is defined as the structural deterioration that can occur as the result of repeated stress/strain cycles caused by fluctuating loads and temperatures. After repeated cyclic loading of sufficient magnitude, microstructural damage can accumulate, leading to macroscopic crack initiation at the most highly affected locations. Subsequent mechanical or thermal cyclic loading can lead to growth of the initiated crack. (MRP-227-A at 3-13, Exhibit NRC000114A)

Q39. What is wear?

A39. [JP][AH] Wear refers to loss of material caused by the relative motion between adjacent surfaces, with the extent determined by the relative properties of the adjacent materials and their surface condition.

Q40. What is thermal aging embrittlement?

A40. [JP][AH] Thermal aging embrittlement (TE) is a loss of fracture toughness, increase in tensile strength and reduction in ductility (the ability to stretch without breaking) of susceptible materials that are exposed to sustained high temperatures. In RVI components, cast austenitic stainless steel (CASS) and precipitation-hardenable (PH) stainless steel are susceptible to thermal embrittlement when exposed to normal operating temperatures. IP2 and IP3 do not have any components made from PH stainless steel.

Q41. How does thermal aging embrittlement occur?

A41. [JP][AH] CASS material is a mix of two distinct crystal structures, or phases. The austenite phase has the same structure as non-cast stainless steel, while the ferrite phase has

a different structure. Metallurgical changes that occur in the ferrite phase cause the embrittlement when exposed to sufficiently high temperature. Experimental work has correlated the loss of fracture toughness with the percent ferrite content in the CASS. (NUREG/CR-4513, Ex. NYS000332)

Q42. What is irradiation embrittlement?

A42. [JP][AH] Irradiation embrittlement (IE) is also referred to as neutron embrittlement. When exposed to high energy neutrons, the mechanical properties of stainless steel and nickel-base alloys can change. Such changes in mechanical properties include increasing yield strength, increasing ultimate strength, decreasing ductility, and a loss of fracture toughness. The irradiation embrittlement aging mechanism is a function of temperature, neutron fluence, and material parameters, such as the chemical composition and thermal processing history of the material.

Q43. What is void swelling?

A43. [JP][AH] Void swelling is defined as a gradual increase in the volume of a component caused by formation of microscopic cavities in the material during high temperature exposure in a neutron environment. These cavities result from the nucleation and growth of clusters of irradiation-produced vacancies in the material microstructure. Helium produced by nuclear transmutations can have a significant impact on the rate of nucleation and growth of cavities in the material. At very high exposure levels, void swelling may produce dimensional changes that exceed the tolerances on a component. Differences in the amount of dimensional change from one part of a component to another may produce significant stresses, and may result in reduced clearances in adjacent parts that are required to move relative to one another.

Q44. What is irradiation-induced creep?

A44. [JP][AH] Creep (or more precisely, secondary creep) is a slow, time and temperature dependent, permanent deformation of materials that can occur when the materials are subjected to stress levels below the yield strength (elastic limit). Irradiation creep occurs when neutron irradiation enhances the creep process.

Q45. What is irradiation-induced stress relaxation?

A45. [JP][AH] Irradiation-induced stress relaxation (ISR) is a gradual loss of preload over time that generally affects threaded fasteners exposed to neutron irradiation. Threaded fasteners are typically tightened, or "preloaded," such that the fastener itself is in tension while the components joined by the fastener are in compression. This keeps the joint tight and prevents loosening due to vibration. Exposure to neutron irradiation can cause a loss of preload over time.

Q46. Are the three types of stainless steel identified as forming the majority of RVI components susceptible to each of the aging mechanisms identified in response to Q30, above?

A46. [JP][AH] No. SCC of the stainless steels used in the RVI is effectively prevented by the water chemistry controls for PWRs, which limit the concentrations of chlorides, fluorides, and dissolved oxygen to very low levels. However, highly stressed components such as those with a structural weld creating weld residual stresses or preloaded bolts may be susceptible to SCC in PWR water chemistry.

Q47. Does every RVI component experience each of the above discussed aging mechanisms?

A47. [JP][AH] No, every RVI component does not experience each of the above discussed aging mechanisms.

Q48. In your previous response, you discuss aging mechanisms. How do aging mechanisms differ from aging effects?

A48. [JP][AH] Aging mechanisms are the fundamental process which causes the degradation. Aging effects are the physical manifestations of the degradation caused by the aging mechanisms. For example, the aging effect of cracking can be caused by fatigue, stress corrosions cracking, irradiation-assisted stress corrosion cracking, etc. Similarly, the aging effect of loss of fracture toughness can be caused by thermal embrittlement or irradiation embrittlement.

Q49. Can you give an example of a component with multiple aging effects covered by the RVI AMP?

A49. [JP][AH] Yes. Per LRA Tables 3.1.2-2-IP2 and 3.1.2.-2-IP3, which contain a line item for each aging effect for each IP2 and IP3 RVI component that is in scope of license renewal (NL-10-063 at 18, Exhibit NYS000313) , baffle-former bolts were determined to be susceptible to the following aging effects cracking (due to IASCC and fatigue), change in dimensions (due to void swelling), loss of preload (due to irradiation stress relaxation), and loss of material. These aging effects are managed by the RVI program, except for loss of material, which is managed by the Water Chemistry Control, Primary and Secondary Program (NL-10-063 at 18, Exhibit NYS000313).

For a full list of aging effects that are applicable to IP2 and IP3 RVI components, LRA Tables 3.1.2-2-IP2 and 3.1.2.-2-IP3 contain a line item for each aging effect for each IP2 and IP3 RVI component that is in scope of license renewal (NL-10-063 at 18-81, Exhibit NYS000313).

Q50. For components that experience more than one aging mechanism, how do these aging mechanisms interact with each other?

A50. [JP][AH] Combinations of aging mechanisms can be either beneficial, deleterious, or have no interaction. Beneficial interactions would be cases in which the effects from one mechanism serve to reduce the severity of the effects from another mechanism. Deleterious interactions would be cases in which the effects from one mechanism serve to exacerbate the severity of the effects from another mechanism. For example, void swelling of two plates joined by a bolt could increase the stress in the bolt, which could in turn make IASCC of the bolt more likely. However, irradiation-assisted stress relaxation could also reduce the stress level in the bolt, thus making IASCC less likely.

When a component is susceptible to more than one aging mechanism that causes the same aging effect, both mechanisms could contribute to the effect. For example, in a component susceptible to both fatigue and IASCC, both mechanisms could contribute to the growth of an existing crack.

Discussion of how potential interactions between aging effects were evaluated in the development of MRP-227-A is addressed by questions Q80-Q133, Q188, and Q198-Q199.

Q51. What are synergistic effects?

A51. [JP][AH] Two aging mechanisms are considered to be synergistic when they interact to cause aging that is more severe than would be predicted based on the simple

additive (or in some cases multiplicative) effect of each mechanism. For example, if neutron embrittlement alone causes a fracture toughness reduction of 20 percent and the thermal embrittlement alone causes a fracture toughness reduction of 25 percent, then a synergistic effect of these two mechanisms hypothetically would result in a reduction of fracture toughness of more than 40% (e.g. 1-(1-.2)(1-.25) =0.4) due to the combination of both thermal and irradiation embrittlement.

THE STAFF'S GUIDANCE FOR ACCEPTABLE AGING MANAGEMENT PROGRAMS

Q52. What is an Aging Management Program (AMP)?

A52. [JP][AH] An AMP is a program that is credited for managing aging effects of systems, structures and components that are within the scope of license renewal and subject to aging management review in accordance with 10 CFR 54.21(a)(1), such that the components' intended functions will be maintained as required by 10 CFR 54.21(a)(3).

Q53. Do the regulations impose specific requirements on an AMP?

A53. [JP][AH] Although 10 C.F.R § 54.21 provides an overall framework that the license renewal application must use to demonstrate the adequacy of planned activities to manage the effects of aging on intended functions, it does not provide specific requirements for AMPs or specific criteria to evaluate AMPs.

Q54. Has the Staff provided any guidance on how AMPs can meet the regulatory requirements for license renewal?

A54. [JP][AH] The Staff has published several documents and revisions to documents that provide guidance on acceptable ways to meet the regulatory requirements, including

NUREG-1801, the Generic Aging Lessons-Learned (GALL) Report, and NUREG-1800, the Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants (SRP-LR).

Q55. What is the purpose of the GALL Report?

A55. [JP][AH] The GALL Report, NUREG-1801, provides, in part, aging management programs (AMPs) that the Staff has determined are acceptable for managing certain aging effects. For example, GALL Report, XI.M2, “Water Chemistry,” describes an effective program for water chemistry control. The GALL Report also serves as a technical basis for the SRP-LR.

The GALL Report was initially published in 2000, and has been periodically revised based on increased knowledge on aging effects and effective aging management approaches that have come from operating experience, and the results of Staff reviews of subsequent license renewal applications. Revision 1 of the GALL Report, NUREG-1801 was issued in 2005 (Accession Nos. ML052110005 and ML052110006) (Ex. NYS00146A-C) (together, “GALL Report Rev. 1”). Revision 2 of the GALL Report, NUREG-1801 was issued in 2010. (Exhibit NYS000147A-D)

Q56. How is the GALL Report used in the review of license renewal applications?

A56. [JP][AH] Both Rev. 1 and Rev. 2 of the SRP-LR indicate that an applicant may reference the GALL Report in a license renewal application to demonstrate that the programs at the applicant’s facility correspond to those reviewed and approved in the GALL Report, and in such a case no further Staff review is required. (SRP-LR, Rev. 1 Exhibit NYS000195, at 2, SRP-LR, Rev. 2, NYS000161, at 2). The GALL Report contains an acceptable way to manage aging effects for license renewal, however, applicants are not required to use the AMPs in the GALL Report. The GALL Report further states that an applicant may propose alternatives for Staff

review in its license renewal application, but use of the GALL Report should facilitate both preparation of a license renewal application by an applicant and timely, uniform review by the NRC Staff. See GALL Report, Vol.1, Rev. 1 at 4 (Exhibit NYS000146A) and GALL Report Rev. 2 at 8. (Exhibit NYS000147A, ADAMS Accession No. ML11349A083)

Q57. What is the purpose of the SRP-LR?

A57. [JP][AH] The SRP-LR, Rev. 1 and Rev. 2, provides guidance to the Staff on how to perform safety reviews of license renewal for nuclear power plant licenses in accordance with 10 C.F.R. Part 54. The principal purposes of the SRP-LR are to ensure the quality and uniformity of the Staff's review, and to present a well-defined base from which the Staff can evaluate the applicant's programs and activities for the period of extended operation. See SRP-LR Rev. 1 and Rev. 2 at iii (Ex. NYS000195 and Ex. NYS000161, respectively).

The SRP-LR was initially published in 2000, and has been periodically revised based on increased knowledge on aging effects and effective aging management approaches from operating experience, and the results of Staff reviews of subsequent license renewal applications. Revision 1 of the SRP-LR, NUREG-1800, Rev. 1, was issued in 2005. Revision 2 of the SRP-LR, NUREG-1800, Rev. 2 was issued in 2010.

Q58. How is the SRP-LR used in the review of AMPs for license renewal applications?

A58. [JP][AH] For an AMP that the applicant claims is "consistent with GALL," the license renewal application provides limited information on the AMP. The SRP-LR notes that the Staff conducts an audit and review at the applicant's facility to evaluate AMPs that the applicant claims to be consistent with the GALL Report. See SRP-LR Rev. 1 and Rev. 2 at 3.0-1 (Ex. NYS000195 and Ex. NYS000161, respectively).

Q59. Has the Staff provided any guidance on evaluating an AMP that has not been specifically addressed in the GALL report or other guidance?

A59. [JP][AH] NRC generic guidance for the contents of AMPs is contained in the Standard Review Plan for Review of License Renewal Applications for Nuclear Plants (SRP-LR), NUREG-1800. Appendix A.1 of the SRP-LR, Revision 2, dated December 2010, contains guidance for plant-specific programs in Branch Technical Position RLSB-1. Appendix A.1 describes AMPs in terms of ten program elements: (1) Scope of the program, (2) Preventive Actions, (3) Parameters Monitored or Inspected, (4) Detection of Aging Effects, (5) Monitoring and Trending, (6) Acceptance Criteria, (7) Corrective Actions, (8) Confirmation process, (9) Administrative Controls, and (10) Operating Experience. GALL Report, Rev. 2, at 6, SRP-LR, Rev. 2 at A.1-3. Guidance is also provided for the information that should be provided for each of these ten elements. EX. NYS000161, SRP-LR, Rev. 2 at A.1-3 to A.1-8].

Q60. Does the NRC provide guidance for acceptable generic AMPs?

A60. [JP][AH] NRC guidance in Chapter XI of the Generic Aging Lessons Learned (GALL) Report, NUREG-1801, provides examples of generic AMPs that have been found to be acceptable for managing aging effects for certain components, or certain aging effects for a number of components. Examples of generic AMPs include XI.M1 ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD; XI.M2 Water Chemistry; and, XI.M31 Reactor Vessel Surveillance.

Q61. How does an applicant identify in their license renewal application that their AMP is the same as an AMP provided in the GALL Report?

A61. [JP][AH] In such a situation, the license renewal application would identify that the AMP is “consistent with GALL” and identify the analogue GALL Report AMP.

Q62. Can an applicant propose an AMP that is similar to but not quite "consistent with GALL"?

A62. [JP][AH] Yes, applications can identify an AMP as "consistent with GALL" and identify enhancements or exceptions between the GALL AMP and the plant AMP. Enhancements and exceptions are generally small deviations from the GALL Report AMP.

Q63. What is an enhancement to an AMP in the GALL Report?

A63. [JP][AH] In some cases, an applicant may choose an existing plant program that does not currently meet all the program elements defined in the GALL Report AMP. If this is the situation, the applicant makes a commitment to augment the existing program to satisfy the GALL Report AMP elements prior to the period of extended operation. This commitment is an AMP "enhancement." An enhancement is a revision or addition to an existing aging management program that the applicant commits to implement prior to the period of extended operation. Enhancements include, but are not limited to, those activities needed to ensure consistency with the GALL Report recommendations. Enhancements may expand, but not reduce, the scope of an AMP. SRP-LR, Rev. 2 at 3.0-3.

Q64. What is an exception to an AMP in the GALL Report?

A64. [JP][AH] Exception are portions of the GALL Report AMP that the applicant does not intend to implement. All exceptions to the GALL Report AMP should be described and justified in the license renewal application. SRP-LR, Rev. 2 at 3.0-3.

Q65. Can applicants propose AMPs that are not described in the GALL Report?

A65. [JP][AH] Yes. Besides using the generic AMPs found to be acceptable in the GALL Report, applicants may also propose a plant-specific AMP. Often an applicant will propose a plant-specific existing AMP because they have an existing program that will adequately manage the aging effects for the in-scope components. In other cases, applicants may propose plant specific programs where either no generic program has been previously evaluated or the limitations on the generic program exclude its use for the specific plant. In any case, plant-specific AMPs are normally specified using the ten elements described in Appendix A.1 of the Standard Review Plan for Review of License Renewal Applications for Nuclear Plants (SRP-LR), NUREG-1800 as discussed previously.

Q66. How does the NRC Staff evaluate the adequacy of an applicant's AMP?

A66. [JP][AH] The Staff's evaluation of the adequacy of an applicant's AMP in managing certain aging effects for particular structures and components is based on its review of the ten program elements that are used to describe each aging management program. The evaluation process is different depending on whether the license renewal application identifies the AMP as consistent with the related AMP from the GALL Report or as a plant-specific AMP.

Q67. How does the NRC Staff evaluate the adequacy of a license renewal applicant's AMP that is identified as "consistent with GALL," including exceptions or enhancements?

A67. [JP][AH] If an applicant for license renewal credits a generic AMP from Chapter XI of the GALL Report, the Staff reviews the applicant's program to ensure all ten elements of the applicant's program are consistent with those of the generic program described in Chapter XI of the GALL Report, considering any exceptions and enhancements that the applicant may have

identified. Beyond those identified in the license renewal application as exceptions or enhancements, differences between the applicant's AMP and the GALL Report AMP are addressed through requests for additional information to ensure that the differences do not adversely affect the AMP's ability to manage the credited aging effects, as required by 10 CFR § 54.21(a)(3). The reviewer documents the acceptability of all exceptions identified in the LRA and all other differences that the Staff identifies.

Q68. How does the NRC Staff evaluate the adequacy of a plant-specific AMP?

A68. [JP][AH] The Staff reviews the applicant's plant-specific program for consistency with the criteria of Branch Technical Position RLSB-1 in the SRP-LR. The Staff reviews the adequacy of the program to meet the criteria of 10 CFR § 54.21(a)(3), such that the intended functions of the structures and components addressed by the AMP will be maintained.

Q69. Does the NRC have guidance for which AMP should be credited for managing a specific aging effect for a specific component?

A69. [JP][AH] Yes. The GALL Report contains tables for each plant system containing line items for each aging effect requiring management. Each line item describes the component, material, environment, aging effect and causal mechanism(s) (in a format of [aging effect] due to [aging mechanism]), recommended aging management program, and whether further evaluation is necessary.

Q70. Does an AMP have to prevent aging effects?

A70. [JP][AH] No. AMPs do not have to prevent or preclude aging effects in order to satisfactorily comply with the NRC regulations. 10 CFR § 54.21(a)(3) states that the license renewal application must demonstrate that the effects of aging will be adequately managed so

that the intended function(s) of the structure or component will be maintained consistent with the CLB for the period of extended operation.

Q71. If an AMP does not prevent aging, how does it protect the components that are subject to the program?

A71. [JP][AH] Many AMPs simply monitor the aging effects via inspections or tests in order to identify when components will no longer be able to fulfill their intended functions and corrective actions are necessary. The SRP-LR refers to this type of AMP as a condition monitoring program. Ex. NYS000161, SRP-LR, Rev. 2 at A.1-1.

Once a component is identified as no longer being able to fulfill its intended functions prior to the next scheduled inspection, the licensee is obligated to take corrective actions. Corrective actions could include repairing the component, replacing the component, or showing that the component will continue to fulfill its intended function despite the inspection results. In practice, AMPs generally have acceptance criteria for inspections that have additional margin so that corrective actions are initiated prior to the component approaching a condition in which it may not be able to fulfill its intended functions, so that repairs or replacement, if required, can be planned prior to loss of operability under the CLB. Schedules for inspections and tests are designed to allow the timely detection of aging effects prior to a component's intended functions being compromised. This process of condition monitoring and corrective actions is commonly instituted in most of the major programs for nuclear power plants and is the underlying criterion for quantifying intervals specified in most maintenance activities.

Q72. Is there guidance regarding which components should be included in an AMP?

A72. [JP][AH] Yes. For generic AMPs provided in the GALL Report, the scope statements for each AMP specify the components which are in the scope of the AMP. Section

A.1.2.3.1 of Appendix A to the SRP-LR, Rev. 2, states that the scope of the AMP should include the structures and components for which the AMP is credited for aging management.

Q73. What structures, systems, and components are within the scope of license renewal?

A73. [JP][AH] The scope of aging management for license renewal is specified by 10 CFR § 54.4, which identifies the plant systems, structures, and components that are in the scope of license renewal as:

> (1) Safety-related systems, structures, and components which are those relied upon to remain functional during and following design-basis events (as defined in 10 CFR 50.49(b)(1)) to ensure the following functions--
>
> (i) The integrity of the reactor coolant pressure boundary;
>
> (ii) The capability to shut down the reactor and maintain it in a safe shutdown condition; or
>
> (iii) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those referred to in § 50.34(a)(1), § 50.67(b)(2), or § 100.11 of this chapter, as applicable.
>
> (2) All nonsafety-related systems, structures, and components whose failure could prevent satisfactory accomplishment of any of the functions identified in paragraphs (a)(1)(i), (ii), or (iii) of this section.
>
> (3) All systems, structures, and components relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the Commission's regulations for fire protection (10 CFR 50.48), environmental qualification (10 CFR 50.49), pressurized thermal shock (10 CFR 50.61), anticipated transients without scram (10 CFR 50.62), and station blackout (10 CFR 50.63).
>
> (b) The intended functions that these systems, structures, and components must be shown to fulfill in § 54.21 are those functions that are the bases for including them within the scope of license renewal as specified in paragraphs (a)(1) - (3) of this section.

Q74. Your response didn't include long-lived and passive as criteria for in-scope structures and components - why is that?

A74. [JP][AH] Although the scope of license renewal per 10 CFR § 54.4 does not discuss long-lived and passive as criteria, these criteria are used to determine the structures and components that are subject to aging management review.

Q75. Are all structures and components that are in the scope of license renewal subject to aging management review?

A75. [JP][AH] No. 10 CFR § 54.21(a)(1) specifies that, for those systems, structures and components that are in scope of license renewal, aging management review is required for those structures and components that (i) perform a license renewal intended function without moving parts or without a change in configuration or properties, and (ii) are not subject to replacement based on a qualified life or specified time period. The process of determining which structures and components are subject to aging management review is called "screening."

Criterion (i) equates to including only those structures and components that are "passive," and criterion (ii) equates to those which are "long-lived." 10 CFR § 54.21 (a)(1)(i) states that "these structures and components include, but are not limited to, the reactor vessel, the reactor coolant system pressure boundary, steam generators, the pressurizer, piping, pump casings, valve bodies, the core shroud, component supports, pressure retaining boundaries, heat exchangers, ventilation ducts, the containment, the containment liner, electrical and mechanical penetrations, equipment hatches, seismic Category I structures, electrical cables and connections, cable trays, and electrical cabinets, excluding, but not limited to, pumps (except casing), valves (except body), motors, diesel generators, air compressors, snubbers, the control rod drive, ventilation dampers, pressure transmitters, pressure indicators, water level

indicators, switchgears, cooling fans, transistors, batteries, breakers, relays, switches, power inverters, circuit boards, battery chargers, and power supplies."

For the purpose of simplicity, the remainder of this testimony will use the colloquial phrase "the scope of license renewal" to include only those structures and components that meet the criteria of 10 CFR § 54.21(a)(1), in lieu of the more cumbersome "subject to aging management review."

STAFF GUIDANCE FOR THE REACTOR VESSEL INTERNALS AMP

Q76. Has the Staff provided any guidance to applicants regarding aging management for RVI?

A76. [JP][AH] Yes. Revision 2 of the GALL Report provided an AMP, GALL Report AMP XI.M16A, "PWR Vessel Internals," and aging management review line items for the RVI. Subsequently, the Staff issued License Renewal Interim Staff Guidance LR-ISG-2011-04, "Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors," on May 28, 2013 (LR-ISG). (Exhibit NRC000524, ADAMS Accession No. ML12270A251). The LR-ISG replaces the guidance for the RVI AMP and the aging management review items that were provided in Revision 2 of the GALL Report, including GALL Report AMP XI.M16A, "PWR Vessel Internals," and represents the current NRC guidance for aging management of PWR RVI.

Q77. Why was it necessary for the NRC to issue interim Staff guidance related to the RVI AMP?

A77. [JP][AH] Revision 2 of the GALL Report was issued in December 2010, after the industry had submitted a topical report, MRP-227, Rev. 0, for NRC Staff review and approval,

but before the issuance of the NRC safety evaluation (SE) on MRP-227 and issuance of the approved version, MRP-227-A. The GALL Report, Rev. 2, AMP XI.M16A, "PWR Vessel Internals," provided new AMR line items and aging management guidance, based on Staff expectations for the inspection protocol that would be provided in the approved version of MRP-227. With the issuance of MRP-227-A, the Staff updated the guidance of AMP XI.M16A through issuance of LR-ISG-2011-04 to ensure consistency of the AMP with MRP-227-A and to update the aging management review line items.

Q78. Please explain the Staff's guidance in LR-ISG-2011-04.

A78. [JP][AH] LR-ISG-2011-04 provides a revision to the AMP in GALL Report Chapter XI.M16A so that the descriptions of the ten program elements reference relevant aspects of MRP-227-A. In addition, LR-ISG-2011-04 provides an update of the aging management review line items that are relevant for PWR RVI.

Q79. What is MRP-227-A?

A79. [JP][AH] MRP-227-A is the NRC-approved version of the industry's topical report that describes inspection and evaluation guidelines for PWR RVI.

Q80. Provide an overview of how MRP-227-A manages aging of the RVI components.

A80. [JP][AH] MRP-227-A manages aging of RVI mainly by condition monitoring through regularly scheduled inspections of components that have been identified as leading indicators for potential aging effects. Components are inspected using a variety of inspection techniques including visual examination, ultrasonic examination, and physical measurements. MRP-227-A is a sampling-based program; in other words, every RVI component is not inspected. Rather, components are selected for inspection based on a combination of

susceptibility to aging effects and the risk of core damage if aging effects were to occur in a particular component. MRP-227-A sorted all RVI components into four different categories that determine if and how the components are subject to inspections under the program. These categories are Primary, Expansion, Existing Programs, and No Additional Measures components. Augmented inspection recommendations are identified for each Primary and Expansion category component. Augmented inspections are any inspections under the AMP that are different or in addition to what would be required by the existing ASME Code, Section XI inspection requirements. An example of an augmented inspection is the baseline volumetric examination using ultrasonic testing (UT) of the baffle-former bolts between 25 and 35 EFPY, with a subsequent examination on a ten-year interval, as specified in Table 4-3 of MRP-227-A (MRP-227-A at 4-27, Exhibit NRC000114B). The only inspection required by the ASME Code, Section XI for these bolts is a visual VT-3 examination once each ten-year interval. The recommendations for the Primary components also identify appropriate intervals for the inspection and a variety of different inspection techniques. Inspection techniques include ultrasonic (UT) testing, EVT-1 enhanced visual examination, and VT-3 visual examination.

Q81. What is a topical report?

A81. [JP][AH] A topical report is a stand-alone report containing technical information about a nuclear power plant safety topic. A topical report improves the efficiency of the licensing process by allowing the NRC Staff to review a proposed methodology, design, operational requirements, or other safety-related subjects that will be used by multiple licensees, following approval, by referencing the approved topical report. The topical report provides an acceptable technical basis for a licensing action. If approved by the NRC, the topical report can be referenced in applications for licensing actions, such as license amendment requests or license renewal applications, by multiple licensees. Topical reports

therefore increase the efficiency of the licensing process by eliminating the need to perform a detailed review of the same topic in multiple similar licensing actions.

