UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	Docket Nos. 50-247-LR and
	)	50-286-LR
ENTERGY NUCLEAR OPERATIONS, INC.	)	
	)	
(Indian Point Nuclear Generating Units 2 and 3)	)	
	)	August 10, 2015

TESTIMONY OF ENTERGY WITNESSES NELSON F. AZEVEDO, ROBERT J. DOLANSKY, ALAN B. COX, JACK R. STROSNIDER, TIMOTHY J. GRIESBACH, RANDY G. LOTT, AND MARK A. GRAY REGARDING CONTENTION NYS-25 (EMBRITTLEMENT)

[redacted]

William B. Glew, Jr., Esq.
ENTERGY NUCLEAR OPERATIONS, INC.
440 Hamilton Avenue
White Plains, NY 10601
Phone: (914) 272-3360
Fax: (914) 272-3242
E-mail: wglew@entergy.com

Kathryn M. Sutton, Esq.
Paul M. Bessette, Esq.
Raphael P. Kuyler, Esq.
MORGAN, LEWIS & BOCKIUS LLP
1111 Pennsylvania Avenue, NW
Washington, DC 20004
Phone: (202) 739-3000
Fax: (202) 739-3001
E-mail: ksutton@morganlewis.com
E-mail: pbessette@morganlewis.com
E-mail: rkuyler@morganlewis.com

COUNSEL FOR ENTERGY NUCLEAR OPERATIONS, INC.

[redacted]

TABLE OF CONTENTS

Page

[redacted]

TABLE OF CONTENTS
(continued)

Page

[redacted]

TABLE OF ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
A/LAI	Applicant/Licensee Action Items
AMP	Aging management program
AMR	Aging management review
ASME	American Society of Mechanical Engineers
ANO	Arkansas Nuclear One
BTP	Branch Technical Position
B.S.	Bachelor of Science
BMI	Bottom-mounted Instrumentation
BWR	Boiling Water Reactor
CASS	Cast Austenitic Stainless Steel
CLB	Current licensing basis
CRDM	Control rod drive mechanism
CRGT	Control Rod Guide Tube
CUF	Cumulative usage factor
CUF_{en}	Environmentally-corrected CUF
DBA	Design basis accident
DOE	Department of Energy
EAF	Environmentally-assisted fatigue
ECCS	Emergency Core Cooling System
EFPY	Effective Full-Power Years
EMA	Equivalent Margins Analysis
EPFM	Elastic-Plastic Fracture Mechanics
EPRI	Electric Power Research Institute
FMECA	Failure Modes, Effects, and Criticality Analyses
FMP	Fatigue Monitoring Program
GALL	Generic Aging Lessons Learned
GSI	Generic Safety Issue
IASCC	Irradiation-assisted stress corrosion cracking
IGSCC	Intergranular stress corrosion cracking
IE	Irradiation Embrittlement
INPO	Institute of Nuclear Power Operations
IP2	Indian Point Nuclear Generating Unit 2
IP3	Indian Point Nuclear Generating Unit 3
IPEC	Indian Point Energy Center
ISI	Inservice inspection
LEFM	Linear Elastic Fracture Mechanics
LRA	License Renewal Application
LWR	Light water reactor
LTOP	Low-temperature overpressure protection
LSCCs	Lower Support Column Caps

[redacted]

TABLE OF ABBREVIATIONS
(continued)

Page

M.B.A.	Master of Business Administration
MeV	Mega-electron volts
MOTA	Material Orientation Toughness Assessment
MRP	Materials Reliability Program
M.S.	Master of Science
NDTT	Nil-ductility Transition Temperature
NEI	Nuclear Energy Institute
NDE	Non-destructive Examination
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NU	Northeast Utilities
NYPA	New York Power Authority
NYS	New York State
PEO	Period of extended operation
P-T	Pressure-Temperature
PTS	Pressurized thermal shock
PWR	Pressurized water reactor
PWROG	Pressurized Water Reactor Owners Group
PWSCC	Primary water stress corrosion cracking
RAI	Request for additional information
RCS	Reactor coolant system
RIS	Regulatory Issue Summary
RPV	Reactor pressure vessel
RPI	Rensselaer Polytechnic Institute
RT_{NDT}	Reference temperature for nil-ductility transition
RVI	Reactor vessel internals
SER	Safety Evaluation Report
SI	Structural Integrity Associates
SRM	Staff Requirements Memorandum
SCC	Stress Corrosion Cracking
SSCs	Systems, structures, and components
SPU	Stretch Power Uprate
SSER	Supplement to the Safety Evaluation Report
SRP-LR	Standard Review Plan for Review of License Renewal
TE	Thermal Embrittlement
TLAA	Time-limited Aging Analysis
TJ	Technical Justification
TR	Topical Report
USE	Upper Shelf Energy
UFSAR	Updated Final Safety Analysis Report
VISA	Vessel Integrity Simulation Analysis
WOG	Westinghouse Owners Group

[redacted]

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	Docket Nos. 50-247-LR and
	)	50-286-LR
ENTERGY NUCLEAR OPERATIONS, INC.	)	
	)	
(Indian Point Nuclear Generating Units 2 and 3)	)	
	)	August 10, 2015

TESTIMONY OF ENTERGY WITNESSES NELSON F. AZEVEDO, ROBERT J. DOLANSKY, ALAN B. COX, JACK R. STROSNIDER, TIMOTHY J. GRIESBACH, RANDY G. LOTT, AND MARK A. GRAY REGARDING CONTENTION NYS-25 (EMBRITTLEMENT)

I. WITNESS BACKGROUND

A. Nelson F. Azevedo ("NFA")

Q1. Please state your full name.

A1. (NFA) My name is Nelson F. Azevedo.

Q2. By whom are you employed and what is your position?

A2. (NFA) I am employed by Entergy Nuclear Operations, Inc. ("Entergy"), the applicant in this matter, as Supervisor of Code Programs at Indian Point Nuclear Generating Units 2 and 3 ("IP2" and "IP3," collectively "Indian Point Energy Center" or "IPEC") in Buchanan, New York.

Q3. Please describe your role in this license renewal proceeding.

A3. (NFA) I am involved in this proceeding as an Entergy witness in connection with the adjudication of this contention, the metal fatigue contention (NYS-26B/RK-TC-1B), and the safety commitments contention (NYS-38/RK-TC-5). During the Track 1 hearings, I was an expert witness on the buried piping and flow-accelerated corrosion contentions (NYS-5 and RK-

TC-2, respectively). My role regarding NYS-25 is to provide testimony based on my supervisory role at IPEC in the management of ASME Code programs at IPEC—specifically the Code programs related to the management of reactor vessel internals ("RVIs") and reactor pressure vessels ("RPVs").

Q4. Please describe your educational and professional qualifications, including relevant professional activities.

A4. (NFA) My professional and educational qualifications are summarized in my *curriculum vitae* (ENT000032). I hold a Bachelor of Science degree ("B.S.") in Mechanical and Materials Engineering from the University of Connecticut, and a Master of Science degree ("M.S.") in Mechanical Engineering from the Rensselaer Polytechnic Institute ("RPI") in Troy, New York. In addition, I have received a Master of Business Administration degree ("M.B.A.") from RPI.

I have more than 30 years of professional experience in the nuclear power industry. During that time, I have held engineering, supervisory, and managerial positions with Northeast Utilities ("NU") for nearly 19 years and with Entergy for more than 14 years. Prior to becoming a Manager at NU in 1998, I was an Engineer at NU for more than ten years, an Engineering Supervisor there for five years and an Engineering Manager for another two years. While in the latter position, I managed five engineering sections and was responsible for implementing numerous engineering programs at Millstone Station, including the fatigue monitoring programs (maintaining the Class 1 fatigue analyses), reactor pressure vessel ("RPV") embrittlement and reactor vessel internals ("RVI") programs. While at NU, I performed several finite element analyses and fatigue analyses for piping systems and for RPVs, pressurizers and steam generators.

[redacted]

Since 2001, I have managed the IPEC engineering section responsible for implementing the American Society of Mechanical Engineers ("ASME") Code programs, including the fatigue monitoring, inservice inspection, inservice testing, boric acid corrosion control, non-destructive examination, steam generators, alloy 600 cracking, RPV embrittlement, and RVI programs. I also am responsible for ensuring compliance with the ASME Code, Section XI requirements for repair and replacement activities at IPEC. Finally, I represent IPEC in various industry organizations, including the pressurized water reactor ("PWR") Owners Group ("PWROG") Materials Subcommittee and the Electric Power Research Institute ("EPRI") Materials Reliability Program ("MRP") Committees.

Q5. Are you familiar with the sections of the IPEC License Renewal Application (Apr. 2007) ("LRA") (ENT00015A-B), and its subsequent revisions, that are relevant to the technical issues raised in NYS-25?

A5. (NFA) Yes. I am familiar with the technical issues related to the management of the effects of aging on the IPEC RVIs and RPVs, including the need to address combinations of aging effects, the need to account for design basis loads in engineering analyses, the elements of the RVI AMP, and fatigue analyses. I have personal knowledge of the development, and subsequent revision, of the portions of the IPEC LRA that address such issues, including the relevant Entergy aging management programs ("AMPs").

In particular, as relevant to NYS-25, I am familiar with Sections 4.2 (Reactor Vessel Neutron Embrittlement), B.1.18 (Inservice Inspection ("ISI") AMP), B.1.38 (Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) AMP), B.1.32 (Reactor Vessel Surveillance AMP), and B.1.42 (Reactor Vessel Internals Program ("RVI

AMP")) of the LRA. As a result, I am familiar with Entergy's plans to manage the effects of aging on RVIs and RPVs at IPEC.

Q6. Please further describe the basis for your familiarity with these aspects of the LRA, and the associated technical issues raised in NYS-25.

A6. (NFA) In my capacity as Supervisor of Code Programs at IPEC, I have been responsible for RPV technical and regulatory issues since 2001, including updating of the reactor coolant system ("RCS") heatup and cooldown curves and RPV surveillance capsule removal and analysis as required by 10 C.F.R. Part 50, Appendix H, as well as the resolution of RCS and RPV structural issues. I have been involved in preparing updates to the LRA and responding to Nuclear Regulatory Commission ("NRC") Staff Requests for Additional Information ("RAIs") and audit questions since Entergy's LRA was submitted in 2007. I also supported Entergy at the Advisory Committee on Reactor Safeguards ("ACRS") meetings for the IPEC LRA held in 2009 and 2015.

In general, I have reviewed various materials in preparing this testimony, including those portions of Entergy's LRA for IP2 and IP3 relating to Entergy's evaluation of the effects of aging on RVIs. I also reviewed NYS' exhibits on NYS-25, which are listed in response to Question 65, below. I reviewed the Atomic Safety and Licensing Board's ("Board's") orders on this contention, including: (1) *Entergy Nuclear Operations, Inc.* (Indian Point Nuclear Generating Units 2 & 3), LBP-08-13, 68 NRC 43 (2008) ("Order Admitting NYS-25"); (2) Licensing Board Memorandum and Order (Ruling on Pending Motions for Leave to File New and Amended Contentions) (July 6, 2011) (unpublished) ("Order Admitting Amended NYS-25"); and (3) Licensing Board Memorandum and Order (Granting Motions for Leave to File Amendments to Contentions NYS-25 and NYS-38/RK-TC-5) (Mar. 31, 2015) (unpublished)

("Order Admitting Amended Contentions"), and the exhibits submitted by the State of New York ("NYS" or "the State") that are relevant to my testimony.

B. <u>Robert J. Dolansky ("RJD")</u>

Q7. Please state your full name.

A7. (RJD) My name is Robert. J. Dolansky.

Q8. By whom are you employed and what is your position?

A8. (RJD) I am employed by Entergy, the applicant in this matter, as a Code Programs Engineer, at IPEC in Buchanan, New York.

Q9. Please describe your role in this license renewal proceeding.

A9. (RJD) I am involved in this proceeding as an Entergy witness in connection with the adjudication of this contention, and the safety commitments contention (NYS-38/RK-TC-5). My role regarding NYS-25 is to provide testimony based on my role as the program owner for the RVI AMP and Steam Generator AMP at IPEC.

Q10. Please describe your educational and professional qualifications, including relevant professional activities.

A10. (RJD) My professional and educational qualifications are summarized in the attached *curriculum vitae* (ENT000522). I hold a B.S. in Aeronautical Engineering from RPI in Troy, New York. I have over 25 years of professional experience in the nuclear power industry. Since 1989, I have been an ASME Code Programs Engineer at IPEC—first with the New York Power Authority ("NYPA") and, later, with Entergy. As an Engineer at NYPA and Entergy, I have been the program owner for, among other programs, the RVI, inservice inspection, inservice testing, steam generators, and alloy 600 cracking programs. Since January 2011, I have been the owner of the IPEC steam generator and RVI AMPs for both units.

[redacted]

Q11. Are you familiar with the sections of the IPEC LRA, and its subsequent revisions that are relevant to the technical issues raised in NYS-25?

A11. (RJD) Yes. I am familiar with the technical issues related to the management of the effects of aging on the IPEC RVIs and RPVs, including the need to address multiple aging effects, the need to account for design basis loads in engineering analyses, the elements of the RVI AMP, and fatigue analyses. I have personal knowledge of the development and subsequent revision of the portions of the IPEC LRA that address such issues, including the relevant Entergy AMPs. In particular, as relevant to NYS-25, I am familiar with Sections 4.2 (Reactor Vessel Neutron Embrittlement TLAA), B.1.18 (ISI AMP), B.1.38 (Thermal Aging and Neutron Irradiation Embrittlement of CASS AMP), B.1.32 (Reactor Vessel Surveillance AMP), and B.1.42 (RVI AMP) of the LRA. As a result, I am familiar with Entergy's plans to manage the effects of aging on RVIs and RPVs at IPEC.

Q12. Please further describe the basis for your familiarity with these aspects of the LRA, and the associated technical issues raised in NYS-25.

A12. (RJD) In my capacity as a Code Programs Engineer at IPEC, I have been responsible for RPV ISI issues since 1989. I was also the IPEC technical lead for the preparation of the sections of the LRA related to ISI for the RVIs, including Sections B.1.18 (ISI) and B.1.38 (Thermal Aging and Neutron Irradiation Embrittlement of CASS). I have been involved in responding to NRC Staff RAIs and audit questions on RVI issues since Entergy submitted the LRA in 2007. I also prepared the RVI Inspection Plan (discussed further below) that Entergy submitted to the NRC in 2011 and prepared a revision to that document in February 2012. I also supported Entergy at the ACRS License Renewal Subcommittee meetings for the IPEC LRA held in 2015.

[redacted]

More specifically, I have been involved in ASME Code-based inspections and evaluations of components and piping at IPEC since 1989. I hold Entergy qualifications for the Section XI repair and replacement, inservice testing, ISI, snubber, and steam generator programs. I hold these qualifications under the IPEC training program, which is accredited by the Institute of Nuclear Power Operations ("INPO"). In addition, I have participated in numerous training programs on code programs and industry evaluations, and have been trained in nondestructive testing, advanced ultrasonic detection, eddy current analysis, and thermal fatigue management. My experience includes oversight of visual, surface, volumetric and eddy current inspections and the supervision of personnel who perform nondestructive examinations. I review and approve condition reports and third-party analyses, including ASME Section XI structural evaluations that may be performed as a result of inspection findings.

I also represent IPEC with respect to Code programs before industry organizations such as the PWR Owners Group. For example, I am a member of the PWR Owners Group Materials Subcommittee, and have attended numerous EPRI steam generator management project meetings. I continue to participate in Entergy and industry programs and meetings to address the latest developments and operating experience in Code-related issues.

Finally, I have reviewed various materials in preparing this testimony, including those portions of Entergy's LRA, for IP2 and IP3 relating to Entergy's evaluation of the effects of aging on RVIs. In addition to the relevant sections of the LRA, I reviewed NYS' exhibits on NYS-25, which are listed in response to Question 65, below. I also reviewed the Board's orders on this contention, including: (1) Order Admitting NYS-25; (2) Order Admitting Amended NYS-25; and (3) Order Admitting Amended Contentions, and the exhibits submitted by NYS that are relevant to my testimony.

C. <u>Alan B. Cox ("ABC")</u>

Q13. Please state your full name.

A13. (ABC) My name is Alan B. Cox.

Q14. By whom are you employed and what is your position?

A14. (ABC) I am now an independent consultant for Entergy. Before my retirement from the company earlier this year, however, I was the Technical Manager of License Renewal with Entergy, the applicant in this matter.

Q15. Please describe your role in this license renewal proceeding.

A15. (ABC) I am involved in this proceeding as an Entergy witness in connection with the adjudication of this contention, the metal fatigue contention (NYS-26B/RK-TC-1B), and the safety commitments contention (NYS-38/RK-TC-5). During the Track 1 hearings, I was an expert witness on the buried piping, cables, and flow-accelerated corrosion contentions (NYS-5, NYS-6/7, and RK-TC-2, respectively). My role regarding NYS-25 is to provide testimony based on my role, as part of Entergy's license renewal services organization, in the development and review of the IPEC LRA.

Q16. Please describe your educational and professional qualifications, including relevant professional activities.

A16. (ABC) My professional and educational qualifications are summarized in my *curriculum vitae* (ENTR00031). Briefly summarized, I hold a B.S. in Nuclear Engineering from the University of Oklahoma and an M.B.A. from the University of Arkansas at Little Rock. I have over 38 years of experience in the nuclear power industry, having served in various positions related to engineering and operations of nuclear power plants during that time. For example, I was licensed by the NRC as a reactor operator in 1981 and as a senior reactor operator in 1984 for Arkansas Nuclear One ("ANO") Unit 1. During operator training and while

[redacted]

serving as a shift technical advisor for both ANO units, I was trained in reactor thermal hydraulics and in plant response to transients and accidents. From 1993 to 1996, I was employed by Entergy as a Senior Staff Engineer at ANO. From 1996 to 2001, I served as the Supervisor, Design Engineering, at ANO. I have previously held a professional engineer's license in the State of Arkansas.

From 2001 to 2015, I worked for Entergy's license renewal services organization, supporting the integrated plant assessment and LRA development for Entergy license renewal projects, as well as projects for other utilities. Specifically, as a member of the Entergy license renewal team, I participated in the development of LRAs for twelve plants owned and operated either by Entergy or other utilities. Since 2001, I have participated in peer reviews for numerous other LRAs for plants throughout the United States. For over ten years, I was a member of the Nuclear Energy Institute ("NEI") License Renewal Task Force. During portions of that time, I served as Entergy's representative on the NEI License Renewal Mechanical Working Group and the NEI License Renewal Electrical Working Group.

Q17. Are you familiar with the sections of the IPEC LRA, and its subsequent revisions that are relevant to the technical issues raised in NYS-25?

A17. (ABC) Yes. I am familiar with the technical issues related to the management of the effects of aging on the IPEC RVIs and RPVs, including the need to address multiple sources of aging effects, the need to account for design basis loads in engineering analyses, the elements of the RVI AMP, and fatigue analyses. I have personal knowledge of the development and subsequent revision of the portions of the IPEC LRA that address such issues, including the relevant Entergy AMPs. In particular, as relevant to NYS-25, I am familiar with Sections 4.2 (Reactor Vessel Neutron Embrittlement), B.1.18 (ISI AMP), B.1.38 (Thermal Aging and

Neutron Irradiation Embrittlement of CASS AMP), B.1.32 (Reactor Vessel Surveillance AMP), and B.1.42 (RVI AMP) of the LRA. As a result, I am familiar with Entergy's plans to manage the effects of aging on RVIs and RPVs at IPEC.

Q18. Please further describe the basis for your familiarity with these aspects of the LRA, and the associated technical issues raised in NYS-25.

A18. (ABC) As Technical Manager for License Renewal, I was directly involved in preparing the IPEC LRA and developing the associated AMPs and commitments. I also have been directly involved in developing and reviewing Entergy's responses to RAIs concerning the LRA and various amendments or revisions to the LRA (principally as they relate to aging management issues), including RAIs regarding RVI and RPV embrittlement issues. I supported Entergy at meetings of the ACRS and its License Renewal Subcommittee for the IPEC LRA held in September 2009, and in March 2009, respectively. I also supported Entergy at the ACRS Subcommittee meeting for the IPEC LRA held in 2015. Accordingly, I have personal knowledge of the development and subsequent revision of the LRA, including the RVI AMP and the RPV Surveillance Program.

I have reviewed various materials in preparing this testimony, including those portions of Entergy's LRA for IP2 and IP3 relating to Entergy's evaluation of the effects of aging on RVIs. In addition to the relevant sections of the LRA, I reviewed NYS' exhibits on NYS-25, which are listed in response to Question 65, below. I also reviewed the Board's orders on this contention, including: (1) Order Admitting NYS-25; (2) Order Admitting Amended NYS-25; and (3) Order Admitting Amended Contentions, and the exhibits submitted by NYS that are relevant to my testimony.

[redacted]

D. **<u>Jack R. Strosnider, Jr. ("JRS")</u>**

Q19. Please state your full name.

A19. (JRS) My name is Jack R. Strosnider, Jr.

Q20. By whom are you employed and what is your position?

A20. (JRS) I am a Senior Nuclear Safety and Licensing Consultant with Talisman International, LLC. Since March 2007, when I retired from the NRC, as discussed below, I have provided consulting services to nuclear utilities and vendors on nuclear safety, performance issues, licensing and inspection activities.

Q21. Please describe your role in this license renewal proceeding.

A21. (JRS) I have been retained by Entergy as an independent technical and regulatory expert in connection with the adjudication of this contention, the metal fatigue contention (NYS-26B/RK-TC-1B), and the safety commitments contention (NYS-38/RK-TC-5). My role regarding NYS-25 is to provide independent expert testimony based on my experience as a senior manager within the NRC, including supervising NRC Staff in engineering, inspection, research, and license renewal-related activities, and to provide technical testimony on the aging management of RVIs, RPVs, steam generators, and other plant components.

Q22. Please describe your educational and professional qualifications, including relevant professional activities.

A22. (JRS) My professional and educational qualifications are summarized in my *curriculum vitae* (ENTR00184). I hold a B.S. and an M.S. in Engineering Mechanics—both from the University of Missouri at Rolla. I also hold an M.B.A. from the University of Maryland. In brief, prior to April 2007, I was employed for 31 years by the NRC. I held numerous senior management positions at the NRC, including Director of the Office of Nuclear Material Safety and Safeguards, Deputy Director of the Office of Nuclear Regulatory Research,

[redacted]

and Director of the Division of Engineering in the Office of Nuclear Reactor Regulation ("NRR"). From 1984 through 1990, I was a supervisor for inspection activities in the NRC's Region I office. I also worked for two years at the Nuclear Energy Agency in Paris, France, which is an intergovernmental organization of industrialized countries that develops guidance and reports on issues that affect nuclear facilities around the world.

I have extensive experience in developing and applying NRC regulations and programs that address the aging of nuclear power plant structures and components, including metal fatigue issues. In addition to serving as the supervisor of inspection activities in the NRC's Region I office from January 1999 through May 2002, I also served as Director of the Division of Engineering in NRR, where I directed engineering reviews and preparation of safety evaluation reports ("SERs") for license renewals. This included developing technical resolutions for first-of-a-kind issues associated with license renewal, including, for example, how to monitor for stress corrosion cracking in the reactor coolant system and how to monitor for void swelling in RVI components.

As it relates to this contention, while Deputy Director of the NRC Office of Nuclear Regulatory Research, I was responsible for research programs related to environmental effects on reactor component cracking, and while a manager in the Office of Nuclear Reactor Regulation, I was responsible for licensing reviews associated with resolution of Generic Safety Issue ("GSI") 190, "Fatigue Evaluation of Metal Components for 60-Year Plant Life," and the evaluation of the effects of fatigue on reactor components. I also was responsible for licensing reviews associated with the integrity of the RPV and monitoring of RVIs. Finally, I have over 30 years of experience with NRC regulatory issues related to primary water stress corrosion cracking ("PWSCC") in steam generators and other reactor components. This experience

[redacted]

includes the development of NRC communications related to PWSCC, the review of inspection programs and the review and performance of PWSCC crack growth and flaw acceptance evaluations.

With respect to aging effects on RPVs, beginning in the 1980s, I was involved in the development of the technical bases for 10 C.F.R. § 50.61, which provides fracture toughness requirements for protection against pressurized thermal shock ("PTS") events. Specifically, I wrote the Vessel Integrity Simulation Analysis ("VISA") computer code, which calculates RPV failure probability for a given PTS event as a function of embrittlement level. While the original VISA computer code is no longer in use today, it spawned the development of numerous similar codes by vendors, consulting organizations, and national laboratories that are in use today, such as OCA-P, VISA-II, PROFMAC-II, OPERA (Persoz *et al.*, 2000), FAVOR and PASCAL.

As an NRC Staff member, I used the VISA code to calculate conditional failure probabilities for a spectrum of PTS events having different temperature and pressure time histories. These conditional failure probabilities were combined with frequencies of PTS events developed from systems analyses and probabilistic risk assessments to produce a curve showing RPV failure frequencies due to PTS events as a function of change in fracture toughness due to embrittlement (as measured by the reference temperature for nil-ductility transition "RT_{NDT}"). During the development of the 10 C.F.R. § 50.61 rule, this curve was used to assess RPV failure frequency associated with the screening criteria in that rule.

Q23. Are you familiar with the sections of the IPEC LRA, and its subsequent revisions that are relevant to the technical issues raised in NYS-25?

A23. (JRS) Yes. I am familiar with the technical issues related to the management of the effects of aging on the IPEC RVIs and RPVs, including the need to address multiple sources

[redacted]

of aging effects, the need to account for design basis loads in engineering analyses, the elements of the RVI AMP, and fatigue analyses. I have personal knowledge of the development and subsequent revision of the portions of the IPEC LRA that address such issues, including the relevant Entergy AMPs. In particular, as relevant to NYS-25, I am familiar with Sections 4.2 (Reactor Vessel Neutron Embrittlement), B.1.18 (ISI AMP), B.1.38 (Thermal Aging and Neutron Irradiation Embrittlement of CASS AMP), B.1.32 (Reactor Vessel Surveillance AMP), and B.1.42 (RVI AMP) of the LRA. As a result, I am familiar with Entergy's plans to manage the effects of aging on RVIs and RPVs at IPEC.

Q24. Please further describe the basis for your familiarity with these aspects of the LRA, and the associated technical issues raised in NYS-25.

A24. (JRS) I have reviewed various materials in preparing this testimony, including those portions of Entergy's LRA for IP2 and IP3 relating to Entergy's evaluation of the effects of aging on RVIs. In addition to the relevant sections of the LRA, I reviewed NYS' exhibits on NYS-25, which are listed in response to Question 65, below. I also reviewed the Board's orders on this contention, including: (1) Order Admitting NYS-25; (2) Order Admitting Amended NYS-25; and (3) Order Admitting Amended Contentions, and the exhibits submitted by NYS that are relevant to my testimony.

E. <u>Timothy J. Griesbach ("TJG")</u>

Q25. Please state your full name.

A25. (TJG) My name Timothy J. Griesbach.

Q26. By whom are you employed and what is your position?

A26. (TJG) I am a Senior Associate at Structural Integrity Associates, Inc. I specialize in technical consulting utilizing state-of-the-art technologies for mitigating and resolving material degradation concerns in nuclear reactor vessels, internals, piping, and other major components.

Q27. Please describe your role in this license renewal proceeding.

A27. (TJG) I have been retained by Entergy as an independent technical expert in connection with the adjudication of this contention and the safety commitments contention, NYS-38/RK-TC-5. My role regarding NYS-25 is to provide independent expert testimony based on my experience developing and implementing aging management strategies for RPVs and PWR RVIs, including work directly related to the generic industry guidelines for managing the effects of aging on RVIs in MRP-227-A, and based on my experience with the metallurgical analysis of aged material properties, and performing stress analyses and fracture mechanics analyses per the ASME Code requirements. *See generally* EPRI, MRP-227-A, Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (Dec. 2011) ("MRP-227-A") (NRC00014A-F).

Q28. Please describe your educational and general professional qualifications, including relevant professional activities.

A28. (TJG) My professional and educational qualifications are summarized in the attached *curriculum vitae* (ENT000617). Briefly summarized, I have more than 40 years of experience in metallurgy and materials engineering, primarily in the nuclear field. I received a B.S. degree and a M.S. degree, both in Metallurgy and Materials Science, from Case Western Reserve University, the last in 1974. I am a member of the American Nuclear Society and ASME. I have served on various ASME Boiler and Pressure Vessel Code committees for over 33 years, I chair the ASME Section XI Working Group on Operating Plant Criteria, which involves setting ASME Code requirements for operating pressure and temperature limits for the prevention of brittle fracture of RPVs. I also am a member of the ASME Section XI Standards Committee.

Q29. Please describe your specific mechanical and structural engineering experience, including experience with analysis of aging effects on RPVs and RVIs.

A29. (TJG) From 1977 to 1982, I was a Principal Engineer with Combustion Engineering. My responsibilities included evaluating the response of nuclear steam supply system components to severe thermal, pressure, and dynamic loads. From 1982 to 1993, I was a Project Manager with EPRI. During that time, I was a member of the Nuclear Safety Analysis Center responsible for developing methodologies to resolve generic safety issues including pressurized thermal shock of reactor pressure vessels. I also managed major EPRI research initiatives to evaluate and develop remedial measures for managing reactor pressure vessel embrittlement.

From 1993 through 2005, I was the Director of Technical Services for ATI Consulting. My responsibilities in this position included assessing nuclear component life, developing aging management strategies for RPVs and PWR RVIs, and applying advanced fracture mechanics methods for severe accident conditions in nuclear vessels and piping systems. From 2006 until now, I have worked as a Senior Associate with Structural Integrity Associates ("SI").

In my current position with SI, I work directly with nuclear utilities to manage reactor vessel integrity issues and develop aging management programs for vessels and internals for extended nuclear plant life. I have worked closely with the EPRI Materials Reliability Program to develop and implement the MRP-227-A inspection and evaluation guidelines for the safety and long-term operation of PWR vessel internals. My experience encompasses the metallurgical analysis of aged material properties, performing stress analyses and fracture mechanics analyses

[redacted]

per the ASME Code requirements, and assuring adherence to the NRC regulations and regulatory requirements for managing aging effects in piping, vessels and RVIs for nuclear plant license renewal.

Q30. Are you familiar with the sections of the IPEC LRA, and its subsequent revisions that are relevant to the technical issues raised in NYS-25?

A30. (TJG) Yes. I am familiar with the technical issues related to the management of the effects of aging on the IPEC RVIs and RPVs, including the need to address multiple sources of aging effects, the need to account for design basis loads in engineering analyses, the elements of the RVI AMP, and fatigue analyses. I have personal knowledge of the development and subsequent revision of the portions of the IPEC LRA that address such issues, including the relevant Entergy AMPs. In particular, as relevant to NYS-25, I am familiar with Sections 4.2 (Reactor Vessel Neutron Embrittlement), B.1.18 (ISI AMP), B.1.38 (Thermal Aging and Neutron Irradiation Embrittlement of CASS AMP), B.1.32 (Reactor Vessel Surveillance AMP), and B.1.42 (RVI AMP) of the LRA. As a result, I am familiar with Entergy's plans to manage the effects of aging on RVIs and RPVs at IPEC.

Q31. Please further describe the basis for your familiarity with these aspects of the LRA, and the associated technical issues raised in NYS-25.

A31. (TJG) I have reviewed various materials in preparing this testimony, including those portions of Entergy's LRA for IP2 and IP3 relating to Entergy's evaluation of the effects of aging on RVIs. In addition to the relevant sections of the LRA, I reviewed NYS' exhibits on NYS-25, which are listed in response to Question 65, below. I also reviewed the Board's orders on this contention, including: (1) Order Admitting NYS-25; (2) Order Admitting Amended

NYS-25; and (3) Order Admitting Amended Contentions, and the exhibits submitted by NYS that are relevant to my testimony.

F. Randy G. Lott ("RGL")

Q32. Please state your full name.

A32. (RGL) My name is Randy G. Lott.

Q33. By whom are you employed and what is your position?

A33. (RGL) I am a Consulting Engineer at Westinghouse Electric Company ("Westinghouse") with over 35 years of experience in nuclear materials and radiation effects.

Q34. Please describe your role in this license renewal proceeding.

A34. (RGL) I have been retained by Entergy as an independent technical and regulatory expert in connection with the adjudication of this contention, the metal fatigue contention (NYS-26B/RK-TC-1B) and the safety commitments contention (NYS-38/RK-TC-5). My role regarding NYS-25 is to provide independent expert testimony based on my experience developing and implementing aging management strategies for RPVs and PWR RVIs, including work on developing the generic industry guidelines for managing the effects of aging on RVIs in MRP-227-A, and based on my experience with and knowledge of Westinghouse's mechanical and structural evaluations of IPEC plant components.

Q35. Please describe your educational and general professional qualifications, including relevant professional activities.

A35. (RGL) My professional and educational qualifications are summarized in the attached *curriculum vitae* (ENT000618). I received a B.S. degree in Nuclear Engineering from the University of Michigan, and a M.S. and Doctor of Philosophy degree in Nuclear Engineering from the University of Wisconsin, the last in 1979. Since joining Westinghouse in 1979, I have been the lead test engineer in the Remote Metallographic (Hot Cell) Facility. In this capacity, I

[redacted]

have been responsible for numerous investigations of materials-related issues in pressurized water reactors. My contributions have provided the basis for the revision of NRC Regulatory Guide 1.99, the development of Westinghouse RPV annealing technology, the safety analysis of reactor tanks at the Savannah River Site, the determination of crack growth rates used in alternative plugging criteria for nuclear steam generators, and the evaluation of RVI performance.

Q36. Please describe your specific nuclear materials engineering experience, including experience with analysis of aging effects on RPVs and RVIs.

A36. (RGL) I have extensive experience with post-irradiation evaluation of reactor components, including RPVs and RVIs. I have supervised testing of RPV surveillance capsules and conducted research programs on irradiation embrittlement and annealing of RPV steels. In addition, I pioneered the application of Master Curve testing to characterize the ductile-to-brittle fracture toughness transition in irradiated RPV steels. I have also conducted numerous test programs on highly irradiated stainless steels, including measurement of tensile, fracture toughness and irradiation-assisted stress corrosion cracking ("IASCC") properties. During my career at Westinghouse, I have participated in the evaluation of aging degradation or failure of numerous reactor components, including bottom-mounted instrumentation ("BMI") flux thimbles, control rod guide tube "split" pins, baffle-former bolts and clevis insert bolts.

For the past eight years, I have been actively involved in the design and implementation of aging management programs for reactor internals. As a member of the MRP Reactor Internals Inspection and Evaluation Guidelines Core Group, I was a contributor to the U.S. industry Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A), including work on aging management strategies for the Westinghouse and Combustion

[redacted]

Engineering plants. These industry recommendations were in turn endorsed in the PWR RVI AMP in the NRC Generic Aging Lessons Learned ("GALL") Report (NUREG-1801).

Q37. Are you familiar with the sections of the IPEC LRA, and its subsequent revisions that are relevant to the technical issues raised in NYS-25?

A37. (RGL) Yes. I am familiar with the technical issues related to the management of the effects of aging on the IPEC RVIs and RPVs, including the need to address multiple sources of aging effects, the need to account for design basis loads in engineering analyses, the elements of the RVI AMP, and fatigue analyses. I have knowledge of the development and subsequent revision of the portions of the IPEC LRA that address such issues, including the relevant Entergy AMPs. In particular, as relevant to NYS-25, I am familiar with Sections 4.2 (Reactor Vessel Neutron Embrittlement), B.1.18 (ISI AMP), B.1.38 (Thermal Aging and Neutron Irradiation Embrittlement of CASS AMP), B.1.32 (Reactor Vessel Surveillance AMP), and B.1.42 (RVI AMP) of the LRA. As a result, I am familiar with Entergy's plans to manage the effects of aging on RVIs and RPVs at IPEC.

Q38. Please further describe the basis for your familiarity with these aspects of the LRA, and the associated technical issues raised in NYS-25.

A38. (RGL) I have reviewed various materials in preparing this testimony, including those portions of Entergy's LRA for IP2 and IP3 relating to Entergy's evaluation of the effects of aging on RVIs. In addition to the relevant sections of the LRA, I reviewed NYS' exhibits on NYS-25, which are listed in response to Question 65, below. I also reviewed the Board's orders on this contention, including the Order Admitting NYS-25, the Order Admitting Amended NYS-25, and the Order Admitting Amended Contentions, and the exhibits submitted by NYS that are relevant to my testimony.

G. Mark A. Gray ("MAG")

Q39. Please state your full name.

A39. (MAG) My name is Mark A. Gray.

Q40. By whom are you employed and what is your position?

A40. (MAG) I am a Principal Engineer in the Primary Systems Design and Repair group at Westinghouse with over 34 years of experience in evaluating nuclear component structural integrity.

Q41. Please describe your role in this license renewal proceeding.

A41. (MAG) I have been retained by Entergy as an independent technical expert in connection with the adjudication of this contention, the metal fatigue contention (NYS-26B/RK-TC-1B), and the safety commitments contention (NYS-38/RK-TC-5). My role regarding NYS-25 is to provide independent expert testimony based on my experience with and knowledge of Westinghouse's fatigue evaluations of IPEC plant components, as well as my experience in structural integrity issues in primary system piping and components, including ASME Code stress and fatigue analysis and environmentally-assisted fatigue ("EAF") evaluations.

Q42. Please describe your educational and general professional qualifications, including relevant professional activities.

A42. (MAG) My professional and educational qualifications are summarized in the attached *curriculum vitae* (ENTR00186). Briefly summarized, I hold a B.S. in Mechanical Engineering, and an M.S. in Mechanical Engineering with a Nuclear Certificate, both from the University of Pittsburgh. I have over 34 years of experience in the nuclear power industry as an employee of Westinghouse. My principal activities at Westinghouse include the evaluation of structural integrity issues in primary system piping and components. This includes the development of plant life extension and monitoring programs and analysis. I have participated in the development and application of transient and fatigue monitoring algorithms and software for the WESTEMS™ Transient and Fatigue Monitoring System, and participated in cooperative efforts with vendors outside Westinghouse in the development of transient and fatigue monitoring systems. I am a member of ASME, the ASME Code Section III Working Group on Piping Design and Working Group on Environmental Fatigue Evaluation Methods, and the EPRI Environmentally-Assisted Fatigue Focus Group. I was also a member of the former EPRI/ASME Environmentally-Assisted Fatigue Expert Panel. I am a registered professional engineer in the Commonwealth of Pennsylvania.

Q43. Please describe your specific mechanical and structural engineering experience, including experience with analysis of fatigue in key reactor components.

A43. (MAG) I have been involved in life extension and license renewal activities at Westinghouse since participating in the first Plant Life Extension pilot study for the Surry Unit 1 nuclear power plant in the mid-1980s. I co-authored the Westinghouse Owners Group ("WOG") Generic Technical Report on Aging Management for Pressurizers, contributed to a similar report covering Reactor Coolant System Piping, and represented Westinghouse before the NRC in their review of the generic reports. I have contributed to the development of transient and fatigue monitoring programs for over a dozen plants. These activities have included overall program development, as well as collection and interpretation of plant historical records and monitoring data for the establishment of baseline fatigue estimates, and identification of improvements to licensee fatigue management programs. I have performed and directed evaluations of the effects of reactor water environment on reactor component fatigue for a number of plants, including IPEC.

In addition, I have extensive experience performing ASME Code evaluations, and in evaluating actual plant transients, including pressurizer surge line stratification (NRC Bulletin 88-11), thermal stratification and cycling (NRC Bulletin 88-08), and pressurizer insurge/outsurge. From 1993 to 1998, I led the WOG program on Mitigation and Evaluation of Pressurizer Insurge and Outsurge Transients. I have led plant-specific activities for evaluation of pressurizer insurge/outsurge transients at a number of plants.

For approximately five years, I was lead engineer for fatigue analysis and fatigue-related issues affecting all Class 1 piping and related systems in U.S. Westinghouse plants. In that capacity, I was responsible for all design fatigue evaluations of Class 1 piping systems and components, as well as evaluation of reported non-design transients for their effects on design requirements. In sum, I have extensive experience in the application of finite element analysis, transfer function, and other techniques to evaluate heat transfer, stress and fatigue of components and structures subjected to complex thermal and mechanical loading conditions.

Q44. Are you familiar with the sections of the IPEC LRA, and its subsequent revisions that are relevant to the technical issues raised in NYS-25?

A44. (MAG) Yes. I am familiar with the technical issues related to the management of the effects of fatigue on IPEC plant components, including the elements of the Entergy AMP that addresses metal fatigue, referred to as the fatigue monitoring program ("FMP"), and fatigue analyses. I have personal knowledge of the development and subsequent revision of the portions of LRA that address the effects of aging due to fatigue, including Sections 4.3 (Metal Fatigue) and Section B.1.12 (FMP).

[redacted]

Q45. Please further describe the basis for your familiarity with these aspects of the LRA, and the associated technical issues raised in NYS-25.

A45. (MAG) I have reviewed various materials in preparing this testimony, including those portions of Entergy's LRA for IP2 and IP3 relating to Entergy's evaluation of the effects of metal fatigue and EAF. In addition to the relevant sections of the LRA, I reviewed NYS' exhibits on NYS-25, which are listed in response to Question 65, below, as they relate to my testimony on this contention.

II. <u>OVERVIEW OF CONTENTION NYS-25</u>

Q46. Are you familiar with Contention NYS-25, as originally proposed by NYS?

A46. (ABC, NFA, JRS, RJD, TJG, RGL) Yes. We have reviewed the "New York State Notice of Intention to Participate and Petition to Intervene" ("Petition"), dated November 30, 2007, available at ADAMS Accession No. ML073400187; the associated Declaration of Dr. Richard T. Lahey, Jr. ("2007 Lahey Decl."), dated November 30, 2007 (NYS000298); and the "New York State Reply in Support of Petition to Intervene," dated February 22, 2008, available at ADAMS Accession No. ML080600444.

NYS-25, as originally proposed, alleged that the LRA "does not include an adequate plan to monitor and manage the effects of aging due to embrittlement of the [RPVs] and the associated internals at both plants, pursuant to 10 C.F.R. § 54.21(a), and an evaluation of time limiting aging analysis [TLAA], pursuant to 10 C.F.R. § 54.21(c)." NYS Petition at 223. The State's initial pleadings focused primarily on the RPV, rather than the RVIs, claiming that the information in the LRA on the time-limited aging analyses ("TLAAs") associated with the RPVs did not include information on "age-related accident analyses," NYS Petition at 224, and that an "intermediate shell in IP2 will not meet the upper shelf energy acceptance criterion of 50ft-lb." *Id*. at 226.

[redacted]

The State's contention as originally proposed also identified certain RVI components as within the scope of its "[c]oncerns over embrittlement," including the core barrel, particularly the "belt-line" region of the reactor core; the thermal shield; and the "baffle plates and formers." Petition at 225; 2007 Lahey Decl. ¶ 15. Dr. Lahey's initial declaration also included the general statement that "RPV internals in IP3 imply operational limits for extended life operations due to the high NDT associated with the predicted irradiation-induced embrittlement." *Id.* ¶ 18.

Q47. Are you familiar with Contention NYS-25, as admitted by the Board on July 31, 2008?

A47. (ABC, NFA, JRS, RJD, TJG, RGL) Yes. The Board admitted NYS-25, finding that: "[w]hether an AMP is necessary to manage the cumulative effects of embrittlement of the RPVs and associated internals is within the scope of this proceeding" and that Dr. Lahey's 2007 Declaration "focuses on specific portions of Entergy's LRA that are, in Dr. Lahey's professional judgment, deficient." Order Admitting NYS-25, LBP-08-13, 68 NRC at 131. We interpret the "associated internals" referenced in that decision to include those RVI components referenced by Dr. Lahey and listed in response to Question 65.

Q48. Did Entergy's original LRA address the aging management of RPVs?

A48. (ABC, NFA, JRS, RJD, TJG, RGL) Yes. Section 3.1.2.2.3 of the LRA identifies that Section 4.2 of the LRA addresses the evaluation of neutron embrittlement TLAAs and that Entergy's associated AMP for managing the loss of fracture toughness due to neutron irradiation embrittlement of the IPEC RPVs is the Reactor Vessel Surveillance Program. *See* LRA at 3.1-7. LRA Section 4.2 describes the evaluation of the four TLAAs related to neutron irradiation embrittlement of the RPVs: (1) the Charpy Upper-Shelf Energy ("USE") TLAA, described in Section 4.2.2 of the LRA; (2) the pressure-temperature ("P-T") limits TLAA, described in

[redacted]

Section 4.2.3; (3) the low temperature overpressure protection ("LTOP") TLAA, described in Section 4.2.4; and (4) the PTS TLAA, described in Section 4.2.5. *See id.* §§ 4.2.2; 4.2.3; 4.2.4; 4.2.5 (ENT00015A).

Q49. Did Entergy's original LRA address the aging management of RVIs?

A49. (ABC, NFA, JRS, RJD, TJG, RGL) Yes. When Entergy submitted the LRA in 2007, the industry was developing generic guidance on managing the effects of aging on PWR RVIs. Thus, in Commitment 30, consistent with the then-current NRC guidance in NUREG-1801, Revision 1, Entergy committed to participate in industry programs for investigating and managing aging effects on RVIs, to evaluate and implement industry programs applicable to RVIs, and to submit an RVI inspection plan not less than 24 months before entering the period of extended operation ("PEO"). *See* LRA at 3.1-7 to 3.1-8, 3.1-9 to 3.1-10 (ENT00015A); *see also* NUREG-1801, Generic Aging Lessons Learned Report, Rev. 1 at 7-30, tbl. 1 (Sept. 2005) ("NUREG-1801") (NYS00146A).

Q50. Following the admission of NYS-25, did Entergy amend its LRA with respect to RPVs?

A50. (ABC, NFA, JRS, RJD) Yes. Entergy submitted several RPV-related amendments to the IPEC LRA. For example, one amendment revised multiple sections of the LRA to provide clarification regarding the RPVs based on a response to RAI 4.2.1-1. *See* NL-08-092, Letter from F. Dacimo, Entergy, to NRC Document Control Desk, "Amendment 5 to License Renewal Application (LRA)," Attach. 3 at 1 (June 11, 2008) ("NL-08-092") (ENT000193). Another amended LRA section 4.2.5 and Commitment 32 to revise the description of how Entergy would address the then-proposed alternate PTS rule. *See* NL-08-127, Letter from F. Dacimo, Entergy, to NRC Document Control Desk, "Additional Information

Regarding License Renewal Application – Structural OE Clarifications, Clarifications for Electrical RAIs and Audit Questions, License Renewal Application Amendment," Attach. 3 at 1 (Aug. 14, 2008) (ENT000619); *see also, e.g.*, NL-08-143, Letter from F. Dacimo, Entergy, to NRC Document Control Desk, "Additional Information Regarding License Renewal Application – Reactor Vessel Fluence Clarification," Attach. 1, Attach. 2 at thirteenth unnumbered page (Sept. 24, 2008) ("NL-08-143") (ENT000231) (providing a RPV fluence clarification and adding LRA Commitment 38); NL-13-122, Letter from F. Dacimo, Entergy, to NRC Document Control Desk, "Reply to Request for Additional Information Regarding the License Renewal Application" Attach. 2 at eleventh, fourteenth, and seventeenth unnumbered pages (Sept. 27, 2013) ("NL-13-122") (NYS000502) (noting closure of RPV-related Commitments 22, 31, and 38 for IP2).

Q51. Has NYS amended NYS-25 to address these amendments to the LRA related to RPVs?

A51. (ABC, NFA, JRS, RJD) No. The State has never amended NYS-25 to address or challenge these amendments related to RPVs.

Q52. Following the publication of the SER in 2009, did Entergy amend its LRA with respect to RVIs?

A52. (ABC, NFA, JRS, RJD, TJG, RGL) Yes. As the industry and NRC promulgated guidance on the management of the effects of aging on PWR RVIs, and consistent with Commitment 30, Entergy submitted an AMP for RVIs on July 14, 2010. *See* NL-10-063, Letter from F. Dacimo to NRC Document Control Desk, "Amendment 9 to License Renewal Application (LRA) – Reactor Vessel Internals Program" (July 14, 2010) ("NL-10-063") (NYS000313). This AMP described Entergy's program to manage the effects of aging on RVIs

[redacted]

using guidance developed from nearly a decade of extensive industry research and set forth in EPRI Materials Reliability Program documents MRP-227 and MRP-228. *See* NL-10-063, Attach. 1 at 82-84 (NYS000313); *see also generally* EPRI, MRP-227, Revision 0, Materials Reliability Program: Pressurized Water Reactor Internal Inspection and Evaluation Guidelines (Dec. 2008) ("MRP-227") (NYS000307); EPRI, MRP-228, Materials Reliability Program: Inspection Standard for PWR Internals (Jul. 2009) ("MRP-228") (NYS000323).

Q53. Did the NRC Staff approve Entergy's LRA in a safety evaluation report?

A53. (ABC, NFA, JRS, RJD) Yes. In October 2009, the NRC Staff issued its Safety Evaluation Report, which approved Entergy's AMPs and TLAA evaluations related to the RPV and RVIs, including Commitment 30, which the Staff found to meet the criteria in the SRP-LR. NUREG-1930, Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3 at 3-273 (Nov. 2009) ("SER") (NYS00326A-F).

Q54. Did the State challenge Entergy's 2010 RVI AMP?

A54. (ABC, NFA, JRS, RJD, TJG, RGL) Yes. On September 15, 2010, the State filed its "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 and Internal Components" ("2010 Motion for Leave"), available at ADAMS Accession No. ML103050402; which included as attachments "Additional Bases for Previously-Admitted Contention NYS-25 (Embrittlement of Reactor Pressure Vessels and Associated Internals)" ("2010 Amended Contention"), and the second "Declaration of Richard T. Lahey" ("2010 Lahey Decl.").

In summary, the State's 2010 filings alleged that the AMP was deficient because it did not: (1) consider the "synergistic" effects of embrittlement and metal fatigue on the RPV and its

RVIs; (2) provide sufficient details about when IPEC will conduct and complete baseline and periodic inspections of the RVIs; (3) include adequate inspection techniques to identify embrittlement issues for certain RVIs; and (4) provide sufficiently specific details or commitments regarding when and how IPEC will implement preventive or corrective actions to address any future embrittlement-related issues.

Q55. Did the State's 2010 Motion for Leave and associated filings raise any new issues related to the RPVs?

A55. (ABC, NFA, JRS, RJD, TJG, RGL) No. The Amended NYS-25 did not raise any new allegations regarding the RPVs.

Q56. Are you familiar with the first Amended Contention NYS-25, as admitted by the Board on July 6, 2011?

A56. (ABC, NFA, JRS, RJD, TJG, RGL) Yes. The Board admitted an amended NYS-25 as pled and as described in response to Question 54, above. *See* 2011 Order Admitting Amended NYS-25 at 27.

Q57. Did the NRC Staff's first supplemental safety evaluation report address the RVI AMP or any issues related to the RPVs? *See* NUREG-1930, Supp. 1, Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3 (Aug. 2011) ("SSER 1") (NYS000160).

A57. (ABC, NFA, JRS, RJD, TJG, RGL, MAG) No. SSER 1 addressed other issues, such as PWSCC in steam generator components and non-RVI-related issues regarding the FMP, but did not include a review of the RVI AMP or any issues related to RPVs, although it did include a cumulative summary of commitments in Appendix A. *See id.* at App. A. The NRC Staff's evaluation of metal fatigue and steam generator-related issues in SSER 1 is addressed in

[redacted]

our testimony on other contentions. *See generally* Revised Testimony of Entergy Witnesses Nelson F. Azevedo, Alan B. Cox, Jack R. Strosnider, Randy G. Lott, Mark A. Gray and Barry M. Gordon Regarding Contention NYS-26B/RK-TC-1B (Aug. 10, 2015) ("Entergy's NYS-26B/RK-TC-1B Testimony") (ENT000679); Revised Testimony of Entergy Witnesses Nelson F. Azevedo, Robert J. Dolansky, Alan B. Cox, Jack R. Strosnider, Timothy J. Griesbach, Barry M. Gordon, Randy G. Lott, and Mark A. Gray Regarding Contention NYS-38/RK-TC-5 (Safety Commitments) (Aug. 10, 2015) (ENT000699).

Q58. Following the admission of the first Amended Contention NYS-25, did Entergy take any action to implement the RVI AMP?

A58. (ABC, NFA, JRS, RJD, TJG, RGL) Yes. Consistent with Commitment 30, Entergy submitted its RVI Inspection Plan on September 28, 2011, two years prior to entering PEO for Indian Point Nuclear Generating Unit 2 ("IP2"). *See* NL-11-107, Letter from F. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "License Renewal Application – Completion of Commitment # 30 Regarding the Reactor Vessel Internals [I]nspection Plan" (Sept. 28, 2011) (NYS000314). Although only IP2 was within two years of entering the PEO, the RVI Inspection Plan covered both units. Entergy submitted the RVI Inspection Plan based on the new AMP in NUREG-1801, Revision 2.

Q59. Thereafter, did Entergy further amend its LRA with respect to RVIs?

A59. (ABC, NFA, JRS, RJD, TJG, RGL) Yes. After EPRI issued the NRC-approved generic industry aging management guidance for RVIs in MRP-227-A (discussed further below), Entergy submitted a revised RVI AMP and Inspection Plan for both IP2 and IP3 based on MRP-227-A on February 17, 2012. NL-12-037, Letter from F. Dacimo, Vice President, Entergy, to NRC Document Control Desk, "License Renewal Application – Revised Reactor Vessel

[redacted]

Internals Program and Inspection Plan Compliant with MRP-227-A" (Feb. 17, 2012) ("NL-12-037") (NYS000496).

Q60. Did the NRC Staff approve Entergy's 2012 RVI AMP?

A60. (ABC, NFA, JRS, RJD, TJG, RGL) Yes. IP2 and IP3 were among the first units in the U.S. fleet to prepare RVI AMPs based on the NRC Staff-approved guidance in MRP-227-A and to have such an AMP reviewed by the NRC Staff as part of an LRA. Given the unique timing circumstances of the IP2 and IP3 LRA, from 2012 through 2014 the NRC Staff issued to Entergy several detailed requests for additional information ("RAI") on this first-of-a-kind AMP. *See* NUREG-1930, Supp. 2, Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3 at B-2 to B-7 (Nov. 2014) ("SSER 2") (NYS000507). Following Entergy's submission of significant additional technical information in response to these RAIs, *see id*., the NRC Staff approved Entergy's revised RVI AMP and Inspection Plan as documented in SSER 2 issued on November 6, 2014. *See id*. at 3-26, 3-59.

Q61. What aging management activities are called for in the approved RVI AMP?

A61. (ABC, NFA, JRS, RJD, TJG, RGL) The RVI AMP monitors the effects of aging on the intended function of the RVIs through periodic and conditional examinations. *See* NL-12-037, Attach. 1 at 3 (NYS000496). The RVI AMP detects and evaluates cracking, loss of material, reduction of fracture toughness, loss of preload and dimensional changes of vessel internals components in accordance with MRP-227-A inspection recommendations and evaluation acceptance criteria. *Id*. at 3-4. The RVI AMP also addresses the remaining elements of an AMP, including preventive and corrective actions, and accounts for applicable future operating experience. *Id*. at 4-9. These activities are discussed in further detail in the balance of our testimony, below.