Q82. What process does the Staff use to review and, if possible, approve topical reports like MRP-227, Rev. 0?

A82. [JP][AH] Generic programs like MRP-227, Rev. 0, are reviewed through the topical report process.

Q83. Can you explain the Staff's review criteria for approving MRP-227, Rev. 0?

A83. [JP][AH] The NRC Staff reviewed MRP-227, Rev. 0, to determine whether its guidance would provide reasonable assurance that the proposed aging management of the subject RVI components would ensure that the RVI components maintain their intended functions during the period of extended operation. The review also considered compliance with license renewal requirements in 10 CFR § 54.21(a)(3) in order to allow licensees or applicants the option of adopting the aging management methodology described in MRP-227, Rev. 0, as the basis for managing age-related degradation in RVI components and incorporating, by reference, the recommended guidelines into their PWR Vessel Internals AMPs (or their equivalents).

Q84. Can you describe the timeline and milestones that led to approval of MRP-227?

A84. [JP][AH] By letter dated January 12, 2009, (Exhibit NRC000508 ADAMS Accession No. ML12017A193)), the Electric Power Research Institute (EPRI) submitted for NRC Staff review and approval Materials Reliability Program (MRP) Report 1016596, Revision 0, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines" (MRP-227). The report was reviewed by a team of NRC Staff experts on reactor vessel internals

integrity issues. In addition to submitting MRP-227, Rev. 0, EPRI provided several supporting technical reports that provided information on the development process for the aging management recommendations in MRP-227, Rev. 0. These reports covered topics such as screening criteria; screening results; failure modes, effects, and criticality analysis (FMECA); and functionality analyses that used techniques such as finite element analysis (FEA). The Staff issued, and EPRI answered, four sets of requests for additional information (RAIs). The Staff also held several public meetings with EPRI to discuss issues related to the review. The Staff issued its initial safety evaluation of MRP-227, Rev. 0, on June 22, 2011. Revision 1 to the Staff's SE was issued on December 16, 2011.

Q85. Did the Staff identify any plant-specific items for resolution related to MRP-227, Rev. 0?

A85. [JP][AH] Based on its review, the NRC Staff identified eight action items that must be addressed by applicants or licensees (Applicant/Licensee Action Items, A/LAIs) in order to apply the methodology of the topical report to aging management of the RVI at a particular plant. These plant-specific action items address topics related to the implementation of MRP-227, Rev. 0, that could not be effectively addressed on a generic basis in MRP-227, Rev. 0, and are identified in Section 4.2 of Revision 1 of the Staff's final SE approving MRP-227, Rev. 0. (Exhibit ENT000230 at 32-34, ADAMS Accession No. ML12089A268)

Q86. Is MRP-227-A an acceptable basis for a generic AMP?

A86. [JP][AH] Yes. Because MRP-227-A is an approved topical report, it provides an acceptable basis for a generic AMP for PWR RVI. Revision 1 of the SE for MRP-227, Rev. 0, concludes that MRP-227, as modified by the conditions and limitations and applicant/licensee action items (A/LAI), provides for the development of an AMP for PWR RVI components within

the scope of MRP-227 that will adequately manage their aging effects such that there is reasonable assurance that they will perform their intended functions in accordance with the CLB during the extended period of operation. See Revision1 SE at 35. The NRC's most current guidance for aging management of PWR RVI is provided in LR-ISG-2011-004 and is based on MRP-227-A, as modified by the action items provided in Revision 1 of the NRC SE.

Q87. What were the steps used to develop the aging management recommendations of MRP-227-A?

A87. [JP][AH] The development of the aging management recommendations in MRP-227-A followed the following steps:

1. Determination of the list of generic components for each vendor RVI design.
2. Determination of the operating conditions for each generic component
3. Screening to identify the aging effects that apply to each component
4. Failure Modes, Effects and Consequences Analysis
5. Functionality Analysis
6. Final Aging Management Strategy determination

Q88. How was the list of generic components in MRP-227-A developed for Westinghouse-design RVI (such as IP2 and IP3)?

A88. [JP][AH] The list of generic components was developed based on a review of plant design drawings and interviews with the designers and analysts. A common set of RVI components was defined for the Westinghouse fleet, with the understanding that there are many design variations within the fleet. As described later, plant-specific variations in the components are addressed in an A/LAI.

Q89. How were the operating conditions for the generic components determined?

A89. [JP][AH] For neutron fluence and heat generation (used to determine internal metal temperatures), pre-existing analyses for representative Westinghouse plants were used as a reference to estimate ranges of these parameters for the fleet. For stress, interviews of designers and stress analysts were used to determine typical levels of normal operating stresses, which were used for analyses because the typically available stress analyses of RVI are based on design-basis accident conditions and not representative of normal operating stresses. However, normal operating stresses are needed for this analysis because they drive aging mechanisms such as SCC over the long term. The designers were also asked if a component had the potential to experience significant fatigue cycles, whether it could undergo wear, and whether it could contain a structural weld.

Q90. Describe the screening process used in the MRP-227-A development.

A90. [JP][AH] Components were screened for eight different aging mechanisms using pre-defined screening criteria. Compared to the mechanisms described in response to Q35 through Q45, the eight mechanisms used in the development of the MRP-227-A combine irradiation-induced creep and irradiation-enhanced stress relaxation. An example of a screening criterion is for stress corrosion cracking of austenitic stainless steel. This material is only considered potentially susceptible to stress corrosion cracking, that is "screened in," if the component stress is greater than or equal to 30-kilo-pounds per square inch (30 ksi), and the component has cold work in excess of 20%. If a component screened in for an aging mechanism, that means the component is potentially susceptible to the mechanism, but not that the mechanism will definitely occur or will become significant enough to prevent the component from performing its intended function.

Q91. Describe the "failure modes, effects and criticality analysis (FMECA)" process.

A91. [JP][AH] The FMECA process was performed by an industry expert panel. In the FMECA process, all possible aging effects and the resulting failure modes, for example fracture or distortion, are identified. The possible effects of a failure on the component, system, and surrounding components are determined. Components were initially placed in "FMECA significance groups" based on a combination of the failure likelihood and damage likelihood (probability of core damage if failure of a given component occurred). The components were ranked by the expert panel considering the extent to which failure might occur due to the degradation mechanism(s) identified, and the consequences of such degradation.

Q92. Describe the "functionality analysis" process.

A92. [JP][AH] A functionality analysis is a more refined assessment performed for certain components or assemblies with the intent to determine the tolerance of these components and assemblies to aging effects. The functionality analyses sometimes uses analytical techniques such as finite element analysis.

Q93. How was the final aging management strategy for each component developed?

A93. [JP][AH] After the screening, FMECA, and functionality analyses, the ranking of the components was adjusted and finalized. The end result was that all components were placed into one of four categories: (1) No Additional Measures, (2) Primary, (3) Expansion, and (4) Existing Programs. Those components that did not screen in for any aging mechanisms, or those for which both the likelihood of failure and core damage risk was low, were assigned to the No Additional Measures category. Other components for which one or more aging mechanisms screened in were generally categorized as either Primary or Expansion category components. When more than one component existed with similar materials, functions, and

screened-in aging mechanisms, the highest overall risk component generally became a Primary component, while the lower risk components became an Expansion component. Components determined to have high or moderate overall risk, but for which activities performed by other existing programs adequately manage the aging effect, were assigned to the Existing Programs category. Once assigned to one of the four categories, inspection techniques were identified as appropriate based on the component and aging mechanism.

Q94. What constitutes the Primary category of components?

A94. [JP][AH] Primary components are those RVI components that have the highest susceptibility to the effects of at least one of the eight aging mechanisms. MRP-227-A generally specifies inspections of Primary components or other aging management activities, such as analyses, with most inspection required within the first two refueling outages from the start of the period of extended operation (PEO). The Primary group also includes components which have shown a degree of tolerance to a specific aging degradation effect, but for which no component with a higher susceptibility exists or for which no component with a higher susceptibility is accessible. (MRP-227-A at 3-15, Exhibit NRC000114A) Inspection findings for a Primary component that exceeds the expansion criteria will result in scheduled inspections of the Expansion components that are linked to that Primary component.

Q95. How are Primary components inspected?

A95. [JP][AH] The Primary components are inspected using a variety of techniques, based on the applicable aging effect and the capabilities of the inspection method. Table 4-3 of MRP-227-A specifies the inspection methods for the Primary components for Westinghouse-design RVI. (pg 4-26 to 4-29 of MRP-227-A).

Q96. What constitutes the Expansion category of components?

A96. [JP][AH] Expansion components are those RVI components that are highly or moderately susceptible to the effects of at least one of the eight aging mechanisms, but for which functionality assessment has shown a degree of tolerance to those effects. The schedule for implementation of aging management inspections for Expansion components will only occur if the findings from the examinations of the Primary components exceed the expansion criteria. (MRP-227-A at 3-15, Exhibit NRC000114A) An example of an Expansion component is the barrel-former bolts. These bolts have a high degree of redundancy and receive lower neutron fluence than the baffle-former bolts, which are the linked primary components.

Q97. How will the Expansion components be inspected?

A97. [JP][AH] If inspection of the Expansion components is necessitated by the findings from the inspection of the Primary components, the Expansion components would be inspected using a variety of techniques, based on the applicable aging effect and the capabilities of the inspection method. Table 4-6 of MRP-227-A specifies the inspection methods for the Expansion components for Westinghouse-design RVI (Ex. NRC000114A-F at p. 4-37).

Q98. What is an Existing Programs Component?

A98. [JP][AH] Existing Program components are those PWR internals that are susceptible to the effects of at least one of the eight aging mechanisms, and existing generic or plant-specific aging management activities are adequate for managing those effects. (MRP-227-A at 3-15, Exhibit NRC000114A)

Q99. Please provide an example of an existing programs component and the corresponding existing program that is credited for managing aging.

A99. [JP][AH] The lower core plate is managed by the ASME Code Section XI requirements. (MRP-227-A at 4-74, Exhibit NRC000114B)

Q100. What constitutes the No Additional Measures category of components?

A100. [JP][AH] No Additional Measures components are those RVI components for which the effects of all eight aging mechanisms are below the screening criteria. Additional components were placed in the No Additional Measures group as a result of FMECA and the functionality assessment. No further action is required by these guidelines for managing the aging of the No Additional Measures components. (MRP-227-A at 3-15, Exhibit NRC000114A)

Q101. How were inspection techniques identified for each component?

A101. [JP][AH] Inspection techniques were identified based on the capability of the inspection technique to detect the expected aging effect(s). For example, periodic enhanced visual (EVT-1) examination is used to detect cracking in the lower core barrel cylinder girth welds because EVT-1 is capable of detecting cracking in that component.

Q102. What are the different types of criteria associated with the inspections findings under the RVI AMP?

A102. [JP][AH] The two types of criteria are acceptance criteria and expansion criteria. Acceptance criteria are used to assess the inspection findings from the Primary component inspections and determine whether corrective actions are required for the component that was inspected. Expansion criteria are used to assess the inspection findings from the Primary component inspections and determine if the Expansion components require inspection.

Q103. Does MRP-227-A define acceptance and expansion criteria for its recommended inspections?

A103. [JP][AH] Yes. Table 5-3 of MRP-227-A provides examination acceptance criteria and expansion criteria for each Primary and Expansion component for Westinghouse-design RVI such as IP2 and IP3. Similar tables are provided for the RVI for the other vendor designs.

Q104. Please provide an example of an acceptance criterion.

A104. [JP][AH] For the lower core barrel cylinder girth welds, the acceptance criteria from MRP-227-A Table 5-3 state that "The specific relevant condition is a detectable crack-like surface indication." In other words, Entergy would have to take appropriate corrective actions if any cracks are detected during the inspection regardless of their size, orientation, or location.

Q105. Please provide an example of an expansion criterion.

A105. [JP][AH] For the lower core barrel cylinder girth welds, the expansion criteria from MRP-227-A Table 5-3 states that "The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the lower core barrel cylinder girth welds shall require that the EVT-1 examination be expanded to include the lower core barrel cylinder axial welds by the completion of the next refueling outage." This puts an explicit limit on how long an applicant could wait before being required to conduct inspections of the applicable expansion components.

Q106. If the MRP-227-A acceptance criteria are exceeded, what happens?

A106. [JP][AH] If the acceptance criteria are exceeded, the results are entered into the plant's corrective action program, which is governed by the quality assurance requirements of

Appendix B to 10 CFR Part 50, for resolution. The licensee would have a number of available options or corrective actions to resolve the finding, including more detailed engineering evaluations in accordance with Section 6 of MRP-227-A, scheduling supplemental inspections, repair, replacement, or seeking a license amendment, when applicable. Part of the corrective actions could result in inspections of additional components even if the expansion criteria were not exceeded.

Q107. If the MRP-227-A expansion criteria are exceeded, what happens?

A107. [JP][AH] If the expansion criteria are exceeded, then MRP-227-A specifies that the inspection of the Expansion component associated with the Primary component must be completed within a specified time frame. For example, if a two-inch crack is found during the EVT-1 inspection of the lower core barrel girth weld, the lower core barrel cylinder axial welds must be inspected by the end of the next refueling outage.

Q108. How do the ASME Code Section XI requirements manage the applicable aging effects for the Existing Programs RVI components?

A108. [JP][AH] The ASME Code, Section XI specifies VT-3 visual examinations for those reactor internals classified as core support structures (Examination Category B-N-3). The ASME Code, Section XI AMP was credited as the existing program for managing specific aging effects only when visual VT-3 examination was judged to be capable of managing the applicable aging effect(s). These effects generally included wear, for which VT-3 is the best suited technique, and cracking when the component was judged to have some tolerance for cracking.

Q109. How are VT-3 visual examinations used under the ASME Code, Section XI?

A109. [JP][AH] Per the ASME Code, Section XI, VT-3 visual examinations are used to determine the general mechanical and structural conditions of components by detecting discontinuities and imperfections, such as loss of integrity at bolted and welded connections, loose or missing parts, debris, corrosion, wear, or erosion, and by identifying conditions that could affect operational or functional adequacy of components. (ASME Code, Section XI, 2007 through 2008 Addenda, IWA-2213 (Jul. 1, 2008), Exhibit NRC000199).

Q110. Could you explain why the RVI AMP does not need additional measures to monitor or manage aging effects in certain components?

A110. [JP][AH] The RVI AMP does not need additional measures to monitor or manage aging effects for certain components because either (1) All aging effects screened out for the components, or (2) The FMECA process determined the safety and economic consequences of failure of the component were sufficiently low that no management of the aging effect is necessary.

Q111. What was the NRC Staff's conclusion regarding the guidance in MRP-227, Rev. 0?

A111. [JP][AH] The Staff found in its final SE of MRP-227, Rev. 0 that MRP-227, as modified by the conditions and limitations and the applicant/licensee actions items summarized in the safety evaluation of MRP-227, provides for the development of an AMP for PWR RVI components within the scope of MRP-227 which will adequately manage their aging effects such that there is reasonable assurance that they will perform their intended functions in accordance with the current licensing basis during the period of extended operation. (Final SE of MRP-227-A, December 16, 2011, p. 35, Exhibit ENT000230).

Q112. Can you briefly explain how the approved version of MRP-227 works to manage aging effects for the RVI?

A112. [JP][AH] MRP-227-A is the NRC-approved version of the industry's topical report for inspection and evaluation of PWR internals. Aging management of the RVI is accomplished by MRP-227-A principally through inspections that focus on examination of the lead components of the RVI (referred to as Primary components in the AMP) for the various aging effects that are anticipated to occur in the RVI. If certain expansion criteria are exceeded, then additional linked Expansion components are inspected. Any inspection findings are resolved through the corrective action program that could result in repair, replacement, or analysis that supports the continued operation of plant with the as-found conditions.

Q113. Why were applicant/licensee action items (A/LAIs) required in the Staff's final SE of MRP-227, Rev. 0?

A113. [JP][AH] The A/LAIs are required to ensure that the aging management activities planned at each plant, based on MRP-227-A, are appropriate. The A/LAIs address topics related to the plant-specific implementation of MRP-227 that could not be effectively addressed on a generic basis in MRP-227.

<u>THE STAFF'S REVIEW OF ENTERGY'S PROPOSED RVI AMP</u>

Q114. Could you provide a brief overview of the IP2 and IP3 RVI AMP including any changes or modifications to the program?

A114. [JP][AH] The IP2 and IP3 RVI AMP consists of a program description, describing the ten elements of the AMP, and an inspection plan. The program description was initially submitted on July 14, 2010 (Exhibit NYS000313), and was revised in the letter dated February

17, 2012 (Exhibit NYS000496). The inspection plan contains tables specifying the inspections for the Primary, Expansion, and Existing Programs components, and tables contain the acceptance and expansion criteria for these components. The Inspection Plan also contains Entergy's proposed resolution of the applicant/licensee action items (A/LAIs) from the Staff's final safety evaluation of MRP-227, Rev. 0. The inspection plan was initially submitted on September 28, 2011 (Exhibit NYS000314), and a revised version consistent with MRP-227-A was submitted on February 17, 2012 (Exhibit NYS000496). Some modifications to Entergy's proposed responses to the action items were made in various responses to requests for additional information.

Q115. Did the Staff's review proceed normally and consistent with the process described in response to Q54-Q77?

A115. [JP][AH] Yes, the review of the IP2 and IP3 RVI AMP was consistent with the process outlined previously in response Q54-Q77. Although the licensee did not indicate that the RVI AMP was consistent with GALL and it was reviewed initially as a plant-specific AMP, the Staff did determine that Entergy's RVI was consistent with GALL AMP XI.M16A as modified by LR-ISG-2011-04.

Q116. Do the materials used in the manufacture of RVI components impact the applicable aging effects and AMP?

A116. [JP][AH] Yes, the materials used in the manufacture of RVI components do impact the applicable aging effects and the AMP.

Q117. Has Entergy proposed an AMP for RVI at Indian Point Unit 2 (“IP2”) and Unit 3 (“IP3”)?

A117. [AH] [JP] Yes. The applicant has proposed to use a common program for IP2 and IP3. Therefore, all discussion of program aspects applies equally to IP2 and IP3.

Q118. Does Entergy’s AMP meet the guidance that the Staff has provided to applicants in LR-ISG-2011-04?

A118. [AH][JP] Yes.

Q119. Please explain how Entergy’s AMP meets the Staff’s guidance?

A119. [AH][JP] Entergy’s AMP meets the Staff guidance because it is consistent with the ten program elements of Chapter XI.M16A, “PWR Internals,” as defined in LR-ISG-2011-04. These elements are (1) scope, (2) preventive actions, (3) detection of aging effects, (4) parameters monitored and inspected, (5) monitoring and trending, (6) acceptance criteria, (7) corrective actions, (8) confirmation process, (9) administrative actions, and (10) operating experience.

Q120. What is the Staff guidance for the scope of RVI AMPs?

A120. [JP][AH] The Staff’s guidance in LR-ISG-2011-04 recommends that the components to be inspected, inspection methods, and inspection schedules be based on MRP-227-A.

Q121. How does Entergy’s AMP, meet the Staff’s guidance for the scope of RVI AMP?

A121. [JP][AH] The AMP is based on MRP-227-A and implements all the Primary, Expansion, and Existing Programs inspections specified for Westinghouse-design RVI in MRP-

227-A. The components to be inspected, inspection methods, and inspection schedules are consistent with those recommended in MRP-227-A. As described in Supplement 2 of the Safety Evaluation Report related to the License Renewal of Indian Point Nuclear Generating Units Nos. 2 and 3 (NUREG-1930, Supp. 2) Ex. NYS000507, the scope of the IP2 and IP3 RVI AMP meets the criteria of LR-ISG-2011-04.

Q122. What is the Staff guidance for preventive actions to be included in RVI AMPs?

A122. [JP][AH] LR-ISG-2011-04 states that the MRP-227-A relies on PWR water chemistry control to prevent or mitigate aging effects that can be induced by corrosive aging mechanisms (e.g., loss of material induced by general corrosion, pitting corrosion, crevice corrosion, or stress corrosion cracking or any of its forms (SCC, PWSCC, or IASCC)). Section XI.M16A of LR-ISG-2011-04 further states that reactor coolant water chemistry is monitored and maintained in accordance with the Water Chemistry Program, as described in GALL AMP XI.M2, "Water Chemistry."

Q123. How does Entergy's AMP meet this guidance?

A123. [JP][AH] Entergy's AMP credits the Water Chemistry Control Program – Primary and Secondary for minimizing the potential for loss of material due to general corrosion, pitting corrosion, and crevice corrosion, and for minimizing the potential for SCC, IASCC, and PWSCC (SSER at 3-15). The Water Chemistry Program is important because it minimizes the concentrations of impurities such as chloride and fluoride ions and dissolved oxygen, thus minimizing the potential for aging mechanisms such as SCC. PWR water chemistry is largely responsible for the very low incidence of SCC in PWR RVI over the first forty years of operation. This is contrasted with the experience in BWRs, which have experienced SCC in welded stainless steel components. The major difference is that BWRs operated with higher dissolved

oxygen levels early in plant life, while PWR water chemistry has always minimized dissolved oxygen. The Staff also found Entergy's Water Chemistry Control Program – Primary and Secondary to be consistent with enhancement, with the criteria of the GALL (NUREG-1930 3-149 and SSER 3-16). Therefore, since Entergy's RVI AMP credited the Water Chemistry Control Program, for preventing the forms of corrosion and cracking listed above, Entergy's AMP meets the Staff's criteria for preventive actions.

Q124. Does Entergy identify any other preventive measures for the RVI AMP?

A124. [JP][AH] Entergy also implemented a low-leakage core design for IP2 and IP3 prior to thirty calendar years of operation, which reduces the potential for irradiation-driven aging mechanisms such as IASCC, IE, void swelling, and ISR.

Q125. What is the Staff guidance for parameters to be monitored or inspected for RVI AMPs?

A125. [JP][AH] LR-ISG-2011-04 Section XI.M16A recommends that for plants with RVI of Westinghouse design, the RVI Program should meet the parameters monitored/inspected criteria consistent with the applicable tables in Section 4, "Aging Management Requirements," of MRP-227-A, which Table 4-3, Westinghouse Plants Primary Components, Table 4-6, Westinghouse Plants Expansion Components, and Table 4-9, Westinghouse Plants Existing Programs Components. For example, MRP-227-A Table 4-3, for the generic component Core Barrel Assembly Upper and lower core barrel cylinder girth welds, specifies periodic enhanced visual (EVT-1) examination. The initial examination must be performed no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval. The required examination coverage is 100 percent of one side of the accessible surfaces of the selected weld and adjacent base metal.

Q126. How does Entergy's AMP meet this guidance?

A126. [JP][AH] Entergy's RVI AMP meets the LR-ISG guidance because the information in the corresponding tables in its RVI Inspection Plan (which implements the RVI AMP) including the applicability (e.g. to IP2 and/or IP3), aging effects to be looked for, linked components for Primary and Expansion components, and the associated examination techniques, examination frequencies, and required examination coverage is consistent with Tables 4-3, 4-6, and 4-9 of MRP-227-A. Ex. NYS000507, SSER at 3-17).

As an example, Table 5-2, "Primary Components for [Indian Point] Units 2 and 3," from NL-12-037, Indian Point Energy Center Reactor Vessel Internals Inspection Plan (Exhibit NY000496), contains line items for the Core Barrel Assembly - Upper and lower core barrel cylinder girth welds. This line item specifies periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval, and a required examination coverage of 100% of one side of the accessible surfaces of the selected weld and adjacent base metal. The required inspection method, schedule and coverage for this component in Entergy's RVI Inspection Plan are identical to those specified for the corresponding generic component in MRP-227-A. The Staff verified that this is true for all the applicable IP2 and IP3 components.

Q127. What is the Staff guidance for detection of aging effects for RVI AMPs?

A127. [JP][AH] With respect to the detection of aging effects, LR-ISG-2011-04, Chapter X1.M16A states, in part, that the inspection methods are defined and established in Section 4 of MRP-227-A and that standards for implementing the inspection methods are defined and established in MRP-228. LR-ISG-2011-04 also describes the inspection methods to be used for detection of various aging mechanisms, all of which are consistent with MRP-227-A.

Q128. How does Entergy's AMP meet this guidance?

A128. [JP][AH] For the Primary and Expansion components, Entergy's AMP meets the criteria of LR-ISG-2011-04 Section X1.M16A because the applicant's program will conduct inspections of these components in accordance with MRP-227-A Tables 4-3 and 4-6. For example, MRP-227-A Table 4-3, for the generic component Core Barrel Assembly Upper and lower core barrel cylinder girth welds, specifies periodic enhanced visual (EVT-1) examination as the inspection method. Table 5-2, "Primary Components for IPEC Units 2 and 3 of NL-12-037, Indian Point Energy Center Reactor Vessel Internals Inspection Plan (Exhibit NYS000507), contains a line items for the Core Barrel Assembly Upper and lower core barrel cylinder girth welds. This line item specifies the inspection method as periodic enhanced visual (EVT-1) examination. Therefore, the required inspection method in Entergy's RVI Inspection Plan is identical to that specified for the corresponding generic component in MRP-227-A. In its review of Entergy's RVI AMP and Inspection Plan, the Staff verified that the inspection methods specified in Tables 5-2, 5-3, and 5-4 of Entergy's RVI Inspection Plan are identical to those specified in Table 4-3, 4-6 and 4-9 of MRP-227-A. Further, because Entergy's RVI Program states that the NDE systems (i.e., the combinations of equipment, procedure, and personnel) used to detect these aging effects will be qualified in accordance with MRP-228, the Staff found this criterion of LR-ISG-2011-04 Section XI.M16A is met (SSER2 at 3-17 to 3-18, Exhibit NYS000507).

Q129. What is the Staff guidance for monitoring and trending for RVI AMPs?