[redacted]

Q62. Did the State file a challenge to Entergy's 2012 RVI AMP?

A62. (ABC, NFA, JRS, RJD, TJG, RGL) Yes. On February 13, 2015, the State filed a second Amended Contention NYS-25. *See* Motion for Leave to Supplement Previously-Admitted Contention NYS-25 (Feb. 13, 2015) ("Second Amended NYS-25 Motion"), *available at* ADAMS Accession No. ML15044A508; New York State February 2015 Supplement to Previously-Admitted Contention NYS-25 (Feb. 13, 2015) ("NYS-25 Supplement"), *available at* ADAMS Accession No. ML15044A507; Declaration of Lisa S. Kwong (Feb. 13, 2015) ("Kwong Declaration") (attaching three documents), *available at* ADAMS Accession No. ML15044A495; and Declaration of Richard T. Lahey, Jr. (Feb. 13, 2015) ("2015 Lahey Declaration"), *available at* ADAMS Accession No. ML15044A492.

Similar to its claims in 2010, the State's Second Amended Contention alleged that the RVI AMP remained deficient because: (1) it does not address or manage the combined, or, as Dr. Lahey puts it, "synergistic" aging effects of embrittlement, fatigue, and other aging mechanisms; (2) it does not "maintain safety margins" during the PEO by, for example, repair or replacement of the RVIs in lieu of an AMP, and does not account for the full range of transient shock loads; (3) certain aging management activities are inadequate, including the lack of preventative or corrective actions, and the failure to submit acceptance criteria for the baffle-former bolt inspections; and (4) the Westinghouse EAF calculations prepared for Indian Point are allegedly inadequate. The Second Amended NYS-25 did not allege any deficiencies in the IPEC LRA regarding the RPVs. The Second Amended NYS-25 did not replace the first Amended Contention NYS-25, nor did it delete outdated or superseded challenges; it merely added new bases, in a cumulative manner, to NYS-25 as earlier amended.

[redacted]

Q63. Are you familiar with the Second Amended Contention NYS-25, as admitted by the Board?

A63. (ABC, NFA, JRS, RJD, TJG, RGL) Yes. On March 31, 2015, the Board admitted the amended contention "without altering or amending the contention as written." Order Admitting Amended Contentions at 10.

Q64. In its prefiled testimony and associated filings, did NYS pursue all of the claims originally raised in NYS-25 regarding RPVs?

A64. (ABC, NFA, JRS, RJD) No. The State's testimony and other filings on this contention have focused exclusively on alleged deficiencies in Entergy's AMP for RVIs, as opposed to issues relating to the RPV. As the State explains, the "focus of Contention 25 is Entergy's deficient AMP for RPVIs." State of New York, Revised Statement of Position, Contention NYS-25 at 17 (June 9, 2015) ("NYS Revised SOP") (NYS000481); *see also* State of New York, Initial Statement of Position, Contention NYS-25 at 10 (Dec. 22, 2011) ("NYS Initial SOP") (NYS000293).

With regard to RPVs, Dr. Lahey briefly alludes to some of his prior claims regarding the RPV when he refers to the "variance" that was "endorsed by the ACRS" to permit continued operation with RPV end-of-life Charpy USE values less than 50 ft-lbs. *See* Report of Dr. Richard T. Lahey, Jr. in Support of Contentions NYS-25 and NYS-26B/RK-TC-1B at 13 (Dec. 20, 2011) ("Report") (NYS000296); *see also* Pre-Filed Written Testimony of Dr. Richard T. Lahey, Jr. Regarding Contention NYS-25 at 28-31 (Dec. 22, 2011) ("Lahey 2011 Testimony") (NYS000294); Revised Pre-filed Written Testimony of Richard T. Lahey, Jr. Regarding Contention NYS-25 (June 9, 2015) ("Revised Lahey Testimony") (NYS000482). In his 2015 testimony, Dr. Lahey also refers to certain documents discussing Branch Technical Position

("BTP") 5-3, regarding the initial fracture toughness of RPV materials, suggesting that certain RPV embrittlement analyses may be non-conservative. *See* Revised Lahey Testimony at 74 (NYS000482). But Dr. Lahey and the State stop short of asserting any specific deficiency in Entergy's LRA regarding the RPVs.

To ensure a complete record, however, our testimony summarizes the information on this topic in the LRA and in the record of the NRC Staff's review of the LRA. We also address issues related to the RPVs in this testimony to the extent that certain concepts applicable to the RPVs may be relevant to an explanation of aging management of the RVIs.

Q65. What statements of position, testimony, and exhibits have you reviewed in preparation for the hearing on NYS-25?

A65. (ABC, NFA, JRS, RJD, TJG, RGL, MAG) At this time, we have reviewed the following documents, filed by the State, to the extent they are relevant to our testimony: NYS000293, NYS Initial SOP; NYS000294, Lahey 2011 Testimony; NYS000295, *Curriculum Vitae* of Dr. Richard T. Lahey, Jr.; NYS000296, Lahey Report; NYS000297, Supplemental Report of Dr. Richard T. Lahey, Jr. in Support of Contentions NYS-25 and NYS-26B/RK-TC-1B ("Supplemental Lahey Report"); NYS000481, NYS Revised SOP; NYS000482, Revised Lahey Testimony; and Exhibits NYS000146A-C, NYS000147A-D, NYS000160, NYS000161, NYS000195, NYS000298, NYS000300, NYS000301, NYS000303 through NYS000342, NYS000370, and NYS000483 through NYS000528.

In reviewing the statements of position, testimony, and reports, we note that, as with the Second Amended Contention NYS-25, the State has not replaced its 2011 submittals with updated materials in 2015, but merely added new information into the record in 2015. Thus, the State and Dr. Lahey's claims are often cumulative and overlapping, redundant in some areas, and

[redacted]

contradictory in others. Accordingly, we have focused our review on the most recent statement of position, testimony, and exhibits, filed on June 9, 2015. Nevertheless, we have reviewed all of these materials and our testimony represents our response to the totality of the State and Dr. Lahey's claims, as they can best be understood.

Q66. What other materials have you reviewed or do you expect to review in the preparation of your testimony?

A66. (ABC, NFA, JRS, RJD, TJG, RGL) We have reviewed numerous documents in preparing this testimony, including, for example, those portions of Entergy's LRA for IP2 and IP3 relating to Entergy's evaluation of the effects of aging on RPVs and RVIs, and the pertinent portions of NRC regulations and guidance documents such as:

- NUREG-1800, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, Rev. 1 (Sept. 2005) ("SRP-LR") (NYS000195);
- NUREG-1800, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, Rev. 2 (Dec. 2010) ("SRP-LR, Rev. 2") (NYS000161);
- NUREG-1801, Generic Aging Lessons Learned Report, Rev. 1 (Sept. 2005) ("NUREG-1801") (NYS00146A-C);
- NUREG-1801, Generic Aging Lessons Learned Report, Rev. 2 (Dec. 2010) ("NUREG-1801, Rev. 2") (NYS00147A-D);
- SER (NYS00326A-F);
- SSER 1 (NYS000160);
- SSER 2 (NYS000507);
- the NRC Staff Safety Evaluation for MRP-227-A (Letter from R. Nelson, NRC, to N. Wilmshurst, EPRI, "Revision 1 to the Final Safety Evaluation of Electric Power Research Institute (EPRI) Report, Materials Reliability Prog[ra]m (MRP) Report 1016596 (MRP-227), Revision 0, 'Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines'" (Dec. 16, 2011) ("SE for MRP-227-A") (ENT000230) ; and
- We will review the NRC Staff's prefiled testimony, statement of position, and exhibits, when those documents are filed.

[redacted]

We also have reviewed EPRI guidance documents regarding the aging management of RVIs, such as MRP-227-A (NRC000114A-F) and other supporting EPRI MRP reports. We also have reviewed other state-of-the-art scientific research on the effects of aging on RVIs, including domestic and foreign industry research, and domestic and foreign activities and operating experience from various laboratories and organizations, including the EPRI MRP, the PWROG, INPO, and the Materials Aging Institute.

We also reviewed Commission and Board decisions in other license renewal proceedings to understand how the Commission has further explained or clarified the scope and content of the applicable regulations and guidance documents as they may relate to license renewals generally and this contention specifically.

Q67. I show you what has been marked as Exhibit ENTR15001. Do you recognize this document?

A67. (ABC, NFA, JRS, RJD, TJG, RGL, MAG) Yes. It is a list of Entergy's exhibits, and includes those documents which we referred to, used, or relied upon in preparing this testimony, ENT00015A-B, ENTR00031, ENT000032, ENT000098, ENTR00184, ENTR00186, ENT000192, ENT000193, ENT000230, ENT000231, ENT000617 through ENT000677, ENT000679, and ENT000699.

Q68. I show you Exhibits ENT00015A-B, ENTR00031, ENT000032, ENT000098, ENTR00184, ENTR00186, ENT000192, ENT000193, ENT000230, ENT000231, ENT000617 through ENT000677, ENT000679, and ENT000699. Do you recognize these documents?

A68. (ABC, NFA, JRS, RJD, TJG, RGL, MAG) Yes. These are true and accurate copies of the documents that we have referred to, used, and/or relied upon in preparing the respective parts of our testimony. In those cases in which we have attached only an excerpt of a document as an exhibit, that is noted on Entergy's exhibit list.

Q69. How do these documents relate to the work that you do as an expert in forming opinions such as those contained in this testimony?

A69. (ABC, NFA, JRS, RJD, TJG, RGL, MAG) These documents represent the type of information that persons within our fields of expertise reasonably rely upon in forming opinions of the type offered in this testimony. Many are documents prepared by government agencies, peer-reviewed articles, or documents prepared by Entergy or the utility industry. We note at the outset that we do not offer legal opinions on the NRC regulations or adjudicatory decisions discussed in our testimony. However, reading those regulations and decisions as technical statements, and using our expertise and experience, we can interpret the meaning of those documents as they relate to how Entergy has addressed the issues raised in this contention. To the extent our testimony provides technical expert opinions on the requirements of NRC regulations, we believe that such opinions will be helpful to the Board inasmuch as they provide to the Board insights into Entergy's and the NRC Staff's processes for complying with the applicable regulations. *See* Licensing Board Order (Denying New York's Motion *in Limine* and Holding Riverkeeper's Motion *in Limine* in Abeyance) at 6 (June 1, 2012) (unpublished).

III. SUMMARY OF DIRECT TESTIMONY AND CONCLUSIONS

Q70. What is the purpose of your testimony?

A70. (ABC, NFA, JRS, RJD, TJG, RGL, MAG) The purpose of our testimony is to demonstrate that NYS-25 lacks merit, and, accordingly, should be resolved in Entergy's favor. In particular, we demonstrate that The IPEC RVI AMP is based on a comprehensive review of available technical information and operating experience, both domestic and foreign. It provides reasonable assurance that the effects of aging on the IP2 and IP3 RVIs will be adequately managed in accordance with applicable NRC regulations, guidance, and precedent. Specifically, the RVI AMP provides reasonable assurance that the IP2 and IP3 RVIs will be maintained consistent with the current licensing basis ("CLB") throughout the PEO. In our professional opinions, NYS-25 fails to show that The IPEC RVI AMP has any deficiency in this regard.

In addition, we explain that the LRA describes the evaluation of four TLAAs related to neutron irradiation embrittlement of the RPVs, and thus provides reasonable assurance that the effects of aging on the IP2 and IP3 RPVs will be adequately managed in accordance with applicable NRC regulations, guidance, and precedent. Accordingly, we conclude that NYS-25 fails to show any deficiency in the LRA pertaining to RPVs.

Q71. Please describe the scope of your testimony.

A71. (ABC, NFA, JRS, RJD, TJG, RGL, MAG) Our testimony identifies and describes the pertinent portions of the IP2 and IP3 LRA related to the management of the effects of aging on RVIs and RPVs. We show that the LRA complies with 10 C.F.R. Part 54 and is fully consistent with the guidance for acceptable AMPs in NUREG-1801, Revision 1 and Revision 2. We also show that MRP-227-A provides a reasonable, NRC-accepted approach for managing the effects of aging on RVIs, *see generally* MRP-227-A (NRC000114A-F), and that The IPEC RVI AMP is consistent with EPRI's MRP-227-A, as accepted by the NRC Staff in its Safety Evaluation. *See* SSER 2 at 3-55 (NYS000507). Thus, there is reasonable assurance that the effects of aging on the IPEC RVIs will be adequately managed so that their intended functions will be maintained consistent with the CLB, throughout the PEO, as required by 10 C.F.R. §§ 54.21(a)(3), 54.21(c)(1)(iii), and 54.29(a).

[redacted]

Q72. Please summarize Dr. Lahey's claims proffered in NYS-25.

A72. (ABC, NFA, JRS, RJD, TJG, RGL, MAG) The key allegations in Dr. Lahey's prefiled Testimony and Report, as summarized by the State, are that:

> (1) Entergy's AMP for RPVIs is not based on an analysis that addresses the critical issue of the synergistic degradation of RPVIs caused by the combination of embrittlement, metal fatigue, irradiation-assisted stress corrosion cracking ("IASCC"), and primary water stress corrosion cracking ("PWSCC"); (2) Entergy's analysis fails to adequately consider the full range of transient shock loads (thermal and decompression) to which RPVIs will be subjected in the event of various postulated accidents, such as a design basis accident ("DBA"), and thus fails to develop a plan which considers those shock loads, and their resultant impact on core coolability, in setting either inspection, acceptance or corrective action criteria; (3) the AMP does not include a commitment to take preventative actions or to implement corrective actions, nor does it provide specific, enforceable acceptance criteria for some components; and (4) the AMP relies on fatigue predictions which are non-conservative and may not accurately predict fatigue-induced component failures.

NYS Revised SOP at 17 (NYS000481). Our testimony demonstrates that these four challenges and Dr. Lahey's various additional, ancillary claims, lack merit.

Q73. Why do you disagree with Dr. Lahey's and the State's claims as set forth in NYS-25?

A73. (ABC, NFA, JRS, RJD, TJG, RGL) As addressed fully below, The IPEC RVI AMP and its supporting Inspection Plan adequately manage the effects of aging on RVIs, and fully address the issues identified by Dr. Lahey. In fact, NYS and Dr. Lahey have largely disregarded the substantial technical bases for the RVI AMP contained in MRP-227-A, its supporting technical reports spanning over a decade, and the plant-specific technical analyses submitted by Entergy for IPEC and reviewed by the Staff in SSER 2. The State and Dr. Lahey have ignored this important record of support, despite its availability through public sources and the mandatory disclosure process. In general, Dr. Lahey raises questions and engages in speculative exercises without considering the considerable efforts of EPRI, the industry, and Entergy to developing the RVI AMP based on operating experience and state-of-the-art research and NRC's rigorous review and approval process.

We also note that Dr. Lahey concludes his 2015 testimony by stating, with reference to his claims regarding "synergistic" degradation and "shock loads," as follows: "I want to stress that during the course of my involvement in these relicensing proceedings I have discovered what I believe to be some important new age-related safety concerns which, to the best of my knowledge, have not been previously considered in relicensing proceedings." Revised Lahey Testimony at 78 (NYS000482). We respectfully disagree with Dr. Lahey. As we explain throughout this testimony, Dr. Lahey has not "discovered" anything new. He has instead overlooked the substantial efforts of the industry and the NRC to fully address the issues he raises in NYS-25.

Q74. Please summarize the bases for your disagreement with Dr. Lahey's proffered claims regarding RVIs.

A74. (ABC, NFA, JRS, RJD, TJG, RGL, MAG) As we fully explain throughout this testimony, and as the NRC Staff concluded in SSER 2, Entergy's LRA for IP2 and IP3 demonstrates that the effects of aging on RVI components, including effects from neutron irradiation embrittlement and other aging mechanisms, will be adequately managed, as required under 10 C.F.R. §§ 54.21(a)(3) and (c)(1)(iii), *see* SSER 2 at 3-26 (NYS000507), and The IPEC RVI Inspection Plan "implements the elements of the RVI AMP in an acceptable manner." *See id*. at 3-59. The following summarizes the principal bases for our disagreement with NYS and Dr. Lahey:

- Contrary to Dr. Lahey's opinions, the IPEC RVI AMP is based on state-of-the-art engineering, systematic evaluation of known and potential degradation mechanisms, the resulting aging effects, and the consequences of those effects on RVIs. This systematic evaluation included consideration of potential "synergisms" involving multiple aging mechanisms. For more than a decade, the EPRI MRP conducted extensive engineering studies to identify the limiting RVI structures, components, and fittings, and prepare aging management guidelines based on those studies. These efforts are documented in MRP-227-A and its many supporting reports that form the basis for the IPEC RVI AMP. Although Dr. Lahey generally demands that a systematic analysis of RVIs be undertaken, he does not identify any disagreements with how EPRI conducted its systematic evaluations.

- The conditions addressed in the development of the MRP-227-A guidelines include the full range of design basis loads. The design basis loads are established in accordance with the CLB and do not change as a function of operating past the original license term. The guidelines are designed to provide reasonable assurance that the RVIs will continue to perform their intended functions, consistent with the CLB—including the consideration of accident loads and seismic loads—through the end of the PEO. Thus, contrary to Dr. Lahey's opinions, the industry, NRC, and Entergy fully considered design basis transient and accident loads as an integral part of the development of the RVI AMP. Once again, Dr. Lahey ignores rather than disputes the substantial technical information on these issues in the record.

- Contrary to the claims of Dr. Lahey and the State, the IPEC RVI AMP includes appropriate preventive and corrective actions, covers an appropriate scope of components, provides appropriate acceptance criteria, and appropriately inspects RVI components in a timely manner. In particular, the RVI AMP provides appropriate and well-defined aging management for the specific RVI components cited by Dr. Lahey, including the baffle-former bolts, clevis insert bolts, and the lower support column caps.

- The EAF evaluations prepared in support of the IPEC LRA, including EAF evaluations of RVI components, are fully documented, conservative engineering analyses that support a finding that the effects of fatigue, including the effects of reactor water environment, will be adequately managed. This issue is addressed in detail in Entergy's testimony on the metal fatigue contention, and the discussion of fatigue evaluations of RVI components is incorporated by reference and not repeated in this testimony. *See* Entergy's NYS-26B/RK-TC-1B Testimony at §§ V.C-E (ENT000679); Section VII.A.9, below.

Thus, as the NRC Staff concluded in SSER 2, Entergy's LRA for IP2 and IP3 demonstrates that the effects of aging on RVI components will be adequately managed, as required under 10 C.F.R. §§ 54.21(a)(3) and (c)(1)(iii), *see* SSER 2 at 3-26 (NYS000507), and The IPEC RVI Inspection Plan "implements the elements of the RVI AMP in an acceptable manner." *See id*. at 3-59.

[redacted]

IV. OVERVIEW OF PART 54 REQUIREMENTS AND GUIDANCE FOR LICENSE RENEWAL

A. License Renewal Regulations

Q75. Please identify and briefly describe the NRC aging management review ("AMR") requirements applicable to IPEC systems, structures, and components ("SSCs").

A75. (ABC, JRS) 10 C.F.R. Part 54 governs the matters that must be considered for purposes of operating license renewal, and in this adjudicatory proceeding. *See* Order Admitting NYS-25, LBP-08-13, 68 NRC at 67-68. Section 54.4 defines the plant SSCs that are within the scope of the license renewal rule based on their intended functions. Part 54 also requires an AMR of in-scope SSCs that are subject to AMR and evaluation of TLAAs.

Q76. How do the NRC regulations define TLAAs?

A76. (ABC, JRS) TLAAs are calculations and analyses that: (1) involve SSCs as delineated in § 54.4(a); (2) consider the effects of aging; (3) involve time-limited assumptions defined by the current (*e.g.*, 40-year) operating term; (4) were determined to be relevant by the licensee in making a safety determination; (5) involve conclusions or provide the basis for conclusions related to the capability of the SSC to perform its intended functions; and are (6) contained or incorporated by reference in the CLB. 10 C.F.R. § 54.3(a).

Q77. How do license renewal applicants evaluate TLAAs for the PEO?

A77. (ABC, JRS) NRC regulations require applicants to either: (1) demonstrate that a TLAA remains valid for the PEO (10 C.F.R. § 54.21(c)(1)(i)); (2) revise the analyses to remain valid for the PEO (10 C.F.R. § 54.21(c)(1)(ii)); or (3) demonstrate that the effects of aging on the intended functions of the SSC will be adequately managed for the PEO (10 C.F.R. § 54.21(c)(1)(iii)). The third option does not rely on demonstrating the validity of the TLAA throughout the PEO prior to issuance of a renewed license; instead, it relies upon an AMP for managing the effects of aging during the PEO.

Q78. What findings must the NRC make to issue a renewed operating license?

A78. (ABC, JRS) 10 C.F.R. § 54.29(a) requires a finding that the applicant has identified and has taken, or will take, actions to address the TLAAs that have been identified for review under 10 C.F.R. § 54.21(c). Specifically, the NRC must find that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the plant's CLB during extended operation. 10 C.F.R. § 54.29(a); *see also id.* § 54.21(c)(1)(iii). Importantly, the standard for this demonstration is one of "reasonable assurance." *See* 10 C.F.R. § 54.29(a); Nuclear Power Plant License Renewal; Revisions, 60 Fed. Reg. 22,461, 22,479 (May 8, 1995) ("Part 54 SOC") (NYS000016) ("the [license renewal] process is not intended to demonstrate absolute assurance that structures or components will not fail, but rather that there is reasonable assurance that they will perform such that the intended functions . . . are maintained consistent with the CLB"). The Commission has recognized that adverse aging effects generally are gradual and thus can be detected by programs that ensure sufficient inspections and testing. *See id.* at 22,475.

Q79. Dr. Lahey refers to the recent discovery of potential nonconservatisms in NUREG-0800, Branch Technical Position ("BTP") 5-3 as a reason why it is "very important to preserve – rather than erode – operational safety margins" as reactors age. What is your response?

A79. (ABC, NFA, JRS, TJG) As a general principal, we agree that it is important to preserve safety margins. But potential non-conservatisms in BTP 5-3 have no relevance to the

[redacted]

MRP-227-A or the IPEC RVI AMP at issue in this contention. BTP 5-3 is NRC guidance related to certain embrittlement calculations for RPVs, not RVIs.

Additionally, this is actually a good example of the level of inherent conservatisms in embrittlement evaluations for RPVs. As Dr. Lahey points out, the industry recently identified a potential non-conservatism in the BTP 5-3 methodology used to establish the unirradiated RT_{NDT} values for older design RPVs when some of the required testing information is not available. As shown in the analysis results presented at the February 19, 2015 NRC public meeting on this topic, other conservatisms and margin in the calculation of unirradiated RT_{NDT} values were more than sufficient to offset the potential non-conservatism identified in the BTP 5-3 methodology. *See* PWROG Presentation, "Material Orientation Toughness Assessment (MOTA) for the Purpose of Mitigating Branch Technical Position (BTP) 5-3 Uncertainties" (Feb. 19, 2015) (ENT000620).

B. <u>License Renewal Guidance</u>

Q80. What guidance documents has the NRC Staff issued to assist applicants in implementing the requirements of 10 C.F.R. Part 54?

A80. (ABC, JRS) The two primary guidance documents issued by the NRC Staff are NUREG-1801, the GALL Report (NYS000146) and NUREG-1800, the SRP-LR (NYS000195).

Q81. Please describe the function of SRP-LR as it relates to AMPs.

A81. (ABC, JRS) The SRP-LR provides guidance to NRC Staff for conducting their review of LRAs. It provides acceptance criteria for determining whether the applicant has met the requirements of the NRC's regulations in 10 C.F.R. § 54.21. *See* SRP-LR § 3.1.2 (NYS000195). For each of the SSCs identified as subject to aging management, one acceptable way to manage aging effects for license renewal is to use an AMP that is consistent with NUREG-1801. *See id.* § 3.0.1.

[redacted]

Q82. Dr. Lahey posits that the NRC's license renewal process and guidance, as set forth in the SRP-LR, is a "highly prescriptive process which does not encourage the discovery of any new aging-related safety concerns during ASLB hearings"? Report at 5 (NYS000296). Do you agree?

A82. (ABC, JRS, TJG) Not at all. The purpose of the SRP-LR is to provide guidance to the NRC Staff who perform safety reviews of applications to renew nuclear power plant licenses in accordance with 10 C.F.R. Part 54. Contrary to Dr. Lahey's statements, the SRP-LR and its companion document, NUREG-1801, explicitly encourage consideration of new aging-related information, primarily through the consideration of operating experience:

> If operating experience or other information indicates that a certain aging effect may be applicable and an applicant determines that it is not applicable to its plant, the reviewer may question the absence of this aging effect unless the applicant has provided the basis for this determination in its license renewal application.

SRP-LR, App. A at A.1-2 (NYS000195). This guidance highlights the need to consider aging effects that have been observed in operating experience or in realistic research or laboratory experiments, as opposed to general theories or speculation. It is consistent with the Commission's desire to avoid transforming the license renewal process into an "open-ended research project," as explained in the original statement of considerations for Part 54. *See* Part 54 SOC, 60 Fed. Reg. at 22,469 (NYS000016). Notably, the SRP-LR and its companion document, NUREG-1801, are periodically revised by the NRC to reflect on-going operating experience and other new technical information.

Q83. Turning to the latter document, please describe the origin and purpose of NUREG-1801.

A83. (ABC, JRS) In an NRC Staff paper, SECY-99-148, "Credit for Existing Programs for License Renewal," dated June 3, 1999, the Staff described options for crediting

[redacted]

existing licensee AMPs to satisfy the requirements of Part 54. By a Staff Requirements Memorandum ("SRM") dated August 27, 1999, the Commission directed the Staff to develop NUREG-1801, which the Staff first issued in 2001, to document its evaluation of existing aging management programs. NUREG-1801 is referenced as a technical basis document in the SRP-LR.

The purpose of NUREG-1801, also referred to as the "GALL Report," is to provide generic AMRs of SSCs in the scope of license renewal. It also identifies and describes AMPs that the NRC Staff has found acceptable for managing the effects of aging on SSCs, based in part on the experience with evaluations of existing programs at operating plants during the initial license period. *See* NUREG-1801, Rev. 1, at 1-2 (NYS00146A). An applicant may reference NUREG-1801 in an LRA to show that its AMPs are consistent with those reviewed and approved in NUREG-1801. *See id.* at 3 (NYS00146A).

The original GALL Report was issued in July 2001. *See* NUREG-1800, Rev. 1 at 4.4-6 (NYS000195). Revision 1 was issued in September 2005. *See* NUREG-1801 at i (NYS00146A). Revision 2 was issued in December 2010. *See* NUREG-1801, Rev. 2 at i (NYS00147A). The revisions reflect further lessons learned from the reviews of LRAs, operating experience, and other public input including industry comments. *See* NUREG-1801, Rev. 2 at 3 (NYS00147A).

Q84. Was NUREG-1801 revised following the preparation, submittal, and NRC Staff review of the IPEC LRA?

A84. (ABC, JRS) Yes. In December 2010, the NRC Staff issued NUREG-1801, Revision 2. This revision was issued more than three years after the IPEC LRA was submitted, and more than a year after the NRC Staff issued its original SER on the IPEC LRA in August

2009. Therefore, Entergy prepared the IPEC LRA using the guidance in NUREG-1801, Revision 1.

Q85. With respect to the issues raised in this contention, are there significant changes in NUREG-1801, Revision 2? If so, please explain.

A85. (ABC, JRS, TJG) Yes, there were significant changes. NUREG-1801, Revision 1 did not provide an AMP for pressurized water reactor ("PWR") RVIs. Instead, it specified that applicants provide a commitment to:

> (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.

NUREG-1801, Revision 1 at IV B2-3 to IV B2-26 (NYS00146B) (in the "Aging Management Programs" column).

In contrast, NUREG-1801, Revision 2 (NYS00147D) contains a new AMP (XI.M16A) addressing PWR RVIs. This new AMP relies on the implementation of EPRI report MRP-227, and applies the guidance in that document.

Q86. Did Entergy revise its LRA to account for the changes in NUREG-1801, Revision 2?

A86. (ABC, JRS, TJG) Yes. As we explain in Section VII.A, Entergy submitted the RVI inspection plan based on the new AMP in NUREG-1801, Revision 2.

Q87. With respect to the RPV embrittlement, are there any significant changes in NUREG-1801, Revision 2?

A87. (ABC, JRS) For RPV embrittlement issues, there are no significant differences between NUREG-1801, Revisions 1 and 2. *See* NRC Regulatory Issue Summary ("RIS") 2011-

[redacted]

05, Information on Revision 2 to the Generic Aging Lessons Learned Report for License Renewal of Nuclear Power Plants (July 1, 2011) (ENT000192).

Q88. Please describe the basic format and content of NUREG-1801, Revision 1.

A88. (ABC, JRS) NUREG-1801, Revision 1 includes tables summarizing various structures and components, the materials from which they are made, the environment to which they are exposed, the relevant aging effects (*e.g.*, cracking due to fatigue, loss of material through pitting, leaching or corrosion), the AMP found to manage the particular aging effect in that component, and whether "further evaluation" is necessary. NUREG-1801, Rev. 1, at 5 (NYS00146A). The evaluation results documented in NUREG-1801, Revision 1 indicate that many existing programs are adequate without change to manage aging effects on particular structures or components for purposes of license renewal. *Id*. at 4. NUREG-1801, Revision 1 also contains recommendations concerning specific areas for which existing programs should be augmented for purposes of license renewal.

Q89. How does the NRC evaluate AMPs in light of NUREG-1801?

A89. (ABC, JRS) The NRC Staff reviews AMPs listed in a license renewal application for consistency with NUREG-1801. *See* SER at 3-1 to 3-5 (NYS00326B). For plant-specific AMPs, the NRC Staff will review whether the applicant's new AMP addresses the ten elements of an AMP, as specified in SRP-LR, Appendix A. *See* SRP-LR, Rev. 2 at A.1-3 (NYS000161).

V. TECHNICAL BACKGROUND ON AGING MANAGEMENT OF RVIs AND RPVs

A. Overview of the RPVs and RVIs

Q90. To provide some background for discussions of Dr. Lahey's claims, please describe the RPVs at IPEC.

A90. (ABC, RJD, NFA, TJG, RGL, JRS) PWRs, such as IP2 and IP3, contain primary coolant under high pressure flowing through the core in which heat is generated by the fission process. The reactor coolant system ("RCS") provides a boundary for containing the coolant under operating temperature and pressure conditions. See LRA at 2.3-2 (ENT00015A).

The RPV, which contains the reactor core and RVIs, is a key part of the reactor coolant pressure boundary. *See id.* at 2.3-3 (ENT00015A). For IP2 and IP3, the RPV components include the shell, top and bottom heads, closure head studs, primary nozzles and safe ends, and control rod drive mechanism ("CRDM") penetrations. *See id.* at 2.3-14. The RPV "beltline" is the region of the RPV predicted to experience sufficient neutron radiation to be considered in the selection of the most limiting material and consists of the area that directly surrounds the effective height of the active core and adjacent regions of the RPV. *See* 10 C.F.R. § 50.61(a).

Q91. Please describe the RVIs at IPEC.

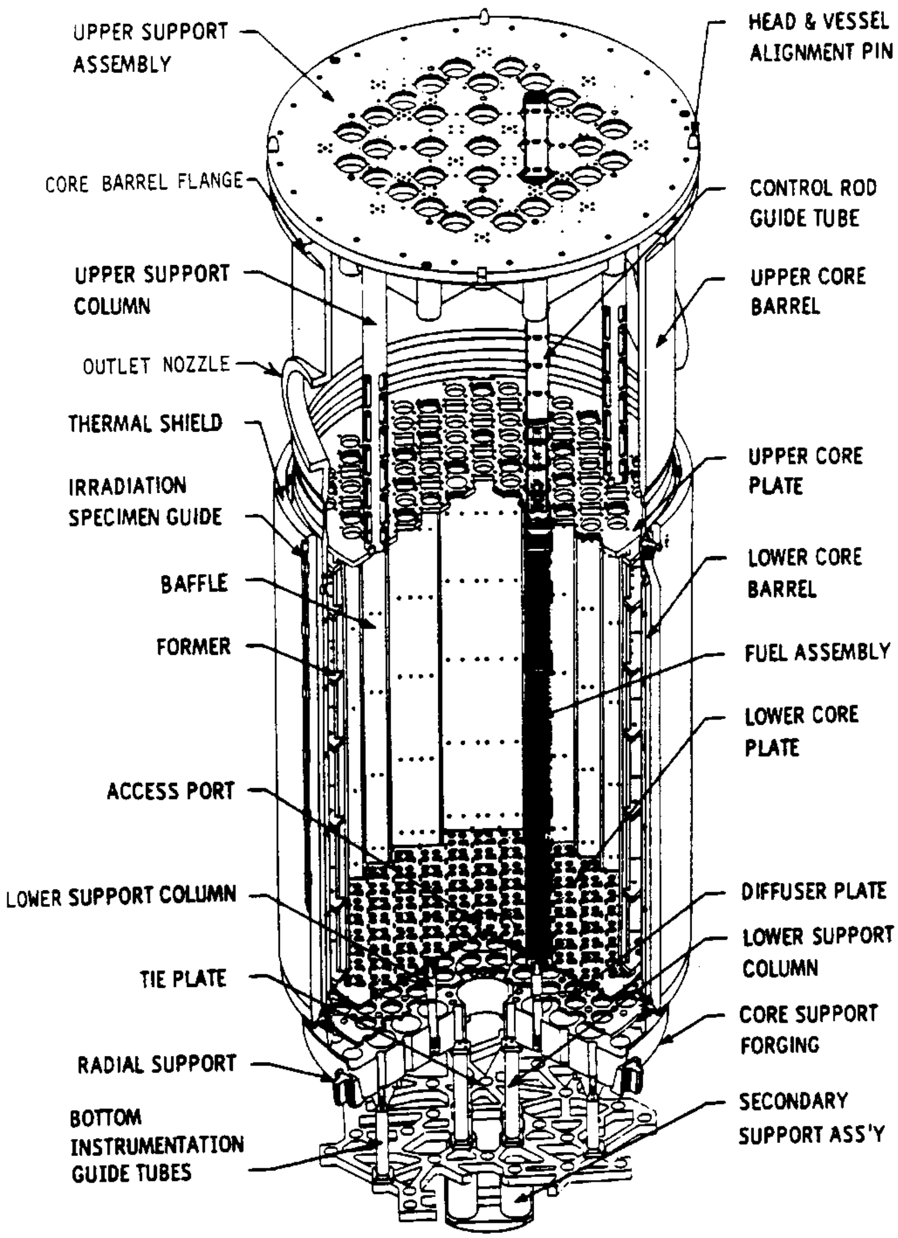
A91. (ABC, RJD, NFA, TJG, RGL, JRS) The RVIs are located inside the RPV. The RVIs for IP2 and IP3 consist of two assemblies: an upper internals assembly and a lower internals assembly. *See* MRP-227-A, encl. at 3-10 to 3-12 (NRC000114A); NL-12-037, Attach. 2 at 4-6 (NYS000496). The major sub-assemblies that constitute the upper internals assembly are the: (1) upper core plate; (2) upper support column assemblies; (3) control rod guide tube assemblies; and (4) upper support plate. *See id.*, Attach. 1 at 4. The lower internals assembly includes the lower core plate, the core barrel, the baffle-former assembly, and other attached components. *See id*. at 5.

Q92. Can you provide a diagram to show the RVIs?

A92. (ABC, RJD, NFA, TJG, RGL, JRS) Yes. Figure 1 provides an overview of the location of the various RVI components within the RPV for a *typical* four-loop Westinghouse PWR similar, but not identical, to IP2 and IP3. *See also* Westinghouse, WCAP-17894-NP, Rev. 1, Component Inspection Details Supporting Aging Management of Reactor Internals at Indian Point Unit 2 at A-2 (Sept. 2014) (ENT000621) (showing IP2-specifc internals, but providing less detail than Figure 1); Westinghouse, WCAP-17901-NP, Rev. 1, Component Inspection Details Supporting Aging Management of Reactor Internals at Indian Point Unit 3 at A-2 (Sept. 2014) (ENT000622) (showing IP3 internals, but providing less detail than Figure 1). Key components that are clearly visible on Figure 1 include the upper core plate, the lower core plate, and the core barrel (which extends from the upper core plate to the lower core support).

[redacted]

Figure 1: Overview of Typical Westinghouse Internals



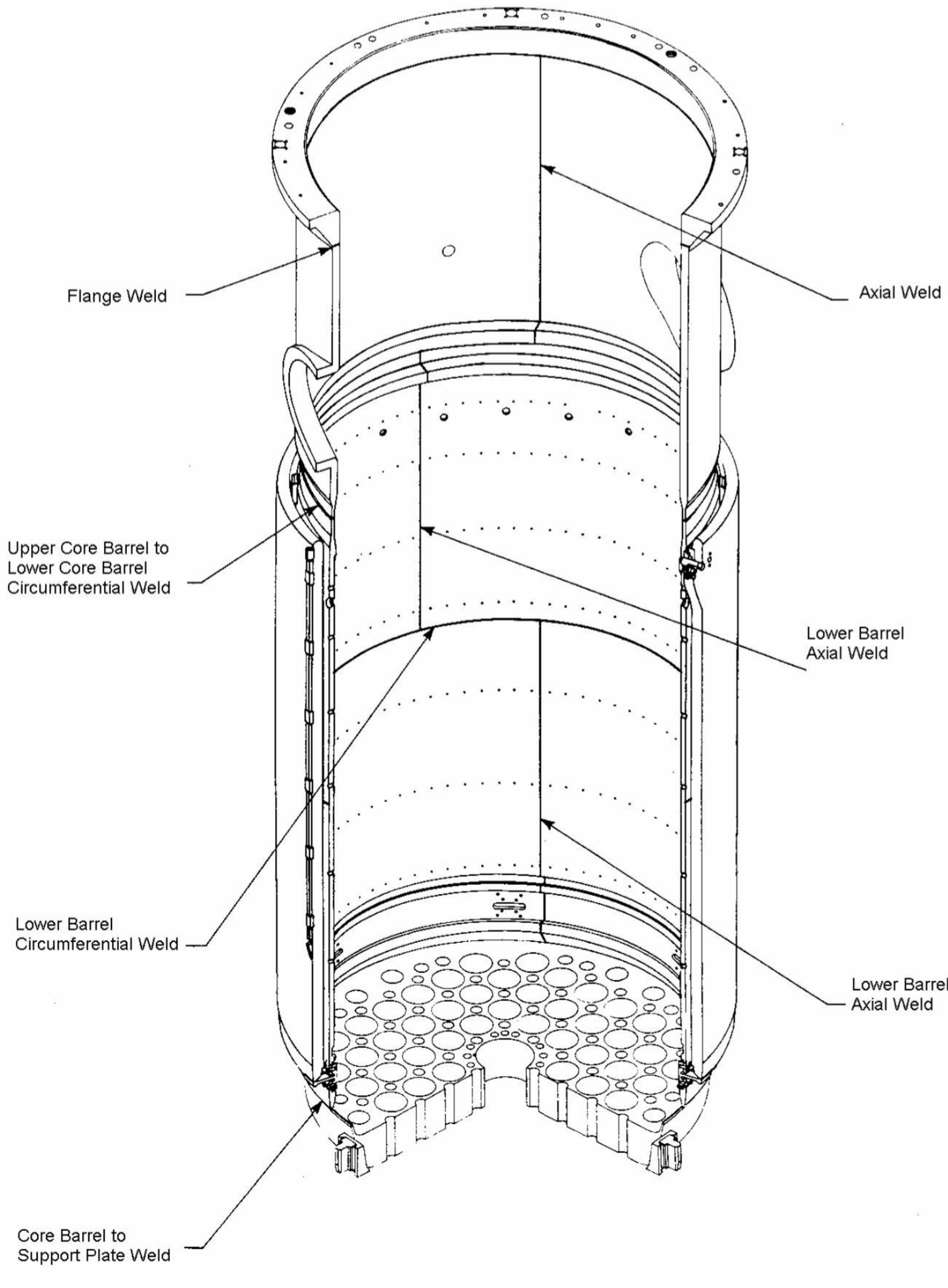
Source: WCAP-14577, Rev. 1-A at 2-39 (NYS000341).

Q93. Which RVIs maintain fuel assembly alignment?

A93. (ABC, RJD, NFA, TJG, RGL, JRS) Fuel assembly alignment (*i.e.*, "core geometry") is maintained by the following RVI components: the upper and lower core plates, baffle plates, baffle formers, the core barrel and the bolts that connect these components. Figure 2 shows the core barrel and its major associated welds in greater detail. Figure 3 shows a slice of the baffle-former assembly that sits within the core barrel, forming an interface between the core and core barrel. In Figure 3, the baffles are the vertical components, while the formers are between the baffles and the core barrel. Figure 3 also shows some of the bolts associated with the baffle-former assembly. We note that the RVI components that maintain fuel assembly alignment are the subject of several specific claims in NYS-25 and will be discussed further below. NL-12-037, Attach. 2 and MRP-227-A also provide a number of additional diagrams showing further details of the RVIs.

[redacted]

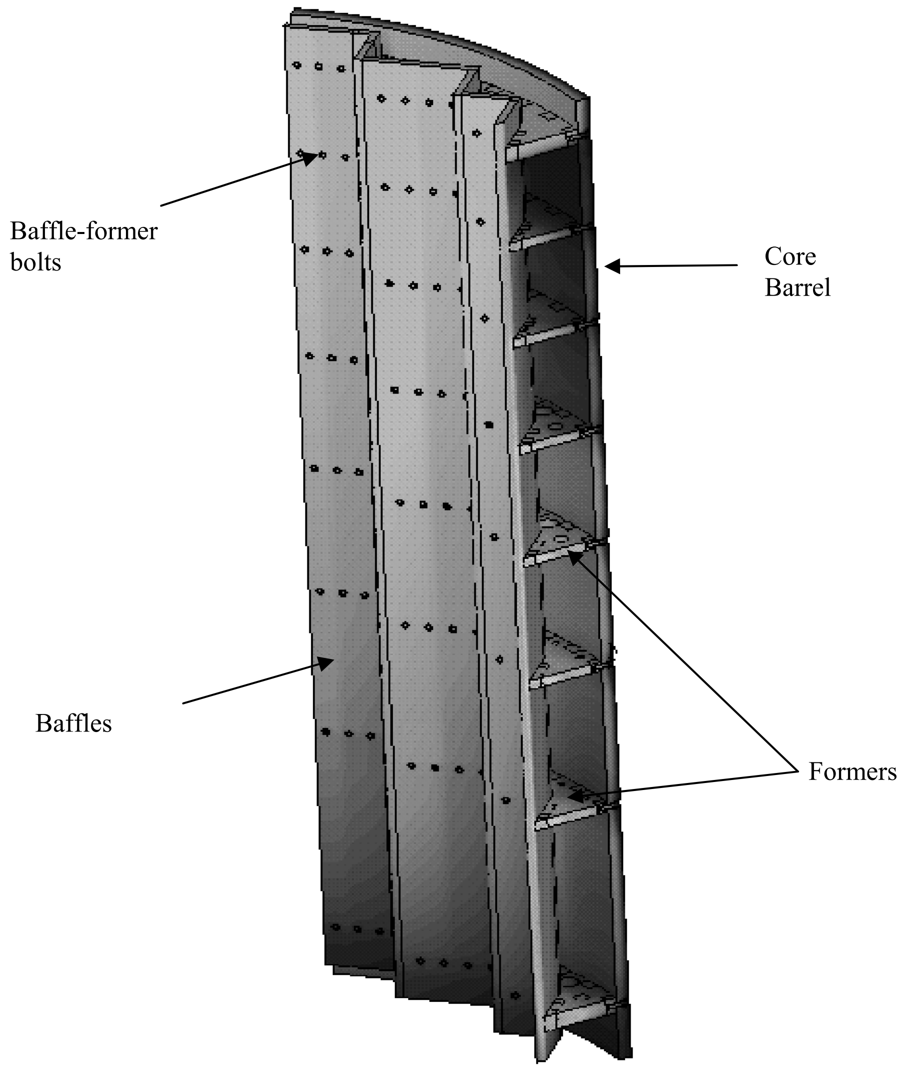
Figure 2: Major Fabrication Welds in Typical Westinghouse Core Barrel



Source: MRP-227-A at 4-61(NRC000114B) ; NL-12-037, Attach. 2 at 8 (NYS000496).

[redacted]

Figure 3: Westinghouse Baffle-Former Assembly (located inside the core barrel)



Adapted from: MRP-227-A at 4-64 (NRC000114B); NL-12-037, Attach 2 at 11 (NYS00496).

Q94. What are the intended functions of the RVIs at IPEC?

A94. (ABC, RJD, NFA, TJG, RGL, JRS) The RVIs direct the coolant flow, support the reactor core, and guide the control rods. LRA at 2.3-3 (ENT00015A). The RVIs do not constitute part of the reactor coolant pressure boundary. *See id.* at 2.3-2; *compare id.* tbls. 3.1.2-1-IP2 and 3.1.2-1-IP3 ("pressure boundary" is the intended function of RPV components) *with id*. tbls. 3.1.2-2-IP2 and 3.1.2-2-IP3 (showing "structural support," "flow distribution," and "shielding" for specified RVIs, but *not* pressure boundary functions); *see also* NL-10-063, Attach. 1 at 18-81 (showing updated Tables 3.1.2-2-IP2 and 3.1.2-2-IP3) (NYS000313).

B. <u>Scope of Components Covered by the RVI AMP</u>

Q95. Does Dr. Lahey correctly describe the RVIs?

A95. (ABC, RJD, NFA, TJG, RGL, JRS) No. He purports to list the various structures, components, and fittings that are included under the term "reactor pressure vessel internals." Revised Lahey Testimony at 11-12 (NYS000482). Dr. Lahey's list, however, differs from, and includes items that are not listed in the State's own exhibit that Dr. Lahey references (NYS000306), and omits items that are listed. As such, Dr. Lahey's description of the RVIs is not complete or accurate.

Q96. Does Dr. Lahey apply incorrect terminology in discussing the IPEC RVIs?

A96. (ABC, RJD, NFA, TJG, RGL, JRS) Yes. Some of his terminology (such as references to the "lower support column and mixer," or "upper mixing vanes") appears to be taken from BWR design or from non-Westinghouse nomenclature and is therefore not relevant to IPEC. Nevertheless, we have sought to interpret and use correct terminology in addressing his concerns.

Q97. Does Dr. Lahey also include RPV components in his description of RVIs?

A97. (ABC, RJD, NFA, TJG, RGL, JRS) Yes. What Dr. Lahey and the State repeatedly refer to as "intermediate shells" are part of the RPV, not the RVIs. Therefore, the effects of aging on these components are not considered in the RVI AMP, but instead under the various portions of the LRA that address the RPV. *See*, *e.g.*, LRA at 4.1-3, tbl. 4.1-1 & 4.1-5, tbl. 4.1-2 (listing TLAAs applicable to the RPVs) (ENT00015B); NUREG-1801, Revision 1 at XI M-102 to M-104 (NYS00146C).

[redacted]

Q98. Where can one find an accurate and complete list of RVIs at IP2 and IP3?

A98. (ABC, RJD, NFA, TJG, RGL, JRS) Exhibit NYS000496, NL-12-037, Attach. 2 at 62-64, provides a complete and correct list of RVI sub-assemblies at IPEC, and breaks those sub-assemblies down into their constituent components.

Q99. On the topic of what components are RVIs and what are not, does Dr. Lahey raise specific concerns about components that are not RVIs (or even part of the RPV)?

A99. (ABC, NFA, JRS, RJD) Yes. In Dr. Lahey's 2011 Report, he asserted that "any [AMP] concerning embrittlement of the [RVIs] should include the control rods . . . ." Report at 20 (NYS000296). But control rods are not subject to AMR for two reasons. First, they perform their intended function with moving parts or a change in configuration. *See* 10 C.F.R. § 54.21(a)(1)(i); LRA at 2.3-14 (ENT00015A). Thus, as the NRC Staff concluded, the control rods are active components not subject to AMR. *See* SER at 2-39 (NYS00326A); Part 54 SOC, 60 Fed. Reg. at 22,477 (NYS000016).

Second, control rods are considered consumables. The NRC has excluded from the license renewal review process those components that are "subject to replacement based on a qualified life or specified time period." 10 C.F.R. § 54.21(a)(1)(ii). At IPEC control rods are replaced approximately every 15 effective full-power years. *See* RIS-2011-07 at 3 (NYS000310); Nuclear Energy Institute, Industry Guidelines for Implementing the Requirements of 10 C.F.R. Part 54 – The License Renewal Rule, Rev. 6, at 52 (Jun. 2005) ("NEI 95-10") (ENT000098) (stating that control rods "do[] not qualify as a TLAA because the design life of control rods is less than 40 years'). Notably, in his 2015 submittal, Dr. Lahey acknowledges that "the control rods . . . can be replaced as required." Revised Lahey Testimony at 13 (NYS000482). However, he suggests that they should still be subject to AMR because "other

[redacted]

associated components" are not normally replaced. *Id*. But he provides no support for his assertion that these other components somehow dictate applicability of AMR to the control rods.

Q100. Does the RVI AMP address the remaining control rod-related components Dr. Lahey discusses?

A100. (ABC, NFA, JRS, RJD) Yes. The control rod guide tube assemblies, including the guide plates (cards) and the lower flange welds are subject to AMR, and the effects of aging on these components are managed through the RVI AMP. *See* NL-12-037, Attach. 2 at 4-5, Attach. 1 at 6-8 (NYS000496). This is clearly shown in the tables reproduced in response to Question 139, below.

Dr. Lahey, therefore, is incorrect when he asserts that Entergy has claimed the "guide tubes, plates, pins and welds" associated with the control rods are not RVIs. Revised Lahey Testimony at 12 (NYS000482); *see also id*. at 13; Lahey 2011 Testimony at 31-32 (NYS000294). Moreover, to the extent that Dr. Lahey suggests that these components are subject to significant neutron irradiation embrittlement, *see* Revised Lahey Testimony at 12 (NYS000482), he is incorrect, because these components are not located in the high-fluence region and therefore are not significantly affected by irradiation embrittlement.

Q101. Dr. Lahey also raises concerns that control rod "stub tube welds" or "J-groove welds" are not within the scope of the RVI AMP. *See* Revised Lahey Testimony at 45-46 (NYS000482). He also claims that these components cannot be fully inspected. *See* *id*. at 45. How are those components addressed in Entergy's LRA?

A101. (ABC, NFA, JRS, RJD) The RPV head penetration nozzle welds, sometimes referred to as the "J-groove welds," are not RVIs (or even part of the RPV) but are instead part of the RPV head. Aging effects applicable to the J-groove welds on the CRDM head penetrations

[redacted]

are managed under the Reactor Vessel Head Penetration Inspection AMP. *See* LRA at 3.1-47 tbl. 3.1.2-1-IP2; *id.* at 3.1-60 tbl. 3.1.2-1-IP3 (ENT00015A); SER at 3-48 (NYS000326B). The adequacy of the Reactor Vessel Head Penetration Inspection AMP is unchallenged in this contention. Nor is there any other contention in this proceeding challenging that program.

In any event, under the Reactor Vessel Head Penetration Inspection AMP, Entergy inspects the J-groove welds using a volumetric examination every outage at both IP2 and IP3 as required by the ASME Code. *See* Cases of ASME Boiler and Pressure Vessel Code, Case N-729-1, "Alternative Examination Requirements for PWR Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds," (Section XI, Division 1) (Mar. 28, 2006) (ENT000623). Entergy performed engineering evaluations to address certain inaccessible areas of these components and submitted requests for relief for IP2 and IP3 demonstrating that the stresses in these regions are low and unlikely to result in crack initiation or crack propagation, and the NRC reviewed these evaluations and approved the relief requests. *See* Letter from N. Salgado, NRC, to Vice President, Operations, Entergy, "Indian Point Nuclear Generating Unit No. 2 – Relief from the Examination Area for Reactor Vessel Head Penetration Nozzles (TAC No. ME1658)" at 1 (Mar. 1, 2010) (ENT000624); Letter from N. Salgado, NRC, to Vice President, Operations, Entergy, "Indian Point Nuclear Generating Unit No. 3 – Relief Requests RR-3-45 and RR-3-46 for Reactor Vessel Head Penetrations Examination (TAC Nos. ME0411 and ME0412)" at 1 (July 8, 2009) (ENT000625). Dr. Lahey provides no critique of Entergy's evaluations of this issue.

[redacted]

Q102. Dr. Lahey raises the possibility of "aggressive corrosion and wasting of the . . . upper head of the RPVs," such as happened at the Davis-Besse plant in 2002. *See* Revised Lahey Testimony at 45-46 (NYS000482). Is that a potential issue for IPEC?

A102. (ABC, NFA, JRS, RJD) The effects of aging on the RPV heads are managed under the Reactor Vessel Head Penetration Inspection AMP, not the RVI AMP. *See* LRA at 3.1-47 tbl. 3.1.2-1-IP2; *id.* at 3.1-60 tbl. 3.1.2-1-IP3 (ENT00015A); SER at 3-48 (NYS00326B). Specifically, under that program, Entergy conducts visual inspections of the outside surface of the RPV head, as required by the ISI program. In any event, Dr. Lahey's references to events at the Davis-Besse plant are misplaced. *See* Letter from P. Milano, NRC, to M. Kansler, Entergy, "Bulletin 2002-01, 'Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity,' 15-Day Response for Indian Pont Nuclear Generating Unit Nos. 2 & 3 (TAC Nos. MB4550 and MB4551)," at 1 (Nov. 12, 2002) (ENT000626) ("The NRC staff . . . has concluded that IP2 and 3 do not appear to have conditions similar to those which led to the degradation at the Davis-Besse Nuclear Power Station."). Thus, Dr. Lahey does not raise a valid concern.

Q103. In 2011, Dr. Lahey referred to NRC Information Notice 2011-13 (June 29, 2011) (NYS000329), as showing that IASCC has led to the failure of control rod blades and a reduction in control rod worth. Report at 15 (NYS000296). How do you respond?

A103. (NFA, JRS, ABC, RJD) First, from a technical perspective, the issues raised in Information Notice 2011-13 do not apply to IPEC. That document discusses cracking in Marathon control rod blades in an aggressive environment from oxidizing BWR water. IPEC—a PWR—does not have control blades at all. IPEC also uses Westinghouse-designed control rods.

[redacted]

Second, as we have previously explained, the control rods are not subject to AMR, and therefore are not within the scope of the AMR or the RVI AMP.

C. <u>Materials Used in the IPEC RVIs and RPVs</u>

1. Overview of Materials Used in IPEC RPVs and RVIs

Q104. What materials are used to construct the RPV and RVIs, and do the differences in materials affect how aging effects are managed for the two types of components?

A104. (RJD, NFA, TJG, RGL, JRS) The IP2 and IP3 RPVs are constructed primarily of low-alloy (carbon) steel, with stainless steel cladding. *See* LRA at 3.1-2, 3.1-43 (ENT00015A).