A129. [JP][AH] LR-ISG-2011-04 Section XI.M16A recommends using the methods of MRP-227-A Section 6 for monitoring, recording, evaluating, and trending the data from the program inspection results. MRP-227-A Section 6 includes recommendations for flaw depth

sizing and for crack growth determinations as well as for performing applicable limit load, linear elastic, and elastic-plastic fracture analyses of relevant flaw indications. LR-ISG-2011-04 Section XI.M16A also states that "examination and re-examinations that are implemented in accordance with MRP-227-A, together with the criteria of MRP-228 for inspection methodologies, inspection procedures, and qualification of inspection personnel, provide timely detection, reporting, and implementation of corrective actions for the aging effects and mechanisms managed by the program."

Q130. How does Entergy's AMP meet this guidance?

A130. [JP][AH] Section 7 of Entergy's AMP, "Corrective Actions," states that any detected condition that fails to meet the examination acceptance criteria must be processed through the corrective action program. Section 7 further states that (1) example methods for analytical disposition of unacceptable conditions are discussed or cited in Section 6 of MRP-227-A and (2) the evaluation methods include recommendations for flaw depth sizing and for crack growth determinations as well for performing applicable limit load, linear elastic, and elastic-plastic fracture analyses of relevant flaw indications. Section 7 further states that these methods or other NRC-approved evaluation methods may be used. The applicant stated in the "Detection of Aging Effects" section of the RVI Program that "[t]he NDE [non-destructive examination] system (i.e., the combinations of equipment, procedure, and personnel) used to detect [the relevant] aging effects will be qualified in accordance with MRP-228." Therefore, the Staff finds that this criterion of LR-ISG-2011-04 Section XI.M16A is met. LR-ISG-2011-04 also states that the program applies applicable fracture toughness properties, including reductions for thermal aging or neutron embrittlement, to the flaw evaluations of the components in cases in which cracking is detected in a RVI component and is extensive enough to warrant a supplemental flaw-growth or flaw-tolerance evaluation. The Staff notes that MRP-227-A Section

6 provides guidance on the fracture toughness properties to be used. Therefore, the Staff finds that the applicant's AMP meets this criterion because it will follow the recommendations of MRP-227-A Section 6.

LR-ISG-2011-04 also states that (1) "[f]or singly represented components, the program includes criteria to evaluate the aging effects in the inaccessible portions of the components and the resulting impact(s) on the intended function(s) of the components," and (2) "[f]or redundant components (such as redundant bolts, screws, pins, keys, or fasteners, some of which are accessible to inspection and some of which are not accessible to inspection), the program includes criteria to evaluate the aging effects in the population of components that are inaccessible to the applicable inspection technique and the resulting impact on the intended function(s) of the assembly containing the components.

This recommendation is implemented in Primary and Expansion category component inspections specified in Tables 5-2 and 5-3 of the RVI Inspection Plan because the applicant's AMP will meets the required inspection coverage from the tables in Section 4.0 of MRP-227-A, which contains criteria for the overall percentage of the total component or redundant population that must be inspected (75%) to claim credit for completing the inspection. Therefore, the Staff found that this recommendation of LR-ISG-2011-04 is met. Based on its review of the LRA, the Staff found the applicant's description of the "monitoring and trending" program element was acceptable because the AMP is consistent with the criteria of LR-ISG-2011-04 Section XI.M16A for this element (SSER2 at pp. 3-18-3-19., Exhibit NYS000507)

Q131. What is the Staff guidance for acceptance criteria for RVI AMPs?

A131. [JP][AH] The Staff's guidance for acceptance criteria is in LR-ISG-2011-04 Section 6 of XI.M16A, which recommends that Section 5 of MRP-227-A (specifically Table 5-3 for Westinghouse-designed RVI) provides the specific examination and flaw evaluation

acceptance criteria for the Primary and Expansion Component examinations. For baffle-former bolts, MRP-227-A Table 5-3 states that the examination acceptance criteria for the UT shall be established as part of the examination's technical justification (technical justification). MRP-228 provides additional guidance on preparation of technical justifications.

Q132. How does Entergy's AMP meet this guidance?

A132. [JP][AH] The Staff found the applicant's AMP consistent with the LR-ISG criteria because the applicant's AMP references Section 5 and Table 5-3 of MRP-227-A with respect to acceptance criteria. However, the IP2 and IP3 RVI Program does not indicate whether a technical justification has been or will be developed for the baffle-former bolts. In a request for additional information response, Entergy stated it had not prepared a technical justification yet and provided a justification for not doing so. In addition, the Staff determined that it does not need to review the technical justification to make a safety finding on the applicant's AMP because (1) MRP-227-A and the Staff's final SE for MRP-227, Revision 0 do not specify that technical justifications must be submitted for Staff review and approval; (2) UT examinations of baffle-former bolts have been performed since the late 1990's, thus, there is reasonable assurance that these examinations can be implemented effectively at IP2 and IP3; and (3) finalizing the technical justification closer to the date of the inspection will allow for the latest UT technology and lessons learned from previous baffle-former bolt examinations to be incorporated. Therefore, the Staff's concern regarding the technical justification was resolved. Additional discussion of the baffle-former bolts inspections can be found in response to Q319. The Staff found the applicant's description of the "acceptance criteria" program element acceptable because the acceptance criteria will be in accordance with the MRP-227-A recommendations, thus meeting the guidance of LR-ISG-2011-04 Section X1.M16A.

Q133. What is the Staff's guidance for corrective actions for RVI AMPs?

A133. [JP][AH] The Staff's guidance for corrective actions from LR-ISG-2011-04 states that any detected conditions that do not satisfy the examination acceptance criteria are required to be dispositioned through the plant corrective action program, which may require repair, replacement, or analytical evaluation for continued service until the next inspection. The Staff's guidance further states that the disposition will ensure that design basis functions of the reactor internals components will continue to be fulfilled for all licensing basis loads and events, and that the implementation of the guidance in MRP-227-A, plus the implementation of any ASME Code requirements, provides an acceptable level of aging management of safety-related components addressed in accordance with the corrective actions of 10 CFR Part 50, Appendix B or its equivalent, as applicable. LR-ISG -2011-04 also states that other alternative corrective actions bases may be used to disposition relevant conditions if they have been previously approved or endorsed by the NRC, and that alternative corrective actions not approved or endorsed by the NRC will be submitted for NRC approval prior to their implementation.

Q134. How does Entergy's AMP meet this guidance?

A134. [JP][AH] The Staff found Entergy's AMP met the Staff's guidance for corrective actions because detected conditions not satisfying the examination acceptance criteria will be processed through the plant's corrective actions programs. Additionally, Entergy's AMP includes provisions to use the methodologies in Section 6 of MRP-227-A or other acceptable alternatives to analytically disposition unacceptable conditions, which is consistent with LR-ISG-2011-04. MRP-227-A, Section 6 describes methods for evaluating conditions such as cracks detected during inspections of RVI. These methods cover supplemental non-destructive evaluation, crack growth rates to be assumed, and techniques for evaluating the structural stability of cracks. During the review of Entergy's AMP, the Staff requested additional

information regarding Entergy’s intent to use topical report WCAP-17096-NP, Rev. 2, “Reactor Internals Acceptance Criteria Methodology and Data Requirements,” (WCAP-17096-NP, Rev. 2, Reactor Internals Acceptance Criteria Methodology and Data Requirements (Dec. 2009), Exhibit NRC000200) as guidance for determining the acceptability of detected conditions. WCAP-17096-NP is a topical report intended by EPRI to be used in conjunction with MRP-227-A, which provides more detailed methods for performing engineering evaluation of conditions detected during MRP-227-A RVI inspections which do not meet the examination acceptance criteria. In its June 14, 2012 response, the applicant indicated it intended to use the guidance of WCAP-17096-NP. However, the NRC Staff has not yet issued a final safety evaluation approving WCAP-17096-NP as of July 1, 2015. Therefore, any use of the WCAP-17096-NP methodology for IP2 or IP3 would be subject to Staff review and approval until this topical report is approved by the NRC Staff.

Q135. What is the Staff guidance for the confirmation process for RVI AMPs?

A135. [JP][AH] The Staff’s guidance for the confirmation process from LR-ISG-2011-04 states that site quality assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the recommendations of Nuclear Energy Institute (NEI) 03-08, “Guidelines for the Management of Materials Issues” (Ref. 11), and the requirements of Appendix B to 10 CFR Part 50 or their equivalent as applicable. LR-ISG-2011-04 further states that the implementation of the guidance in MRP-227-A, in conjunction with NEI 03-08 and other guidance documents, reports, or methodologies cited in this AMP, provides an acceptable level of quality and an acceptable basis for confirming the quality of inspections, flaw evaluations, and corrective actions.

Q136. How does Entergy's AMP meet this guidance?

A136. [JP][AH] Entergy's AMP meets this guidance because the attributes of the applicant's confirmation process (including conformance to Appendix B to 10 CFR 50), which are generic to all AMPs for IP2 and IP3, were approved in NUREG-1930, and because the RVI AMP also meets the guidance of NEI 03-08, the Staff finds that the applicant's confirmation process is consistent with LR-ISG-2011-04.

Q137. What is the Staff guidance for administrative controls for RVI AMPs?

A137. [JP][AH] The Staff's guidance for administrative controls for RVI AMPs from LR-ISG-2011-04 states that the administrative controls for these types of programs, including their implementing procedures and review and approval processes, are implemented in accordance with the recommended industry guidelines and criteria in NEI 03-08, and are under existing site 10 CFR 50 Appendix B Quality Assurance Programs, or their equivalent, as applicable. LR-ISG-2011-04 further states that the evaluation in Section 3.5 of the NRC's SE, Revision 1, of MRP-227 provides the basis for endorsing NEI 03-08, and that this includes endorsement of the criteria in NEI 03-08 for notifying the NRC of any deviation from the inspection and evaluation (I&E) methodology in MRP-227-A and for justifying the deviation no later than 45 days after its approval by a licensee executive.

Q138. How does Entergy's AMP meet this guidance?

A138. [JP][AH] Entergy's AMP meets this guidance because the administrative controls applicable to the RVI AMP are in accordance with Appendix B to 10 CFR 50 and the RVI AMP meets the NEI 03-08 implementation requirements, including notification of the NRC of deviations from MRP-227-A.

Q139. What is the Staff guidance for Operating Experience for RVI AMPs?

A139. [JP][AH] The Staff's guidance for Operating Experience for RVI AMPs in LR-ISG-2011-04 states that the review and assessment of relevant operating experience for its impacts on the program, including implementing procedures, are governed by NEI 03-08 and Appendix A of MRP-227-A, and that, consistent with MRP-227-A, the reporting of inspection results and operating experience is treated as a "Needed" category item under the implementation of NEI 03-08. LR-ISG-2011-04 further states that the program is informed and enhanced when necessary through the systematic and ongoing review of both plant-specific and industry operating experience, as discussed in Appendix B of the GALL Report, which is documented in LR-ISG-2011-05, "Ongoing Review of Operating Experience."

Q140. How does Entergy's AMP meet this guidance?

A140. [JP][AH] Entergy's AMP meets this guidance because the applicant (a) demonstrated that it is aware and appropriately responds to operating experience, thus meeting the LR-ISG-2011-04 for a systematic and ongoing review of plant-specific and industry operating experience and (b) conforms to the guidance of NEI 03-08 with regard to reporting of operating experience. Further, the Staff found that the applicant has appropriately responded to past operating experience such as that related to SCC of split pins and the recent OE related to clevis insert bolts. This was demonstrated by Entergy's replacement of split pins in response to industry experience with SCC failures of the pins, and Entergy's evaluation of recent OE related to clevis insert bolts.

Q141. What is the RVI Inspection Plan?

A141. [JP][AH] The RVI Inspection Plan is a document describing how the inspections and other aging management activities of Entergy's RVI AMP will be implemented.

Q142. How does the RVI Inspection Plan relate to Entergy's RVI AMP?

A142. [JP][AH] The RVI AMP description only describes the ten elements of the RVI AMP to demonstrate compliance with the GALL Report, while the RVI Inspection Plan contains the details of the inspections of the Primary, Expansion, and Existing Programs components for IP2 and IP3. The RVI Inspection Plan also describes how Entergy proposed to resolve each A/LAI.

Q143. Of the 8 A/LAI's identified in the Staff's SE for MRP-227-A, which are applicable to IP2 and IP3?

A143. [JP][AH] A/LAI 1,2,3,5, 7 and 8 are applicable to IP2 and IP3.

Q144. Why don't A/LAI 4 and A/LAI 6 apply to IP2 and IP3?

A144. [JP][AH] A/LAI 4 and A/LAI 6 only apply to Babcock & Wilcox design RVI, not Westinghouse-design RVI, and therefore, are not applicable to IP2 and IP3.

Q145. Please summarize what is required by A/LAI 1?

A145. [JP][AH] A/LAI 1 requires applicants or licensees to assess its design and operating history and demonstrate the applicability of MRP-227-A to its plant (Exhibit ENT000230 at p. 32).

Q146. Why did the Staff include A/LAI 1 in the final SE of MRP-227, Rev. 0?

A146. [JP][AH] The Staff included A/LAI 1 because there were certain assumptions regarding operating conditions, such as neutron fluence, temperature, and stress that were used in the screening, FMECA, and functionality analysis of the generic RVI components. The

operating conditions used were based on a representative plant and were not bounding for all plants

Q147. How did Entergy resolve A/LAI 1?

A147. [JP][AH] Entergy used guidance issued by the EPRI MRP in letter MRP 2013-025 (Enclosure to MRP Letter 2013-025, MRP-227-A Applicability Guidelines for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs (Oct. 4, 2013), Exhibit NRC000198) to resolve this issue. MRP 2013-025 provides quantitative criteria based on core power density (related to fuel management) and geometry. These criteria were found acceptable by the NRC Staff in a safety evaluation of WCAP-17780-P. (Exhibit NRC000215)). If the plant meets the quantitative criteria, it can be reasonably assured that the assumptions made in developing MRP-227-A regarding neutron fluence and internal metal temperature of RVI components adequately represent the plant. MRP 2013-025 also provides guidance for assessing whether a plant has components with 20% or greater cold work. Entergy demonstrated that IP2 and IP3 met the core power density and geometry related criteria for Westinghouse-design RVI, and also demonstrated that IP2 and IP3 have no components with 20% or greater cold work. In addition to the above criteria, Entergy also demonstrated the three basic applicability criteria from MRP-227-A, Section 2.4 are met, which are: (1) 30 years of operation with high leakage core loading patterns (fresh fuel assemblies loaded in peripheral locations) followed by implementation of a low-leakage fuel management strategy for the remaining 30 years of operation; (2) base load operation, i.e., typically operates at fixed power levels and does not usually vary power on a calendar or load demand schedule; and (3) no design changes beyond those identified in general industry guidance or recommended by the original vendors.

Q148. What was the Staff finding on Entergy's resolution of A/LAI 1?

A148. [JP][AH] Since Entergy adequately addressed the two factors for which the Staff determined that additional plant-specific information was necessary to verify applicability of MRP-227-A to IP2 and IP3 – (1) cold work induced stress; and, (2) fuel management – by confirming that IP2 and IP3 comply with the criteria defined in the guidance document, the Staff determined Entergy adequately resolved A/LAI 1. (SSER2 at p. 3-34, Exhibit NYS000507)

Q149. Please summarize what is required by A/LAI2.

A149. [JP][AH] A/LAI 2 requires an applicant or licensee to review all the RVI components within the scope of license renewal at its plant, and identify whether all these components correspond to a generic component for the applicable vendor design. If any components are found within the scope of license renewal for the plant that do not have a corresponding generic component, the plant-specific RVI program must be modified to include these plant-specific components. (MRP-227, Rev. 0 SE at 32-33, Exhibit ENT000230)

Q150. Why did the Staff include A/LAI 2 in its final SE of MRP-227, Rev. 0?

A150. [JP][AH] The Staff included A/LAI 2 to address the possibility that some plants would have RVI components not included in the generic list of components, which would need to be evaluated on a plant-specific basis to determine if aging management activities were needed. (MRP-227, Rev. 0 SE at 16, Exhibit ENT000230)

Q151. How did Entergy resolve A/LAI2?

A151. [JP][AH] Entergy reviewed the IP2 and IP3 components within the scope of LR against the generic list of components for Westinghouse-design RVI in technical report MRP-191 Revision 0, "Materials Reliability Program: Screening, Categorization and Ranking of

Reactor Internals of Westinghouse and Combustion Engineering PWR Designs," (Exhibit NYS000321-00-BD01). Entergy determined there were no IP2 and IP3 components that were not equivalent to a generic component for Westinghouse-design RVI, although there were some components with differences in the specified material. The material differences were generally the use of a different grade of austenitic stainless steel than assumed for the generic component. Entergy evaluated the components made from different materials to determine if any changes in aging management were necessary, and determined no changes in aging management were necessary. (SSER2 at 3-34-3-35 NYS00507)

Q152. What was the Staff finding on Entergy's resolution of A/LAI 2?

A152. [JP][AH] The Staff found the applicant's response to A/LAI 2 acceptable because the applicant verified that all of the IP2 and IP3 components within the scope of LR are covered by MRP-191, which the Staff confirmed, and the Staff determined via an audit of the applicant's evaluation of the material differences that the applicant's evaluation was acceptable. (SSER2 at p. 3-35, Exhibit NYS000507)

Q153. Please summarize what is required by A/LAI 3.

A153. [JP][AH] A/LAI 3 requires an applicant or licensee to perform plant-specific analysis to justify the adequacy of its plant-specific programs. (MRP-227, Rev. 0 SE at p.33. Exhibit ENT000230)

Q154. Why did the Staff include A/LAI 3 in its final SE of MRP-227, Rev. 0?

A154. [JP][AH] The Staff included A/LAI 3 in its final SE of MRP-227, Rev. 0 because the report indicated certain components were addressed by plant-specific programs, thus no

specific guidance on aging management of these components was provided in the report. (MRP-227, Rev. 0 SE at pp. 16-17 Exhibit ENT000230)

Q155. How did Entergy resolve A/LAI 3?

A155. [JP][AH] For Westinghouse-design RVI such as those of IP2 and IP3, the only component to which A/LAI 3 applies is the guide tube support pins (split pins). Entergy's plant-specific program addressed SCC of split pins by replacement with new pins made from SCC-resistant materials.

For IP2, the split pins were replaced in the 1990's with Alloy X-750 pins. This is the same alloy as the original split pins, but the replacement pins had a different heat treatment that makes them more resistant to SCC. The IP3 split pins were replaced in 2009 with cold-worked Type 316 stainless steel pins, which operating experience has shown are highly resistant to SCC. Entergy indicated that it intended to replace the IP2 pins with cold-worked Type 316 stainless steel pins in 2016. To address Staff concerns regarding the possibility of SCC of the IP2 pins during the time prior to replacement, Entergy performed a statistical model based on operating experience that showed a very low probability of split pin failures prior to the planned replacement date. Entergy also made a commitment (Commitment 50) to replace the IP2 pins during the t2016 refueling outage. (SSER2 at 3-36 to 3-38, Exhibit NYS000507)

Q156. What was the Staff finding on Entergy's resolution of A/LAI 3?

A156. [JP][AH] The Staff found that the applicant's plant-specific program for split pins was acceptable because the existing pin replacement date has been justified. The date is formalized through a license commitment that will be made a requirement through license condition on issuance of the license renewal. The split pins resistance to SCC will be enhanced through the use of improved materials. (SSER2 at p. 3-38, Exhibit NYS000507)

Q157. Please summarize what is required by A/LAI 5.

A157. [JP][AH] A/LAI 5 requires an applicant or licensee to identify plant-specific acceptance criteria to be applied when performing the physical measurements required by MRP-227-A. (MRP-227, Rev. 0 SE at pp. 33-34 Exhibit ENT000230)

Q158. Why did the Staff include A/LAI 5 in its final SE of MRP-227, Rev. 0?

A158. [JP][AH] The Staff included A/LAI 5 because MRP-227, Rev. 0, did not include a description of the physical measurement techniques or the acceptance criteria. Therefore, the Staff needed this information to be provided on a plant-specific basis. (MRP-227, Rev. 0 SE at pp. 21-22 Exhibit ENT000230)

Q159. How did Entergy resolve A/LAI 5?

A159. [JP][AH] The only component for which physical measurements are specified for Westinghouse-design RVI such as those of IP2 and IP3 is the hold-down spring. For IP2 and IP3, Entergy stated the acceptance criteria are a function of spring height as a function of time relative to the required hold-down force. The decrease in the hold-down spring height is assumed to occur linearly over time. The approach linearly interpolates the required minimum spring height at the time of measurement between the spring height at startup and the minimum spring height at 60 years. If the first spring height measurement is less than the required height, successive (additional) measurements of spring height will be performed or the hold-down spring will be replaced.

Since details of the calculation of the acceptance criteria were considered proprietary, the Staff performed an audit of Entergy's calculation. Based on the audit, the Staff understood how the acceptance criteria was calculated. Further, the Staff reviewed information during the

audit justifying the conservatism of assuming a linear decrease in hold-down spring height as a function of time. (SSER2 at pp. 3-38-3-39, Exhibit NYS000507) [redacted] (U.S. Nuclear Regulatory Commission Office of Nuclear Reactor Regulation Division of License Renewal Aging Management Audit Report (Oct. 6, 2014), Exhibit NRC000216).

Q160. What was the Staff finding on Entergy's response to A/LAI 5?

A160. [JP][AH] The Staff found Entergy's response to A/LAI 5 acceptable because Entergy provided a plant-specific acceptance criterion for the remaining compressible height or hold-down force of the hold-down spring. (SSER2 at p. 3-39, Exhibit NYS000507).

Q161. Please summarize what is required by A/LAI 7.

A161. [JP][AH] A/LAI 7 requires an applicant or licensee to perform a plant-specific analysis of CASS RVI components to demonstrate the components will remain capable of performing their intended functions during the period of extended operation. The analyses must account for the potential loss of fracture toughness of the components due to both thermal

embrittlement and irradiation embrittlement, as applicable. The requirement to perform the functionality analysis is focused on components that require aging management according to the guidance of MRP-227-A. (MRP-227, Rev. 0 SE at pp. 34 Exhibit ENT000230).

Q162. Why did the Staff include A/LAI 7 in its final SE of MRP-227, Rev. 0?

A162. [JP][AH] The Staff included A/LAI 7 due to concerns that potential reductions in fracture toughness due to TE and IE were not adequately addressed by the guidance of MRP-227-A. The Staff was concerned that the combined effect of TE and IE could reduce fracture toughness in some CASS components to a level at which the components could fail due to pre-existing fabrication flaws or service induced flaws that are too small to be detected by the inspection technique recommended by MRP-227-A, under normal or design-basis loading. (MRP-227, Rev. 0 SE at p. 23 Exhibit ENT000230)

Q163. How did Entergy resolve A/LAI 7?

A163. [JP][AH] Entergy identified that the only components requiring a response to A/LAI 7 for IP2 and IP3 are the lower support columns. Only the upper portion of the lower support columns, the column cap, is made from CASS. The lower support columns are an Expansion component in MRP-227-A, and the associated (linked) Primary component is the control rod guide tube lower flange welds. The lower support columns are shown in Figure 5 and the control rod guide tube CRGT lower flange welds are shown in Figure 3. Figure 10 shows a detail of the construction of typical lower support columns.

Figure 4-34
Typical Westinghouse core support column. Core support column bolts fasten the top of the support column to the lower core plate

Figure 10 – Lower Support Columns, from MRP-227-A Figure 4-34 (Exhibit NRC000114B)

The Staff's final SE of MRP-227, Rev. 0 described three possible methods for applicants or licensees to perform functionality analyses of CASS components:

- Functionality analyses for the set of like components or assembly-level functionality analyses, or

- Component level flaw tolerance evaluation justifying that the MRP-227 recommended inspection technique(s) can detect a structurally significant flaw for the component in question, taking into account the reduction in fracture toughness due to irradiation embrittlement and thermal embrittlement; or

- Applying screening criteria for the component's material to demonstrate that the components are not susceptible to either thermal embrittlement or irradiation embrittlement, or the synergistic effects of thermal embrittlement and irradiation embrittlement combined. The Staff's final SE of MRP-227, Rev. 0 stated that for assessment of CASS materials, the licensees or applicants for LR may apply the criteria in the NRC letter of May 19, 2000, "License Renewal Issue No. 98-0030, Thermal Aging Embrittlement of Cast Stainless Steel Components" (NRC ADAMS Accession No. ML003717179) (Exhibit NRC000213) as the basis for

> determining whether the CASS materials are susceptible to the thermal aging embrittlement mechanism. However, this document restricts application of the screening criteria to components with neutron fluence below 1x1017 n/cm2, which make these criteria inapplicable for many RVI components.

(MRP-227, Rev. 0 SE at p. 23 Exhibit ENT000230).

Entergy screened the lower support columns for IP2 and IP3 for TE susceptibility using the delta ferrite content calculated from the actual chemical composition of the columns, which was obtained from plant construction records. The lower support columns for IP2 and IP3 are made from Type CF-8 stainless steel, which is a low-molybdenum content grade of cast stainless steel. Low-molybdenum CASS grades are less susceptible to TE than high-molybdenum CASS grades such as Type CF-8M. Entergy determined the lower support columns at IP2 and IP3 were not susceptible to TE. (SSER2 at p. 3-41, Exhibit NYS000507) Entergy used the criteria from "License Renewal Issue No. 98-0030, *Thermal Aging Embrittlement of Cast Stainless Steel Components*" (NRC ADAMS Accession No. ML003717179)(Exhibit NRC000213). These screening criteria would allow low-molydenum CASS with delta ferrite content of 20 % or less to be screened out (not susceptible) to TE. Since portions of the lower support column will have neutron fluence at the end of life greater than $1x10^{17}$ n/cm^2, the Staff did not accept Entergy's screening. Entergy did determine the columns are susceptible to IE, using a screening criterion of $6.7x10^{20}$ n/cm^2 (1 dpa) as the fluence above which the CASS materials is susceptible to IE.