The RVIs, in contrast, are made of wrought austenitic stainless steel, other stainless steels including Cast Austenitic Stainless Steel ("CASS"), or nickel-based alloys. *See id*. at 3.1-3. Tables 3.1.2-2-IP2 and 3.1.2-2-IP3 in Entergy's LRA, as updated by the RVI AMP, provide specific information on the materials used in the various RVI components. *See* NL-10-063, Attach. 1 at 18-81 (NYS000313). As we will explain in detail, the materials used in the RVIs exhibit less temperature-dependent changes in unirradiated mechanical properties than low-alloy materials used in the RPV. The threshold fluence for irradiation embrittlement is much higher than the threshold in the RPV steels and the RVI steels do not exhibit the ductile-to-brittle transition that characterizes the RPV steels.

Q105. Further describe the different types of materials used in the IPEC RVIs and their physical characteristics

A105. (NFA, RJD, TJG, RGL, JRS) The various categories of materials that are used in the IPEC RVIs are listed in Attach. 2 to NL-12-037 at 62-64 (NYS000496), and are described, in turn, below. In general, the materials used in the RVIs are tougher than low-alloy materials at all temperatures, showing a more gradual, continuous increase in toughness with increasing temperature.

<u>*Wrought Austenitic Stainless Steels*</u>: The majority of the IPEC RVI components are fabricated from wrought austenitic stainless steels. *See* NL-12-037, Attach. 2 at 62-64 (NYS000496). These materials were chosen for their high strength, high fracture toughness, corrosion resistance, and the ability to withstand high temperatures and fluences. These austenitic stainless steel materials do not generally exhibit a transition temperature from ductile to brittle behavior. *See* G. Was, FUNDAMENTALS OF RADIATION MATERIALS SCIENCE: METALS AND ALLOYS; PART III: MECHANICAL EFFECTS OF RADIATION DAMAGE at 689-90 (2007) ("Was Text") (ENT000627). Furthermore, even though an increase in strength and decrease in toughness do occur when exposed to neutron irradiation, these materials retain their resistance to fast fracture within the operating temperature range of interest for PWRs. *See id.* The only exceptions are wrought austenitic stainless steel materials with high amounts of cold-working (> 20%), *see id.*, but these materials are not present at IPEC, *see* NL-14-117, Letter from F. Dacimo, Vice President, Entergy, to NRC Document Control Desk, Reply to Request for Additional Information Regarding the License Renewal Application, Indian Point Nuclear Generating Unit Nos. 2 & 3, Attach. 2 at 2-3 (Sept. 8, 2014) (NYS000506).

<u>*Cast Austenitic Stainless Steels*</u>: Some of the RVI components are CASS materials which contain varying amounts of delta ferrite. *See, e.g.*, NL-12-037, Attach. 2 at 63 (NYS000496) (identifying CASS as the material for the upper support column bases). Delta ferrite is a separate phase that forms within the material upon cooling of certain alloy types of CASS materials. The ferrite phase is critical in determining the mechanical properties and corrosion resistance of CASS materials. The presence of this delta ferrite in a primarily austenitic structure is why

[redacted]

CASS materials are often referred to as "duplex" stainless steels. CASS components include the lower support columns in some Westinghouse plants such as IP2 and IP3. CASS materials with high delta ferrite can experience some loss of ductility at low temperatures outside the range of concern, although they typically do not exhibit a well-defined transition temperature and may be screened out from susceptibility. *See* I. Tylek and K. Kuchta, Mechanical Properties of Structural Stainless Steels, 4-B TECHNICAL TRANSACTIONS | CIVIL ENGINEERING 59, 71-72 (2014) (ENT000628); Letter from C. Grimes, NRC, to D. Walters, NEI, "License Renewal Issue No. 98-0030, 'Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components,'" Encl. at 7 (May 19, 2000) (ENT000629).

At IP2 and IP3, the only CASS materials located in a high-fluence region are the lower support column caps ("LSCCs"). *See* NL-12-037, Attach. 2 at 62-64 (NYS000496). The upper instrumentation conduits and supports, upper support column assemblies, and the lower support casting, shown in this exhibit are in low-fluence regions. The IP2 and IP3 LSCCs are not made of high delta ferrite materials, so there is no concern about a potential transition from ductile to brittle behavior for these CASS components. *See* NL-14-013, Letter from F. Dacimo, Entergy to NRC Document Control Desk, "Additional Information Regarding the License Renewal Application – Action Item 7 from MRP-227-A," Attach. 1 at 3, 4 (Jan. 28, 2014) (NYS000503).

Austenitic Stainless Steel Welds: Austenitic stainless steel welds also contain a small percentage of delta ferrite and display some of the same characteristics as CASS or duplex stainless steels—that is, reduced toughness at low temperatures outside the range of concern, and with no defined transition temperature.

Nickel Alloys: Finally, some IPEC RVIs components are made of nickel alloys or other alloy materials, none of which exhibit a transition temperature over the temperature range of interest. *See* NL-12-037, Attach. 2 at 62-64 (NYS00496). For example, the guide tube support pins, support pin nuts, and clevis insert bolts at IP2 are fabricated from precipitation-hardened Alloy X-750. *See id*. The clevis insert bolts at IP3 are Alloy X-750, while the clevis inserts at both units are Alloy 600. *See id*. These precipitation-hardened materials do not exhibit a transition temperature, as low-alloy ferritic materials do. *See* W.J. Mills, *Effect of Temperature on the Fracture Toughness Behavior of Inconel X-750*, in ASTM STP 733: FRACTOGRAPHY AND MATERIALS SCIENCE 98, 98-114 (L.N. Gilbertson and R.D. Zipp, eds., 1981) (ENT000630). Moreover, RVI components made of these materials are all located outside regions with the highest neutron fluence. Therefore, these specific components at IPEC do not experience significant irradiation effects. *See* EPRI, MRP-191, Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design at 3-8, tbl. 3-6 (Nov. 2006) ("MRP-191") (NYS000321).

2. Potential Embrittlement of RVI and RPV Materials

Q106. Dr. Lahey and the State raise concerns about irradiation embrittlement. Please provide a description of neutron irradiation embrittlement.

A106. (NFA, TJG, RGL, JRS) During the operation of a nuclear plant, the fracture toughness of materials can decrease as a result of high-energy neutrons emanating from the reactor core and impacting the RPV beltline region, defined in response to Question 90, above. Over time, this process reduces the ability of the RPV materials to resist the unstable propagation of a pre-existing crack (*i.e.*, it reduces their "fracture toughness"). *See* Was Text at 647-651 (ENT000627). The process, whereby the fracture toughness of a material is reduced by neutron flux, is known as neutron irradiation embrittlement. *See id*. at 643-44.

The degree of exposure to neutrons is normally expressed in terms of "fluence," which is the number of high-energy neutrons (*i.e.*, with an energy level > 1 Mega-electron volts ("MeV")) that have struck a square centimeter of a material. *See* EPRI, MRP-175, Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values at 3-4 (Dec. 2005) ("MRP-175") (ENT000631). For low-alloy RPV materials, neutron exposures above $1x10^{17}$ n/cm^2 (E > 1 MeV) require a material surveillance program under 10 C.F.R. Part 50, Appendix H. In contrast, the MRP-175 threshold fluence for irradiation embrittlement in austenitic stainless steels is $1x10^{21}$ n/cm^2 (E>1 MeV). *See* MRP-175 at 2-8 (ENT000631).

The temperature at the beginning of transition from ductile to non-ductile behavior is the reference temperature for nil-ductility transition, or "RT_{NDT}". The RT_{NDT} for pressure vessel material is established in accordance with the testing procedures specified in the ASME Code, Article NB-2330. *See* ASME Boiler & Pressure Vessel Code, Section III, Article NB-2000, "Material" at 22-23 (1998) (ENT000632).

Although stainless steel and nickel alloy RVI materials are also subject to irradiation embrittlement, they do not undergo a ductile-to-brittle transition or fail by brittle cleavage even though the neutron exposure levels are much higher than those of the vessel. *See* Was Text at 685-689 (ENT000627). Therefore, there is no measured RT_{NDT} for the austenitic stainless steels or nickel-based alloys. However, at fluences above the MRP-175 screening threshold, it is recognized that these austenitic stainless steels will experience decreases in fracture initiation toughness and in the resistance to ductile tearing. *See id*. at 686-687. These effects have been explicitly considered in the MRP-227-A guidelines and in the RVI AMP implementation at IPEC. *See generally* MRP-227-A (NRC00114A-F).

Q107. Dr. Lahey suggests that RVIs, as well as RPVs, can transition from ductile to brittle temperature behavior, and become subject to brittle fracture. *See* Revised Lahey

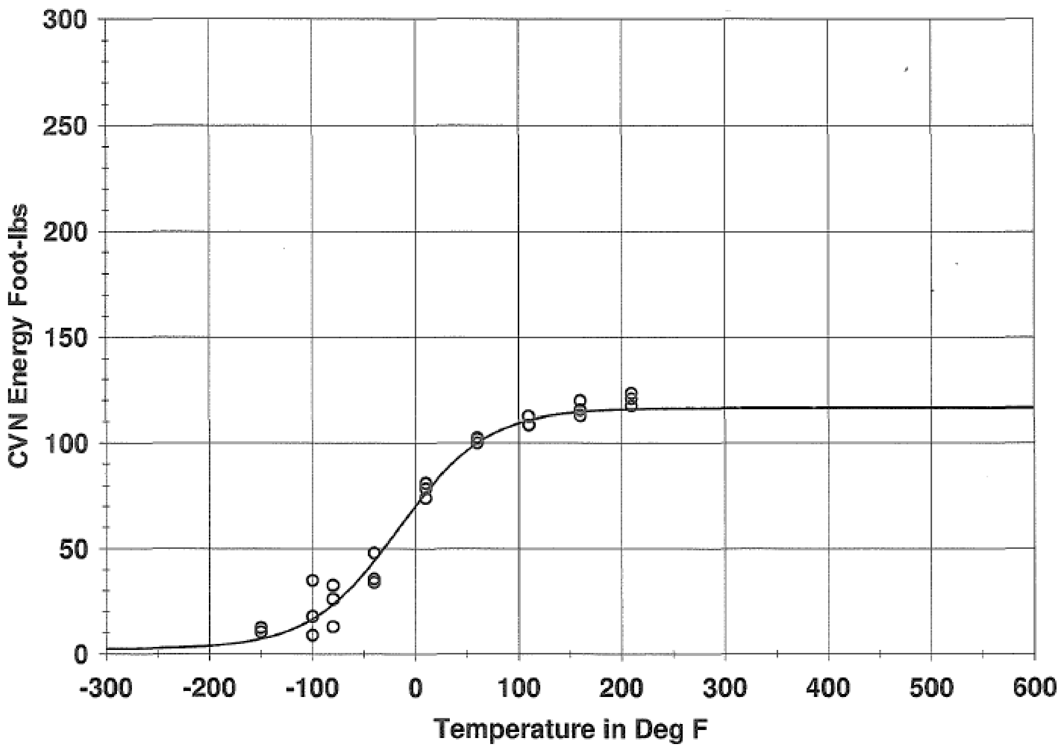
[redacted]

Testimony at 22-23 (NYS000482). Please further describe the concept of transition temperature.

A107. (NFA, TJG, RGL, JRS) Exposure to neutron fluence results in a change in mechanical properties and a decrease in the energy required to break a "Charpy" specimen. *See* Was Text at 664-666 (ENT000627). The Charpy impact test measures the energy required to break a small notched steel sample with a swinging pendulum. The energy required to break the Charpy specimen can then be correlated to the material fracture toughness. Charpy tests performed over a range of temperatures have established that fracture behavior of the low alloy RPV steels has three levels: an "upper shelf" (higher temperatures where metals exhibit tough, ductile behavior); a "lower shelf" (lower temperatures where metals exhibit brittle behavior); and a transition range (temperatures between the upper and lower shelves where the metal's behavior turns from ductile, or fracture resistant, to brittle at the lower temperatures). *See generally* Was Text (ENT000627).

To illustrate these points, consider the graph of unirradiated Charpy impact energy for the low-alloy RPV steel against increasing temperature, in Figure 4. This graph is taken from a Letter from P. Anderson, Entergy, to NRC Document Control Desk, "Documentation for Pressurized Thermal Shock Evaluation Meeting" (June 2, 2010) ("PTS Letter") (ENT000633). Charpy impact energy, measured in ft-lbs, is a measure of ductility, with higher values indicating a greater ability to resist fracture (*i.e.*, higher fracture toughness). The figure shows that the Charpy impact energy value for an unirradiated RV material begins at a low value, referred to as the lower shelf region; and gradually increases through what is known as the transition region to a region of relatively high toughness known as the "upper shelf." When low-alloy materials are irradiated, the transition region in which the Charpy impact energy increases from the lower to the upper shelf shifts to higher temperatures, and, at the same time, the level of upper shelf impact energy decreases. This means that the material has become less resistant to both brittle and ductile fracture within the range of test temperatures.

Figure 4: From IP2 Unirradiated Surveillance Capsule Report



Source: PTS Letter, Attach. 3, App. G (ENT000633).

Q108. As a practical matter, does the concept of transition temperature apply to the RVIs at IPEC?

A108. (NFA, TJG, RGL, JRS) No. Over the range of operating temperatures associated with a nuclear reactor, the concept of transition temperature does not apply to the wrought or cast stainless steels and nickel alloy materials used in the IPEC RVIs. This is because, despite exhibiting some loss of fracture toughness, these materials do not exhibit a change from ductile-to-brittle behavior at lower temperatures or a well-defined transition from a cleavage fracture "lower shelf" to a fully ductile "upper shelf" in the temperature region of interest for normal, transient, or postulated accident conditions at the plant. *See* Was Text at 643-644 (ENT000627). By contrast, as described above, the low-alloy materials that constitute the IPEC RPV do exhibit a transition from low to high toughness over the temperature range of interest in normal, transient, or postulated accidence conditions and that shifts with neutron irradiation. Therefore, the degrees of change in the transition temperature as a measure of irradiation embrittlement are relevant only to the low-alloy RPV materials, not to the austenitic stainless steel and nickel-based alloy RVIs used at IPEC.

In summary, to the extent Dr. Lahey is suggesting that the RVIs at IPEC could experience a transition from ductile behavior and become susceptible to brittle fracture, *see, e.g.*, Revised Lahey Testimony at 22, 23, 28-29 (NYS000482), we disagree.

Q109. Dr. Lahey suggests that given the higher fluence experienced by some RVIs in comparison to the RPVs, that the RVIs "suffer a lot more radiation damage and embrittlement" than the RPVs and that, in general, the RVIs are "highly embrittled." Revised Lahey Testimony at 26-27 (NYS00482). Do you agree?

A109. (TJG, RGL, JRS) No. Given the differences in RVI and RPV materials we have just discussed, Dr. Lahey's general comparison based on fluence alone is invalid. Overall, the RPV low alloy steels are far more susceptible to irradiation effects than the RVI stainless steels. For RVI components that were potentially susceptible to the effects of irradiation embrittlement, the MRP-191 screening process used the MRP-175 threshold fluence values to identify and provide guidelines for engineering evaluations for such components. *See* MRP-191 at v (NYS000321); MRP-175, App. A (ENT000631).

Q110. On this topic, Dr. Lahey has stated that the end-of-life "Charpy impact Upper Shelf Energy (USE) for some thermally-aged cast stainless steel in-core components

[redacted]

could be as low as 28 ft-lb$_f$, which is well below the acceptable ASME code-specified minimum of 50 ft-lb$_f$ . . . ." Report at 13 (NYS000296) (citing WCAP-14577, "License Renewal Evaluation: Aging Management for Reactor Internals," Rev. 1-A at 3-13 (2001) ("WCAP-14577") (NYS000341)). How do you respond?

A110. (TJG, JRS, RGL, RJD, NFA) Dr. Lahey is incorrect. The 50 ft-lb limit, referenced by Dr. Lahey, applies to low-alloy steels in the pressure boundary, but lower values are allowed with adequate justification. *See* 10 C.F.R. Part 50, Appendix G. RVIs are not fabricated from low-alloy steels and the ASME Code does not specify a 50 ft-lb limit for RVI materials. Moreover, WCAP-14577, cited by Dr. Lahey, found 28 ft-lbs to be acceptable for 60 years of operation for CF-8 cast stainless steel internal components. *See* WCAP-14577 at 3-13 (NYS000341) (citing the 28 ft-lb value and concluding that the effects of thermal aging are "not significant" for CF-8 cast stainless steel).

Q111. Overall, do you agree with Dr. Lahey that "the synergistic interactions between radiation-induced embrittlement, corrosion-induced cracking, and fatigue-induced degradation mechanisms have not been considered"? Revised Lahey Testimony at 15 (NYS000482).

A111. (RGL, TJG) No. Dr. Lahey's discussion about the relationships between degradation mechanisms is difficult to follow because he seems to mix references to three types of crack growth. To clarify the three separate mechanisms, they are: (1) fracture toughness properties change with irradiation of materials, *see*, *e.g.*, *id*. at 22 (stating "radiation-induced damage results in a decrease in fracture toughness and ductility"); (2) SCC is affected by irradiation which may cause crack initiation and growth, *see*, *e.g.*, *id*. at 15 (referring to "radiation enhanced corrosion-induced cracking"); and (3) potential effects on fatigue cracking and growth from neutron irradiation, *see*, *e.g.*, *id*. at 59 (citing "recent studies" that allegedly "show the extreme sensitivity of crack growth rate and fracture toughness to irradiation").

As to the first issue, we have already explained in response to Question 108, above, that the RVIs do not exhibit the ductile-to-brittle transition. However, the elastic-plastic fracture resistance of the irradiated austenitic stainless steels is expected to decrease, and these effects are recognized and addressed through the engineering evaluation methodologies specified in Chapter 6 of MRP-227-A.

To the extent Dr. Lahey is concerned with increased susceptibility to SCC due to irradiation, Revised Lahey Testimony at 46 (NYS000482), that possibility is by definition IASCC. *See* Was Text at 678 (ENT000627). IASCC is specifically managed through the MRP-227-A inspection programs. *See* MRP-227-A at 2-1 (NRC00114A); *see also* Westinghouse, WCAP-17096-NP, Rev. 2, Reactor Internals Acceptance Criteria Methodology and Data Requirements at 3-2 (Dec. 2009) ("WCAP-17096") (ENT000635).

Finally, to the extent that Dr. Lahey means that the effects of embrittlement, including loss of toughness, make existing cracks in the affected RVI components less resistant to *fatigue* crack growth, we again disagree with him. Fatigue and irradiation do not interact "synergistically." For example, as explained in MRP-175, "[t]he work of several researchers suggest that neutron irradiation does not result in a further reduction in fatigue properties and in some cases suggests an improvement." MRP-175 at D-3 (ENT000631); *see also* Draft NUREG/CR-6909, Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials, Rev. 1, Draft Report for Comment at 9 (Mar. 2014) (NYS000490A-B). This is discussed further in Sections IV.A.3 and V.E.2 of Entergy's testimony on the metal fatigue contention. *See generally* Entergy's NYS-26B/RK-TC-1B Testimony § IV.A.3 (ENT000679).

Moreover, fatigue is one of the eight age-related degradation mechanisms evaluated during the development of the guidelines in MRP-227-A. These inspection activities are in addition to, not in lieu of, fatigue analyses under the FMP.

3. Design Basis Pressure and Thermal Loads

Q112. Dr. Lahey suggests that RVIs could be subject to "pressure and/or thermal shock loads." Revised Lahey Testimony at 16 (NYS000482). What is a pressurized thermal shock ("PTS") event?

A112. (JRS, NFA, TJG, RGL) While Dr. Lahey's statements are not clear, it appears that he is referring to PTS loads. PTS events are transients resulting in severe over-cooling concurrent with or followed by significant pressure increase in the RPV. *See* NUREG-1874, Recommended Screening Limits for Pressurized Thermal Shock (PTS) at xi (March 2010) (ENT000637). Such events include, among others, a pipe break in the primary pressure circuit, a stuck-open valve in the primary pressure circuit that later re-closes (causing re-pressurization), or a break in a main steam line. *See id*. PTS events are typically characterized by an initiating event, such as a LOCA, leading to cold safety injection water coming in contact with the RPV wall causing large thermal stresses, followed by a repressurization of the primary system resulting in sustained high pressure-retaining stresses (referred to in the ASME Code as "membrane" stresses). If the RPV is excessively embrittled, then the thermal and pressure stresses could propagate a pre-existing crack that may continue through the RPV wall. *See id*.

Q113. How do NRC licensees ensure that RPVs have sufficient fracture toughness to withstand such events?

A113. (JRS, NFA) In order to ensure an acceptably low probability of RPV failure due to PTS events, the NRC developed the RT_{PTS} screening criteria in 10 C.F.R. § 50.61. The screening criteria effectively define a limiting level of embrittlement, where the potential for

[redacted]

RPV failure due to a PTS event is deemed to be acceptably low. In addition, in 2010, the NRC issued an alternative set of requirements to 10 C.F.R. § 50.61, codified in Section 50.61a. Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events; Final Rule, 75 Fed. Reg. 13 (Jan. 4, 2010) ("Alternate PTS Rule").

Q114. Are RVIs susceptible to PTS in the same manner as RPVs?

A114. (TJG, JRS, RGL, NFA) As shown in response to Question 94, RVIs have no pressure retaining function. A PTS transient, therefore, does not subject the RVI components to the sustained membrane stresses characteristic of the effects of a PTS event on an RPV. NUREG-1806, Vol. 1, Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61) at xix (Aug. 2007) (ENT000638) (noting that PTS events can cause a "repressurization of the RPV"). [redacted]

Q115. Where are the applicable CLB design basis conditions for the IPEC RVIs documented?

A115. (RGL, RJD, NFA, JRS) Design basis transients and loads are defined in the CLB for IPEC. These loading conditions are used as the basis for normal, anticipated transient and design basis accident analyses to demonstrate that design margins per the ASME Code are met, and must be used for engineering evaluations under the corrective action program. Normal and anticipated loads are used to perform design fatigue analyses. Importantly, because these loads are part of the CLB for IP2 and IP3 and do not change with time, they are not undergoing revision as part of the license renewal process for IPEC.

Specifically, design basis transients are identified on a plant-specific basis in Chapter 4 of the Updated Final Safety Analysis Reports ("UFSAR") for IP2 and IP3. *See, e.g.*, Entergy, IP3 FSAR Update, Revision 20 § 4 (NYSR0013D); Entergy, IP2 FSAR Update, Revision 25 § 4 (ENT000634). Design basis accident loads are described in Chapter 14 of the UFSARs, for both IPEC Units. *See, e.g.*, Entergy, IP3 FSAR Update, Revision 20 § 14 (NYSR0013I); Entergy, IP2 FSAR Update, Revision 25 § 14 (ENT000634). All Service Level loads were reviewed and updated during the 2004 power uprate project re-analyses, and are documented in Chapters 3 (Nuclear Steam Supply Systems and Auxiliary Equipment Design Transient) and 6 (Safety Analysis) of the respective engineering reports. *See* WCAP-16156 §§ 3, 6 (ENT000639); WCAP-16211 §§ 3, 6 (ENT000640); *see also id.* and WCAP-16156 at 5.2-17 [redacted].

Q116. Are the differences between RVIs and RPVs, in materials, functions, and loads reflected in different methods used to manage the effects of aging for license renewal?

A116. (RJD, NFA, TJG, RGL, JRS) Yes. The differences in materials (low-alloy steel RPVs and stainless steel RVIs) and functions (pressure boundary function for RPVs, other functions for RVIs), and loads (including PTS loads in the design basis for RPVs, but not for RVIs) are reflected in different license renewal requirements and guidance, and ultimately in the separate and distinct AMPs and TLAA evaluations in Entergy's LRA discussed throughout our testimony. In particular, because the RPV is a primary pressure boundary, it is sensitive to large pressure transients. Thermal shocks coincident with large pressure transients are a particular

[redacted]

concern in the RPV both because they can be additive and because they can lower the vessel temperature and potentially challenge material at the lower shelf, or low-ductility region. It is therefore important that the vessel operate in the ductile region, and the NRC imposes, for example, the requirements of 10 C.F.R. § 50.61, discussed in response to Question 113. The effects of embrittlement on RVIs, however, are not evaluated in TLAAs, but are managed through the AMPs specified in Staff guidance: primarily the RVI AMP, along with Inservice Inspection, Thermal Aging and Neutron Irradiation Embrittlement of CASS and Water Chemistry Control programs. *See generally* NL-12-037 (NYS000496); SSER 2 (NYS000507).

Q117. With these same differences in mind, please comment on Dr. Lahey's observation that "radiation-induced embrittlement of the RPVs and their associated internals is an important age-related safety concern." Revised Lahey Testimony at 46 (NYS000482).

A117. (TJG, RGL, JRS) By blending RPVs and RVIs in the same sentence, Dr. Lahey incorrectly implies that the effects of irradiation on these materials are similar. Radiation-induced embrittlement of the RPVs is an important safety concern and has been recognized as such in the nuclear industry. This aging effects is thoroughly addressed in regulations and extensive regulatory guidance and is being managed at IPEC.

However, embrittlement of the RPV low-alloy steel material and embrittlement of the austenitic stainless steel materials used in the RVIs cause different effects, raise different concerns, and require different aging management strategies. The radiation-induced embrittlement of the RVI is much less a concern than embrittlement of the RPV because they: (a) are less susceptible to brittle failure; (b) are not part of the reactor coolant pressure boundary; and (3) have a level of redundancy in their design. Nevertheless, as we will explain, the RVI

[redacted]

AMP provides reasonable assurance that the effects of aging due to embrittlement on the IPEC RVIs will be adequately managed so that their intended functions will be maintained consistent with the CLB, throughout the PEO.

VI. REGULATORY GUIDANCE ADDRESSING MANAGEMENT OF THE EFFECTS OF AGING ON RVIs

A. NRC Staff Guidance Regarding Management of the Effects of Aging on RVIs

Q118. Since the issuance of NUREG-1801, Revision 2, has the NRC Staff issued interim staff guidance on the aging management of PWR RVIs?

A118. (ABC, JRS, TJG) Yes. The NRC published a notice of availability for the final License Renewal Interim Staff Guidance on this topic in the *Federal Register* on June 3, 2013. *See* Final Interim Staff Guidance LR-ISG-2011-04; Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors, 78 Fed. Reg. 33,120 (June 3, 2013). The NRC Staff developed LR-ISG-2011-04 to update its guidance in NUREG-1801, Rev. 2 based on the conclusions of the NRC's revised safety evaluation on MRP-227-A. *See* Final 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) ("LR-ISG-2011-04") (ENT000641). LR-ISG-2011-04 revised the recommendations in NUREG-1801, Rev. 2 and the NRC Staff's acceptance criteria and review procedures to ensure consistency with MRP-227-A and provide a framework to adequately address age-related degradation and aging management of RVI components during the PEO.

Q119. What is MRP-227-A?

A119. (ABC, JRS, TJG, RGL) MRP-227-A is the NRC-approved version of EPRI's guidance on the aging management of RVIs. EPRI developed MRP-227, Revision 0, which was submitted to the NRC in January 2009. *See* MRP-227, Rev. 0 (NYS00307A-D). As explained

[redacted]

further below, the NRC Staff has reviewed and approved MRP-227, and EPRI has now issued an approved version, titled MRP-227-A.

Q120. How does the current NRC Staff guidance address aging management of RVIs for plants such as IPEC?

A120. (ABC, JRS, TJG) The guidance in LR-ISG-2011-04 addresses the aging management of PWR RVIs, including those at IPEC, by calling for the implementation of an AMP following the recommendations of MRP-227-A.

B. MRP-227 and MRP-227-A Are the Result of a Systematic Evaluation of the Effects of Aging on RVIs

Q121. You mentioned that the NRC Staff reviewed and approved MRP-227. Please summarize the NRC Staff's review.

A121. (ABC, JRS, TJG, RGL) After nearly three years of review, including several sets of RAIs, the NRC Staff issued its Safety Evaluation for MRP-227, Revision 0, in June 2011. *See* Letter from R. Nelson, NRC, to N. Wilmshurst, EPRI, "Final Safety Evolution of EPRI Report, Materials Reliability Program Report 1016596 (MRP-227), Revision 0, 'Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines' (TAC No. ME0680)" (Jun. 2011) ("Safety Evaluation for MRP-227, Revision 0") (NYS000309). The NRC Staff's safety evaluation identified seven Topical Report Conditions and eight Applicant/Licensee Action Items ("A/LAI") that must be addressed by applicants and licensees on a plant-specific basis. *See id.* encl. at 24-30). In the Safety Evaluation for MRP-227, Revision 0, the Staff requested that EPRI publish an "accepted" version of MRP-227 (*i.e.*, a version addressing the Topical Report Conditions, A/LAIs, RAIs, and RAI responses in a single document), to be designated "MRP-227-A." *Id*. at 1.

Q122. When did EPRI issue MRP-227-A?

A122. (ABC, JRS, TJG, RGL) EPRI issued the revised document, designated MRP-227-A, in December 2011. *See generally* MRP-227-A (NRC000114A-F).

Q123. Who prepared MRP-227-A?

A123. (TJG, RGL) MRP-227-A was prepared by the EPRI Materials Reliability Program ("MRP"), through its Reactor Internals Inspection and Evaluation Guidelines Core Writers' Group. The late Dr. Robert Nickell was the Principal Investigator for the project, with technical support provided by EPRI and vendor representatives, including ourselves. Nuclear utility oversight and review was provided by several industry representatives. The fifteen members of the Core Writers' Group, listed on page v of MRP-227-A, collectively have hundreds of years of collective experience in the commercial nuclear power industry. The effort to prepare MRP-227 began in the late 1990s and culminated in the publication of the final document in December 2008, the submittal of MRP-227, Revision 0, to the NRC Staff for review and approval in January 2009, and the issuance of the NRC-approved MRP-227-A in December 2011. *See* SE for MRP-227-A at 1 (ENT000230).

Q124. How was MRP-227-A developed?

A124. (TJG, RGL) The development of MRP-227-A proceeded in four steps: (1) development of screening criteria for the applicable aging mechanisms; (2) screening of RVI components based on susceptibility to degradation; (3) functionality analysis and failure modes, effects, and criticality analyses ("FMECA"), which resulted in the "binning" of components into different risk severity and inspection categories; and (4) development of the inspection and evaluation guidelines and flaw evaluation methodology. *See* SE for MRP-227-A, encl. at 4; *see also* MRP-227-A (NRC000114A-F) at 1-1.

[redacted]

Q125. Does MRP-227-A consider potential combinations of aging effects?

A125. (TJG, RGL) Yes. As explained further below, the screening process explicitly included consideration of potential combinations of aging effects. The aging management guidelines in MRP-227-A are supported by numerous underlying EPRI MRP technical studies, covering topics from aging degradation mechanisms and resulting effects, categorization of components, aging management strategies, acceptance criteria, and other topics. These technical studies document the considerable body of operating experience, state-of-the art research, and laboratory experiments that underpin the MRP-227-A guidelines. Based on the supporting MRP reports, MRP-227-A provides comprehensive aging management guidelines, detailing inspections to detect the effects of aging (individually or in combination), methods to evaluate such aging effects, and considerations for repair or replacement of degraded components.

This process of screening, categorization, functionality assessment, and then development of an aging management strategy and inspection and examination guidelines is illustrated in Figure 2-2 from MRP-227-A (NRC000114A).

Q126. You mentioned that MRP-227-A is supported by numerous underlying technical studies. What are the principal supporting documents for MRP-227-A?

A126. (TJG, RGL) Section 8 of MRP-227-A provides a list of principal supporting documents referenced in the creation of MRP-227-A. *See* MRP-227-A at 8-1 to 8-2 (NRC000114C). Although this list is not comprehensive, the most significant supporting or related studies in the development of the MRP-227-A guidelines include:

- EPRI, MRP-232, Materials Reliability Program: Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals (Dec. 2008) ("MRP-232") (ENT000642). *See also* EPRI, MRP-232, Revision 1, Materials Reliability Program: Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals (Dec. 2008) ("MRP-232, Rev. 1") (ENT000643).

- EPRI, MRP-230, Materials Reliability Program: Functionality Analysis for Westinghouse and Combustion Engineering Representative PWR Internals (Oct. 2009) ("MRP-230") (ENT000644).
- MRP-228 (NYS000323). *See also* EPRI, MRP-228, Rev. 1, Materials Reliability Program: Inspection Standard for PWR Internals – 2012 Update (Dec. 2012) (ENT000645).
- EPRI, MRP-210, Materials Reliability Program: Fracture Toughness Evaluation of Highly Irradiated PWR Stainless Steel Internal Components (Dec. 2007) ("MRP-210") (ENT000646).
- MRP-191, EPRI, Materials Reliability Program: Screening, Categorization and Ranking of Reactor Internals of Westinghouse and Combustion Engineering PWR Designs (Nov. 2006) (NYS000321).
- MRP-175 (ENT000631).
- EPRI, MRP-134, Materials Reliability Program: Framework and Strategies for Managing Aging Effects in PWR Internals (June 2005) (ENT000647).
- WCAP-17096-NP, Rev. 2 (ENT000635).

Q127. Does Dr. Lahey discuss the information presented in these supporting reports?

A127. (TJG, RGL) Only to an extremely limited extent. The State has submitted some of these documents as exhibits, and Dr. Lahey has referenced some of these documents in his past reports (but without disputing any of the information in them). The Revised Lahey Testimony contains only one citation to one of these documents, but only to generally point out that RVIs "can experience" more neutron fluence than RPVs. *See* Revised Lahey Testimony at 27 (NYS000482). Thus, Dr. Lahey does not directly address the decade or more of engineering work that led to MRP-227-A. Most importantly, he does not challenge the technical adequacy of that work or the voluminous underlying technical supporting documents.

Q128. Dr. Lahey states that "an adequate inspection plan for RPV internals is a necessary, but not sufficient, means of assuring safe extended plant operations. Indeed, a

systematic evaluation of the degraded RPV internals is needed to identify the limiting structures, components and fittings that need to be repaired or replaced before the onset of extended operations." Revised Lahey Testimony at 51 (NYS000482) (emphasis in original). How do you respond?

A128. (TJG, RGL, NFA, RJD, JRS) As we have just shown, and as we will explain further throughout this testimony, the guidelines in MRP-227-A are based on a systematic evaluation of degradation mechanisms (including multiple concurrent mechanisms), the resulting aging effects (including combinations of effects), and consequences that identified the limiting RVI structures, components, and fittings. This evaluation is documented in the voluminous supporting reports to MRP-227-A, including most pertinently the Failure Modes, Effects, and Criticality Analysis ("FMECA") for Westinghouse components in MRP-191. *See* MRP-191 at 6-1 to 6-27 (NYS000321).

Q129. Please explain the overall approach to aging management set forth in MRP-227-A?

A129. (ABC, JRS, TJG, RGL) Based on a considerable body of research and operating experience, MRP-227-A provides aging management guidelines, defines inspections to detect the effects of aging, and recommends methods to evaluate aging effects. *See* MRP-227-A at 1-1 (NRC000114A). In order to monitor the potential effects of aging on PWR RVIs, including the potential combined effects, MRP-227-A divides RVI components into four groups with different aging management activities specified for each group (Primary, Expansion, Existing Programs, and No Additional Measures) depending on: (1) the relative susceptibility to and tolerance of applicable aging effects, and (2) the existence of other programs that manage the effects of aging on those components. *See id.* at 3-12 to 3-16. MRP-227-A also contains specific, conservative

[redacted]

examination acceptance criteria in Section 5 that can be used to determine when a particular examination result must be entered into the plant corrective action program. *See* MRP-227-A at 5-1 to 5-23 (NRC000114B). Section 6 of MRP-227-A describes the corrective action program options and the process to address situations when examination acceptance criteria are not met. *See id*. at 6-1 to 6-11. The corrective action program considers several potential disposition paths, including more detailed examination, engineering evaluation, repair or replacement. *See id*. at 6-2 to 6-3; NL-12-037, Attach. 1 at 8 (NYS000496) (referencing Section 6 of MRP-227-A).

C. <u>The NRC Staff Reviewed and Approved MRP-227-A as a Topical Report</u>

Q130. Turning back to the NRC Staff's review and approval of MRP-227-A, what were the Staff's conclusions?

A130. (ABC, JRS, TJG, RGL) After reviewing changes made to MRP-227, Rev. 0 in response to NRC staff comments, the NRC Staff issued a revised Safety Evaluation endorsing MRP-227-A as a topical report in December 2011. SE for MRP-227-A at 1 (ENT000230). The Staff concluded that MRP-227-A "provides for the development of an AMP for PWR RVI components . . . which 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 [PEO]." *Id.*, at 35 (ENT000230). Further, the Staff concluded that:

> Any applicant may reference MRP-227 as modified by this SE and approved by the NRC, in a LRA or other licensing action to satisfy the requirements of 10 CFR 54.21(a)(3) for demonstrating that the effects of aging on the RVI components, within the scope of MRP-227, will be adequately managed. The staff also concludes that, upon completion of plant-specific action items set forth in Section 4.0, referencing the NRC-approved version of MRP-227 in a LRA and summarizing the AMP contained in MRP-227 in a FSAR supplement will provide the staff with sufficient information to make necessary findings required by 10 CFR 54.29(a)(1) for RVI

[redacted]

> components within the scope of MRP-227, as approved by the NRC.

Id.

Q131. Did the NRC Staff review the use of VT-3 examinations for certain specified RVI components as part of its review of MRP-227-A?

A131. (TJG, RJD, ABC, JRS) Yes. The NRC Staff reviewed the use of VT-3 examinations (discussed further in response to the next question, below), as well as the other non-destructive examination ("NDE") techniques proposed in MRP-227-A, and endorsed EPRI's proposed examination methods, including VT-3 examinations for specified components, finding them to be "implemented by well established standard procedures." SE for MRP-227-A at 7 (ENT000230).

Q132. When Dr. Lahey suggests that "there are significant shortcomings of this technique to detect material cracking, degradation, or wear prior to failure, as has been noted by USNRC staff," Revised Lahey Testimony at 62 (NYS000482), do you agree?

A132. (TJG, RJD, ABC, JRS) No. An NRC Staff non-concurrence statement advocated for the use of EVT-1 as opposed to VT-3 inspections for certain aging effects. The non-concurrence asserted that "cracking which may occur due to fatigue, SCC, and IASCC mechanisms in RVI components are not amenable to discovery using the VT-3 method in a timely manner." Letter from J. Sipos to Administrative Judges (Apr. 29, 2011), first unnumbered attachment, R.L. Tregoning, Reasons for Non-concurrence on "Draft Safety Evaluation for the Electric Power Research Institute's Topical Report (TR) Materials Reliability Program (MRP) Report 1016596 (MRP-227), Revision 0, 'Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines'" at 3 (undated) (NYS000370).

The NRC Chief, Vessels and Internals Integrity Branch responded to the non-concurrence. The key observations in that response are as follows:

- "[A]lthough VT-3 examinations have not traditionally been credited for identifying cracking, they have been proven to be capable of doing so by operating experience."
- "[T]he quality of inspections achieved using the VT-3 inspection standard from the American Society of Mechanical Engineers Code has improved over the years as critical parameters (surface cleaning, lighting, character height specification for qualification, etc.) have been refined."
- "In addition, the staff notes that the components for which VT-3 examinations have been credited are very flaw tolerant, either because of the size of the component in question and the length of flaw required to begin to postulate the potential for failure, or because a group of like components is considered in which multiple like components must fail in order to compromise the functionality of the group . . . ."
- "VT-3 examinations for cracking were not specified in MRP-227 in cases where the data would potentially be used in a fracture mechanics analysis to demonstrate the structural integrity of the vessel internals."

M. Mitchell, Chief, Vessels and Internals Integrity Branch, Response to Non-Concurrence Regarding Safety Evaluation for Topical Report MRP-227, "Pressurized Water Reactor Internals Inspection and Evaluation Guidelines" at 2 (undated) (ENT000648). We believe this response fully explains why the use of VT-3, as specified in MRP-227-A, is appropriate.

VII. ENTERGY'S LICENSE RENEWAL APPLICATION ADEQUATELY ADDRESSES AGING MANAGEMENT OF THE RVIs AND RPVs

A. Entergy's RVI AMP for IPEC

1. Overview of the IPEC RVI AMP and Inspection Plan

Q133. Please summarize the IPEC RVI AMP as it stands today.

A133. (ABC, NFA, RJD, JRS, TJG, RGL) As explained in Section VI.B, the IPEC RVI AMP, as updated, relies upon the extensive industry research documented in MRP-227-A and MRP-228, and in the many reports supporting those documents. *See* NL-12-037, Attach. 1 at 3-4 (NYS000496); NL-10-063, Attach. 1 at 82-84 (NYS000313); *see also generally* MRP-227-A at

[redacted]

8-1 to 8-2 (NRC000114C); MRP-228 (NYS000323). The RVI AMP is designed to manage the effects of aging applicable to RVIs at IP2 and IP3, such that there is reasonable assurance that those effects will be adequately managed throughout the PEO, consistent with NRC regulations. This includes the pertinent combinations of aging effects. The RVI AMP has three principal components: (1) examinations and other inspections, along with a comparison of data to examination acceptance criteria, as defined in MRP-227-A and MRP-228; (2) resolution of indications that exceed examination acceptance criteria by entering them into the applicant's Corrective Action Program; and (3) monitoring and control of reactor primary coolant water chemistry based on industry guidelines. *See generally* NL-12-037, Attach. 1 (NYS000496).

Q134. Please summarize The IPEC RVI Inspection Plan.

A134. (ABC, NFA, RJD, JRS, TJG, RGL) The RVI Inspection Plan provides additional details on the inspections to be conducted under the RVI AMP, including: (1) the type of examinations; (2) the level of examination qualification; (3) the schedule of initial inspection and frequency of subsequent inspections: (4) the criteria for sampling and coverage; (5) the criteria for expansion of scope if unanticipated indications are found; (6) the acceptance criteria; (6) the methods for evaluation of examination results that do not meet the acceptance criteria; (7) provisions to update the program based on industry-wide results; and (8) contingency measures to repair, replace, or mitigate, beyond the information set forth in the RVI AMP. *See* NL-12-037, Attach. 2 at 1 (NYS000496).

Q135. Has the NRC Staff approved The IPEC RVI AMP and Inspection Plan for IPEC?

A135. (ABC, RJD, NFA, TJG, RGL, JRS) Yes. As part of the NRC's review process, the NRC Staff issued RAIs on several topics, and Entergy provided significant additional

[redacted]

technical information in response. *See* SSER 2 at 3-13 to 3-59 (NYS000507). Appendix B of SSER 2 provides a chronology of correspondence between Entergy and NRC Staff related to the RVIs. The NRC Staff documented its review and approval of the Indian Point RVI AMP and Inspection Plan in SSER 2 on November 6, 2014. *Id*. at 6-1.

In particular, the NRC Staff concluded that all ten elements of the Indian Point RVI AMP are consistent with LR-ISG-2011-04 and are therefore acceptable. *Id*. at 3-15, 3-17 to 3-22, and 3-26. The NRC Staff further concluded that Entergy "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." *Id*. at 3-26. With respect to Entergy's Inspection Plan, the NRC Staff concluded that it too is consistent with the RVI inspection and evaluation guidelines in MRP-227-A, and that Entergy adequately addressed all of the A/LAIs and Topical Report Conditions. *See id*. at 3-59.

Q136. Overall, what is your assessment of the IPEC RVI AMP?

A136. (JRS, TJG, RGL) The IPEC RVI AMP is consistent with industry guidance, relies on appropriate preventive actions, and is supported by a comprehensive Inspection Plan designed to identify aging effects that could potentially impact component function. The RVI AMP includes appropriate corrective actions in the event flaws or other aging effects are found. It properly addresses applicable aging effects, including combinations of effects. The program will be refined and enhanced based on industry research and operating experience, consistent with Entergy's program for evaluation of operating experience. It is consistent with the relevant NRC-accepted industry guidance in MRP-227-A. For these reasons, and for the reasons discussed throughout this testimony, the Entergy RVI AMP provides reasonable assurance that

IPEC RVI components will continue to perform their intended functions, consistent with the CLB, during the PEO, in accordance with 10 C.F.R. § 54.21(a)(3).

2. Inspections Under the RVI AMP

Q137. How does the IPEC RVI AMP monitor the potential effects of aging?

A137. (RJD, TJG, RGL) Following the guidance in MRP-227-A, The IPEC RVI AMP separates PWR RVI components into four groups with different aging management strategies specified for each group (Primary, Expansion, Existing Programs, and No Additional Measures) depending on: (1) the relative susceptibility to and tolerance of applicable aging effects, and (2) the existence of other programs that manage the effects of aging on those components. *See* MRP-227-A at 3-15 to 3-16 (NRC000114A). This categorization is not dependent on analyzing the behavior of the individual components under accident loads. Rather, the inspection categorization process evaluated possible component failure under accident loads and if the assumed failure could impact a design basis function the component was assigned to an inspection category using the appropriate inspection techniques and frequency of inspections. *See* MRP-227-A at 4-1 to 4-79 (NRC000114A-B).

Q138. What is the scope and periodicity of the IPEC RVI inspections?

A138. (RJD, TJG, RGL) Based on the guidance in MRP-227-A, the RVI AMP specifies the methods, extent, and frequency of inspections at IPEC. The necessary inspections are specified in Table 5-2 (primary components), Table 5-3 (expansion components), and Table 5-4 (existing program components). *See* NL-12-037, Attach. 2 at 37-51 (NYS000496); *see also* MRP-227-A at 4-26 to 4-29, 4-37 to 4-39, & 4-74 (NRC000114B). Specifically, the fifth column of Table 5-2, prominently labeled "Examination Method/Frequency," lists the applicable examination frequencies for each of the IPEC Primary components. NL-12-037, Attach. 2 at 37-42 (NYS000496).

[redacted]

These tables contain columns describing the component, any particular applicability requirement for that component (*i.e.*, which plants have the component), the degradation effect to be detected, the examination method and frequency, the examination coverage, and any linkage between the Primary and Expansion components. *See id*. at 37-51; *see also, e.g.*, MRP-227-A at 4-26 to 4-29, 4-37 to 4-39, & 4-74 (NRC000114B).

The schedule for Expansion components, for which a functionality assessment has shown a degree of tolerance for aging effects and/or a lower probability of degradation, depends on the findings of Primary component examinations. *See* NL-12-037, Attach. 2 at 19 (NYS000496); *see also* MRP-227-A at 3-15 (NRC000114A). The remaining RVI components are either covered by inspections under other, existing programs, or are not expected to experience significant aging degradation and, therefore require no additional measures. *See* NL-12-037, Attach 2 at 19, 51 (NYS000496); *see also* MRP-227-A at 3-15, 4-74 (NRC000114A-B).

Q139. Dr. Lahey has suggested that Entergy has not provided sufficient details about its inspection schedule. *See* Revised Lahey Testimony at 48-49 (NYS000482). Can you provide more detail on how often are these inspections conducted at IPEC?

A139. (RJD, TJG, RGL) Yes. Table 5-2 of Attach. 2 to NL-12-037, which is based on Table 4-3 of MRP-227-A, specifies the required timing of the first-time inspections and subsequent intervals for the primary components in the RVI AMP. Table 1, below, summarizes this information from NL-12-037, Attachment 2. For most components, the first planned inspections at IPEC are scheduled for two refueling outages from the beginning of the PEO; *i.e.*, the spring of 2016 for IP2, and the spring of 2019 for IP3.

[redacted]

Table 1: Summary of Inspections for Primary, Expansion, and Existing Programs Components

Item	Effect (Mechanism)	Expansion Link / Primary Link / Reference	Examination Method / Frequency
Primary Components			
Control Rod Guide Tube Assembly			
Guide plates (cards)	Loss of Material (Wear)	None	Visual (VT-3) no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.
Lower flange welds	Cracking (SCC, Fatigue) Aging Management (IE and TE)	Bottom-mounted instrumentation (BMI) column bodies, Lower support column bodies (cast), Upper core plate Lower support, casting	Enhanced visual (EVT-1) examination to determine the presence of crack-like surface flaws in flange welds no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a ten-year interval.
Core Barrel Assembly			
Upper core barrel flange weld	Cracking (SCC)	Core barrel outlet nozzle welds	Periodic enhanced visual (EVT-1) no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a ten-year interval.
Upper & lower core barrel cylinder girth welds	Cracking (SCC, IASCC, Fatigue)	Upper and lower core barrel cylinder axial welds Lower core barrel cylinder girth weld expands to lower support column bodies (cast)	Periodic enhanced visual (EVT-1) no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a ten-year interval.
Lower core barrel to lower support casting weld (At IPEC this weld is the lower core barrel to lower support casting weld. IPEC does not have a lower core barrel flange)	Cracking (SCC, Fatigue)	None	Periodic enhanced visual (EVT-1) no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a ten-year interval.
Baffle Former Assembly			
Baffle-edge bolts	Cracking (IASCC, Fatigue) that results in • Lost or broken locking	None	Visual (VT-3), with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.

Item	Effect (Mechanism)	Expansion Link / Primary Link / Reference	Examination Method / Frequency
	devices • Failed or missing bolts • Protrusion of bolt heads Aging Management (IE and ISR) Void Swelling effects on this component [are] managed through management of void swelling on the entire baffle-former assembly.		
Baffle-former bolts	Cracking (IASCC, Fatigue) Aging management (IE and ISR) Void swelling effects on this component is managed through management of void swelling on the entire baffle-former assembly.	Lower support column bolts, Barrel-former bolts	Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination on a ten-year interval.
Assembly (Includes: Baffle plates, baffle edge bolts and indirect effects of void swelling in former plates)	Distortion (Void Swelling), or Cracking (IASCC) that results in • Abnormal interaction with fuel assemblies • Gaps along high fluence baffle joint • Vertical displacement of baffle	None	Visual (VT-3) examination to check for evidence of distortion, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.

[redacted]

Item	Effect (Mechanism)	Expansion Link / Primary Link / Reference	Examination Method / Frequency
	plates near high fluence joint • Broken or damaged edge bolt locking systems along high fluence baffle joint		
Alignment & Interfacing Components			
Internals hold down spring	Distortion (Loss of Load) Note: This mechanism was not strictly identified in the original list of age-related degradation mechanisms.	None	Direct measurement of spring height within three cycles of the beginning of the license renewal period. If the first set of measurements is not sufficient to determine life, spring height measurements must be taken during the next two outages, in order to extrapolate the expected spring height to 60 years.
Thermal Shield Assembly			
Thermal shield flexures	Cracking (Fatigue) or Loss of Materials (Wear) that results in thermal shield flexures excessive wear, fracture or complete separation	None	Visual (VT-3) no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.
Expansion Components			
Upper Internals Assembly			
Upper core plate	Cracking (Fatigue, Wear)	Control rod guide tube (CRGT) lower flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.
Lower Internals Assembly			
Lower support casting	Cracking Aging Management (TE in Casting)	Control rod guide tube (CRGT) lower flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.
Core Barrel Assembly			
Barrel-former bolts	Cracking (IASCC, Fatigue)	Baffle-former bolts	Volumetric (UT) examination. Re-inspection every 10 years following initial inspection.

Item	Effect (Mechanism)	Expansion Link / Primary Link / Reference	Examination Method / Frequency
	Aging Management (IE, Void Swelling and ISR)		
Core barrel outlet nozzle welds	Cracking (SCC, Fatigue) Aging Management (IE of lower sections)	Upper core barrel flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.
Upper & lower core barrel cylinder axial welds	Cracking (SCC, IASCC) Aging Management (IE)	Upper and lower core barrel cylinder girth welds	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.
Lower Support Assembly			
Lower support column bolts	Cracking (IASCC, Fatigue) Aging Management (IE, and ISR)	Baffle-former bolts	Volumetric (UT) examination. Re-inspection every 10 years following initial inspection.
Lower support column bodies (cast)	Cracking (IASCC) including the detection of fractured support columns Aging Management (IE)	Control rod guide tube ("CRGT") lower flanges Lower core barrel cylinder girth weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.
Bottom Mounted Instrumentation System			
Bottom-mounted instrumentation (BMI) column bodies	Cracking (Fatigue) including the detection of completely fractured column bodies Aging Management (IE)	Control rod guide tube (CRGT) lower flanges	Visual (VT-3) examination of BMI column bodies as indicated by difficulty of insertion/withdrawal of flux thimbles. Re-inspection every 10 years following initial inspection. Flux thimble insertion/withdrawal to be monitored at each inspection interval.
Existing Program Components			
Core Barrel Assembly			

[redacted]

Item	Effect (Mechanism)	Expansion Link / Primary Link / Reference	Examination Method / Frequency
Core barrel flange	Loss of Material (Wear)	ASME Code Section XI	Visual (VT-3) examination to determine general condition for excessive wear.
Upper Internals Assembly			
Vertical sections of tophat	Cracking (SCC, Fatigue)	ASME Code Section XI	Visual (VT-3) examination.
Lower Internals Assembly			
Lower core plate	Cracking (IASCC, Fatigue) Aging Management (IE)	ASME Code Section XI	Visual (VT-3) examination of the lower core plates to detect evidence of distortion and/or loss of bolt integrity.
Lower core plate	Loss of Material (Wear)	ASME Code Section XI	Visual (VT-3) examination.
Bottom Mounted Instrumentation System			
Flux thimble tubes	Loss of Material (Wear)	NUREG-1801, Rev. 1	Surface (ET) examination.
Alignment & Interfacing Components			
Clevis insert bolts	Loss of Material (Wear)	ASME Code Section XI	Visual (VT-3) examination.
Upper core plate alignment pins	Loss of Material (Wear)	ASME Code Section XI	Visual (VT-3) examination.

Source: NL-12-037, Attach. 2 at 37-51 (NYS000496), as modified by NL-14-093, Letter from F. Dacimo, Vice President, Entergy, to NRC Document Control Desk, Reply to Request for Additional Information Regarding License Renewal Application Indian Point Nuclear Generating Unit Nos. 2 & 3 (Aug. 5, 2014) ("NL-14-093") (NYS000505).

Q140. Dr. Lahey asserts that the AMP, "as set forth in NL-10-063 [NYS000313] lacks sufficient details to know when the baseline inspections of the RPV and its internals will begin and end, and the scope of these inspections. Thus, it is not possible to know whether the proposed baseline inspections will be comprehensive and adequate." Revised Lahey Testimony at 48-49 (NYS000482). How do you respond?

A140. (TJG, RGL, RJD, ABC, NFA) As we have just shown, there is no lack of detail on the scope and schedule for inspections in the RVI AMP. Although Dr. Lahey and the State call for baseline inspections, they do not explain why the designated schedule for any particular

[redacted]

component might be inadequate. Instead of alleging particular deficiencies, Dr. Lahey and the State vaguely assert that the available information lacks sufficient detail to be critiqued

Q141. What is the technical basis for the inspection schedules in MRP-227-A?