Using the delta ferrite values for the IP2 and IP3 lower support columns, the Staff applied alternate interim screening criteria developed by the NRC Staff (NRC Position on Aging Management of CASS Reactor Vessel Internal Components, (June 11, 2014), Exhibit NRC000201), and determined that the IP2 and IP3 lower support columns could be screened out for TE. (SSER2 at p. 3-45, Exhibit NYS000507) According to the Staff's updated criteria, low-molybdenum statically cast CASS with a ferrite content less than 15 percent can be

screened out for TE and any synergistic effects of TE and IE, and this low-molybdenum statically cast CASS is only susceptible to IE at fluences greater than 1×10^{21} n/cm^2 (1.5 dpa). This ferrite value represents a reduction in the ferrite content from the screening values from the Staff's May 19, 2000, letter to accommodate possible combined effects of thermal and neutron embrittlement. The ferrite content of the IP2 and IP3 column caps meets this criterion because all the materials used to fabricate the column caps have calculated ferrite content less than 15 percent.

In addition to screening for TE and IE, Entergy also provided information for the lower support columns from original fabrication demonstrating that there is a low probability that the columns have original fabrication defects. Specifically, a liquid penetrant examination and radiography were performed during original fabrication, and no surface breaking defects were found. Entergy also determined that service-induced cracking was very unlikely because (1) The operating stresses in the columns are too low to cause IASCC at the expected neutron fluence levels for the columns, and (2) Entergy performed fatigue analyses of the columns showing cumulative usage factors adjusted for the effects of the reactor water environment (CUF_{en}) of less than 1.0 for the life of the plant, which means that fatigue crack initiation is not expected during the life of the plant. (SSER2 at p. 3-43, Exhibit NYS000507)

In response to NRC Staff concerns regarding whether the associated primary component for the lower support columns, the control rod guide tube lower flange welds, was a good predictor for IASCC and IE, Entergy added an additional plant-specific Primary component linked to the lower support columns. This component is the core barrel lower girth welds, which are exposed to similar neutron fluence levels as the upper portion of the lower support columns and also are susceptible to IASCC and IE. Although the CRGT lower flange welds screened in for IE and IASCC, they experience much lower neutron fluence levels and thus are not as good of a predictor for these aging mechanisms for the lower support columns.

Q164. What was the Staff finding on Entergy's resolution of A/LAI 7?

A164. [JP][AH] The Staff found that Entergy adequately addressed A/LAI 7 based on the following: (1) Entergy evaluated the CASS components of the RVI (the lower support columns (column caps)); (2) Entergy screened the column caps for TE and IE using plant-specific materials data and determined that the column caps are not susceptible to TE (and the Staff confirmed the results of the screening using its own screening criteria); (3) Entergy provided information on fabrication NDE demonstrating that pre-existing flaws are unlikely to exist in the column caps; (4) Entergy provided information on the expected stresses and neutron fluence for the column caps that demonstrated that service-induced cracking due to IASCC is unlikely; and (5) Entergy modified its RVI Program to include a link to a lead component that is an appropriate predictor of IASCC and IE for the column caps, with an appropriate schedule for performing the Expansion inspection if necessary. Therefore, the Staff found that the information provided by Entergy provides reasonable assurance that the functionality of the column caps will be maintained during the period of extended operation for IP2 and IP3. (SSER2 at 3-47, Exhibit NYS000507).

Q165. What does A/LAI 8 require?

A165. [JP][AH] A/LAI 8 requires applicants or licensees to make a submittal for NRC review and approval to credit their implementation of MRP-227-A as an AMP for the RVI components at their facility. (MRP-227, Rev. 0 SE at p. 34 Exhibit ENT000230). The information required is detailed in Section 3.5.1 of the MRP-227, Rev. 0 SE at 25-26, Exhibit ENT000230. The action item requires items (1) and (2) for all applicants or licensees and items 3-5 for those applicants submitting LRAs after the issuance of the final SE for MRP-227, Rev. 0. Although Entergy submitted the LRA for IP2 and IP3 before the issuance of the final SE for

MRP-227, Rev. 0, Entergy chose to provide the information for all five items. The items are summarized as follows:

1. An AMP for the facility that addresses the 10 program elements as defined in NUREG-1801, Revision 2, AMP XI.M16A.

2. An RVI Inspection Plan addressing all the applicable A/LAI's.

3. A Final Safety Analysis Report (FSAR) supplement containing a summary description of the RVI AMP and any TLAAs related to RVI. This is required for all AMPs and TLAAs by 10 CFR 54.21(d).

4. Any changes to technical specifications required to manage aging of the RVI, as required by 10 CFR 54.22.

5. Identify all analyses in the current licensing basis that are TLAAs related to RVI, as required in general for TLAAs by 10 CFR 54.21(c)(1). A/LAI 8 also stated that for those cumulative usage factor (CUF) analyses that are TLAAs, the applicant may use the PWR Vessel Internals Program as the basis for accepting these CUF analyses in accordance with 10 CFR 54.21(c)(1)(iii) only if the RVI components in the CUF analyses are periodically inspected for fatigue-induced cracking in the components during the period of extended operation. The periodicity of the inspections of these components shall be justified to be adequate to resolve the TLAA. Otherwise, acceptance of these TLAAs shall be done in accordance with either 10 CFR 54.21(c)(1)(i) or (ii), or in accordance with 10 CFR 54.21(c)(1)(iii) using the applicant's program that corresponds to the GALL Report, Revision 2, AMP X.M1, "Metal Fatigue of Reactor Coolant Pressure Boundary Program." To satisfy the evaluation requirements of ASME Code Section III, Subsections NG-2160 and NG-3121, the existing fatigue CUF analyses should include the effects of the reactor coolant system water environment.

Q166. Why did the Staff include A/LAI 8 in its final SE of MRP-227, Rev.0?

A166. [JP][AH] The Staff included A/LAI 8 in the final SE of MRP-227, Rev. 0 to ensure applicants or licensees include sufficient information in submittals referencing MRP-227-A, so that the Staff can determine whether applicants or licensees will implement the guidance of MRP-227-A appropriately. (MRP-227, Rev. 0 SE at p. 25 Exhibit ENT000230)

Q167. How did Entergy resolve A/LAI 8?

A167. [JP][AH] Entergy provided an RVI AMP, RVI Inspection Plan, and FSAR Supplement describing the RVI AMP, thus meeting the requirements of Items 1-3 of A/LAI 8. Entergy stated that no TS changes were required related to the RVI AMP, thus Item 4 was not applicable. For Item 5, the LRA contained a TLAA for fatigue of RVI, with CUFs reported in the LRA. The original RVI fatigue CUFs did not account for the effects of the reactor water environment. Entergy resolved Item 5 via a license renewal commitment (Commitment 49) to update these CUFs to account for the effects of the reactor water environment, and management of the fatigue TLAAs for RVI using its Fatigue Monitoring Program, which is consistent with the guidance of the GALL Report, Revision 2, AMP X.M1, "Metal Fatigue of Reactor Coolant Pressure Boundary Program." Entergy's calculation of the environmentally adjusted CUFs, or CUF_{en} values, will use methods consistent with those that are acceptable according to the GALL Report, Revision 2, Section X.M.1, "Fatigue Monitoring." (SSER2 at 3-49-3-53, Exhibit NYS000507). Commitment 49 has been completed for IP2. Ex. NRC000184.

Q168. What was the Staff's finding on Entergy's resolution of A/LAI 8?

A168. [JP][AH] The Staff found Entergy acceptably resolved A/LAI 8 because it provided all the required information items. For Item 5, the Staff found Entergy's resolution acceptable based on its commitment (Commitment 49) to recalculate the RVI CUFs using NRC-approved methodology to account for environmental effects, and also because it will use its Fatigue Monitoring Program, which is consistent with the Staff's guidance, to manage the fatigue TLAA as specified for Item 5, when fatigue is not managed via inspections. (SSER2 at 3-51-3-53, Exhibit NYS000507). Commitment 49 has been completed for IP2. Ex. NRC000184.

Q169. How will Primary components be inspected under Entergy's RVI AMP?

A169. [JP][AH] For the Entergy RVI AMP, Table 5-2 of the RVI Inspection Plan specifies the inspection methods for the Primary components. (pg. 37 to 42 of Ex. NYS000496). The Staff found these inspections were all consistent with the Primary component inspections specified in MRP-227-A for Westinghouse-design RVI. (SSER2 at 3-18, Exhibit NYS000507)

Q170. How will the Expansion components be inspected under Entergy's RVI AMP?

A170. [AH][JP] For the Entergy RVI AMP, Table 5-3 of the RVI Inspection Plan specifies the inspection methods for the Expansion components. (pg. 43 to 50 of Ex. NYS000496)

Q171. How are the Existing Programs components inspected under Entergy's RVI AMP?

A171. [AH][JP] The inspections of the Existing Programs components under Entergy's AMP are listed in Table 5-4 of the RVI Inspection Plan. (pg. 51 of Ex. NYS000496). The Staff found these inspections are consistent with the Existing Programs inspection specified in MRP-227-A for Westinghouse-design RVI. (SSER2 at 3-17 and 3-59, Ex. NYS000507)

Q172. Are there any differences in the way Entergy is implementing its RVI AMP compared to the generic industry program described in MRP-227-A?

A172. [AH][JP] As discussed in the responses to Q166-Q167 related to A/LAI 7, Entergy added an additional Primary link to the core barrel girth weld for the lower support columns in order to provide a more appropriate lead component with respect to the relevant aging mechanisms for the lower support columns, irradiation embrittlement and IASCC. Otherwise, the Primary, Expansion and Existing Programs components, inspection scope,

methods, and schedules that are specified under Entergy's RVI AMP are identical to those specified in MRP-227-A for generic Westinghouse RVI.

Q173. Based on the MRP-227-A and IP construction and history, did Entergy assign the RVI components into the appropriate categories?

A173. [JP][AH] Yes.

Q174. In your expert opinion, does Entergy's AMP meet the Commission's regulations?

A174. [JP][AH] Entergy's AMP meets the Commissions regulation at 10 CFR 54.21(a)(3), which requires, for each structure and component subject to aging management review (AMR), that applicants for license renewal 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. Our opinion is based on the facts that:

- Entergy submitted an RVI AMP based on NRC-approved topical report MRP-227-A. On the basis of its review of the applicant's RVI AMP, the Staff concluded that the applicant has demonstrated 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, as required by 10 CFR 54.21(a)(3), because the AMP is consistent with the appropriate guidance of LR-ISG-2011-04. The Staff also reviewed the FSAR supplement for this AMP and concludes that it provides an adequate summary description of the program, as required by 10 CFR 54.21(d).(SSER at 3-26)
- The Staff found that Entergy's RVI Inspection Plan implements the elements of Entergy's RVI AMP in an acceptable manner because: (1) the applicant's program is consistent with the generic RVI inspection and evaluation guidelines of MRP-227-A; (2) the applicant adequately addressed all of the A/LAIs of the final SE for MRP-227, Rev. 0, that are applicable to Westinghouse-designed RVI or generically to all NSSS designs; and (3) the RVI Inspection Plan addresses the conditions of the final SE for MRP-227, Revision 0. (SSER at 3-59)

<u>DISCUSSION OF DR. LAHEY'S TESTIMONY</u>

Q175. Are you familiar with the testimony, reports, and opinions of New York's witness, Dr. Lahey?

A175. [JP][AH] Yes.

Q176. Do you agree with Dr. Lahey's opinions regarding the adequacy of the AMP for reactor vessel internals?

A176. [JP][AH] No.

Q177. Please identify the portions of NYS-25 and Dr. Lahey's testimony in support of NYS-25 that in your expert opinion are not correct?

A177. [JP][AH] Dr. Lahey identifies a number of areas of concern that are not material to the Staff's analysis, the regulatory requirements, or the scope of a license renewal proceeding including: (1) Synergistic Effects, (2) Design Basis and Accident Loads, (3) Preemptive Replacement, (4) Use of VT-3, and (5) Fatigue Error Analysis.

Q178. Why do you disagree with Dr. Lahey's concerns regarding synergistic effects?

A178. [JP][AH] We primarily disagree with Dr. Lahey's concerns regarding synergistic effects because (1) the lead or Primary components to be inspected under Entergy's program are those components that would be most likely to be affected by synergistic effects if they exist, and (2) Entergy's AMP inspects for cracking and the inspections are insensitive as to whether cracking results from a single mechanism or multiple synergistic mechanisms.

Q179. Why do you disagree with Dr. Lahey's concerns regarding design basis and accident load failure analysis?

A179. [JP][AH] The scope of the license renewal proceeding is narrowly defined by the Commission. Under that narrow scope, the NRC does not revisit design basis accidents, accident analysis, and other portions of the current licensing basis ("CLB"). In our opinion, Dr. Lahey's assertion regarding the need for new accident analysis challenges the Commission's careful determination to limit issues that are material to the license renewal decision. As proposed and designed, the RVI AMP is structured to manage the aging effects such that the intended function of the components will be preserved during the period of extended operation, i.e. from 40 to 60 years. It accomplishes this task by establishing an inspection plan for the relevant components that is able to identify potential aging effects prior to any loss of function, through appropriate schedules and conservative acceptance criteria. Once potential aging effects are identified, the applicant would put the results into its corrective actions program and appropriately disposition the issue through additional analysis, repair, or replacement that could include license amendments, when necessary.

Q180. Why do you disagree with Dr. Lahey's assertion that preemptive replacement of all RVI components is necessary?

A180. [JP][AH] While we agree that components should be repaired or replaced prior to losing the capability to perform their intended functions, as part of corrective actions if found to be necessary, it is unnecessary and inappropriate to require preemptive replacement while those components remain safe to operate and are within the allowable operating limits of the CLB.

Q181. Why are VT-3 inspections adequate techniques for identifying the potential for aging in RVI components?

A181. [JP][AH] VT-3 is the most appropriate inspection method for some aging effects, such as wear and distortion. VT-3 is only used for detection of cracking in redundant populations of components, in which failures of individual components can be tolerated, or in very robust components where a single crack cannot lead to loss of function. Further, VT-3 examination is not the only examination technique used and is not used for a majority of Primary and Expansion components. Enhanced visual testing (EVT-1) is used to inspect for cracking in non-redundant components, and ultrasonic testing (UT) is also used to detect cracking in baffle-former bolting which has a known history of degradation, and in which cracking cannot be detected by visual examination prior to bolt failure.

Q182. Why do you disagree with Dr. Lahey's concerns regarding the lack of an error analysis in the fatigue analyses?

A182. [JP][AH] Fatigue analyses are deterministic in nature and use conservative assumptions that result in adequate and ample margin in the results to account for any potential uncertainties in the analysis. For example, (1) fatigue analyses assume that all actual transients are as severe as the corresponding design transients, when in reality actual transients are typically less severe in terms of rate of temperature change and temperature range, (2) fatigue analyses assume a bounding number of each transient type for the life of the plant, while monitoring of these numbers of cycles by the Fatigue Monitoring Program ensures the actual cycle counts are less than the bounding number, and (3) design factors are applied on stresses as a part of the fatigue analyses.

ANALYSIS OF SYNERGISTIC EFFECTS

Q183. Please elaborate on your reasons for disagreement with Dr. Lahey's assertion that Dr. Entergy's Amended and Revised RVI Program is inadequate because it does not address the synergistic effects of various aging mechanisms, such as radiation-induced embrittlement, corrosion-induced cracking and fatigue-induced degradation.

A183. [JP][AH] We disagree with Dr. Lahey's concerns regarding synergistic effects because (1) the Primary components to be inspected under Entergy's program are those components that are most likely to be affected by synergistic effects if they exist; (2) Entergy's AMP inspects for and will detect cracking, whether a single aging mechanism or multiple or synergistic aging mechanisms contribute to the cracking; (3) embrittlement alone will not cause failure without the presence of a crack, and the inspections performed by Entergy's AMP are sufficient to detect cracking; (4) Dr. Lahey has not identified any test results or operating experience which demonstrate that synergistic effects are significant for PWR RVI, and existing laboratory test data on synergistic effects is inconclusive; and (5) the industry RVI program in which Entergy is participating is a living program which shares operating experience among all the PWR licensees, thus, any occurrence of unexplained or accelerated degradation due to synergistic effects will be identified, and adjustments to the industry guidance and the Entergy AMP will be made to ensure continued integrity of RVI across the fleet.

Q184. How does Entergy's RVI AMP account for the components most likely to experience synergistic aging effects?

A184. [JP][AH] If a component was subject to both embrittlement and any type of cracking mechanism, including fatigue cracking, SCC, or IASCC, this would be considered in

the FMECA process. Components that are subject to both embrittlement and cracking, in which component failure was deemed likely, and for which failure of the component could cause core damage were typically identified as Primary or Expansion category components.

Therefore, in Entergy's RVI Inspection Plan, the components most likely to experience aging effects and most likely to impact safe operations are inspected first as Primary components.

Q185. Provide an example of a component expected to undergo multiple aging effects.

A185. [JP][AH] An example of such a component is the core barrel girth welds, which are susceptible to irradiation embrittlement, and cracking due to IASCC, fatigue and SCC.

Q186. How are the core barrel girth welds categorized in Entergy's AMP?

A186. [JP][AH] These welds are Primary components that will be inspected using enhanced visual examination (EVT-1) within two refueling outages of the beginning of the period of extended operation (PEO), and every ten years thereafter.

Q187. Would potential synergistic effects change how aging is managed?

A187. [JP][AH] No. When more than one cracking mechanism screened in, the MRP-227 process assumed all of the mechanisms were active, although some may have been considered more likely. The inspection recommendations of MRP-227-A were based on identification of cracks from any source and were not based on crack growth and fracture mechanics analyses. If a crack is found by the inspection, corrective actions would have to consider the effects from any mechanism that could grow the crack, e.g. fatigue and IASCC.

Q188. Are there any other ways the AMP handles synergistic aging effects?

A188. [JP][AH] Yes. Entergy's AMP, as amended by various responses to NRC Staff requests for additional information adequately addresses A/LAI 7 from the NRC Staff's final SE of MRP-227, Rev. 0. A/LAI 7 was included by the Staff specifically to address the possibility of a synergism between TE and IE for CASS material, causing greater loss in fracture toughness. Entergy addressed this effect by using NRC guidance on screening of CASS material to identify that the ferrite content is low enough to exclude thermal embrittlement as a potential synergistic contributor to embrittlement with irradiation embrittlement.

In addition, the interaction of corrosion and fatigue crack growth is addressed by the use of environmental adjustments to cumulative fatigue usage factors for components subject to significant fatigue cycles and exposed to reactor coolant. Entergy has committed to handle this effect by recalculating its cumulative fatigue usage factors for RVI components using environmental adjustment factors.

Q189. Can embrittled RVI components fail without the presence of a crack?

A189. [JP][AH] No. Some type of crack-like defect is necessary for failure to occur.

Q190. How are the crack size that could cause failure of a component and the material fracture toughness related?

A190. [JP][AH] As the material fracture toughness, or resistance to crack growth, decreases, the size of the crack necessary to cause failure of a component also decreases.

Q191. How is fracture toughness characterized?

A191. [JP][AH] In less ductile materials, such as carbon steel tested at low temperatures, fracture toughness is typically characterized by the plane-strain fracture

toughness parameter K_{IC}, which is the value of stress intensity at which unstable crack growth begins. The applied stress intensity is determined using an equation which, in its simplest form, can be expressed as, $K= \sigma\sqrt{\pi a}$ where σ is the applied stress, and "a" is the crack dimension.

In more ductile materials, such as austenitic stainless steels and carbon steel at higher temperatures, fracture toughness is characterized using J-integral versus crack extension (J-R) tests, which measure the energy absorbed as a function of crack extension for a specimen that is subjected to an increasing load. Ductile behavior is exhibited in J-R tests when the curve has a positive slope that is when additional energy is required to continue growth of the crack. The value of J near the onset of crack extension is known as J_{IC}. (ASM Handbook, Vol. 19, Fatigue and Fracture, Figure 14 at 595 (1996), Exhibit NRC000202 at 398-399).

Q192. What is the effect of embrittlement on the fracture properties of stainless steel?

A192. [JP][AH] Embrittlement, caused either by irradiation or thermal aging in the case of cast austenitic stainless steels, reduces the amount of energy required to extend a crack by a given amount. However, the stainless steel will still typically fail by ductile tearing. Figure 11, below, shows some examples of the effect of neutron irradiation on the J-R curves of austenitic stainless steels.

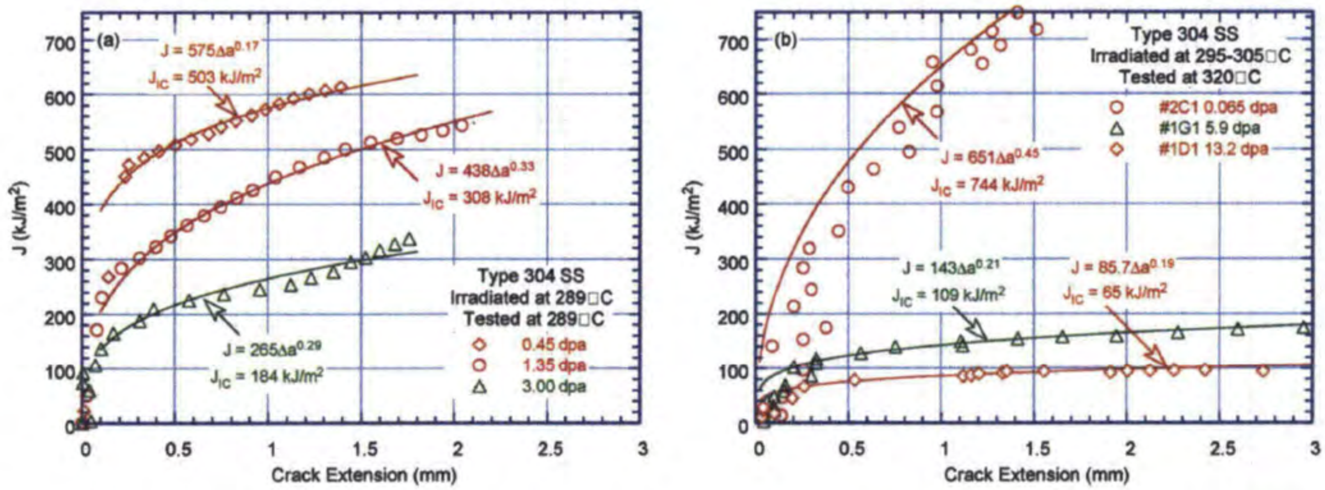

Figure 56. Change in fracture toughness J_{IC} as a function of neutron exposure for SSs (Refs. 11,122).

Figure 11 – Effect of Irradiation on J-R Curves – From NUREG/CR-7027 Figure 56 at 68, Exhibit NYS000487

Q193. What is the effect of fatigue on the structural integrity of a component?

A193. [JP][AH] If a component experiences cyclic or fatigue loading, there will be no effect on the structural integrity unless a fatigue crack initiates. At a given level of cyclic or alternating stress, a certain number of cycles is required for fatigue crack initiation. If that number of cycles has not been experienced yet by the component, then no fatigue crack will be present. Therefore, the component's structural integrity will not be affected.

Q194. Is inspection for cracking an effective means of managing aging effects such as embrittlement, given the fact that synergistic effects could exist?

A194. [JP][AH] Yes, inspection that is capable of detecting cracking is effective in managing aging effects because the effect of embrittlement is to reduce the resistance to crack growth, and thus inspection that can exclude the presence of cracks also excludes the adverse effects of embrittlement on component structural integrity. Entergy's AMP specifies enhanced

visual examination (EVT-1), ultrasonic examination, or VT-3 inspection for cracking (only for redundant components). In accordance with MRP-228; "Materials Reliability Program: Inspection Standard for PWR Internals," (Exhibit NYS000323), which is referenced by MRP-227-A (MRP-227-A at 4-4, Exhibit NRC000114A), equipment resolution for EVT-1 examinations must be demonstrated by resolving a character (letter) with a height of 0.044 inches (MRP-228 at 2-12, Exhibit NYS000323). Operating experience has demonstrated that ultrasonic examination is effective for detecting cracking in baffle-former bolts (MRP-227-A at 4-5, A-3 Exhibit NRC000114A, Exhibit NC000114C). Although VT-3 examination has less stringent resolution requirements, VT-3 is only used for detecting cracking in redundant components where failure of individual components can be tolerated, or in those with some tolerance for cracking, as detailed in our responses to Q245 through Q252.

Q195. What is your opinion of Dr. Lahey's concern regarding the synergistic effect of fatigue and embrittlement?

A195. [JP][AH] With respect to Dr. Lahey's concern about the potential synergistic effect of fatigue and embrittlement, we agree that embrittlement may have an effect on the fatigue properties of the stainless steel material, because embrittlement changes the mechanical properties of the material. However, these changes do not always make fatigue cracking more likely because they can be beneficial to the fatigue life of the material.

Q196. How does irradiation affect the fatigue properties of stainless steel?

A196. [JP][AH] Irradiation embrittlement increases both the ultimate tensile strength and yield strength of the stainless steel materials with which the RVI are constructed (NUREG/CR-7027 at 94, Exhibit NYS000487). NUREG/CR-6909 "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials," summarizes data on fatigue life of

irradiated stainless steels. (NUREG/CR-6909 at 9-11, Exhibit NYS000490A). The results are inconclusive, i.e. they do not demonstrate a pronounced synergistic effect.

Q197. What is a fatigue limit?

A197. [JP][AH] For high-cycle fatigue, the fatigue limit is the level of cyclic stress below which the part can survive an essentially infinite number of cycles without failure. Therefore, thermal embrittlement and irradiation embrittlement should actually increase the resistance to fatigue crack initiation and growth.

Q198. Dr. Lahey identifies a significant amount of research currently being conducted to expand our understanding of aging effects. Does that research indicate any consensus regarding the synergistic interaction of aging effects?

A198. [JP][AH] There is no consensus for most synergistic interactions that have been hypothesized. However, regarding the interaction of fatigue and corrosion caused by the reactor water environment (environmental fatigue), this synergistic effect is relatively well understood, and NRC guidance exists to address this effect.

Q199. Based on your review of synergistic effects, is there any data that would contradict Dr. Lahey's theories regarding synergistic effects?

A199. [JP][AH] Yes. NUREG/CR-6909, "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials," (Exhibit NYS000490) contains data relevant to the interaction of irradiation embrittlement and fatigue. In addition, NUREG/CR-7184 Chen, et al., Crack Growth Rate and Fracture Toughness Tests on Irradiated Cast Stainless Steels (Revised December 2014) (ML14356A136) (Exhibit NYS000488 A/B), describes the results of recent NRC-sponsored testing of CASS material subject to both thermal aging and irradiation. Also,

NUREG/CR-7027, "Degradation of LWR Core Internal Materials due to Neutron Irradiation," (Exhibit NYS000487) contains data on fatigue crack growth rates of irradiated stainless steel.