A141. (TJG, RGL) The inspection schedules in MRP-227-A were developed based on the same research and operating experience that we have previously summarized in response to Question 126, above. In addition, a number of PWRs have already entered the PEO and have conducted inspections that provide additional baseline information for PWRs yet to enter the PEO. *See, e.g.*, Letter from D. Pelton, NRC, to E. McCartney, NextEra Energy Point Beach, "Point Beach Nuclear Plant, Units 1 and 2 – Staff Assessment of Reactor Vessel Internals Inspection Plan Based on MRP-227-A (TAC Nos. ME8235 and ME8236)," Encl. at 3 (Mar. 30, 2015) (ENT000649) (noting inspections at Point Beach Nuclear Plant); G. Gardner, Chair, EPRI Reactor Internals Working Group, Recent Materials Inspections of PWR Reactor Internals at slides 19-26 (Mar. 2015) (ENT000650) (summarizing RVI inspections through Fall 2014). And further, under its operating experience program, Entergy will monitor the results of RVI inspections at other plants and take necessary actions to assess and address inspection findings relevant to IPEC.

Q142. Did the NRC Staff confirm the adequacy of the periodicity of inspections?

A142. (ABC, JRS, RGL, RJD) Yes. The NRC Staff, in its Safety Evaluation for MRP-227-A, acknowledged the justification for the timing of the initial PEO and subsequent inspections, and found the inspection intervals acceptable. *See* SE for MRP-227-A at 18, 21 (ENT000230).

[redacted]

3. The RVI AMP Manages the Effects of Aging Regardless of the Underlying Aging Mechanism

Q143. Does the RVI AMP focus primarily on identifying aging mechanisms or managing aging effects?

A143. (ABC, JRS, TJG, RGL) The NRC's license renewal process has long focused on aging "effects," rather than aging "mechanisms." Since 1995, when the NRC promulgated its revised license renewal rules, the NRC has emphasized that the identification of individual aging mechanisms is not required as part of the license renewal review. *See* Part 54 SOC, 60 Fed. Reg. at 22,463 (NYS000016). Instead, the regulations in 10 C.F.R. Part 54 concentrate on ensuring that important SSCs will continue to perform their intended functions during the PEO. *See id*. Thus, the inspections conducted under the RVI AMP look for evidence of any of the aging effects of concern, and appropriate action is taken if any relevant conditions related to those effects are discovered, regardless of their cause. In this regard, we note that The IPEC RVI AMP implements the program set forth in MRP-227-A. Although it considers potential RVI degradation mechanisms that have been identified through operating experience and relevant laboratory testing, MRP-227-A focuses inspections on managing the resulting aging effects, and this is consistent with NRC Staff guidance. *See* LR-ISG-2011-04 at 3 (ENT000641).

Q144. What specific aging effects are addressed in the RVI AMP?

A144. (RJD, TJG, RGL, JRS) Consistent with MRP-227-A, Section 3.2, the RVI AMP addresses the following eight age-related degradation mechanisms and their associated effects:

1. Stress corrosion cracking ("SCC")
2. Irradiation-assisted stress corrosion cracking ("IASCC")
3. Wear
4. Fatigue
5. Thermal aging embrittlement
6. Irradiation embrittlement (also referred to as neutron embrittlement)
7. Void swelling and irradiation growth
8. Thermal and irradiation-enhanced stress relaxation or irradiation-enhanced creep

[redacted]

See NL-12-037, Attach. 1 at 6 (NYS000496). This list includes all of the aging mechanisms identified by Dr. Lahey on page 11 of his Report and on page 15 of his Revised Prefiled Testimony—and more.

For each of the eight mechanisms, MRP-227-A identifies the resulting aging effect, which will then be managed through inspections under the MRP-227-A guidelines. Notably, in most cases, the key effects are cracking, dimensional changes, or wear, but in all cases, as explained below, the inspections specified in MRP-227-A are designed to detect potential aging effects applicable to each RVI component, regardless of the underlying mechanism. Therefore, as the IPEC RVI AMP is based on MRP-227-A, contrary to Dr. Lahey's claims, the IPEC RVI AMP does not "fail[] to consider how those interacting degradation mechanisms will impact the . . . RPV internals." Revised Lahey Testimony at 49 (NYS000482).

Q145. Do these aging effects impact all RVI components equally?

A145. (RJD, TJG, RGL, JRS) No. For example, items 2, 6, 7, and 8 in the previous response are all irradiation-induced mechanisms that are directly related to high levels of neutron irradiation, and therefore only applicable to the subset of RVIs subject to such an environment. The resulting aging effects will only impact those RVIs that are closest to the core and adjacent to the active core region, and therefore experience relatively high neutron fluence. Thus, the IPEC RVI AMP and Inspection Plan consider the relative susceptibility of particular RVI components to multiple degradation mechanisms that may result in, or contribute to, combined aging effects that could impact functionality of the RVIs.

Q146. Dr. Lahey claims that inspections under the RVI AMP cannot detect embrittlement. For example, he points to a footnote in Table 3-3 of MRP-227-A, which states that "[t]here are no recommendations for inspection to determine embrittlement

level because these mechanisms cannot be directly observed." Revised Lahey Testimony at 38 (NYS000482). Thus, according to Dr. Lahey, the "level of degradation due to embrittlement of RPV internal components, fittings and structures, and their ability to withstand fatigue and shock loads cannot be determined using the inspection techniques proposed in MRP-227-A." ***Id.*** **at 38-39. How do you respond?**

A146. (TJG, RGL, JRS, RJD) Note 1 for Tables 3-2 and 3-3 of MRP-227-A correctly points out that no recommendations for inspection to determine embrittlement level are contained in the guidance because these mechanisms cannot be directly observed. *See* MRP-227-A at 3-23 & 3-26 (NRC000114A). But embrittlement is only an issue for RVIs if there is a crack. *See generally* MRP-210 at 3-1 to 3-29 (ENT000646). Therefore, while it is not possible to detect the level of embrittlement directly through visual inspection, the guidance in MRP-227-A provides for inspections that detect the manifestation of significant thermal aging or neutron-irradiation embrittlement—specifically, the potential growth of a pre-existing defect.

Once a defect is discovered, its ability to withstand fatigue and combinations of both normal and accident loads is evaluated by either fracture mechanics analysis or a structural analysis (*i.e.*, an engineering evaluation) using the lower bound fracture toughness; *i.e.*, the evaluation assumes a bounding level of embrittlement of the material. *See* MRP-227-A at 6-4 (NRC000114B). Thus, the program has compensated for any inability to directly determine the level of embrittlement through a conservative assumption employed during evaluation of inspection findings. Thus, reasonable assurance that the effects of aging will be adequately managed is provided without the need for direct observation or measurement of the level of embrittlement.

[redacted]

Q147. Dr. Lahey suggests that the RVI AMP is deficient because it only "indirectly" monitors for embrittlement of RVIs, rather than doing so "directly," and that this is effectively a program that will detect failures after they occur. Revised Lahey Testimony at 37 (NYS000482); *see also id.* at 54. Is that a deficiency in the RVI AMP?

A147. (TJG, RGL, JRS, RJD) No. We understand that Dr. Lahey considers the monitoring of RVI components "through visual or volumetric inspection techniques that look for cracking," to be "indirect," and that he suggests that only destructive testing qualifies as "direct" monitoring. Revised Lahey Testimony at 37 (NYS000482). But, flaw evaluations in MRP-210, using lower-bound fracture toughness values, demonstrated that RVI components are only threatened by flaws that are significantly larger than those that are readily detectable through the prescribed visual or volumetric inspection methods. *See generally* MRP-210 at 3-1 to 3-29 (ENT000646). If a defect is detected by the required examinations, then the defect must be dispositioned through engineering evaluation under the corrective action process. Again, the engineering evaluation will assume the most limiting properties of the material; *i.e.*, the lower bound fracture toughness. *See* MRP-227-A at 6-4 (NRC000114B). This is consistent with the approach in MRP-227-A, as approved by the NRC Staff, and provides reasonable assurance of continued functionality without the need for direct measurement of the level of embrittlement.

4. RVIs Are Robust and Highly Failure Tolerant

a. PWR RVIs Are Robust

Q148. In general, are PWR RVIs highly susceptible to aging effects?

A148. (JRS, TJG, RGL) The primarily austenitic stainless steel RVIs—and the remaining RVI components that are constructed of other damage-resistant and flaw-tolerant materials—have performed well in service at many plants for thousands of reactor years, with very little adverse operating experience. *See* MRP-227-A, App. A (NRC000114C). These

[redacted]

materials have been shown to be very resistant to aging effects. *See id*. For example, the operating experience summary in Appendix A of MRP-227-A shows a high level of flaw tolerance and very few indications of aging effects that might compromise the continued function of PWR RVIs. *See id*. These findings are supported by the analytical effort reported in MRP-210 (ENT000646), discussed above, which shows the high level of flaw tolerance exhibited by these RVI components, even considering lower bound embrittled fracture toughness properties. Thus, as shown in MRP-210, there is reasonable assurance that continued functionality will not be impaired before the effects of aging can be detected.

Q149. How would you compare PWR RVIs and BWR RVIs in terms of susceptibility to the effects of aging?

A149. (JRS, TJG, RGL) It is well-established that because of differences in operating environments and chemistry, boiling water reactors ("BWRs") are generally more susceptible to cracking due to IASCC and intergranular stress corrosion cracking ("IGSCC") than PWRs. For example, over four decades of operating experience, we have seen instances of premature IGSCC in the welds of recirculation piping and BWR RVI components, such as core shrouds, that have not been observed in PWRs. *See* T. Griesbach and B. Gordon, Materials Aging Management Programs at Nuclear Power Plants in the United States § 3.2 (Oct. 2007) (ENT000651). Largely due to this well-understood adverse operating experience for BWR RVIs, the industry has adopted a proactive approach in developing enhanced periodic examination requirements for PWR RVIs, represented by MRP-227-A. These enhanced examinations, while focused on aging effects from mechanisms that are plausible and relevant, are designed to detect effects that may be generated by one mechanism or multiple mechanisms well before any potential loss of functionality or safety concern arises.

[redacted]

Q150. Is there operating experience regarding the effects of aging on RVIs developed from existing activities under the ASME Code ISI Program?

A150. (JRS, TJG, RGL) Yes. The ASME Code Section XI periodic ISIs for PWR RVIs, focused primarily on core support structures and consisting of general condition VT-3 visual examinations, have effectively detected the few instances of PWR RVI degradation that have occurred to date. For example, industry experience with the inspection of baffle-former bolts has been documented in Appendix A of MRP-227-A. *See* MRP-227-A, App. A (NRC000114C). These bolts are considered to be leading indicators for Westinghouse RVIs for the combination of irradiation-induced stress relaxation, void swelling, and irradiation-assisted stress corrosion cracking. *See* MRP-232, Rev.1 at 4-84 to 4-85 (ENT000643). Very few cracked or failed baffle-former bolts have been detected during these examinations and, in most cases, no cracked or failed bolts were detected at all. MRP-227-A, App. A at A-3 to A-4 (NRC000114C).

Q151. In 2011, Dr. Lahey suggested that because of the power uprates of 3.26% for IP2 in 2004 and 4.85% for IP3 in 2005, that the "plants are already being driven harder than their original designs envisioned." Report at 8 n.1 (NYS000296). Would you expect that power uprates at IP2 and IP3 would have a significant impact on the aging of RVIs?

A151. (ABC, NFA, JRS, TJG, RGL, RJD) No. As an initial matter, Dr. Lahey provides no evidence in support of his vague concern and identifies no specific problem in the RVI AMP as a result of the Stretch Power Uprates ("SPUs"). As part of the SPUs performed at IP2 and IP3, Entergy evaluated the effect of changes due to SPU on critical RVI components, and the NRC approved Entergy's evaluation. *See* Letter from P. Milano, NRC, to M. Kansler, Entergy, "Indian Point Nuclear Generating Unit No. 2 - Issuance of Amendment Re: 3.26 Percent Power

Uprate (TAC No. MC1865)" (Oct. 27, 2004) ("SER for IP2 Uprate") (ENT000652); Letter from P. Milano, NRC, to M. Kansler, Entergy, "Indian Point Nuclear Generating Unit No. 3 - Issuance of Amendment Re: 4.85 Percent Stretch Power Uprate and Relocation of Cycle-Specific Parameters (TAC No. MC3552)" (Mar. 24, 2005) ("SER for IP3 Uprate") (ENT000653).

Although IPEC power uprates implemented in 2004 and 2005 have resulted in increased reactor power, the resulting neutron irradiation is significantly less than originally predicted at the time of plant design because of the fuel management (*i.e.*, fluence reduction) strategies implemented over the years, as we will discuss in response to Question 203. Thus, the SPUs have a relatively small impact on the functionality or aging effects on RVIs.

b. Management of Aging Effects on Baffle-Former Bolts

Q152. On the topic of baffle-former bolts, Dr. Lahey criticizes Entergy's inspections as a "wait-and-see approach." Revised Lahey Testimony at 55-56 (NYS000486). How do you respond?

A152. (TJG, RGL, NFA, RJD) Entergy is not proposing a wait-and-see approach for baffle-former bolts. Entergy is planning to inspect 100% of the baffle former bolts at IP2 in Spring 2016 and at IP3 in Spring 2019, with subsequent examinations on ten-year intervals. *See* NL-12-037, Attach. 2 at 40 (NYS000496). To prepare for these inspections, as explained in SSER 2, the UT examination acceptance criteria for the baffle-former bolts will be developed as part of the technical justification ("TJ") for the inspections. The TJ must be developed by six months prior to the first inspections at each unit. *See* SSER 2 at 3-20 (NYS000507) (citing NL-12-089, Letter from F. Dacimo, Entergy, to NRC Document Control Desk, "Reply to Request for Additional Information Regarding the License Renewal Application" (June 14, 2012) (NYS000497)). As the Staff stated in SSER 2, this is acceptable because, as Entergy explained: (1) the Staff's Safety Evaluation for MRP-227-A does not specify that TJs must be submitted to the Staff for review and approval; (2) UT examinations of baffle-former bolts have been performed since the 1990s, so there is reasonable assurance that these examinations can be effectively implemented at Indian Point; and (3) finalizing the TJ closer to the date of inspections will allow the latest UT technology to be used and lessons learned to be incorporated. *See* SSER 2 at 3-20 (NYS000507). Dr. Lahey does not address or dispute the reasons for the Staff's conclusion.

Q153. What conclusion do you draw regarding the tolerance of baffle-former bolts to the effects of aging?

A153. (JRS, TJG, RGL) When the ISI inspection results are considered in combination with the existing analyses of acceptable baffle-former bolt patterns—that is, the relatively few baffle-former bolts needed to assure continued functionality and safety of the entire bolted assembly—we conclude that the overall tolerance of these RVI assemblies to aging effects, even when subjected to a combination of effects, remains high. The Westinghouse and IPEC analyses are discussed further below.

Q154. How do you respond to Dr. Lahey's assertion that Entergy has not developed acceptance criteria for the baffle former bolts? *See* Revised Lahey Testimony at 56 (NYS000482).

A154. (TJG, NFA, RJD, JRS) The examination acceptance criterion for individual baffle-former bolts will be no defect that can be detectable via UT (*i.e.*, a defect exceeding 30% of the bolt cross-sectional area). *See* WCAP-17096 at E-18 (ENT000635). The TJ will merely demonstrate that the UT inspections at IPEC will be capable of detecting such cracking. Similar TJs have been prepared at other plants.

Q155. Is Entergy required to provide all acceptance criteria for all components in its RVI AMP?

A155. (ABC, JRS) No. The SRP-LR explains:

> Acceptance criteria could be specific numerical values, or could consist of *a discussion of the process for calculating specific numerical values of conditional acceptance criteria* to ensure that the structure and component intended function(s) will be maintained under all CLB design conditions.

NUREG-1800, Rev. 1, at A.1-6 (NYS000195) (emphasis added).

Q156. Has an analysis been conducted for baffle-former bolts in Westinghouse 4-loop plants to determine their ability to withstand design basis loads, considering synergistic effects?

A156. (TJG, RGL, NFA, RJD) Yes. [redacted]

[redacted]

Q157. Would you agree that an acceptable baffle-former bolting pattern analysis is a "temporary, short-term 'fix'" that is not adequate for "shock" loads, as Dr. Lahey describes it? Revised Lahey Testimony at 47-48 (NYS000482).

A157. (TJG, RGL, NFA, RJD) No. An engineering evaluation, such as documented in WCAP-15270, is not a "temporary, short-term fix." Instead, WCAP-15270 demonstrates the capability to continue to operate and maintain full functionality, including the ability to withstand the full range of design basis loads.

Q158. Is Entergy preparing a more detailed, IPEC-specific minimum bolting pattern analysis?

A158. (RGL, NFA, RJD) Yes. Entergy has contracted with Westinghouse to perform a more realistic plant-specific minimum bolting pattern analysis for IPEC. This evaluation will consider design basis loads for IP2 and IP3, including the dynamic effects and blowdown loads from pipe breaks of various sizes, low cycle thermal fatigue loads, high cycle flow induced vibration loads, and seismic loads. If inspections reveal degradation of baffle-former bolts, then the minimum baffle-former bolting pattern will be used as the basis for engineering evaluations to determine the acceptability of the baffle-former bolts following the required UT examinations from MRP-227-A. *See* MRP-227-A at 6-9 to 6-11 (NRC000114B).

Q159. Will the new minimum bolting pattern analysis be prepared in accordance with an NRC-approved methodology?

A159. (RGL, NFA, RJD) Yes. WCAP-17096 provides standard methodologies for developing engineering evaluations for Westinghouse RVIs. *See* WCAP-17096 (ENT000635). For the development of a minimum baffle-former bolting pattern analysis, WCAP-17096 points to Westinghouse, WCAP-15030-NP-A, Westinghouse Methodology for Evaluating the

[redacted]

Acceptability of Baffle-Former-Barrel Bolting Distributions under Faulted Load Conditions (Mar. 2, 1999) ("WCAP-15030-NP-A") (ENT000655). *See* WCAP-17096 at 7-1, C-3 (ENT000635). The NRC Staff has reviewed and approved WCAP-15030-NP. *See* WCAP-15030-NP-A at v-vi (ENT000655). Westinghouse will use the methodology approved by the NRC Staff to develop the plant-specific minimum baffle-former bolting pattern evaluations for IP2 and IP3. This evaluation is being prepared in advance of the spring 2016 refueling outage at IP2, when the baffle-former bolts will be inspected.

Q160. How do you respond to Dr. Lahey's reference to Entergy's statement that, "[a]s with other U.S. commercial power plants, cracking of baffle-former bolts is recognized as a potential issue," EPRI's acknowledgment of observed cracking of baffle-former bolting in European PWRs, and experience with degradation of split pins? *See* Revised Lahey Testimony at 47 (NYS000482).

A160. (TJG, RGL, NFA, RJD) Dr. Lahey's comments regarding operating experience with degraded bolting and pins do not show any deficiency in the RVI AMP, but instead simply illustrate that Entergy is appropriately using operating experience in the RVI AMP.

For the baffle-former bolts, as Dr. Lahey states, *Entergy has acknowledged* that the MRP considered the U.S. and international operating experience in the development of the inspection scope, methods and frequency in MRP-227-A. MRP-227-A, Appendix A, at A-3 to A-4 (NRC000114C). This experience also was the primary impetus for WCAP-15270, WCAP-15030, and for the plant-specific evaluation underway for IPEC. *See* WCAP-15270 at 1-1 (ENT000654). Thus, Entergy is addressing this operating experience through its inspections of baffle-former bolting as part of the IPEC RVI AMP. While Dr. Lahey demands "replace[ment] of the degraded bolts," Revised Lahey Testimony at 56 (NYS000482), there is no evidence that

Entergy's plans to manage such potential degradation through inspections, analysis, and, as necessary, replacement do not meet the regulatory requirement to adequately manage (rather than preclude) the effects of aging.

As for the control rod guide tube alignment pins (split pins), Entergy is again addressing this operating experience. It replaced the split pins at IP3 in 2009 with cold-worked Type 316 stainless steel, which is a significant improvement over the previous material, Alloy X-750. Entergy has committed to replace the split pins at IP2 in 2016. NL-14-067 Attach. 1 at 3-4 (NYS000504) and SSER 2 at 3-36 to 3-38 (NYS000507). Dr. Lahey raises no dispute with Entergy's plans regarding split pins.

c. Management of Aging Effects on Clevis Insert Bolts

Q161. Dr. Lahey suggests that Entergy should replace another set of components, the clevis insert bolts. Revised Lahey Testimony at 56-57 (NYS000482). What are the clevis insert bolts?

A161. (TJG, RGL, NFA, RJD) The clevis inserts are designed to limit the tangential motion between the lower end of the core barrel and the vessel. There are six clevis components welded to the inside of the RPV. Each clevis insert is fit to the welded clevis to provide alignment with the radial keys on the core barrel. The clevis insert bolts help hold the insert in place. *See* Westinghouse, Technical Bulletin TB-14-5, Reactor Internals Lower Radial Support Clevis Insert Cap Screw Degradation at 2 (Aug. 25, 2014) ("TB-14-5") (ENT000656).

Q162. Are clevis insert bolts safety-related components?

A162. (TJG, RGL, NFA, RJD) No. Once the core barrel is installed, clevis insert motion is restrained and the bolts are not required to keep the insert in place during plant operation. *See* TB-14-5 at 2-4 (ENT000656). As long as the clevis inserts (not the bolts) remain in place, the overall safety function of the lower radial supports will not be affected by bolt failure, even under LOCA or other design basis loading conditions. *See id.*

Q163. Does Entergy inspect the clevis insert bolts at IPEC?

A163. (TJG, RGL, NFA, RJD) Yes. The clevis insert bolts are treated as an Existing Program component under the RVI AMP, because they are periodically inspected once every ten-year interval under the ASME Code, Section XI program per Table IWB-2500-1. *See* NL-12-037, Attach. 2 at 51 (NYS000496). Entergy last inspected the clevis insert bolts at IP2 in 2006 and at IP3 in 2009. *See* NL-14-067, Attach. 1 at 5 (NYS000504).

Q164. Dr. Lahey cites operating experience from another Westinghouse plant in 2010 to criticize Entergy's alleged proposed "wait-and-see approach" for the clevis insert bolts. Specifically, he suggests that given the extent of degradation found at the other plant, Entergy should not have proposed "to manage the aging degradation of clevis insert bolts with visual (VT-3) inspections rather than pre-emptive replacement." Revised Lahey Testimony at 56-57 (NYS000482). How do you respond?

A164. (TJG, RGL, NFA, RJD) As discussed in the operating experience summary in MRP-227-A, in 2010 certain damaged clevis insert bolts were detected at a Westinghouse-designed reactor. *See* MRP-227-A, Appendix A, at A-2 (NRC000114C). The degraded clevis insert bolts were fabricated from an X-750 alloy with a heat treatment that is not used in the clevis insert bolts at IP2 and IP3. *See also* NL-13-122, Attach 1 at 9 (NYS000502).

The NRC Staff issued an RAI to Entergy on this issue during the review of the RVI AMP. In its September 27, 2013 RAI response, Entergy demonstrated that the existing periodic inspections under the ASME Code Section XI program are adequate for IPEC. SSER 2 at 3-23 to 3-25 (NYS000507); *see also* NL-13-122, Attach 1 at 4 (NYS000502). This conclusion is based on two primary considerations. First, there is inherent design redundancy in the lower radial support system. The overall ability of the system to perform its intended function, even under seismic and LOCA conditions, will not be compromised by failure of clevis insert bolts. *See* SSER 2 at 3-24 (NYS000507). Second, the clevis insert bolts at IP2 and IP3 are not in the most susceptible heat-treatment condition for PWSCC. *See id*.

Q165. Dr. Lahey discusses the alleged "high rate of failure (about 60% of the total bolts were damaged) and low rate of visual detection (only about 24% of the damaged bolts were detected)" in the 2010 operating experience involving clevis insert bolts. Revised Lahey Testimony at 56-57 (NYS000482). Has Entergy considered that issue in developing the clevis insert bolt inspections?

A165. (RGL, NFA, RJD) Yes. Beyond the information in SSER 2, Westinghouse further evaluated this operating experience in Technical Bulletin TB-14-5 issued on August 25, 2014. *See generally* TB-14-5 (ENT000656). As explained in TB-14-5, the primary function of the clevis inset bolt is to draw the clevis insert into the matching reactor vessel lug and hold it in place. However, as we have previously explained, the clevis insert bolts are not part of the load path and the failure of the bolts simply does not degrade the ability of the clevis inset to transfer load from the barrel to the vessel. *See id*. In the reported operating experience, there was no indication that bolt failure had led to displacement of the clevis insert itself.

For this reason, volumetric inspections of the clevis insert bolts were not required, and visual examinations (VT-3), consistent with ASME Code, Section XI, Table IWB-2500-1, remain sufficient.

Q166. Dr. Lahey further states that in the evaluation of the clevis insert bolt operating experience, Entergy assumed, in its RAI response on the clevis insert bolts, "that

[redacted]

all other components will be functioning according to their design specifications, and does not consider the fact that the other components may also be undergoing degradation from various interacting aging mechanisms." Revised Lahey Testimony at 57-58 (NYS000482). How do you respond?

A166. (RGL, NFA, RJD) Entergy is appropriately managing the effects of aging on the clevis insert bolts at IPEC. We disagree with Dr. Lahey because, first, as the NRC Staff noted in SSER 2, there is a high degree of redundancy in the lower radial support system, of which the clevis inserts are a part. *See* SSER 2 at 3-24 (NYS000507). Entergy is not required to assume (without evidence) that other components that are within the scope of the RVI AMP or another AMP also are degraded when it evaluates the functionality of the clevis insert bolts. *See* NRC Inspection Manual, Manual Chapter 2516, Policy and Guidance for the License Renewal Inspection Program at 2 (Aug. 13, 2013) (ENT000657) ("Postulated failures that could result from system interdependencies that are not part of the CLB and that have not been previously experienced need not be considered as part of a [LRA]."). Second, as we stated, Entergy is periodically inspecting the clevis insert bolts at IPEC. And third, as we have shown, the adverse operating experience Dr. Lahey cites is not applicable to the clevis insert bolts at IPEC, so there is no basis for additional volumetric inspections at IPEC.

5. The RVI AMP Addresses Lower Support Column Caps and Combinations of Aging Effects from Multiple Degradation Mechanisms

a. Potential Combinations of Aging Effects

Q167. Is it possible for multiple aging effects to impact RVI components in a synergistic manner?

A167. (RJD, TJG, RGL, JRS) In general, yes. MRP-227-A considers that possibility. As we have explained, the inspections conducted under the RVI AMP will detect the effects of

[redacted]

aging regardless of the underlying cause or causes and trigger any necessary corrective actions. *See, e.g.*, MRP-175, (ENT000631); MRP-191 (NYS000321); MRP-232 (ENT000642).

Q168. How did the studies supporting MRP-227-A consider potential combinations of aging effects?

A168. (TJG, RGL) During the development of MRP-227-A, the Reactor Internals Focus Group of the EPRI MRP, including participants from Westinghouse, Combustion Engineering, AREVA, and others, developed a set of standard screening criteria that were used to identify components with one or more potential aging mechanisms and how those effects could combine to affect functionality. *See generally* MRP-175 (ENT000631); MRP-230 (ENT000644). The resulting aging effects were first examined individually, and then also in combination to determine whether the combined effects could have negative effects on the function of the components. Components with one or more identified degradation mechanisms were categorized depending on the potential consequences of degradation (*i.e.*, they were categorized based on the effects). *See* MRP-191 § 6 (NYS000321) (discussing the failure modes, effects, and criticality analyses, "FMECA"). This "waterfall strategy" identified the locations for aging effects. The waterfall strategy is illustrated in Figure 6-1 from MRP-134 (ENT000647).

The work documented in MRP-175 to identify thresholds for aging effects was then used to develop the screening and categorization results documented in MRP-191 (NYS000321), which in turn provide the technical basis for the functionality analysis in MRP-230 (ENT000644) and ultimately the examinations specified in MRP-227-A.