Q200. Dr. Lahey cites a statement from NUREG/CR-6909 (Exhibit NYS000490 at p. 11) that "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," to support his opinion that the synergistic effect between fatigue and embrittlement is poorly understood. What is your opinion of this statement?

A200. [JP][AH] Dr. Lahey has taken the statement out of context by eliminating the clause that precedes the portion that he quoted. The entire quote is "Although some small-scale laboratory fatigue ε–N (strain versus number of cycles) 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." (NUREG/CR-6909 at 11, Exhibit NYS000490A) Based on the fatigue test results of irradiated stainless steels reported in NUREG/CR-6909, we agree that the effect of neutron irradiation on fatigue life is not conclusively demonstrated. However, this does not mean that it is "poorly understood," just that the results do not conclusively demonstrate an effect. In addition, the issue of the effect of neutron irradiation on fatigue life needs to be separated from the effects of the PWR primary water environment. We agree that the PWR primary water environment decreases the fatigue life of stainless steels. We also note that NUREG/CR-6909 does not recommend any adjustment to fatigue life based on neutron fluence levels.

Q201. In your opinion, do the results reported in NUREG/CR-7184 support a synergistic effect of IE and TE?

A201. [JP][AH] No. The results showed that the loss of fracture toughness in CASS material exposed to conditions that would cause IE and TE simultaneously (i.e, simultaneous irradiation and thermal aging) was less than the sum of the loss of fracture toughness due to IE and TE individually. Therefore, the fracture toughness results of NUREG/CR-7184 do not support a synergistic effect of IE and TE for CASS.

Q202. Do the test data in NUREG/CR-7027 show that irradiation accelerates fatigue crack growth rates in stainless steel?

A202. [JP][AH] No. NUREG/CR-7027 showed that actual crack growth rates measured for irradiated Type 304 stainless steel were no greater than those predicted using a correlation for non-irradiated Type 304 stainless steel, as shown by Figure 12 below. (NUREG/CR-7027 at 54, Exhibit NYS000487). The specimens were irradiated to 2.16 dpa, approximately equal to a neutron fluence of 1.45×10^{21} n/cm2. Based on generic estimates of fluence for screening purposes in MRP-191, this fluence is similar to the fluence that could be experienced by portions of the core barrel outboard of the active core, and the bottom of the upper core plate. (MRP-191 Table 4-6 at 4-22 through 4-29, Exhibit NYS000321-00-BD01). Therefore, there does not appear to be a synergistic effect of irradiation and fatigue at moderate fluence levels experienced in many parts of the RVI.

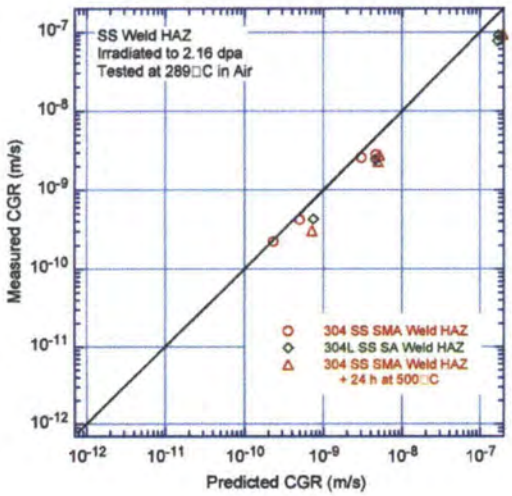

Figure 45.
CGR data under cyclic loading for irradiated SS weld HAZ materials in air at 289°C (Ref. 11).

Figure 12 – Crack Growth Rate of Irradiated Type 304 Stainless Steel Weld Heat Affected Zone, Measured on Irradiated Material vs. Predicted using a Correlation for Non-irradiated Stainless Steel (From Figure 45, NUREG/CR-7027 at 54, Exhibit NYS000487)

Q203. Dr. Lahey cites two statements to support his opinion that embrittlement decreases the resistance to crack propagation, "the effects of embrittlement, especially loss of fracture toughness, make existing cracks in the affected materials and components less resistant to growth," from the USNRC Letter, Grimes to Newton, at 16 (Feb. 10, 2001) (Ex. NYS000324), and "irradiation embrittlement decreases the resistance to crack propagation" [Westinghouse Owners Group WCAP-14577 Rev. 1-A Report, at 3-2 (March 2001) (Exhibit NYS000341)]. Do these statements support the existence of a synergistic effect of fatigue and embrittlement?

A203. [JP][AH] No, neither of these statements relates to a synergistic effect of fatigue and embrittlement. Both statements refer to the decreased resistance of irradiated material to ductile growth of cracks, not the initiation or growth of fatigue cracks. WCAP-14577 Rev. 1-A, at 3-2 (March 2001) (Exhibit NYS000341), states that:

> embrittlement is evidenced by increases in yield strength and tensile strength, and a decrease in ductility and fracture toughness. 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. Any pre-existing flaws or defects of significant size would have been prevented by quality assurance (QA) procedures during plant construction.

We note that the key concept is that a cracking mechanism is needed to initiate and grow a crack to a critical size. Such a cracking mechanisms could include fatigue, and SCC. There is no conclusive evidence that embrittlement accelerates either the initiation or the rate of growth due to these cracking mechanisms. For IASCC, the effects of irradiation embrittlement are embedded in the characterization of the mechanism.

Q204. If previously unknown synergistic effects became active in PWR RVI, how would the existence of such effects be identified and communicated?

A204. [JP][AH] Plant-specific RVI inspections, such as those under Entergy's RVI AMP, are part of an overall industry inspection program for PWRs conducted according the guidance of MRP-227-A. If synergistic effects of aging mechanisms were to occur, the resulting degradation will likely be found in at least one plant in the fleet. The inspection implementation requirements for MRP-227-A require each U.S. PWR to submit a summary report of all inspections and monitoring, items requiring evaluation, and new repairs, to the MRP Program Manager within 120 days of the completion of an outage during which PWR internals within the scope of MRP-227 are examined. The summary of results from all the PWR plants are compiled into a biennial report for the benefit of the fleet, the regulator, and other industry stakeholders. (MRP-227-A at 7-3, Exhibit NRC000114C) This reporting mechanism is used to disseminate operating experience to other PWRs, so if unexpected degradation occurs, Entergy

will be made aware of it and needed adjustments to the Entergy RVI AMP will be made. Industry's PWR RVI inspection program is a living program that will continue to evolve based on the combined operating experience of all the plants performing inspections. We understand it is industry's intent to modify its inspection program as necessary to adjust for operating experience.

Q205. In your expert opinion, does Entergy's RVI AMP address the potential for synergistic aging effects in an adequate manner?

A205. [JP][AH] Yes, Entergy's RVI AMP has appropriate measures in place to adequately manage the potential for synergistic aging effects through regularly scheduled inspections of leading components likely to experience detectable aging effects first.

ANALYSIS OF SHOCK LOADS

Q206. Are you familiar with Dr. Lahey's concerns regarding the ability of the reactor vessel internal components to perform during a design basis accident?

A206. [JP][AH] Yes. Dr. Lahey appears to advocate that the applicant be required to revisit the design-based and other shock loads on the RVI components rather than developing AMPs that are designed to detect and, if necessary, correct aging issues of components prior to them losing their ability to perform their intended functions.

Q207. Please summarize the reasons that you disagree with Dr. Lahey's opinion that Entergy's Amended and Revised RVI Program is inadequate because it does not address the potential consequences of shock loads resulting from design basis accidents on severely fatigued and embrittled components.

A207. [JP][AH] We disagree with Dr. Lahey's statement for several reasons. First, shock loads from design basis accidents are currently part of the CLB for the plant and are used to develop the acceptance criteria necessary for each component to perform its intended function. As such, it is not necessary to revisit CLB design-basis accidents in order to determine the appropriate acceptance criteria. Second, even embrittled RVI components will not fail without the presence of a crack. The inspections to be performed under Entergy's RVI AMP provide reasonable assurance that no cracks will be present in RVI components.

Q208. What data supports your opinion that the fracture toughness will not become so low that the RVI components can fail without the presence of a crack?

A208. [JP][AH] MRP-227-A recommends the use of the J-R curve parameters based on the test data in BWRVIP-100-A, "BWR Vessel and Internals Project, Updated Assessment of the Fracture Toughness of Irradiated Stainless Steel for BWR Core Shrouds." (EPRI 1021001NP/BWRVIP-100NP, Final Report, Rev. 1, BWR Vessel and Internals Project, Updated Assessment of the Fracture Toughness of Irradiated Stainless Steel for BWR Core Shrouds (Oct. 2010), Exhibit NRC000203) (MRP-227-A at 6-3 to 6-4, Exhibit NRC000114C). Figure 6-1 of MRP-227-A shows many J-R curves from BWRVIP-100-A (MRP-227-A at 6-6, Exhibit NRC000114C), most of which have considerable ductile crack extension, as demonstrated by the fact that the curves have a positive slope. (see Q191 for a description of the J-R test) Regarding this same figure, the nonproprietary version of BWRVIP-100-A notes that "[t]he curves in Figure 2-1 indicate a trend where there is a high potential for ductile crack extension

and additional load carrying capacity at fluences lower than 1E21 ($1x10^{21}$ n/cm^2), and lower potential for ductile crack extension at fluences greater than 1E21. Each curve shown in Figure 2-1 indicates some degree of ductile crack extension, including base and weld materials irradiated at 8E21 n/cm^2." (Ex, NRC000209 at 2-3,).

BWRVIP-100, Rev. 1 is an updated version of the BWRVIP-100-A report that contains some additional fracture toughness data from operating BWRs. The additional test data in BWRVIP-100, Rev. 1 extends the fluence range over which irradiated stainless steel shows primarily ductile behavior up to a fluence of $2x10^{21}$ n/cm^2. (Ex NRC000203, Rev. 1 at 2-20,). Even for higher neutron fluence levels between $3x10^{21}$ n/cm^2 and $6x10^{21}$ n/cm^2, the majority of the specimens exhibit ductile behavior. (Ex. NRC000203 at 2-19 to 2-20,). Although most of the test materials (60 of 71 specimens) were irradiated in operating BWRs (Ex. NRC000203 at 2-16,), the stainless steel materials are similar to those used in PWR RVI and the radiation environment is similar enough in BWRs and PWRs that the material response should be similar, given the same neutron fluence.

By comparison, the peak neutron fluence for the IP2 and IP3 core barrel girth welds is predicted to be $2.51x10^{21}$ n/cm^2 (SSER2 at 3-46, Ex. NYS000507) at the end of the period of extended operation. Based on an examination of neutron fluence estimates for generic Westinghouse RVI components in MRP-191, the majority of components in Westinghouse-design RVI will have neutron fluence at 60 years of less than $3x10^{21}$ n/cm^2 (5 dpa). (MRP-191 Table 4-6, at 4-22 to 4-29, Ex. NYS000321). Therefore, the majority of RVI components in IP2 and IP3 should still be expected to retain ductility through the period of extended operation.

Q209. What components of the IP2 and IP3 RVI will experience high neutron fluence?

A209. [JP][AH] The components that are predicted to have neutron fluence greater than $3x10^{21}$ n/cm^2 (5 dpa) at 60 years in Westinghouse-design RVI mainly include those

associated with the baffle-former assembly, lower core plate, bottom mounted instrumentation, and the lower support columns (MRP-191 at pp.4-22 to 4-29, Exhibit NYS000321-00-BD01). Although the generic screening value of neutron fluence for the core barrel is 1×10^{21} to 1×10^{22}, as previously noted, the plant-specific peak fluence for IP2 and IP3 for the core barrel is less than 3×10^{21} n/cm2. (SSER2 at 3-46, Exhibit NYS000507)

Q210. What inspections will be performed on these high fluence components under Entergy's AMP?

A210. [JP][AH] The baffle-former assembly and baffle-former bolts are subject to inspection as Primary components in Entergy's AMP in accordance with MRP-227-A (NL-12-037 Attachment 1 at pp. 40-41, Exhibit NYS000496), while the lower core plate is subject to inspection as an Existing Programs component under Entergy's AMP (NL-12-037 at p. 51, Exhibit NYS000496). MRP-227-A requires ultrasonic (UT) inspection of the baffle-former bolts for cracking. Therefore, cracks in baffle-former bolts should be detected if they exist.

Q211. Which high-fluence components will not be inspected?

A211. [JP][AH] The lower support columns and bottom mounted instrumentation columns are categorized as Expansion components, so they will only be inspected if degradation in the associated Primary component is detected. (MRP-227-A at 4-38 and 4-39, Exhibit NRC000114B).

Q212. How does Entergy's AMP ensure these high-fluence Expansion components will not fail due to cracking?

A212. [JP][AH] (MRP-191 at 6-13, Ex. NYS000321) The lower support columns are addressed by Entergy's response to A/LAI 7, which resulted in additional actions to demonstrate

that the intended functions of the columns would be maintained through the period of extended operations, considering the predicted embrittlement (SSER2 at 3-40 through 3-47, Ex. NYS000507). The condition of the bottom mounted instrumentation columns are monitored when the flux thimble tubes are inserted and withdrawn, such that difficulty in insertion of the instruments will be indicative of possible degradation of the columns (NL-12-037 at p. 49, Ex. NYS000496).

Q213. Can the RVI still perform its function if some of these "Expansion" high-fluence components fail?

A213. [JP][AH] Yes. Both the lower support columns and the bottom mounted instrumentation columns are redundant components. Further, the FMECA process determined failure of a BMI column does not threaten safe shutdown (MRP-191, Table 6-5 at 6-13, Exhibit NYS000321). For the lower support columns, PWROG-14048-P, Rev. 0, "Functionality Analysis: Lower Support Columns," December 2014 (PWROG-14048-P, Rev. 0, Functionality Analysis: Lower Support Columns (Dec. 2014), Exhibit NRC000204), is a technical report providing a generic analysis of lower support column functionality in Westinghouse-design RVI. Although this report was not referenced by IP2 and IP3, it provides a proof of principle that the lower support system for Westinghouse-design RVI can tolerate a large number of failed lower support columns and still support safe shutdown. While the report remains under NRC review for use generically, it is instructive of one potential method demonstrating this issue.

Q214. Are there any non-redundant Expansion components, and if so what happens if they fail?

A214. [JP][AH] Yes. Non-redundant Expansion components include the upper core plate, upper and lower core barrel axial welds, lower support casting, and core barrel outlet

nozzle welds. The upper core plate could jeopardize safe shutdown if it were to fail. Failure of core barrel axial welds does pose a safe shutdown risk. The FMECA analysis showed the core barrel outlet nozzle welds pose an economic risk only. The lower support casting would jeopardize safe shutdown if it failed (MRP-191 Table 6-5 at 6-16, Exhibit NRC000321).

Q215. Are non-redundant Expansion components that can jeopardize safe shutdown likely to fail?

A215. [JP][AH] No. None of these components are likely to fail, either due to conditions that create a lower risk of embrittlement, lower stresses, or both.

For the Upper Core Plate: Since the upper core plate has multiple holes, it is likely that cracking between these holes is self-limiting. This component screened out for irradiation embrittlement due to an estimated maximum upper core plate fluence of less than 1×10^{21} n/cm^2 (MRP-191 Table 4-6 at 4-23). However, additional irradiation analyses conducted by Westinghouse showed the fluence could slightly exceed the irradiation embrittlement screening limit. ("Evaluation Of WCAP-17780-P, "Reactor Internals Aging Management MRP-227-A Applicability For Combustion Engineering [(CE)] And Westinghouse [Electric Company (Westinghouse)] Pressurized Water Reactor Designs," and MRP-227-A, Applicability Guidelines For CE And Westinghouse Pressurized Water Reactor Designs, at 5, ADAMS Accession No. ML14309A484, Exhibit NRC000215). However, since the predicted fluences are only slightly above the screening value, it is unlikely this component would have a severe loss of toughness. The fluence screening limit for irradiation embrittlement is neutron fluence greater than or equal to 1×10^{21} n/cm^2 (1.5 dpa) (MRP-191 Table 3-6 at 3-8)

For the Core Barrel Axial Welds: It is likely that cracks in these welds would be self-limiting since these welds run between the girth welds and do not run the full height of the core barrel. Further, these welds were categorized as Expansion because they are less risk-

significant than the core barrel girth welds, which are the linked Primary component for the axial welds. We note that detailed analyses in MRP-232 [redacted] (Exhibit NRC000225B at 4-86,) The core barrel girth welds were originally categorized as Expansion components with the upper core barrel flange weld as the linked Primary component. However, due the fact that the upper core barrel flange weld is susceptible to SCC but not IASCC, the Staff was concerned the upper flange weld was not a good lead component for IASCC relative to the girth welds. Therefore, the Staff imposed a condition in its final SE of MRP-227, Rev. 0, requiring these welds be elevated to the Primary category.

For the Lower Support Casting: The neutron fluence for lower support casting is low (<1020 n/cm2), MRP-191 Table 5-6 at 4-28, Exhibit NYS000321), so irradiation embrittlement screened out, and low-molybdenum CASS was used, so it is unlikely to have severe thermal embrittlement. The lower support casting is also a very massive component, and primarily loaded in compression.

Q216. Can the RVI still perform its functions if any components fail?

A216. [JP][AH] Yes. Many of the components are redundant components, such as the baffle-former bolts. For these bolts, previous analyses have shown that less than 30 percent of the total number of baffle-former bolts need to be intact for structural integrity of the baffle-former assembly (MRP-227-A at A-3, Ex. NRC000114C). The FMECA process determined that failure of the BMI columns would not prevent safe shutdown, and that the probability of core damage associated with failure of the BMI columns was low.

Q217. Can RVI components function with cracks, as long as the cracks are not too big?

A217. [JP][AH] Yes, there are several examples of analyses of the capability of RVI not to fail even with cracks, under normal and design basis loading. This type of analysis is known as a flaw tolerance analysis.

Q218. What are the general characteristics of a flaw tolerance analysis?

A218. [JP][AH] Such analyses typically consider the size and orientation of flaws that are expected to exist in the components, the flaw detection capability of flaws by NDE, and the expected fracture toughness of the materials. An initial flaw size is assumed, typically based on the largest flaw that could not be detected by the inspections performed on the components. A final flaw size is determined by performing a crack growth analysis, by adding to the initial flaw size crack growth due to all applicable crack growth mechanisms (such as fatigue and SCC). A fracture mechanics evaluation is performed on the final flaw, which considers all normal and design basis loading conditions. This evaluation determines if the final flaw size is stable under these loadings.

Q219. Can you provide specific examples of flaw tolerance analyses performed for RVI?

A219. [JP][AH] Yes, two such analyses are contained in MRP-210, “EPRI 1016106, Final Report, Materials Reliability Program: Fracture Toughness Evaluation of Highly Irradiated PWR Stainless Steel Internal Components (MRP-210) (Dec. 2007), (Exhibit NRC000218), and Technical report PWROG-14048-P, Rev. 0, “Functionality Analysis: Lower Support Columns,” December 2014, (Exhibit NRC000204).

Q220. Can you elaborate on these two reports?

A220. [JP][AH] Yes. MRP-210 (Exhibit NRC000218), [redacted]

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

[redacted]

[redacted]

[redacted]

Q221. Is there any operating experience involving inspection of RVI?

A221. [JP][AH] Yes, Several PWRs have performed initial inspections of RVI following the guidance of MRP-227-A. These PWRs have all operated for more than 40 years and the initial MRP-227-A inspections were performed during the first or second refueling outage during the period of extended operation. Results of these inspections are summarized in a presentation entitled “Recent Materials Inspections of PWR Reactor Internals,” by Glenn Gardner of Dominion, at the 2015 Regulatory Information Conference. (Regulatory Information Conference Presentation Slides, “Recent Materials Inspections of PWR Reactor Internals” (Mar. 2015), Exhibit NRC000207).

Q222. What does this operating experience tell us about the probability of cracks existing in PWR RVI?

A222. [JP][AH] The results summarized in the presentation indicate no cracking has been found, with the exception of some cracking of bolts, about 1.5% of Westinghouse baffle-former bolts. Over 9000 inches of irradiated welds have been inspected. Many of these inspections have been enhanced visual (EVT-1) examinations, such as the inspections of the core barrel welds in Westinghouse-design RVI. Therefore, cracking has not been identified to date in the majority of PWR internals components. Details on inspections performed up to May 2014 were provided to the NRC Staff in Biennial Report of Recent MRP-227-A Reactor Internals Inspection Results, May 12, 2014, MRP Letter 2014-09. (ADAMS Accession Nos.

ML14135A383, ML14135A384, ML14135A385) (MRP Letter 2014-09, Biennial Report of Recent MRP-227-A Reactor Internals Inspection Results (May 12, 2014), Exhibits NRC000208A-E).

Q223. In your view, does the available information support Dr. Lahey's assertion that Entergy's Amended and Revised RVI Program is inadequate because it does not address the potential consequences of shock loads resulting from design basis accidents on severely fatigued and embrittled components?

A223. [JP][AH] To summarize our view, the available information does not support Dr. Lahey's assertion. We do not expect RVI components at IP2 and IP3 to have end-of-life fracture toughness that is so low that they could fracture under normal or design-basis loads with tiny or microscopic cracks or no cracks. This is supported by several analyses demonstrating the flaw tolerance of RVI components with end-of-life embrittlement. In addition, RVI components that would have the most severe loss of fracture toughness, other than the BMI columns, are subject to periodic inspection or additional evaluations to ensure functionality under Entergy's AMP. However, failure of the BMI columns does not prevent safe shutdown and poses a low risk of core damage. Further, operating experience from RVI inspection performed in accordance with MRP-227-A has revealed no cracking, except in a low percentage of bolts.

Q224. Are there any other reasons that you disagree with Dr. Lahey's opinion that Entergy's Amended and Revised RVI Program is inadequate because it does not address the potential consequences of shock loads?

A224. [JP][AH] Yes. Dr. Lahey implies a new analysis involving time-limited assumptions should be generated. In his testimony, Dr. Lahey states: "Indeed, NUREG-1800, Rev. 2 (Table 4.1-3) (Ex. NYS000161) identifies reduced fracture toughness of reactor vessel

internals as a candidate for a time limited aging analysis." (Lahey PFT at 23, Ex. NYS000482). Per 10 CFR Part 54, there are six criteria for TLAA's (see response to Q16), one of which is that these analyses are contained or incorporated by reference in the CLB at the time of the application for license renewal. No analyses existed or were incorporated by reference in the CLB related to loss of fracture toughness of RVI, and thus no new analysis is required by the regulations nor reasonable.

Q225. Do you agree with Dr. Lahey's statement in his testimony that RVI components will be "severely fatigued"?"

A225. [JP][AH] No. The use of the term "severely fatigued" is imprecise from an engineering perspective. If Dr. Lahey means that RVI components have initiated cracks caused by fatigue, then such cracks are identifiable by the very inspections that Entergy plans to employ as a part of its RVI AMP. If Dr. Lahey means that RVI components have a high fatigue usage with a CUF close to 1.0, then it is unlikely that a fatigue crack will have initiated, or if one has initiated then it would be identified by inspection, and hence there is no impact of the "severe fatigue" on the integrity of the RVI components. In either case, Dr. Lahey has not provided any evidence to support his claims that the RVI components are "severely fatigued" or that such a condition would impact the structural integrity of the RVI components without the presence of detectable cracks.

Q226. Are you familiar with the concept of the current licensing basis ("CLB")?

A226. [JP][AH] Yes.

Q227. Please explain the CLB and its relationship to license renewal?

A227. [JP][AH] The CLB interacts with license renewal in several ways. As defined in 10 CFR § 54.3:

> Current licensing basis (CLB) is the set of NRC requirements applicable to a specific plant and a licensee's written commitments for ensuring compliance with and operation within applicable NRC requirements and the plant-specific design basis (including all modifications and additions to such commitments over the life of the license) that are docketed and in effect. The CLB includes the NRC regulations contained in 10 CFR parts 2, 19, 20, 21, 26, 30, 40, 50, 51, 52, 54, 55, 70, 72, 73, 100 and appendices thereto; orders; license conditions; exemptions; and technical specifications. It also includes the plant-specific design-basis information defined in 10 CFR § 50.2 as documented in the most recent final safety analysis report (FSAR) as required by 10 CFR § 50.71 and the licensee's commitments remaining in effect that were made in docketed licensing correspondence such as licensee responses to NRC bulletins, generic letters, and enforcement actions, as well as licensee commitments documented in NRC safety evaluations or licensee event reports.

The CLB is significant to license renewal because 10 CFR § 54.21(a)(3) specifies that, for each structure and component subject to aging management review (that is, long-lived passive components in the scope of license renewal), applicants 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. In addition, the LRA must contain an evaluation of time-limited aging analyses that are part of the CLB.

Time-limited aging analyses, or TLAAs, are analyses that meet all six of the following criteria as defined in 10 CFR § 54.3:

> (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.

Q228. Is there a TLAA related to the capability of RVI to resist shock loads resulting from design-basis accidents?

A228. [JP][AH] No. There is a structural analysis of the capability of the RVI to resist loads resulting from design-basis accidents, which is summarized in the IP2 and IP3 FSARs. However, this analysis was not identified as a TLAA in the LRA.

Q229. Describe the structural design analyses performed for the IP2 and IP3 RVI?

A229. [JP][AH] For IP2, the FSAR, Rev. 25 Chapter 14 at 117 (Indian Point, Unit 2, Updated Final Safety Analysis Report (UFSAR), Rev. 25, Chapter 14 - Safety Analysis (2014), Exhibit NRC000206), states that "the reactor internal components of Indian Point Unit 2 are not ASME Code components, because Sub-section NG of the ASME Boiler and Pressure Code edition applicable to this unit did not include design criteria for the reactor internals since its design preceded Subsection NG of the ASME Code. However, these components were originally designed to meet the intent of the 1971 Edition of Section III of the ASME Boiler and Pressure Vessel Code with addenda through the Winter 1971." For IP3, the FSAR, Chapter 16 at 18 (Exhibit NRC000224), states that the stress criteria established in Section III of the ASME Boiler and Pressure Vessel Code, Article 4, were adopted as a guide for the design of the internals and core with exception of those fabrication techniques and materials which were not covered by the Code, such as the fuel rod cladding. The FSAR, Chapter 16 at 18 also states that the components were designed under the basic principles of:

1) Maintaining distortions within acceptable limits,

2) Keeping the stress levels within acceptable limits, and

3) Preventing fatigue failures.

In the ASME Code, Section III design process, which the design of the IP2 and IP3 RVI essentially followed, the stresses and strains experienced by components must not exceed the allowable stresses and strains for the material, under all normal and design basis loading. However, the ASME Code, Section III design process does not include the assumption of any flaws such as cracks in components, instead using multipliers on loads to account for imperfections in the components. The nondestructive examinations performed during fabrication and during preservice and inservice inspections are effective in identifying flaws in the components.

Q230. Do the designs of the RVI for IP2 and IP3 have limits on the allowable deflection during design basis conditions?

A230. [JP][AH] Yes. For IP2 and IP3, the acceptance criteria for RVI under design-basis accidents are based on limiting permanent deflection of several RVI components, including the core barrel, the upper core internals, the control rod guide tubes, and flux thimbles (Exhibit NRC000206, Rev. 25 Ch. 14 at 114,), (IP3 FSAR Rev. 04 Chapter 14 (2011), at 129, Exhibit NRC000223). The allowable deflections for the core barrel is an inward deflection of 4.1 inches or an outward deflection of 1.0 inches (Exhibit NRC000206 Table 14.3-14, Ch. 14 at 162,), (Exhibit NRC000223 at Table 14.3-11, 196). Although the axial deflection allowed for the upper internals is smaller (0.100 inch), the upper internals will accumulate less neutron fluence than the core barrel and thus will maintain higher fracture toughness.

In addition, the reactor internals are designed to accommodate a complete loss of the normal support of the lower internals that results in a core drop. The IP2 FSAR Ch. 3 at 39

(Indian Point, Unit 2, Updated Final Safety Analysis Report (UFSAR), Rev. 25, Chapter 3 – Reactor (2014), Ex. NRC000205) states that:

> in the event of downward vertical displacement of the internals, energy absorbing devices limit the displacement by contacting the vessel bottom head. The load is transferred through the energy devices of the internals. The energy absorbers, cylindrical in shape, are contoured on their bottom surface to the reactor vessel bottom head geometry. Their number and design are determined so as to limit the forces imposed to less than yield. Assuming a downward vertical displacement, the potential energy of the system is absorbed mostly by the strain energy of the energy absorbing devices.
>
> The free fall in the hot condition is on the order of 0.50-in. and there is an additional strain displacement in the energy absorbing devices of approximately 0.75-in. Alignment features in the internals prevent cocking of the internals structure during this postulated drop. The control rods are designed to provide assurance of control rod insertion capabilities under this assumed drop of internals condition. The drop distance of about 1.25-in. is not enough to cause the tips of the shutdown group of rod cluster control assemblies to come out of the guide tubes in the fuel assemblies.

The IP3 FSAR describes an identical design feature (IP3 FSAR Rev. 04 Chapter 3 (2011) at 40-41, Exhibit NRC000222). Therefore, the core barrel could theoretically suffer a complete circumferential break causing a core drop, and safe shutdown could still be accomplished.

Q231. You mention “allowable limits” for RVI component deflection that are contained in the FSAR. Do these allowable limits represent a condition that would indicate a loss of function of the component?

A231. [JP][AH] No, the “allowable limits” generally contain a margin between the “No loss-of-function limit” in the FSAR and the limit that is permitted within the design.

Q232. If the RVI components at IP2 or IP3 were found to be degraded, for example with a finding of cracks, how would the presence of the cracks be evaluated compared to the design of the RVI component?

A232. [JP][AH] One way would be to perform structural analyses of the RVI components, to compare the structural response of the degraded RVI component to the allowable limits in the FSAR, as described previously. This analysis would consider the configuration of the degraded RVI component, including the size of the crack and the material properties (such as the strength and fracture toughness) of the RVI component material, including irradiation effects, if applicable. The component would be acceptable for continued operation if the component were found to continue to satisfy the allowable limits.

Q233. Do the regulations applicable to license renewal require a re-evaluation of the design basis accident analyses?

A233. [JP][AH] No, 10 C.F.R. Part 54 is clear that the focus of the license renewal review is to assess the adequacy of aging management to ensure that the intended functions(s) will be maintained consistent with the CLB for the period of extended operations, and not that the CLB must be re-evaluated including design-basis accident analysis. One exception exist for the consideration of a TLAA assumption in the analysis under 10 C.F.R. § 54.21(c).

Q234. In your expert opinion, if the RVI AMP is implemented appropriately, would it identify and take appropriate corrective actions to ensure the CLB is preserved during the period of extended operation?

A234. [JP][AH] Yes, if implemented appropriately, the Entergy RVI AMP would identify and take appropriate corrective actions to ensure that the CLB is preserved during the period of extended operations.

ANALYSIS OF DR. LAHEY ASSERTION REGARDING PREEMPTIVE REPLACEMENT

Q235. Please elaborate on your reasons for disagreement with Dr. Lahey's assertion that Entergy's Amended and Revised RVI Program is inadequate because it does not provide for proactive repair or replacement of RVI components, but rather relies on a "wait and see" approach for aging management that relies on detection of cracks or other aging effects prior to part repair or replacement.

A235. [JP][AH] We agree that Entergy's RVI Inspection Plan relies primarily on inspections to assess the condition of the RVI components rather than preemptive repairs or replacement. The use of condition monitoring by inspections is entirely consistent with the requirements for license renewal, which state that 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."

Note that the regulations do not require proactive repair or replacement but rather that the effects of aging do not affect the ability to perform intended functions. Branch Technical Position RLSB-1, "Aging Management Review – Generic," Appendix A of NUREG-1800, Rev. 2, indicates that aging management programs are generally of one of four types: prevention, mitigation, condition monitoring, and performance monitoring. Condition monitoring programs inspect for the presence and extent of aging effects (NUREG-1800, Rev. 2 at A1-1). Therefore, managing aging via a condition monitoring program is consistent with both the relevant NRC regulations and NRC guidance related to aging management.

Although Entergy's RVI AMP is a condition monitoring program, it does not preclude repairing or replacing RVI components if warranted by operating experience. An example is the replacement of the split pins at IP2 and IP3 with SCC-resistant materials, in response to

industry experience in other Westinghouse-design RVI with failures of split pins due to SCC. In addition, repair or replacement of degraded RVI components would be among the corrective actions that Entergy could implement, if necessary.

Q236. Does Dr. Lahey indicate any preference for replacing aging components over additional inspection or analysis?

A236. [JP][AH] Dr. Lahey clearly favors proactive or preemptive repair and replacement as the preferred approach to ensure continued safe operation of IP2 and IP3, and preferably before the units enter the period of extended operation.

Q237. Is it appropriate or necessary to require the preemptive replacement of all components that may be subject to aging effect?

A237. [JP][AH] No, it is not neither appropriate nor necessary to require the preemptive replacement of all components. While the preemptive replacement of components subject to aging effects would be one way to effectively manage aging effects, it is not the only acceptable method and may be undesirable when the entire issue is examined holistically. For an AMP like the RVI AMP, inspections can be and are determined to be adequate for managing aging effects in various related programs proposed by the applicant, including programs that have not been challenged by New York. Applicants could use an analysis similar to 10 C.F.R 54.21(c) (TLAAs) to show that a component will maintain its function throughout the period of extended operation. As such, it would be inappropriate to eliminate alternative methods for compliance due to a preference for preemptive replacement.

Q238. Do NRC regulations require preemptive replacement of components?

A238. [JP][AH] No. NRC regulations do not require preemptive replacement of components for license renewal. The NRC regulation at 10 CFR § 54.21(a)(3) requires that the effects of aging be appropriately managed and does not state that preemptive replacement is the only way to manage aging. The NRC issued guidance that identifies condition monitoring programs as one class of AMPs that may be used to manage the effects of aging. (Ex. NYS000161, NUREG 1800, Appendix A.1).

Q239. Does operating experience suggest widespread replacement of RVI components is needed?

A239. [JP][AH] No. As discussed previously in response to Q221 through Q222, the inspections to date have identified little or no occurrence of aging effects leading to cracks or degradation such that the intended functions of the RVI might be jeopardized.

Plants with Westinghouse-designed RVI that have performed these baseline inspections include Ginna (41.6 years/33.51 EFPY), Surry Unit 1 (40 years/29.6 EFPY), Surry Unit 2 (40 years/30.04 EFPY), Point Beach, Unit 1 (42 years/33.64 EFPY), Point Beach, Unit 2 (41 years/34 EFPY), H. B. Robinson, Unit 2 (43 years/31.4 EFPY), and Prairie Island, Unit 2 (39 years, 34 EFPY, baffle-former bolts only).

Q240. Dr. Lahey's testimony mentions control rod guide tube support pins (split pins) as the only RVI components at IP2 and IP3 that have been replaced. Do you agree with this?

A240. [JP][AH] Yes.

Q241. Why were split pins replaced at IP2 and IP3?

A241. [JP][AH] Split pins were replaced because there was operating experience with split pin failures due to SCC in the 1980's. Additionally split pins are difficult or impossible to inspect without disassembly of the upper internals. Therefore, the licensee decided to replace the split pins with SCC-resistant materials rather than to periodically disassemble the upper internals to inspect them.

Q242. Dr. Lahey mentions that split pins will be replaced for the second time at IP2. Why are the IP2 split pins being replaced again?

A242. [JP][AH] The IP2 split pins were replaced initially in the 1990's with a more SCC-resistant version of the same alloy used in the original split pins, Alloy X-750. However, industry experience subsequently showed that even improved X-750 material could experience SCC and that cold-worked Type 316 stainless steel was a better material choice for the pins. Therefore, Entergy plans to replace the IP2 split pins with cold-worked Type 316 stainless steel pins. Cold-worked Type 316 stainless steel was used for the initial replacement of IP3's split pins.

Q243. Dr. Lahey uses the fact that Entergy does not plan to preemptively replace the baffle-former bolts as an example of Entergy's "wait-and-see attitude" toward RVI degradation. Do you agree with this assertion?

A243. [JP][AH] No. Based on the low incidence of baffle-former bolt failures (1.5%) and the large degree of redundancy in the baffle-former assembly, it is unreasonable to contend that preemptive replacement is warranted. Three-loop Westinghouse design PWRs like IP2 and IP3 have on the order of 1000 baffle-former bolts in the baffle-former assembly, so the failure of, for example 15 out 1000 bolts is unlikely to impair the functionality of the baffle-former assembly.

ANALYSIS OF VISUAL AND OTHER INSPECTION TECHNIQUES

Q244. Please elaborate on your reasons for disagreement with Dr. Lahey's assertion that Entergy's Amended and Revised RVI Program is inadequate because it relies on VT-3 inspection techniques for many components, which Dr. Lahey claims is inadequate for detecting cracking, degradation, or wear prior to failure.

A244. [JP][AH] For several reasons, we disagree with Dr. Lahey's assertion that Entergy's RVI AMP is inadequate because it relies on VT-3 visual examinations rather than ultrasonic inspection. First, Entergy's AMP does not primarily rely on VT-3 inspections, as EVT-1 and UT inspections are also used. VT-3 inspection is specified only for four of ten Primary components, one of nine Expansion components, and six of seven Existing Programs components. Since many of the Existing Programs components credit the ASME Code, Section XI Inservice Inspection Program, which specifies VT-3 inspections for core support structures, it is not surprising that VT-3 is used for many components in this category. Also, VT-3 visual examination is proposed by Entergy when it is the most appropriate method for detecting the degradation mechanism of interest and this method is consistent with MRP-227-A recommendations. Inspection methods in MRP-227-A were chosen based on effectiveness in detecting the relevant aging effects. For example, VT-3 is an effective and appropriate method for detecting loss of material due to wear. In the ASME Code definition of VT-3, wear is one of the conditions that is relevant for this method. Components that are inspected for wear using VT-3 in Entergy's RVI AMP include clevis insert bolts and the upper core plate alignment pins. Operating experience regarding cracking of clevis insert bolts at another plant, and why VT-3 inspection remains appropriate for the inspection of the clevis insert bolts, is addressed in Q286 through Q295

Also, VT-3 is specified for detecting cracking only for components that are a part of a redundant population such that failures of individual components can be tolerated, or for components which have lower safety consequences of failure. VT-3 visual inspection has proven to be effective in finding evidence of failures of these types of redundant components. Since the ASME Code, Section XI specifies VT-3 for RVI, prior to implementation of the industry program in accordance with MRP-227-A, no other examination method was generally used for RVI, with the exception of ultrasonic testing of some bolts, particularly baffle-former bolts. Appendix A to MRP-227-A lists numerous of examples of degradation detected visually including degradation in clevis insert bolts, baffle former bolting at one Westinghouse-design plant, flow-induced vibration failures of thermal shields, wear of radial keyways and clevis inserts, and baffle-to-baffle bolts in a Babcock & Wilcox-design RVI (MRP-227-A at A-2 to A-6 Exhibit NRC000114C).

Q245. What are the types of components in Entergy's RVI AMP that will be inspected via VT-3 visual examinations?

A245. [JP][AH] The components inspected via VT-3 visual examination in Entergy's AMP include the following:

- Primary Components: Control Rod Guide Tubes – Guide Plates (Cards), Baffle-Former Assembly – Baffle-Edge Bolts, and Thermal Shield Assembly – Thermal Shield Flexures;
- Expansion Components: Bottom Mounted Instrumentation System – Bottom Mounted Instrumentation (BMI) Column Bodies; and
- Existing Programs Components: Upper Internals Assembly – Upper Support Ring or Skirt, and Lower Internals Assembly – Lower Core Plate.

Q246. Why is VT-3 an appropriate inspection method for the Control Rod Guide Tube Assembly – Guide Plates (Cards)?

A246. [JP][AH] VT-3 is an appropriate inspection method for the guide cards because VT-3 is an appropriate inspection method for loss of material due to wear, which is the aging effect of concern for this component. In addition, EPRI has issued interim guidance for guide card inspections. These inspections quantify the amount of wear to the guide card slots using scaled measurements on photographs.

Q247. Why is VT-3 an appropriate inspection method for the baffle-edge bolts?

A247. [JP][AH] VT-3 is an appropriate inspection method for the baffle-edge bolts because, although these bolts screened in for cracking due to IASCC and fatigue, the baffle-edge bolts are not required for structural integrity of the baffle-former assembly, as noted in the response to Request for Additional Information (RAI) #2 of RAI Set 1 related to the Staff's review of MRP-227, Rev. 0,
(MRP-227-A, Exhibit NRC000114C, at PDF page 35). VT-3 is also the most appropriate method to detect irradiation stress relaxation, which can be indirectly detected by looking for signs that the bolted joint has loosened. [redacted] In addition the baffle-edge bolts prevent separation of baffle plates at the corner. Separation could be detected via the VT-3 examination.

Q248. Why is VT-3 an appropriate inspection method for the baffle-former assembly?

A248. [JP][AH] VT-3 is the most appropriate inspection method for the baffle-former assembly, because the main aging effect of concern is distortion due to void swelling. The baffle-former assembly consists of the baffle plates, formers, baffle-former bolts, and baffle-

edge bolts. The baffle-former bolts and baffle-edge bolts have separate specified inspections in Entergy's AMP. The focus of the baffle-former assembly inspection is the baffle plates, the baffle-edge bolts and the indirect effects of void swelling in the former plates. The former plates are not directly accessible for visual inspection, but swelling of the formers would be detectable via distortion of the baffle plates. VT-3 is specified to inspect for the aging effect of distortion due to the mechanism void swelling, or cracking due to IASCC that results in either (1) abnormal interaction with fuel assemblies, (2) gaps along high fluence baffle joint, (3) vertical displacement of baffle plates hear high fluence joints, or (4) broken or damaged edge bolt locking systems along high fluence baffle joints. VT-3 visual examination is more effective than ultrasonic testing for detecting evidence of distortion such as gaps between plates. Although IASCC is screened in, detailed irradiation and stress analysis showed that IASCC is not an issue for the baffle plates, because the stresses are sufficiently low. (MRP-227-A, MRP-227 Roadmap, at B49, Exhibit NRC000114F).

Q249. Why is VT-3 an appropriate inspection method for the thermal shield flexures?

A249. [JP][AH] VT-3 is an appropriate inspection method for the thermal shield flexures because these are redundant components. Since complete fracture of one thermal shield flexure can be tolerated, VT-3 is an appropriate inspection method.

Q250. Why is VT-3 an appropriate inspection method for the Bottom Mounted Instrumentation System – BMI Column Bodies?

A250. [JP][AH VT-3 is appropriate for the BMI column bodies because failure of a BMI column body represents an economic risk only and does not jeopardize safe shutdown per the FMECA results (MRP-191, Table 6-5 at 6-13, Exhibit NYS000321), and because, although categorized as an Expansion component, the BMI column bodies are only inspected if there is

difficulty of insertion/withdrawal of flux thimbles. Therefore, there would already be an indication of a problem prior to conducting the VT-3 inspection.

Q251. Why is VT-3 an appropriate inspection method for the Upper Internals Assembly, and the Upper Support Ring or Skirt?

A251. [JP][AH] VT-3 is an appropriate inspection method for the upper Internals Assembly, Upper Support Ring or Skirt[2] because the FMECA analysis indicated only economic consequences of failure of this component. (MRP-191 Table 6-5 at 6-13, Exhibit NYS000321) Further, although SCC and fatigue both screened in for this component, SCC is extremely unlikely based on operating experience, and fatigue analysis indicates fatigue crack initiation is not expected. Although SCC screened in for this component due to the presence of structural welds, operating experience in PWR internals has shown no SCC of austenitic stainless steel except in threaded fasteners. Therefore, SCC is unlikely for this component. [redacted]

Q252. Why is VT-3 an appropriate inspection method for the Lower Internals Assembly – Lower Core Plate?

[2] Table 5-4 of NL-12-037 at p. 51, notes that IPEC has a tophat design, which includes no support ring or skirt, and the vertical sections of the tophat will be inspected.

A252. [JP][AH] VT-3 is an appropriate inspection method for the lower core plate (LCP) because the applicable cracking mechanisms for this component have been shown to be unlikely by analysis, or are being managed by another AMP, and VT-3 is the most appropriate method for detecting the most likely aging effect of loss of material due to wear. The LCP screened in for cracking due to IASCC and fatigue. Other screened in aging effects include wear and IE. Detailed irradiation and stress analyses showed that IASCC was unlikely because the stresses are too low. (MRP-227-A, MRP-227 Roadmap, at B49, Exhibit NRC000114F) Entergy also stated that it would use its Fatigue Monitoring Program to manage the effects of fatigue on the RVI rather than the RVI Program, (SSER2 at 3-52, Exhibit NYS000507). Entergy made Commitment 49 to recalculate all the CUF values for the IP2 and IP3 RVI to account for environmental effects, and to take corrective actions as necessary if the resulting CUF_{en} values exceed 1.0, which could include repair or replacement of components (SSER2 at 3-52, Exhibit NYS000507). Therefore, fatigue is not an issue since fatigue cracks are not expected to initiate if CUF_{en} values remain below 1.0, or Entergy will take corrective actions. Based on the above, since neither IASCC nor fatigue are expected to initiate cracking in the LCP, in our opinion VT-3 visual inspection of the LCP is sufficient, and is appropriate to detect evidence of loss of material due to wear, which is more likely.

ANALYSIS OF FATIGUE ERROR ANALYSIS

Q253. Please elaborate on your reasons for disagreement with Dr. Lahey's opinion that the CUF_{en} calculations for RVI are inadequate because some of the resulting CUF_{en} values are close to 1.0, and no error analysis has been performed, and that, therefore, components could have fatigue cracking since error in the CUF_{en} analysis could result in actual values greater than 1.0.

A253. [AH][GS] First, to clarify semantics, we 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. 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 Section IIII, 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.

Furthermore, Staff does not agree with Dr. Lahey's testimony and use of the citation 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"). 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 (k_{eff}) of the system is less than criticality criterion. *See* 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}. *See* Spent Fuel Report at 9 through 16 (Ex. NYS000348). The report cited by Dr. Lahey is therefore consistent with our testimony 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.

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.

Q254. Dr. Lahey also questions the use of WESTEMS^{TM} because the Staff had indicated that it should not be used when conducting analyses under the ASME Subsection NB-3600. Did the Staff indicate that WESTEMS^{TM} should not be used for any licensing applications?

A254. [AH][GS] Yes, in certain situations. 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} reanalyses. Furthermore, he is also concerned about the analytical framework employed by the WESTEMS™ computer code. See Revised Lahey at 72 (Ex. NYS000482). 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 Safety Evaluation Report (NUREG-1930, Supplement 1, August 2011, Exhibit NYS000160, "SER Supp. 1"):

> …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.

See Revised Lahey at 71 and 72 (Ex. NYS000482).

This is not an accurate restatement of SER Supp. 1. 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 ASME Section III NB-3600 fatigue analyses. See SER Supp. 1 at 4-2 and 4-3 (Ex. NYS000160). As stated in SER Supp. 1, Entergy provided the two commitments (Commitments 44 and 45) regarding use of WESTEMS™ based on documented concerns by the Staff in a separate review related to the AP1000 design certification application. See 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 Regulatory Issue Summary (RIS) 2011-14, Metal Fatigue Analysis Performed by Computer Software, (December 2011) (Ex. NRC000112) ("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 related to the WESTEMS™ code, Commitment 44 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. See 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 Commitment 45 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 internal components. Therefore, the

WESTEMS^{TM} NB-3600 module was not used by Entergy in their fatigue calculations for reactor vessel internal components. Instead, Westinghouse used another module of WESTEMS^{TM} based on the methodology of ASME Section III NB-3200. Therefore, Dr. Lahey's testimony regarding the NB-3600 module of the WESTEMS^{TM} computer code is not relevant with respect to Entergy's CUF_{en} calculations. Furthermore, the Staff noted that the WESTEMS^{TM} NB-3600 module was not used in the reanalyses described in WCAP-17199-P and WCAP-17200-P because Entergy indicated that the calculations were performed in accordance with ASME Section III NB-3200. See WCAP-17199-P and WCAP-17200-P 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^{TM} fatigue software during the Salem license renewal process. The Staff's review of Salem Nuclear Generating Station's use of WESTEMS^{TM} 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^{TM} by Salem Nuclear Generating Station. See 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 ASME Section III 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. See NUREG-2101 at 3-168 and 3-169 (Ex. ENT000195). No further concerns were identified with the method in which the WESTEMS^{TM} NB-3200 module performs fatigue calculations in accordance with ASME Section III 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.

Q255. With respect to Entergy's use of WESTEMS™ for Indian Point, was that an acceptable use of the computer code?

A255. [AH][GS] Yes. Entergy's use of WESTEMS™ for Indian Point used the ASME Section III NB-3200 module of the WESTEMS™ computer code. As described in the answer to the previous question, the Staff finds the use of this module acceptable.

Q256. Dr. Lahey asserts that the fatigue analysis for pressurizer spray nozzle, regenerative heat exchanger, RV control rod housing, CRDMS Upper Joint Canopy are inadequate. Are any of these components part of the reactor vessel internal AMP?

A256. [JP][AH] No these items are not in the RVI, and are not addressed by the RVI AMP.

Q257. Dr. Lahey raises similar concerns regarding reactor coolant pressure boundary components. Are reactor coolant pressure boundary components part of the reactor vessel internal AMP?

A257. [JP][AH] No, these items are not in the RVI, and are not addressed by the RVI AMP.

ANALYSIS OF OTHER ISSUES RAISED IN DR. LAHEY'S TESTIMONY

Q258. Do you agree with Dr. Lahey's identification of additional components that should be monitored in the RVI AMP?

A258. [JP][AH] We disagree with Dr. Lahey's opinion that control rods should be considered RVI, as stated in Dr. Lahey's testimony at p.13 (Lahey PFT at 13, Exhibit NYS000482).

Q259. Why are the control rod drive mechanisms and the control rods not part of the AMP for reactor vessel internals?

A259. [JP][AH] Control rods are not subject to aging management because they are not long-lived and passive components and thus they are outside the scope of aging management for license renewal purposes.

Q260. What is the difference between active and passive components?

A260. [JP][AH] Per 10 CFR § 54.21(a)(1), components and structures that are subject to an aging management review are those

(i) That perform an intended function, as described in § 54.4, without moving parts or without a change in configuration or properties.

(ii) That are not subject to replacement based on a qualified life or specified time period.

Components that perform an intended function, as described in § 54.4, without moving parts or without a change in configuration or properties, are commonly referred to as "passive." Components that perform their intended functions with moving parts or a change in

configuration or properties are commonly referred to as "active."

Q261. Are the control rod drive mechanisms subject to other regular inspections?

A261. [JP][AH] Control rods are subject to surveillance testing per the Technical Specifications of both IP2 and IP3. Rod drop testing is performed for each rod prior to criticality after each reactor head removal per Surveillance Requirement 3.1.4.3. Rod freedom of movement is also tested every 92 days (quarterly) per Technical Specification surveillance requirement 3.1.4.2. Any problems with control rod operability would be identified by these tests. (IP2 TS 3.1.4, exhibit NRC000228, IP3 TS 3.1.4 Exhibit NRC000229).

Q262. In Dr. Lahey's testimony the issue of boric acid leaks from the control rod drive penetrations is one of his concerns, is there a program that is already being performed to monitor for this occurrence?

A262. [JP][AH] Yes.

Q263. What program monitors the control rod drive mechanism penetrations for boric acid leakage?

A263. [JP][AH] The Reactor Vessel Head Penetration Inspection Program (LRA Section B.1.31) monitors the control rod drive mechanism penetrations in the reactor pressure vessel head for leakage using bare metal visual inspections of the top of the RPV head to detect boric acid leakage, and nonvisual (ultrasonic or eddy current examinations) from under the RPV head to detect cracks in the penetrations. (NUREG-1930 at 3-48, Exhibit NYS00326B-00-BD01) In addition, the Boric Acid Corrosion Prevention Program (LRA Section B.1.5) complements the Reactor Vessel Head Penetration Inspection Program by performing visual inspection of the reactor vessel head at locations specified by IP2-specific and IP3-specific plant procedures.

(NUREG-1930 at 3-50, Exhibit NYS00326B-00-BD01) The Staff noted that these procedures provide general guidance for performing the system walkdowns and bare metal visual examinations of both the IP2 and IP3 upper RV closure heads and other ASME Code Class 1 components for evidence of boric acid leakage, boric acid residues, or signs of corrosion. In NUREG-1930 Section 3.0.3.1.12, the NRC Staff found that Entergy's Reactor Vessel Head Penetration Inspection Program was consistent with GALL AMP XI.M11A, "Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Water Reactors."

Q264. Dr. Lahey implies that the proposed aging management of the control rods and their associated guide tubes, plates and welds is inadequate because of coverage limitations of the nondestructive testing related to the J-groove welds. (Lahey PFT at 45, Exhibit NYS000482). Do you agree with Dr. Lahey's concerns?

A264. [JP][AH] No. The inspections identified by Dr. Lahey are outside the scope of the RVI AMP. In NUREG-1930, the NRC Staff noted that it reviewed safety evaluations that granted relief for reduced inspection coverage for IP2 and IP3 and stated that the inspection bases granted in these safety evaluations are acceptable because they are in conformance with the required inspection bases that are defined in the "detection of aging effects" and "monitoring and trending" program elements in GALL Report AMP XI.M11A. (NUREG-1930 at 3-49 to 3-50, Exhibit NYS00326B-00-BD01).

Q265. Do you agree with Dr. Lahey's assessment of Entergy's ordering new reactor vessel closure heads and his speculation the reason for replacing the closure heads was due to safety concerns from reduced inspection coverage?

A265. [JP][AH] No. The reactor pressure vessel closure heads concerns identified by Dr. Lahey are outside the scope of the RVI AMP. Entergy is fully in compliance with all NRC

requirements, including aging management requirements, for the reactor pressure vessel closure heads. Because degradation has been found in the J-groove welds cited by Dr. Lahey of many plant original reactor vessel heads, licensees typically choose replacement of reactor pressure vessel closure heads for economic reasons, since less frequent volumetric inspection is required for the more SCC-resistant penetration materials, including J-groove welds used in the replacement heads.

Q266. Can you explain the difference between an inspection program and analysis program for AMPs?

A266. [JP][AH] Yes. An inspection program specifies inspections, such as visual and nondestructive examinations, to detect evidence of aging degradation such as cracking, wear, or distortion. An analysis program would involve analysis or calculations to demonstrate that components would remain capable of performing their intended functions at the end of plant life, considering the expected aging degradation, such as loss of fracture toughness, and does not require inspections to verify component condition.

Q267. Do you agree with Dr. Lahey's discussion of the transition of materials fracture behavior from ductile failure to brittle failure as impacting the performance of the RVI including the vessel interior wall and associated internals?

A267. [JP][AH] No.

Q268. What part of that analysis do you disagree with?

A268. [JP][AH] The RVI are constructed from austenitic stainless steels and nickel-based alloys which do not experience a ductile-to-brittle transition.

Q269. Do you agree with Dr. Lahey's analysis regarding the failure and relocation of the reactor vessel internal components subject to transient and accident loads as one reason that Entergy's RVI AMP is inadequate?

A269. [JP][AH] No.

Q270. Can you explain your concerns with Dr. Lahey's analysis?

A270. [JP][AH] Yes. The RVI components are unlikely to relocate during a design basis accident.

Q271. Please explain why the RVI are not likely to relocate in a design basis accident in a manner that would preclude or impact core coolability?

A271. [JP][AH] By the nature of design basis analysis, potentially affected structures and components, such as the RVI, are designed to withstand the loading conditions that result from the accidents, thus forming, in part the current licensing basis for the structures and components. The purpose of aging management is to maintain intended functions of structures and components consistent with the CLB. Therefore, relocation of RVI components should not occur.

The types of catastrophic failure of major RVI components that could cause core relocation are very unlikely. The inspections of RVI components that will be performed under the RVI AMP in the period of extended operation provide reasonable assurance that degradation such as cracking is not present, or if present, will be detected prior to threatening the structural integrity of the RVI. Thus the likelihood of core relocation is minimal.

Q272. Dr. Lahey raises a number of concerns with nondestructive evaluation (NDE), do you share his concerns?

A272. [JP][AH] No.

Q273. What are Dr. Lahey's concerns with NDE and NDT?

A273. [JP][AH] Dr. Lahey's two main concerns with NDE seem to be the use of VT-3 for detection of cracking, and the fact that Entergy has not developed inspection acceptance criteria for baffle-former bolts, instead relying on future preparation of a Technical Justification for the baffle-former bolts.

Q274. Why don't you share Dr. Lahey's concern about the use of VT-3?

A274. [JP][AH] With respect to VT-3, it is our opinion that it is used appropriately in Entergy's AMP because it is used when (1) it is the most appropriate technique for the most likely aging mechanism, (2) for inspection of components for which cracking is unlikely, or are redundant, and (3) has proven effective for detecting degradation in RVI. Our opinion on the VT-3 concern is detailed in our response to question Q244 through Q252.

Q275. Why don't you share Dr. Lahey's concern that a technical justification has not yet been prepared for baffle-former bolts?

A275. [JP][AH] We do not share Dr. Lahey's concern because, in our opinion, we do not need to review the technical justification prior to the inspections.

Q276. What is a technical justification, as it relates to the UT examination of baffle-former bolts?

A276. [JP][AH] According to MRP-228, "Materials Reliability Program: Inspection Standard for PWR Internals," Final Report (1016609) (July 2009), (Exhibit NYS000323) the Technical Justification is a required written report providing the detailed explanation of the examination process, including the theory of the examination technique as applied to reactor internals inspections, the essential variables of the procedure, other influential parameters important to the overall performance of the examination system, and field experience and/or mockup demonstrations supporting the capabilities of the NDE system. The Technical Justification includes a description of the UT procedure qualification, the active or anticipated material damage mechanisms and the NDE process used to interpret the results of the applied examination technique. (MRP-228 at 2-1, Exhibit NYS000323)

Q277. What is Dr. Lahey's concern related to the technical justification for baffle-former bolts?

A277. [JP][AH] Dr. Lahey's concern with the technical justification for baffle-former bolts seems to be that the detailed acceptance criteria will not be developed until shortly before the inspections in 2019 and 2021 for IP2 and IP3, respectively.

Q278. In your opinion, why do the detailed acceptance criteria not need to be reviewed by the Staff to have reasonable assurance that aging of the baffle-former bolts will be adequately managed?

A278. [JP][AH] As noted in SSER2, the Staff determined that it does not need to review the technical justification to make a safety finding on the applicant's AMP because (1) MRP-227-A and the Staff's final SE for MRP-227, Rev. 0, do not specify that technical justifications

must be submitted for Staff review and approval; (2) UT examinations of baffle-former bolts have been performed since the late 1990's, thus, there is reasonable assurance that these examinations can be implemented effectively at IP2 and IP3; and (3) finalizing the technical justification closer to the date of the inspection will allow for the latest UT technology and lessons learned from previous baffle-former bolt examinations to be incorporated. (SSER2 at 3-20, Exhibit NYS000507)

Q279. If baffle-former bolts were found to have cracks, what corrective actions would be taken?

A279. [JP][AH] The Corrective Actions element of Entergy's AMP, states that any detected condition that fails to meet the examination acceptance criteria must be processed through the corrective action program, and that example methods for analytical disposition of unacceptable conditions are discussed or referenced in Section 6 of MRP-227-A. Per the guidance of MRP-227-A, bolts are not rejected based on flaw sizing but rather upon flaw detection. Therefore any bolt with a relevant indication is assumed to be non-functional. Section 6.4 of MRP-227-A recommends for bolted assemblies, such as the baffle-former assembly, that assembly-level functionality analyses should be performed to show the assembly will remain functional, taking into account the number and location of flawed bolts. Several licensees with Westinghouse-design RVI have developed minimum pattern analyses for baffle-former bolts

Q280. Are you concerned that not having a technical justification prepared at the present time presents a safety issue related to the integrity of the RVI?

A280. [JP][AH] No. It has been previously demonstrated that UT examination can detect flaws in baffle-former bolts in Westinghouse-design RVI. The percentage of flawed bolts is expected to be low based on operating experience, and the applicant will take appropriate

corrective actions, either by demonstrating the structural adequacy of the baffle-former assembly, or by replacement of bolts as necessary.

Q281. Are you familiar with the term baseline inspections?

A281. [JP][AH] Yes.

Q282. Please explain what a baseline inspection is and its use?

A282. [JP][AH] The term "baseline" is typically used to refer to initial inspections of components against which future inspection results would be compared. In ASME Section XI, a preservice inspection is required for new components that will be subject to UT examination for subsequent inservice inspections (ISI). The preservice inspection would also be done via UT.

Q283. Is Entergy performing baseline inspections for the RVI AMP?

A283. [JP][AH] Yes. Entergy's AMP and MRP-227-A require baseline inspections of most components within two refueling outages after the beginning of the period of extended operation (PEO). Inspection schedules for a few components have a different timeline and basis, such as baffle-former bolts, for which the initial inspection will be between 25 and 35 effective full power years (EFPY).

Q284. When are those inspections scheduled?

A284. [JP][AH] The initial inspections under the RVI AMP for IP2 and IP3 would occur no later than 2017 and 2019, respectively, for those components recommended to be inspected within two refueling outages of the beginning of the PEO.

Q285. Are those appropriate dates to conduct the initial inspection of RVI under the RVI AMP?

A285. [JP][AH] The dates are appropriate because they meet the requirement to perform the initial inspections under the RVI AMP no later than the second refueling outage following the start of the PEO.

Q286. Are you familiar with the operating experience concerning failures of clevis insert bolts that occurred at another Westinghouse-design plant?

A286. [JP][AH] Yes. Bolts failed in several clevis inserts at D.C. Cook, Unit 1. The apparent cause analysis determined the most likely cause of bolt failure was primary water stress corrosion cracking (PWSCC), a form of SCC affecting nickel-based alloys that can occur in normal PWR reactor coolant water chemistry. (SSER2 at 3-25)

Q287. Describe the configuration and function of the clevis insert bolts.

A287. [JP][AH] The clevis insert bolts attach the clevis inserts, which are a nickel-based alloy, to lugs on the reactor vessel. The clevis insert bolts are cap screws and are retained in place with a welded locking bar such that, even if the bolt fails, it will not become a loose part. Dowel pins go through holes in the clevis inserts into the vessel lug which are capable of supporting the weight of the clevis inserts. The lugs are made from low-alloy steel similar to the reactor pressure vessel plates and are welded to the vessel interior. The clevis inserts and bolts are part of the lower radial support system. Figure 11 shows the location of the clevis insert and Figure 12 shows a detail of how the radial keys on the core barrel interface with the clevis inserts. Figure 13 shows the configuration of the bolts and dowel pins in a D.C. Cook, Unit 1, clevis insert; IP2 and IP3 have a similar configuration. Figure 14 shows a clevis insert bolt head as it appears during the VT-3 visual inspection.

Q288. How were the clevis insert bolt failures at D.C. Cook, Unit 1 detected?

A288. [JP][AH] During the March 2010 10-year in-service inspection (ISI) of the reactor vessel at D.C. Cook Unit 1, an anomaly was observed in the lower radial support clevis inserts during a visual VT-3 examination. At D.C. Cook Unit 1, each clevis insert features eight cap screws and two dowel pins, which provide for retention of the insert. At a total of seven (out of 48) bolt locations, there was visual evidence that the bolt head had detached from the shank. On one of the clevis dowel pins, the lock welds on the pin had broken and the pin was slightly rotated and displaced deeper into the insert. Reviews of video images from the previous 10-year ISI vessel inspection showed no indications of wear, fractures, or other anomalies with the clevis insert bolts or dowel pins at any location. (Westinghouse InfoGram IG-10-1, Reactor Internals Lower Radial Support Clevis Insert Cap Screw Degradation (Mar. 31, 2010), Exhibit NRC000219).

Q289. Describe the design functions of the lower radial support system.

A289. [JP][AH] The main design function of the lower radial support system (LRSS), of which the clevis insert bolts are a part, is the prevention of tangential or rotational motion of the lower internals assembly while permitting axial displacement and differential radial expansion. These supports are designed to (1) prevent excessive tangential displacement of the lower internals during seismic events and loss-of-coolant accident (LOCA) conditions (2) to limit displacements and misalignments in order to avoid overstressing the core barrel, (3) to ensure that the control rods can be freely inserted. (SSER2 at 3-24, Ex. NYS000507)

Q290. Did the clevis insert bolt failures lead to a loss of functionality?

A290. [JP][AH] No. The failed bolts were detected visually, thus alerting the licensee that there was a problem before the function of the LRSS was compromised. (SSER2 at 3-25, Ex. NYS000507)

Q291. Is there a risk of loss of functionality if the clevis insert bolts fail?

A291. [JP][AH] No. Due to the tight clearance between the clevis inserts, RPV lugs, and the radial keys on the core barrel that interface with the clevis inserts, the clevis inserts would remain in place unless the core barrel was removed. At that point, if a clevis insert became dislodged, it would be visually observed. Also, there is a high degree of redundancy because there are six radial supports spaced evenly around the reactor vessel interior.

Q292. Under Entergy's RVI AMP, how will the clevis insert bolts be inspected?

A292. [JP][AH] The clevis insert bolts will be inspected by visual VT-3 examination as an Existing Programs component. The VT-3 examination is conducted as part of the inservice inspection of the core support structures under the ASME Code, Section XI, Inservice Inspection Program.

Q293. Is Entergy's inspection of the clevis insert bolts capable of detecting broken bolts?

A293. [JP][AH] Yes. Entergy confirmed its VT-3 inspection directly views the bolt heads, dowel pins, and locking bars. In the plant that experienced bolt failures, failed bolts were detected via anomalies in the appearance of the bolt heads, dowel pins and locking bars. (SSER2 at 3-25, Ex. NYS000507)

Q294. Has Entergy ever inspected the clevis insert bolts?

A294. [JP][AH] Yes, VT-3 examinations of the clevis insert bolts were completed in 2006 for IP2 and 2009 for IP3 with no recordable indications, meaning that no degradation was observed. (SSER2 at 3-25, Exhibit NYS000507)

Q295. In your opinion, is it safe to operate the IP2 and IP3 reactors if clevis insert bolts are not preemptively replaced?

A295. [JP][AH] Yes. Our opinion is based on the fact that failure of a large number of bolts would not prevent the function of the LRSS from being performed, because, due to the tight clearance between the radial keys and clevis inserts, the clevis inserts would be retained in place. Also, due to the redundancy in the LRSS, all six clevis inserts are not needed for the overall function of alignment and restricting tangential motion from being performed. Further, VT-3 examination has been demonstrated to be capable of detecting failures of clevis insert bolts, if a significant portion of a plant's bolts are degraded.

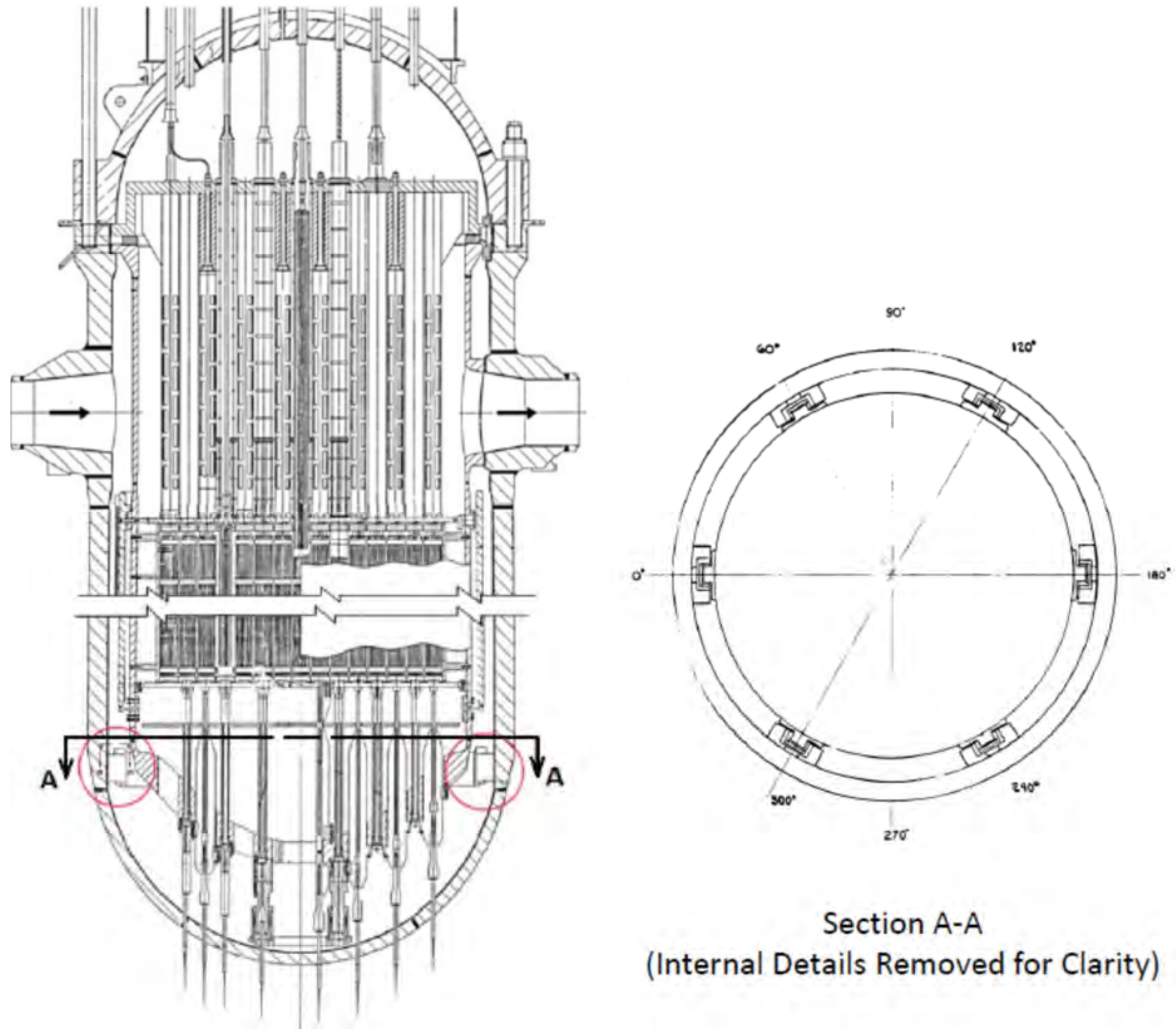

Figure 13 – Left – reactor pressure vessel cross-section showing clevis insert location; Right – Section showing plan view of clevis insert locations in lower radial support system. From Wilson, B., “Industry and NRC Coordination Meeting Materials Programs Technical Exchange: Clevis Insert Bolt Update,” slide presentation to Pressurized Water Reactor Owners Group (PWROG), June 2014, ADAMS Accession No. ML14163A520, at 3 (Exhibit NRC000220)

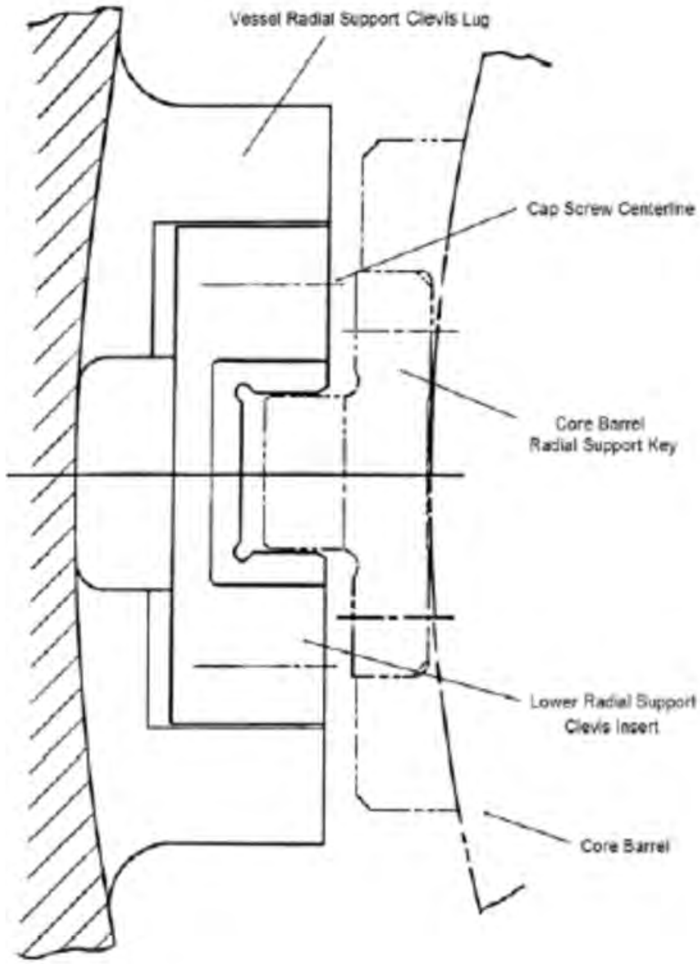

Figure 14- Plan view of clevis insert and interfacing components. From Wilson, B., "Industry and NRC Coordination Meeting Materials Programs Technical Exchange: Clevis Insert Bolt Update," slide presentation to Pressurized Water Reactor Owners Group (PWROG), June 2014, ADAMS Accession No. ML14163A520. (Exhibit NRC000220)

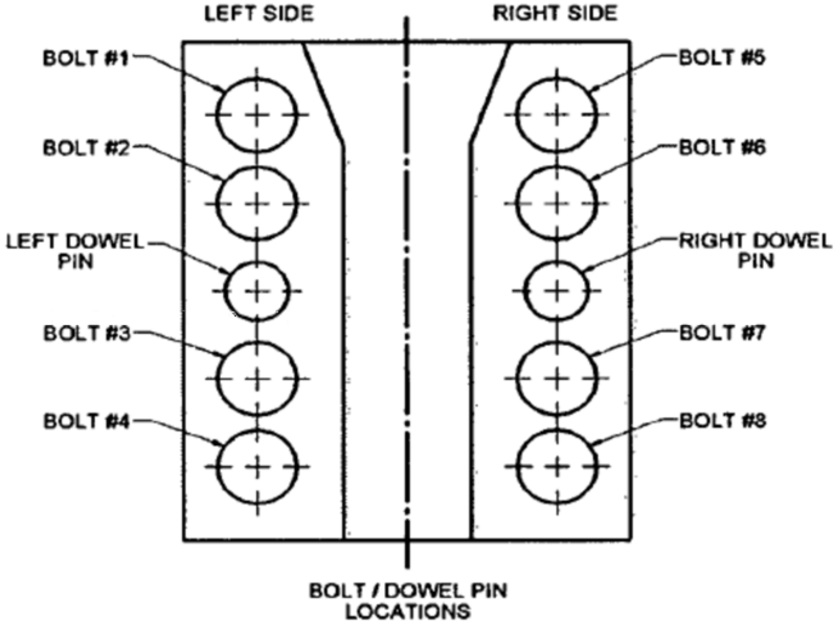

Figure 15 - Schematic of Typical Bolt Configuration for Each Clevis Insert at D.C. Cook, Unit 1 (Figure 2 From Enclosure 3: Babcock & Wilcox Report, S-1473-002, Revision 0, "Examination of Clevis Bolts Removed from D. C. Cook Nuclear Plant". Part 1 of 3. (ADAMS Accession No. ML14253A318) (Exhibit NRC000221A-F)

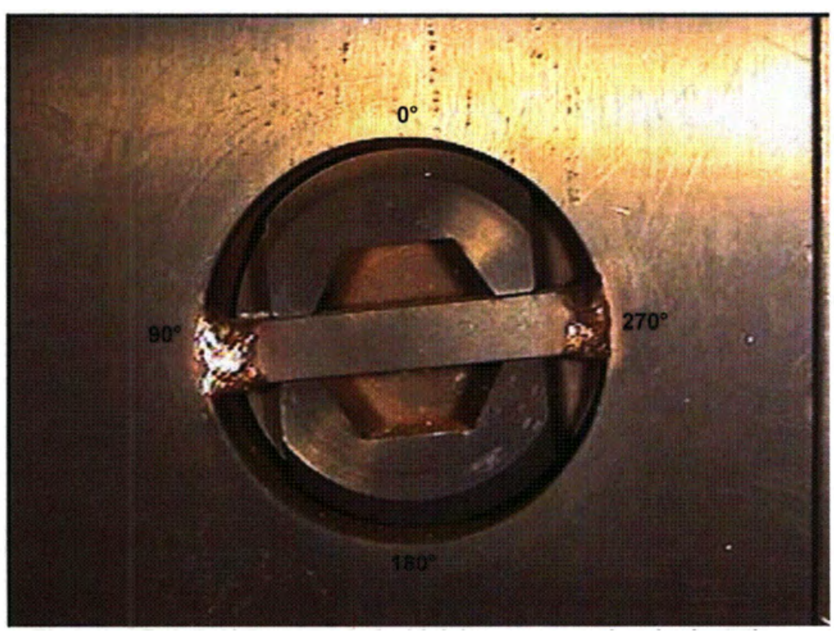

Figure 16 - Typical Appearance of Clevis Insert Bolt Head from VT-3 Visual Inspection Video at D.C. Cook, Unit 1 (Figure 6 From Enclosure 3: Babcock & Wilcox Report, S-1473-002, Revision

0, "Examination of Clevis Bolts Removed from D. C. Cook Nuclear Plant". Part 1 of 3. (ADAMS Accession No. ML14253A318) (Exhibit NRC000221A-F)

Q296. Are you familiar with Dr. Lahey's concerns regarding the core support columns?

A296. [JP][AH] Yes. Dr. Lahey's concerns regarding the core support columns can be summarized as follows:

1. The assumption that only a limited number of columns could be flawed is nonconservative.

2. The columns are inaccessible for inspection and VT-3 visual inspection offered "no meaningful information regarding the structural integrity of the columns."

3. Entergy's assumption that the lower support columns are subject to mainly "nominal operating stresses" is flawed because these components are subject to a range of normal and transient stresses.

4. Entergy's conclusion that irradiation-induced cracking is unlikely is contradicted by recent studies, including NUREG/CR-7184, which shows the "extreme sensitivity" of crack growth rate and fracture toughness to irradiation.

5. The selection of the lower core barrel girth weld as an alternate Primary component for the lower support columns, is not appropriate because the weld is a much different type of component, with respect to design, material and loading, than the core support columns.

Q297. Do you disagree with any Dr. Lahey's assertions regarding the core support columns?

A297. [JP][AH] We disagree with all of Dr. Lahey's assertions regarding the core support columns.

Q298. Could you provide a brief description of the core support columns referenced by Dr. Lahey.?

A298. [JP][AH] In Entergy's RVI Inspection Plan, the component is referred to as the lower support assembly, lower support column bodies (cast). (NL-12-037 at 48, Exhibit NYS000496) In response to a request for additional information, Entergy clarified that only the upper portion of the lower support column bodies, known as the column caps, is made from CASS. (NL-12-134 at 8, Exhibit NYS000498) The lower support columns are part of the lower support structure, shown in Figure 5, and details of the column construction are shown in Figure 9.

Q299. Please explain your disagreements with Dr. Lahey's assertion regarding of the likelihood of the core support columns containing a significant number of undetected flaws prior to installation.

A299. [JP][AH] We disagree with Dr. Lahey's assertion that the assumption that only a limited number of columns could be flawed is nonconservative based on (1) non-destructive testing results during fabrication, (2) operating experience in PWRs, and (3) fatigue crack initiation analysis.

Q300. Could you explain how the non-destructive testing results conducted during fabrication support the assumption that the core support columns are unlikely to contain many flaws?

A300. [JP][AH] Entergy demonstrated the likelihood of significant manufacturing defects in the columns is low based on the nondestructive examination that the columns were subject to during fabrication, which included liquid penetrant (PT) and radiography (RT)

examination. (SSER2 at 3-43). Those test results showed no surface breaking defects in any of the columns.

Q301. What has operating experience demonstrated regarding the core support columns?

A301. [JP][AH] Operating experience in PWRs has shown CASS material has not experienced any degradation due to any form of SCC, including IASCC. Appendix A of MRP-227-A summarizes operating experience with PWR RVI degradation, and does not note any experience with failures of CASS material (MRP-227-A at A-1 through A-7). For SCC, the screening criteria of MRP-191 only screen in CASS for SCC if the ferrite content is less than 5%, combined with stress greater than 35 ksi (MRP-191 at 3-2), neither of which applies to the lower support columns at IP2 and IP3. The delta ferrite content in CASS is known to increase its resistance to SCC. In addition, the normal operating stresses on the columns are compressive. Normal operating stresses are the long-term sustained stresses that drive SCC, as opposed to transient stresses, which affect the component only for a brief period of time. SCC does not affect components not subject to tensile stresses.

Q302. How does the fatigue crack initiation analysis support the Staff's conclusion that the program is adequate?

A302. [JP][AH] Entergy also demonstrated fatigue crack initiation is not expected during the life of the plant because it updated the fatigue analyses of the columns to account for the effects of the reactor coolant environment and the environmentally adjusted cumulative usage factor (CUF_{en}) for the columns is less than 1.0 at the end of the PEO. (SSER2 at 3-43)

Q303. Please explain why you disagree with Dr. Lahey's assertion that the columns are inaccessible for inspection.

A303. [AH][JP] We disagree with Dr. Lahey's assertion that the columns are inaccessible for inspection because the columns could be accessed through the flow holes in the LCP or the flow holes in the lower support casting with the right type of camera. Special camera delivery tooling may need to be built to accomplish this. However, EPRI considered this when developing the inspection recommendations in MRP-227-A because it specified an EVT-1 visual examination for the columns as the Expansion inspection.

Q304. Do you agree with Dr. Lahey' assertion that VT-3 visual inspection offered "no meaningful information regarding the structural integrity of the columns?

A304. [JP][AH] We agree that the VT-3 examinations of the columns conducted in the past under Entergy's ASME Code, Section XI inservice inspection program provided little or no meaningful information with respect to the structural integrity of the columns, because only the portions of the columns below the dome lower support plate (specifically the exterior bottom of the core barrel) were inspected. The portion of the lower support column bodies below the dome lower support plate is the end of the column body that extends past the lower support column nut. The portion of the lower support column bodies that was inspected was the wrought lower section. (Indian Point Nuclear Generating Unit Nos. 2 & 3 - Reply to Request for Additional Information Regarding the License Renewal Application, NL-13-122, dated September 27, 2013 (ADAMS Accession No. ML13277A007, Exhibit NYS000502, at 4). Since the bottom end of the lower support columns are retained by a nut in the dome lower support plate, it is not clear that if the upper CASS portion failed, that any anomaly would be noted from below the dome lower support plate, since the column may be retained in place by the nut.

Q305. Do you agree with Dr. Lahey's assertion that Entergy's assumption that the lower support columns are subject to mainly "nominal operating stresses" is flawed?

A305. [JP][AH] We disagree with Dr. Lahey's assertion that Entergy's assumption that the lower support columns are subject to mainly "nominal operating stresses" is flawed. Although these components can be subject to a range of normal and transient stresses, the conditions that would primarily drive the aging degradation effects of interest for the lower support columns (e.g., cracking and irradiation embrittlement) are normal, steady state operating conditions, not transient conditions.

Q306. Why would the primary driver for degradation be normal, steady state operating conditions?

A306. [JP][AH] While the columns are subject to transient conditions during normal operating transients such as startup and shutdown, and expected operating occurrences such as reactor trips, these transient conditions are short term. Comparatively, the amount of time spent at normal, steady-state operating conditions is much greater. Therefore, the potential for aging mechanisms such as SCC, IASCC, and irradiation embrittlement is properly assessed from the normal, steady-state operating conditions.

Q307. What type of stresses are the columns subject to under normal, steady-state operating conditions?

A307. [JP][AH] Under normal steady-state operating conditions, the columns are subject to mainly compressive stresses due to the nature of their design.

Q308. What about fatigue cracking, isn't that caused by transient conditions?

A308. [JP][AH] Yes. However, the number of transient cycles is limited and is tracked by the Fatigue Monitoring Program to ensure design-basis assumptions regarding the number of occurrences of each transient are not exceeded. Since Entergy has shown the fatigue CUF_{en} of the columns will remain below one through the end of the licensed operating period when considering these design-basis fatigue cycles, fatigue cracking is not expected to initiate.

Q309. Please explain your disagreements with Dr. Lahey's assertion that Entergy's conclusion that irradiation-induced cracking is unlikely is contradicted by recent studies?

A309. [JP][AH] We disagree with Dr. Lahey's assertion that Entergy's conclusion that irradiation-induced cracking is unlikely is contradicted by recent studies, including NUREG/CR-7184, which shows, in his words, the "extreme sensitivity" of crack growth rate and fracture toughness to irradiation.

Q310. Please summarize the content of NUREG/CR-7184.

A310. [JP][AH] NUREG/CR-7184 by Chen, et al., Crack Growth Rate and Fracture Toughness Tests on Irradiated Cast Stainless Steels (Revised December 2014) (ML14356A136) (Exhibit NYS000488 A/B), describes the results of recent NRC-sponsored testing of CASS material subject to both thermal aging and irradiation. Therefore, the test results could theoretically be used to assess the potential synergistic effect of IE and TE.

Q311. Why do you disagree with Dr. Lahey's assertion that NUREG/CR-7184 contradicts Entergy's conclusion that irradiation-induced cracking of CASS components is unlikely?

A311. [JP][AH] We disagree with Dr. Lahey's assertion because (1) The materials and test conditions for the results reported in NUREG/CR-7184 are not representative of the IP2 and IP3 CASS components, and (2) The results of the tests are inconclusive and tend to refute rather than support the existence of a synergistic effect on both fracture toughness and crack growth rate sensitivity.

Q312. Why are the materials tested in NUREG/CR-7184 not representative of those at IP2 and IP3?

A312. [JP][AH] The materials tested in NUREG/CR-7184 have much higher ferrite content than the CASS material in the IP2 and IP3 lower support columns. The IP2 and IP3 lower support columns all had delta ferrite content of 15% or less, while the materials tested in NUREG/CR-7184 have delta ferrite contents of 23% and greater. The sensitivity of CASS material to TE is largely a function of the delta ferrite content. In addition, one of the materials tested was a high-molybdenum grade of CASS, CF-8M. High-molybdenum CASS material is known to be more prone to loss of fracture toughness due to TE than low-molybdenum CASS material such as Type CF-8, which was used to fabricate the IP2 and IP3 lower support columns. Therefore, the materials documented in NUREG/CR-7184 would be expected to be much more likely to have loss of fracture toughness due to TE than the IP2 and IP3 lower support column materials.

Q313. Why were the test conditions reported in NUREG/CR-7184 not representative of the operating conditions of the IP2 and IP3 components?

A313. [JP][AH] The materials reported on in NUREG/CR-7184 were only irradiated to a neutron fluence of 0.08 dpa. This fluence level is only about one-fiftieth of the peak fluence predicted for the lower support columns at 60 years.

Q314. What did NUREG/CR-7184 show with regard to a synergistic effect of irradiation and thermal aging on fracture toughness of CASS?

A314. [JP][AH] With respect to fracture toughness, NUREG/CR-7184 does not prove the existence of a synergistic effect. The report found that CASS material that was already thermally aged, had relatively little additional loss of toughness due to irradiation. Figure 98 of NUREG/CR-7148 shows that the fracture toughness of CF-8 material that is both thermally aged and irradiated is only about 20-30% lower than the toughness of material that was either only thermally aged or only irradiated. Using data from Table 16 of NUREG/CR-7184, for the CF-3 material, the fracture toughness decreased by 36% due to irradiation alone, and the fracture toughness decreased 47% due to thermal aging alone. Under both thermal aging and irradiation, the fracture toughness decreased 64%. A synergistic effect would be expected to cause a loss of fracture toughness greater than would be caused by the simple combination of the individual effects. In this case the product of the remaining fracture toughness from the individual effects, would be (1-0.36) x (1-0.47) = 0.34, or a 66% decrease. Therefore, the actual decrease in toughness under both effects was (slightly) less than would be predicted by the product of the two effects. The results for the CF-8 specimen indicate a fracture toughness loss due to thermal aging alone of 56%, loss due to irradiation alone of 63%, and a loss of fracture toughness due to both of 66%, which is only a few percent greater than the loss due to irradiation alone. The product of the individual effects would be (1-0.56) x (1-0.63) = 0.16, or an

84% decrease, which is much greater than the 66% decrease actually observed. (NUREG/CR-7814 at 136, Exhibit NYS000488B) Therefore, since the observed decrease in fracture toughness is less than would be predicted due to the product of the individual effects, the fracture toughness results of NUREG/CR-7184 do not prove a synergistic effect exists; rather, they suggest the opposite is the case.

Q315. Does NUREG/CR-7184 demonstrate that the combination of irradiation and thermal aging would make cracking worse?

A315. [JP][AH] No. With respect to cracking, in the summary discussion, NUREG/CR-7184 states that only moderate crack growth rates in the range of 10^{-11} m/s were recorded in the SSC tests of CASS specimens, regardless of their thermal aging history or irradiation conditions. By comparison, MRP-227-A Section 6.0 recommends the use of the boiling water reactor hydrogen water chemistry (BWR HWC) crack growth rate curve for all PWR IASCC and SCC analyses until generic curves are established for IASCC and SCC in PWR environment. The BWR HWC curve is for wrought stainless steel and weld materials that are not susceptible to thermal aging. Comparing the crack growth rate from Figure 97 of NUREG/CR-7184 to the BWR HWC crack growth rate of Figure 6-4 of MRP-227-A (MRP-227-A at 6-7, Exhibit NRC000114B), at the same applied stress intensity factor (K) level of 20 mega-pascals-square root meters (MPa√m), the predicted crack growth rate from NUREG/CR-7184 would be $4x10^{-11}$m/s, while for the BWC HWC curve, the predicted crack growth rate would be $2.75x10^{-10}$ m/s. Therefore, thermal aging or irradiation did not increase the crack growth rates in these CASS materials, and in fact the crack growth rate is lower than the rates used for wrought stainless steel materials, which are used for most RVI components.

Q316. Please explain your disagreements with Dr. Lahey's assertion that the selection of the lower core barrel girth weld as an alternate Primary component for the lower support columns is not appropriate?

A316. [JP][AH] We disagree with Dr. Lahey's assertion that the selection of the lower core barrel girth weld, as an alternate Primary component for the lower support columns, is not appropriate because the weld is a much different type of component, with respect to design and material and loading, than the core support columns. While we agree that the lower core barrel girth weld is fabricated from significantly different materials than the lower support columns, it experiences a similar environment in terms of radiation and temperature. Further, the weld also shares some of the same aging mechanisms that are driven by neutron irradiation, such as IASCC and IE. In that respect, in our opinion the weld is a more appropriate predictor of these mechanisms than the original Primary component that is linked to the lower support columns, the control rod guide tube (CRGT) lower flange welds. The CRGT lower flange welds are expected to experience lower neutron fluence levels, thus did not screen in for IASCC. The estimated neutron fluence for the CRGT lower flanges, according to MRP-191, is $7x10^{20}$ – $1x10^{21}$ n/cm^2 (1-1.5 dpa). (MRP-191 Table 4-6 at 4-22). The lower core barrel girth welds are expected to have tensile stress due to weld residual stress, unlike the high-fluence portion of the lower support columns. Therefore, the lower core barrel girth weld is considered more likely to have cracking due to IASCC than the lower support columns. (SSER2 at 3-46) There is no CASS component in the IP2 or IP3 RVI that has predicted neutron fluence levels closer to the lower support columns, which could be selected as a Primary component for the lower support columns.

Q317. What is your opinion of Entergy's evaluation of integrity of the lower support columns, considering the possible effects of both irradiation embrittlement and thermal embrittlement?

A317. [JP][AH] In our opinion, Entergy provided sufficient information to provide reasonable assurance that cracks in columns are unlikely, and that the functionality of the lower support columns and the lower support system will be maintained through the end of plant life. This information included:

- Screening the column caps for TE and IE using plant-specific materials data and determining that the column caps are not susceptible to TE (and the Staff confirmed the results of the screening using its own screening criteria)
- Information on fabrication NDE demonstrating that pre-existing flaws are unlikely to exist in the column caps
- Information on the expected stresses and neutron fluence for the column caps that demonstrated that service-induced cracking due to IASCC is unlikely
- Entergy modified its RVI Program to include a link to a lead component that is an appropriate predictor of IASCC and IE for the lower support columns, with an appropriate schedule for performing the Expansion inspection if necessary. (SSER2 at 3-47)

Q318. What is your opinion of Dr. Lahey's assertion that "… the RPV internals will be much more embrittled than the RPV walls, which have historically been the focus of USNRC embrittlement concerns." (Lahey PFT June 9, 2015, p. 28 lines 22-23 and p. 29 line 1, Ex. NYS000482)

A318. [JP][AH] We disagree with this assertion. Although we agree with Dr. Lahey's statement that neutron fluence levels in some regions of the RVI will be up to four orders of

magnitude greater than the peak neutron fluence expected in the RPV beltline, the materials of construction of the RVI are very different and respond to irradiation differently. The RPV is constructed of low-alloy steel. Low-alloy steels begin to experience embrittlement at neutron fluences as low as 1×10^{17} neutrons/square centimeter. In low-alloy steels, the greatest rate of embrittlement occurs between neutron fluences of 1×10^{17} n/cm^2 and 8×10^{19} n/cm^2. Additional neutron fluence will cause incrementally smaller changes in fracture toughness. The RVI are constructed mainly of austenitic stainless steels. Austenitic stainless steels do not experience any significant embrittlement below a neutron fluence of 6.7×10^{20} n/cm^2, equivalent to 1 displacement-per-atom (dpa). (NUREG/CR-7027, p. 65, Exhibit NYS000487). This is approximately ten times the largest expected fluence for PWR RPVs at 60 years.

Q319. Do you have other concerns regarding Dr. Lahey's assertions regarding RVI component embrittlement?

A319. [JP][AH] Yes, we have other concerns. Another important distinction between the RVI and RPV is that the RVI are not pressure boundary components and do not experience the high pressure and thermal stresses that are experienced by the RPV during normal operating transients such as heatup and cooldown. Further, many of the RVI components exposed to the highest neutron fluence are not those relied upon for core support, thus they do not experience high loading levels.

The baffle-former assembly is directly adjacent to the active core and thus will accumulate the highest neutron fluence. However, the baffle-former assembly has a high degree of structural redundancy and robustness. Baffle plates are secured to the supporting former plates using approximately 1000 bolts in both the IP2 and IP3 RVI. While some bolts have experienced cracking due to IASCC, the overall failure rate of these bolts has only been around 1.5%. Structural analyses of minimum bolting patterns in Westinghouse-design RVI

have shown only a small fraction of the bolts are necessary to withstand the anticipated normal and accident loads. For example, MRP-227-A Appendix A indicates that a proactive minimum bolt pattern replacement at two reactors having 1088 total baffle-former bolts replaced a total of 276 bolts at one reactor and 203 bolts at the other reactor. (MRP-227-A at A-3, Exhibit NRC000114C) The lower core plate in Westinghouse-design RVI also experiences high neutron fluence. However, the lower core plate is a very thick, robust component. The core barrel is a key core support structure in Westinghouse-design RVI. The peak neutron fluence in the core barrel lower girth weld for IP2 and IP3 was calculated to be 4 dpa (SSER at 3-46).

Finally, to require the same standard for fracture toughness for the reactor pressure vessel and the RVI is not appropriate, since the reactor pressure vessel is part of the reactor coolant pressure boundary, one of the main defense in depth measures against release of radioactivity. A through wall crack is not tolerable in the reactor coolant pressure boundary and would cause a loss of the pressure retaining function of the RCPB, while the RVI have considerable structural redundancy. Many RVI components could still perform their design functions with through-thickness cracks, such as the core barrel and baffle plates.

Q320. Dr. Lahey mentions the pressurized thermal shock analysis for the IP3 RPV in his testimony. Is there a problem with Entergy's PTS analysis for IP3?

A320. [JP][AH] There is no problem with Entergy's PTS analysis for IP3. Dr. Lahey correctly notes that the limiting plate for IP3 is predicted to exceed the applicable screening criterion by 9.9 deg F. (Lahey PFT at 42, Exhibit NYS000482) However, Dr. Lahey did not mention that this prediction is for 60 years (54 effective full power years). Per NUREG-1930, IP3 is predicted to exceed the PTS screening criterion approximately 9 years after it begins its period of extended operation. (NUREG-1930 at 4-17, Exhibit NYS00326E-00-BD01). As documented in NUREG-1930, Entergy made Commitment #32 stating:

> As required by 10 CFR 50.61(b)(4), IP3 will submit a plant-specific safety analysis for plate 62803-3 to the NRC three years prior to reaching the RT PTS screening criterion. Alternatively, the site may choose to implement the revised PTS (10 CFR 50.61) rule when approved, which would permit use of Regulatory Guide 1.99, Revision 3. (NUREG-1930 at 4-17 Exhibit NYS00326E-00-BD01)

Subsequent to the publication of NUREG-1930, the alternate PTS Rule 10 CFR 50.61a was published, which is the “revised PTS rule” referred to in the commitment. Entergy therefore has the option to perform an analysis in accordance with 10 CFR 50.61a to fulfil Commitment #32. IP3 is currently in compliance with the regulation at 10 CFR 50.61. The NRC Staff concluded in NUREG-1930 that [t]he applicant's commitment will ensure that the PTS-related aging effects for IP3 will be managed during the period of extended operation, pursuant to 10 CFR § 54.21(c)(1)(iii). (NUREG-1930 at 4-17 Exhibit NYS00326E-00-BD01).

Q321. What would be the outcome if Entergy did not fulfill Commitment 32?

A321. [JP][AH] Compliance with 10 CFR 50.61 is a regulatory requirement. If it did not submit its revised analysis, Entergy would be in violation of 10 CFR 50.61. Therefore, enforcement action would be taken by the NRC, which could include plant shutdown until Entergy could demonstrate that IP3 is in compliance with 10 CFR 50.61..

Q322. From your testimony Entergy has made several commitments that relate to the issues discussed in Dr. Lahey’s testimony, including Commitments 32, 44, 45, 49 and 50. Is it your expectation that Entergy will complete these commitments?

A322 [AH] Yes. As an example, Commitment 49 has already been completed for IP2. Ex. NRC000184.

Q323. Are you familiar with the assertions in NYS-38/RK-TC 5 regarding the unenforceability of commitments?

A323. [AH] Yes.

Q324. Do you agree with New York's assertions that whether these license renewal commitments would not be enforceable by the Staff, New York, or the public?

A324. [AH] No. The assertions fail to account for the license conditions that are imposed as a condition of issuing the renewed license that would incorporate the license renewal commitments, including Commitments 41 and 42, in the IP2 and IP3 FSAR.

Q325. Can you explain why New York is mistaken regarding the enforceability of the commitments mentioned in your testimony?

A325. [AH] New York's analysis is flawed for several reasons, including the failure to consider: (1) the impact of license conditions imposed as part of issuing the renewed license, and (2) the Commission's regulations that impose controls regarding the method for making changes to the FSAR, including license amendments under 10 C.F.R. § 50.90.

Q326. Could you explain the license conditions that are imposed as a condition of license renewal?

A326. [AH] Yes. The license conditions vary from plant-to-plant depending on the circumstances of each plant. There are, however, several license conditions that are imposed for each renewed license. The Staff indicated in its SER that it intended to impose these license conditions as a condition license renewal. For example in the recently issued license renewal for Callaway, the Staff imposed:

(17) <u>License Renewal License Conditions</u>

(a) The information in the Final Safety Analysis Report (FSAR) supplement, submitted pursuant to 10 CFR 54.21(d), is henceforth part of the FSAR which will be updated in accordance with 10 CFR 50.71(e). As such, the licensee may make changes to the programs and activities described in the FSAR supplement, without prior Commission approval, provided the licensee evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

(b) The licensee's FSAR supplement submitted pursuant to 10 CFR 54.21(d), as revised during the license renewal application review process, and as revised in accordance with license condition 2.C.(17)(a), describes certain programs to be implemented and activities to be completed prior to the period of extended operation.

1. [Licensee] shall implement those new programs and enhancements to existing programs no later than April 18, 2024.

2. [Licensee] shall complete those designated inspection and testing activities, as noted in Appendix A of the [Safety Evaluation Report for License Renewal], no later than April 18, 2024, or the end of the last refueling outage prior to the period of extended operation, whichever occurs later.

3. [Licensee] shall notify the NRC in writing within 30 days after having accomplished item (b)1 above and include the status of those activities that have been or remain to be completed in item (b)2 above.

While the exact wording of the license condition for IP2 and IP3 may vary slightly, it will establish essentially the same limitations on each license.

Q327. Can you explain what the license condition that the Staff would impose on the renewed license does?

A327. [AH] The license condition accomplishes several related issues. First, the condition will require Entergy to update the FSAR with the proposed AMPs and other commitments made during the license renewal review process. It requires the programs to be implemented by a date certain, or end of the last refueling outage prior to the period of extended

operations. Second, it establishes the dates for certain inspections and testing activities to be completed, which can vary from "prior to the period of extended operation" to specific dates for inspections scheduled during the license renewal period. Third, it requires the licensee to notify the Staff when it completes the implementation of the AMP and testing that are required prior to period of extended operation. This notification allows the Staff to schedule the IP71003 inspection for the plant, which is implemented to verify completion of commitments. Finally, the requirement to incorporate these programs as part of the licensee's FSAR places any changes to the programs under the Commission's regulatory control and change process. The licensee would need to evaluate any proposed changes to the FSAR under 10 C.F.R. § 50.59 and, depending on the proposed change and evaluation, make changes administratively or through a formal license amendment.

Q328. Can you explain how the Commission's regulations act to control changes to the license renewal commitments?

A328. [ALH] As discussed above, the license condition requires the licensee to place the programs described in its license renewal commitments in its FSAR. Once part of the FSAR, changes are controlled under the Commission regulations at 10 C.F.R. § 50.59. While the full evaluation process for determining whether a particular change can be made by the licensee alone or whether it would require a license amendment is complex, the licensee is required to document any determination regarding decisions to modify the FSAR without a licensee amendment. Changes to the FSAR are submitted to Staff on a regular basis for review. Should members of public or New York conclude that the licensee has made changes to its FSAR improperly, they could petition the Director of NRR to take action under 10 C.F.R. § 2.206. The Staff, similarly, could take enforcement action if changes were made to the FSAR by the licensee but required a license amendment. Alternatively, members of the public or New

York would have an opportunity to challenge any change to the FSAR submitted as a license amendment under 10 C.F.R. § 50.90 through the normal hearing process.

Q329. Based on your review of the Entergy's RVI AMP, contention NYS-25 and NYS-38, and the exhibits and testimony of Dr. Lahey, what is your expert opinion regarding the adequacy of Entergy's proposed RVI AMP?

A329. [JP][AH] Entergy's currently proposed RVI AMP for Indian Point Units 2 (IP2) and 3 (IP3) and inspection plan provides a fully developed and effective condition monitoring program that provides for timely inspections of the reactor vessel internals components that are most likely to have aging degradation, thus is capable of managing the aging effects that may be experienced by RVI components through the period of extended operation. As documented in the SSER, the Staff concluded that Entergy's AMP is consistent with the approved program as documented in License Renewal Interim Staff Guidance LR-ISG-2011-04, "Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors,", and satisfies the Commission's regulatory requirements including 10 C.F.R. § 54.29(a).

Q330. Does this conclude your testimony?

A330. [JP][AH][GS] 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 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)	)	
	)	

AFFIDAVIT OF JEFFREY C. POEHLER

I, Jeffrey C. Poehler, 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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ENTERGY NUCLEAR OPERATIONS, INC.	)	Docket Nos. 50-247/286-LR
	)	
(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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