Q169. As you noted, Dr. Lahey expresses an "over-arching" concern that the NRC allegedly considers "various aging mechanisms" in "silos," so that the effects of fatigue, embrittlement, and corrosion are considered separately, without considering "synergistic

[redacted]

interactions" between these mechanisms. Revised Lahey Testimony at 14-15; *see also* Report at 6 (NYS000296). How do you respond?

A169. (ABC, JRS, TJG, RGL, RJD, NFA) Dr. Lahey's concern is without foundation. As an initial matter, the NRC's license renewal regulations in 10 C.F.R. Part 54 focus on the evaluation and management of aging effects, rather than individual aging mechanisms. The Commission articulated this intent clearly in the Statements of Consideration for the current Part 54. *See* Part 54 SOC, 60 Fed. Reg. at 22,469 (NYS000016). Thus, rather than addressing the individual mechanisms—such as various types of fatigue, embrittlement, or corrosion—the NRC's license renewal process is aimed at managing the effects of fatigue, embrittlement, and corrosion. Consistent with the license renewal rule, MRP-227-A considered the individual underlying aging mechanisms, but focuses its inspections on the manifestations of aging effects, regardless of the cause or causes. *See id.* at 22,463.

In addition, Dr. Lahey's concerns appear to be based on speculation that such issues have not been addressed, rather than on a review of the work that led up to MRP-227-A. As we will explain in response to Question 172, during the development of MRP-227-A, EPRI appropriately considered combinations of aging effects, including potential "synergistic" effects that could affect the RVIs. As the NRC Staff concluded in its Safety Evaluation for MRP-227-A, EPRI considered: "individual *or synergistic*" effects of thermal aging or neutron irradiation embrittlement, and "loss of preload due to either individual *or synergistic* contributions from thermal and irradiation-enhanced stress relaxation . . . ." SE for MRP-227-A at 4 (emphasis added) (NRC000114A).

Thus, we disagree with Dr. Lahey that the RVI AMP—or the license renewal process in general—inappropriately considers various aging mechanisms in "silos" without considering the combined effects of multiple mechanisms. In short, Dr. Lahey does not dispute *how* EPRI addressed his over-arching concern—he only incorrectly asserts that he has "discovered" this important new issue. Revised Lahey Testimony at 78 (NYS000482). This misconception is a fundamental reason why NYS-25 lacks merit.

Q170. Has Dr. Lahey identified any combination of aging effects that EPRI did not consider in developing MRP-227-A?

A170. (RGL, TJG) No. Dr. Lahey has not identified any combinations of aging effects that could result from multiple aging mechanisms that are different from the effects already managed through MRP-227-A and the RVI AMP.

Q171. Dr. Lahey asserts that the Department of Energy ("DOE") the NRC, and national laboratories have "recently embarked on an ambitious R&D program to understand and resolve" his concerns regarding the synergistic aging effects on nuclear plant components. *See, e.g.*, Revised Lahey Testimony at 17 (NYS000482) (citing DOE, Light Water Sustainability Program, Material Aging and Degradation Technical Program Plan (Aug. 2014) ("MAaD Program Plan") (NYS000485)). Do these new research programs suggest any deficiency in MRP-227-A or the IPEC the RVI AMP?

A171. (ABC, JRS, TJG, RGL) No. The "ambitious R&D program" Dr. Lahey describes is intended to address the long-term challenges and research needs for operating nuclear plants beyond 60 years, not beyond 40 years. *See* MAaD Program Plan at iii-iv (NYS000485). The IPEC LRA is not seeking to extend operations beyond 60 years.

The activities of the U.S. Department of Energy's Light Water Reactor Sustainability Program similarly address the challenges of second, not first, license renewal. This is clear from

[redacted]

the very first page of the Executive Summary of the State's exhibit. *Id.* at iii ("Extending reactor service to beyond 60 years will increase the demands on materials and components.").

In any event, the MRP-227 inspection and evaluation guidelines were designed to accommodate the uncertainties associated with areas where research remains ongoing. For example, there is ongoing research to develop improved void swelling models. *See* Jean Smith, Irradiated Materials Testing, INSIDE MRP at 2 (Winter 2012) (ENT000658). But the void swelling calculations used in MRP-230 and MRP-232 to evaluate this effect were based on aggressive assumptions about neutron doses and heat generation rates in the baffle-former structure that emphasized the impact of swelling. *See* MRP-232 at 1-1 (ENT000642). In another example, although there was ongoing research on the combined effects of thermal aging and irradiation on fracture toughness of CASS, in the generic guidelines in MRP-227-A, CASS components are simply assumed to be embrittled for purposes of engineering evaluations. *See* MRP-191 at 5-3 (NYS000321).

Ultimately, the fact that certain research is ongoing is not an indication of any deficiency in an AMP. In fact, it is a sign of a healthy, constantly-improving program.

Q172. Based on EPRI's analysis in MRP-227-A, what combinations of aging effects are PWR RVIs susceptible to?

A172. (TJG, RGL) As documented in the various studies supporting MRP-227-A, during the screening and evaluation process the MRP considered potential interactions between the eight degradation mechanisms discussed in Section 3.2 of MRP-227-A, and the potential for combined aging mechanisms. *See*, *e.g.*, MRP-191 at 6-6 (NYS000321) (showing that a component with multiple screened-in degradation mechanisms would have a Component Failure

Likelihood of at least medium). A few examples illustrate specific combinations the MRP identified.

First, Irradiation-Assisted Stress Corrosion Cracking ("IASCC"), as its name implies, itself is caused by a combination of SCC and irradiation. Therefore materials or components that are subject to IASCC are also subject to irradiation embrittlement ("IE") and potentially to void swelling and irradiation-induced stress relaxation. These mechanisms do not occur in isolation.

A second example is fatigue in combination with the various types of SCC, where the common resulting effect is cracking and the combination, if it were to occur, may cause more rapid crack growth. *See*, *e.g.*, MRP-227-A at 6-3 (NRC000114B) (showing that fracture mechanics evaluations consider whether flaw growth is due to fatigue, SCC, or a combination).

A third example, explained in MRP-175, is that irradiation-enhanced stress relaxation of threaded fasteners is a potential cause of fatigue and wear. *See* MRP-175 at 2-5, 2-9 to -11 (ENT000631). Also in MRP-175, EPRI applied a reduced fluence threshold for IE to cast stainless steel components to account for a hypothetical possibility of combined interactions between IE and thermal embrittlement ("TE"). *See id.* at 2-8.

In a fourth example, Westinghouse created a detailed finite element model of a representative Westinghouse core baffle/former/barrel structure and simulated 60 years of reactor operation, in order to better understand the interaction of the irradiation-related degradation mechanisms in the RVIs. *See* MRP-230 § 3.1 (ENT000644). The results were used a basis for the MRP-227 inspection guidelines. *See* MRP-232 § 4.2 (ENT000642).

Q173. Are all potential combinations of aging mechanisms "synergistic"?

A173. (TJG, RGL) No. Dr. Lahey broadly implies that the synergy between aging mechanisms may actually have a greater (*i.e.*, worsening) effect than the sum of the individual mechanisms alone. *See* Revised Lahey Testimony at 17 (NYS000482). This overlooks the fact that the "synergistic" effect may in some cases have less of an effect, or an improvement, in the material's resistance to aging. As we explained in response to Question 111, above, fatigue and irradiation embrittlement, for example, contribute to potential aging effects in very different ways, and in some cases irradiation changes the strength properties and can increase the fatigue life of RVI materials.

Q174. During the review of MRP-227-A, did the NRC Staff ask EPRI to explain how it addressed "synergistic" effects?

A174. (TJG, RGL, JRS) Yes. During its technical review of MRP-227, the NRC Staff specifically requested additional information on how the program accounts for "synergistic" effects. MRP-227-A, Attachment, Request for Additional Information (RAI) # 4 at 4 (Aug. 30, 2010) (NRC000114D). EPRI responded by explaining that:

> potential susceptibility to the effects from multiple degradation mechanisms was considered by: (1) identifying such combinations during the initial screening based on known interactions (e.g., irradiation-induced stress relaxation of bolt pre-load combined with either wear or fatigue); (2) FMECA [failure modes, effects, and criticality analyses] expert elicitation of combined effects that resulted in greater consequences; and (3) recommending examinations capable of detecting relevant conditions caused by more than one degradation mechanism or effect.

MRP-227-A, Attachment, RAI Set 4 Final Responses at 20 (Oct. 29, 2010) (NRC000114D). As EPRI further explained in the same RAI response:

> [F]or many components subject to more than one degradation mechanism with moderate or significant effects, the final inspection category recommendation reflected the need for the inspection to detect an effect common to more than one degradation mechanism (e.g., cracking caused by IASCC and fatigue). And, in a few cases, the final inspection category recommendation reflected the need for an inspection capable of detecting more than one effect during the same examination (e.g.,

> distortion caused by void swelling; gross cracking and material separation caused by IASCC).

Id. Thus, as EPRI explained in 2010, the RVI AMP specifies inspections that address the underlying aging mechanisms and, more importantly, the resulting aging effects—such as cracking, loss of material, loss of fracture toughness, dimensional changes, and loss of preload—regardless of the particular underlying aging mechanism or *combination of mechanisms*.

And as previously noted, in its 2011 Safety Evaluation for MRP-227-A, the NRC Staff reached the same conclusion, noting that EPRI considered potential degradation mechanisms, including various combinations of effects. *See* SE for MRP-227-A at 4 (emphasis added) (NRC000114A). Dr. Lahey does not acknowledge this information or explain why EPRI's considerable efforts to identify and address interactions between aging mechanisms (and their resulting effects) are inadequate.

b. Management of Aging Effects on Lower Support Column Caps

Q175. Did the NRC Staff impose any additional requirements for licensees to consider multiple aging mechanisms for certain components on a plant-specific basis?

A175. (RGL, TJG, NFA, RJD, JRS) Yes. As previously noted, as part of the Safety Evaluation for MRP-227-A, the NRC Staff imposed several Topical Report conditions and A/LAIs requiring additional inspections to address how one or more degradation mechanisms that could affect certain components, and required a plant-specific evaluation of possible combined aging effects on susceptible components. *See* SE for MRP-227-A at 32-34 (NRC000114A). For example, A/LAI 7 requires a plant-specific evaluation to demonstrate that susceptible CASS RVI components will maintain their functionality during the PEO, considering possible loss of fracture toughness due to TE, IE, and other mechanisms. *Id*. at 34.

[redacted]

Q176. Did the NRC Staff issue RAIs to Entergy on this issue during its review of the RVI AMP?

A176. (RGL, TJG, NFA, RJD) Yes. During the NRC Staff's review of the RVI AMP, the Staff issued a series of RAIs to Entergy on this issue, asking for consideration of the potential combined effects of not only TE and IE, but also IASCC in CASS components. *See* SSER 2 at 3-41 to 3-47 (NYS000507); *see also* Letter from R. Kuntz, NRC, to Vice President, Operations, Entergy, "Request for Additional Information for the Review of the Indian Point Nuclear Generating Unit Nos. 2 and 3, License Renewal Application," Encl. (May 15, 2012) (ENT000659); Letter from J. Daly, NRC, to Vice President, Operations, Entergy, "Request for Additional Information for the Review of the Indian Point Nuclear Generating Unit Nos. 2 and 3 License Renewal Application, Set 2013-01 (TAC Nos. MD5407 and MD5408)," Encl. (Feb. 6, 2013) (ENT000660); Letter from K. Green, NRC, to Vice President, Operations, Entergy, "Request for Additional Information for the Review of the Indian Point Nuclear Generating Unit Nos. 2 and 3 License Renewal Application, Set 2013-04 (TAC Nos. MD5407 and MD5408)," Encl. (July 26, 2013) (ENT000661); Letter from K. Green, NRC, to Vice President, Operations, Entergy, "Request for Additional Information for the Review of the Indian Point Nuclear Generating Unit Nos. 2 and 3 License Renewal Application, Set 2014-02 (TAC Nos. MD5407 and MD5408)," Encl. (Apr. 9, 2014) (ENT000662).

In response to these RAIs, Entergy showed that the CASS lower core support columns (column caps) for IPEC are not susceptible to TE, so they are not susceptible to the combined effects of IE and TE together, and provided further justification for how RVI AMPs will manage the potential aging effects on this component. *See* NL-12-134, Letter from F. Dacimo to NRC Document Control Desk, "Reply to Request for Additional Information Regarding License

[redacted]

Renewal Application" (Sept. 28, 2012) (NYS000498); NL-13-052, Letter from F. Dacimo to NRC Document Control Desk, "Reply to Request for Additional Information Regarding License Renewal Application" (May 7, 2013) (NYS000501); NL-13-122 (NYS000502); NL-14-093 (NYS000505). The NRC Staff approved this approach in SSER 2. *See* SSER 2 at 3-47 (NYS000507).

Q177. Dr. Lahey cites a recent report from Argonne National Laboratory, which, acknowledges that "with respect to [CASS components], that 'a combined effect of thermal aging and irradiation embrittlement could reduce the fracture resistance even further to a level neither of these degradation mechanisms can impart alone,'" and stating that "'no data are available at present with regard to the combined effect of thermal aging and irradiation embrittlement' on CASS." Revised Lahey Testimony at 18 & 20 (NYS000482) (quoting Chen, *et al.*, "Crack Growth Rate and Fracture Toughness Tests on Irradiated Cast Stainless Steels," NUREG/CR-7184 at xv (Revised Dec. 2014)) (NYS00488A-B). Do you agree with Dr. Lahey?

A177. (TJG, RGL, RJD) We agree with Dr. Lahey that this is an area of ongoing generic research, but not with his assertion that no data are available. A significant amount of research and attention has already been focused on the combined effects of TE and IE in CASS materials, and the impact for PWR internals is still being addressed both on a plant-specific basis and on a generic basis through the combined efforts of the BWRVIP and the EPRI MRP. Those groups have recently provided a joint statement of position to the NRC Staff summarizing those research findings. *See generally* Letter from A. McGehee & T. Hanley, EPRI, to NRC Document Control Desk, BWRVIP 2015-025, "Project No. 704 – Summary of Industry Position on Screening Criteria for Thermal and Irradiation Embrittlement for PWR and BWR Reactor

Internals Fabricated of Cast Austenitic Stainless Steel" (Mar. 9, 2015) (ENT000663). This is a key reason why the NRC Staff identified the need for further evaluation of CASS components in A/LAI 7, and ultimately required Entergy to further address the potential for thermal and irradiation embrittlement of CASS components, as explained below.

Q178. Are the potential combined effects of thermal and irradiation embrittlement on CASS components an issue for IPEC?

A178. (TJG, RGL, RJD) No. Dr. Lahey cites Chopra, O.K., "Degradation of LWR Core Internal Materials Due to Neutron Irradiation," NUREG/CR-7027 (Dec. 2010) (NYS000487) in support of his concerns on this point. This study considered worst case assumptions from bounding materials (*e.g.*, high-molybdenum, grade CF-8M) present in either PWRs or BWRs. The IPEC RVIs, however, do not include CF-8M materials. In fact, the maximum ferrite content for the IP2 CASS material (the LSCCs) is 14.6%, and the maximum ferrite content for the IP3 LSCCs is 11.8%. *See* NL-14-013, Attach. 1 at 3, 4 (NYS000503). These values are below the 20% ferrite screening value which is the level required for a material to experience a measurable decrease in the fracture toughness as a result of TE, and below the NRC Staff's conservative 15% screening criterion for CASS RVI materials. *See* Email from J. Holonich, NRC, to K. Amberge, EPRI, et al., "Summary Tables for CASS Position," Attach. at 1 (Mar. 12, 2014, 1:23 PM) (ENT000664); *see also* SSER 2 at 3-44 to -45 (NYS000507). In comparison, many of the heats evaluated in NUREG/CR-7027 had delta ferrite content greater than 20% and as high as 42%.

6. The RVI AMP Addresses Appropriate Design Basis Loads, Including Seismic and LOCA Loads

Q179. Dr. Lahey raises the possibility that, given the "synergistic interactions" he describes, RVI components "may not be able to survive the shock loads associated with

[redacted]

significant seismic events or the pressure and/or thermal shock loads induced by various accidents and severe operational transients." Revised Lahey Testimony at 15, 16 (NYS000482); *see also, e.g.*, Report at 6-7, 15, 17, 19-20 30-31 (NYS000296); Lahey 2011 Testimony at 17-18, 36, 38 (NYS000294) (raising similar issues). How does the RVI AMP address this possibility?

A179. (RGL, TJG, JRS, NFA, RJD, ABC) It is not entirely clear what Dr. Lahey means by the term "shock loads." If his concern is with loads that are greater than or different from those specified in the CLB for IP2 and IP3, or with scenarios that are beyond the plants' licensing bases, there is no requirement to address such loads in The IPEC RVI AMP. But if his concern is with loads that are within the CLB of IP2 and IP3, then such loads are fully addressed in the RVI AMP.

Q180. How do MRP-227-A and the IPEC RVI AMP engineering evaluation process address the potential impact on the RVIs of accident loads such as those resulting from a design basis accident LOCA, seismic event, or other significant accident or transient which might occur at the end of the PEO?

A180. (TJG, RGL, RJD, NFA, JRS) The MRP-227-A inspection and evaluation guidelines are intended to detect conditions that may impair the continued functionality of the RVIs, under CLB loads, including LOCA and seismic loads. Indeed, the very purpose of the program is to provide reasonable assurance that the RVIs will continue to perform their intended functions, consistent with the CLB—including the consideration of accident loads—through the end of the PEO. *See* 10 C.F.R. §§ 54.21(a)(3), 54.21(c)(1)(iii), and 54.29(a).

Specifically, the effect of irradiation on stainless steel is to generally increase strength. *See* MRP-175, App. F § F.1 (NYS000319). The detrimental impact of irradiation embrittlement

[redacted]

is on the decreased ability of the material to tolerate cracking. *See id*. § F.2. Without crack formation, both RVI austenitic steels and RPV ferritic steels are strengthened by irradiation (*i.e.*, the yield and ultimate stresses increase with dose). *See* Was Text at 581-582 (ENT000627). Therefore, the MRP-227-A guidelines specify inspections of key irradiated components to assure that there are no cracks that could lead to failure and loss of functionality under transient loads. *See, e.g.*, MRP-227-A at 4-26 (control rod guide tube assembly lower flange welds); 4-29 (thermal shield flexures) (NRC00114B).

In summary, the intent of the MRP-227 inspection program is to identify cracking that could degrade the ability of the component to withstand CLB loads. Without the presence of cracking, the ability of the irradiation-strengthened material to withstand normal and accident loads is not degraded. If a degraded component is discovered, then MRP-227-A requires the explicit evaluation of CLB loads, including "shock loads" such as acoustic loads and rarefraction waves due to a LOCA in an engineering evaluation. *See* MRP-227-A § 6 (NRC00014B).

Q181. Does Dr. Lahey address or critique this approach to the consideration of CLB loads?

A181. (TJG, RGL) No. Dr. Lahey merely states, without support, that synergistic aging effects and "shock loads" "have not been considered." Revised Lahey Testimony at 15-16 (NYS000482); *see also, e.g.*, 2015 Lahey Declaration at 13 ¶ 19 (NYS000483) ("New York's main concerns . . . have simply been ignored"). In fact, it is New York State and Dr. Lahey that have ignored the decade or more of engineering work that underpins the RVI AMP.

Q182. Dr. Lahey further states that "[a]n inspection-based approach to aging management, such as the one developed by the nuclear industry in MRP-227 and condoned by USNRC in MRP-227-A, is useful but it fails to account for the possibility that highly

embrittled and fatigued RVI components may not have signs of degradation that can be detected by an inspection, but such weakened components could nonetheless fail as a result of a severe seismic event or thermal or pressure shock load." Revised Lahey Testimony at 39-40 (NYS000482). How do you respond?

A182. (TJG, JRS, RGL, RJD) We disagree. Dr. Lahey is positing that incipient failure can occur in RVI components with no visible cracks or no early indication of material weakness. As we have discussed, the materials used in RVI components are resilient, and designed in a redundant fashion. *See* Section VII.A.4, Further, as we have shown throughout our testimony, the RVI AMP is a prioritized inspection program based on the systematic evaluation that culminated in MRP-227-A. That systematic evaluation identified the most susceptible components.

Failure of a component without a pre-existing crack is governed by the mechanical properties of the material, the yield and ultimate strength, in particular. Irradiation increases the yield and ultimate strengths. *See* Was Text at 581-582 (ENT000627). Thus, while the final failure may show less ductility due to the level of embrittlement, the increased yield and ultimate strengths actually increase the load carrying capacity of the component such that CLB design margins will be maintained.

Dr. Lahey provides no evidence for his claim that undetectable degradation is lurking and can result in catastrophic failure of these resilient materials with no warning. From a microstructural and metallurgical standpoint, he provides no hypothesis—much less any evidence—for *how* the flaw-tolerant RVI materials can become so brittle that they can fail without any visible cracks under design basis loads. Furthermore, for many of the components

Dr. Lahey raises concerns about, there are detailed functionality analyses showing that the components are designed in a highly redundant manner. *See* Section VII.A.4, *supra*.

Q183. Dr. Lahey further alleges that "the applicant's reactor safety analyses implicitly assume that the reactor core will maintain a coolable geometry during emergency core cooling system ("ECCS") operation subsequent to a DBA LOCA, notwithstanding the degradation and possible deformation or relocation of various RVI components and potential flow blockages and degraded core cooling which may result." Revised Lahey Testimony at 54 (NYS000482). Do you agree?

A183. (TJG, RGL, NFA, RJD, JRS) No. Entergy did not "assume" these facts. The need for RVI components to maintain functionality including structural support and core coolability under design basis loading conditions is a CLB requirement that was a starting point for the development of the inspection program in MRP-227-A. *See* Response to Question 180, above.

Q184. Dr. Lahey raises the possibility that a severe earthquake or shock load on RVIs could lead to "core blockage," 2011 Testimony at 17 (NYS000294), or to "the loss of a coolable core geometry." Revised Lahey Testimony at 29 (NYS000482); *see also* Report at 16 (NYS000296). How does the RVI AMP address this possibility?

A184. (TJG, RGL, NFA, RJD) As we explain throughout our testimony, the inspections specified in the RVI AMP and Inspection Plan are designed to maintain design margins and to provide assurance that material properties are maintained such that failures will not occur due to design basis loads such as those from a severe earthquake or other accident loads; *i.e.*, under the scenarios posited by Dr. Lahey.

Q185. Do accident loads need to be considered as a contributor to the effects of aging on RVI components?

A185. (NFA, TJG, RGL, JRS) No. Aging is a gradual, long-term degradation of a component resulting from sustained environmental conditions (*e.g.*, applied loads and residual stresses). *See* Part 54 SOC, 60 Fed. Reg. at 22,475 (NYS000016). One-time loads from accidents or other events, such as pipe breaks and seismic events, do not contribute to aging. This is why the ASME Code, Section III excludes accident loads from fatigue evaluations. *See* ASME Code, Section III, Article NB-3000, "Design" § NB-3200 (1989) (NYS000349).

The ASME Code, Section III compares accident loads, such as large break LOCAs and large main steam line breaks, to the stress allowables to ensure that the affected components remain capable of performing their intended safety function during and after the event. *See id*.

NRC Staff guidance further explains that postulated accident loads, such as LOCA or seismic loads, need not be considered as a contributor to aging effects. As stated in the SRP-LR:

> The applicable aging effects to be considered for license renewal include those that could result from normal plant operation, including plant/system operating transients and plant shutdown. *Specific aging effects from abnormal events need not be postulated for license renewal*.

NUREG-1800, Rev 1, App A, A.1.2.1 (NYS000195). Thus, loads associated with a postulated LOCA which is a design basis accident need not be considered as a contributor to aging effects. This position is logical because LOCAs are rare events that are independent of the age of the plant. Additionally, if an emergency or faulted condition should occur, then the licensee would be required to evaluate its impact on affected components and take appropriate corrective actions prior to returning the unit to operation. But regardless of the age of the components, the potential for their failure as a result of accident loads is addressed, as explained in this section.

[redacted]

7. The RVI AMP Uses Appropriate Inspection Techniques

Q186. What inspection techniques are used in the RVI AMP?

A186. (RJD, NFA, JRS, TJG, RGL) MRP-227-A identifies inspection techniques for those PWR RVI components that are most susceptible to the aging effects of concern and have the highest risk associated with failure. Based on MRP-227-A, four types of inspection techniques are specified in the IPEC program:

- **Volumetric (ultrasonic, or "UT")** examination is specified in order to detect indications in the most potentially-affected bolting, in particular the baffle-former bolts. Functionality can be confirmed by physical measurements where, for example, loss of material caused by wear may occur. *See* NL-12-037, Attach. 2 at 25-26 (NYS000496).

- **Enhanced visual examination, (or "EVT-1")** is specified for components that require the detection of very tight, surface-breaking defects.

- **Visual ("VT-1")** examination is specified for components that are generally tolerant to surface-breaking defects, but where the length of those surface-breaking defects needs to be determined with some degree of accuracy.

- **Visual ("VT-3"**) examination, *i.e.*, visual inspections used to detect indications and imperfections, is specified when only the general condition of the RVI component is required, in accordance with the relevant conditions described in the ASME Code Section XI, IWB-3520.2. *See* ASME Boiler & Pressure Vessel Code, Section XI, Subsection IWB, "Requirements for Class 1 Components of Light-Water Cooled Plants" § 3520.2 (2010) (ENT000665).

The standards for deployment of these inspection techniques and the necessary qualification requirements for both equipment and personnel are given in MRP-228 at 2-11 to 2-15 (NYS000323).

Q187. For which types of components or purposes are these techniques used?

A187. (RJD, NFA JRS, TJG, RGL) Section 4.2 of MRP-227-A explains that the different non-destructive examination NDE methods are suitable, depending upon: (1) tolerance of the component functionality to the progression of particular effects, (2) accessibility of the component by the equipment needed for the examination, and (3) suitability of the equipment for detecting the particular effect. *See* MRP-227-A at 4-3 (NRC000114A). The selected methods are consistent with those specified in the NRC-approved versions of ASME Code Section XI. *See id*. at 4-3. When visual examinations are specified for particular RVI components in MRP-227-A, the relevant conditions are described in Column 3 of the tables in Section 4.

Entergy will qualify visual testing methods used to manage cracking in RVIs in accordance with MRP-228. *See* NL-12-037, Attach. 1 at 6 (NYS000496). Consistent with MRP-228, the non-destructive examinations will be qualified to detect the aging effect being monitored. Entergy will continue to monitor operating experience and adjust inspection techniques as necessary. *See* NL-12-037, Attach. 1 at 9 (NYS000496) ("this program will account for applicable future operating experience during the period of extended operation.")

Q188. What is the basis for the adequacy of these inspection techniques?

A188. (RJD, NFA JRS, TJG, RGL) The basis for the adequacy of these inspection techniques is described in the companion document to MRP-227-A, which is MRP-228. This document describes the standards to be met by each specified examination method, including the requirements for Technical Justification, in accordance with Article 14 of the ASME Code Section V, and such details as the lighting, camera speed, and surface cleanliness for remote visual examinations. Section 3 of MRP-228 also goes into detail on the requirements for length sizing accuracy. The standards in MRP-228 reflect the latest information in regulatory documents such as NUREG/CR-6943. *See* NUREG/CR-6943, A Study of Remote Visual Methods to Detect Cracking in Reactor Components (Oct. 2007) (ENT000666).

In addition, NUREG/CR-6943 addresses visual examinations, including remote visual examinations, and describes the characteristics of flaws to be detected in nuclear reactor components, in particular such critical characteristics as the crack opening displacement. For

[redacted]

example, Section 3 describes the types of equipment that are used for remote visual examinations, such as cameras, and discusses the equipment parameters that need to be controlled in order to detect certain kinds of flaws with any degree of certainty. The information contained in NUREG/CR-6943 and its predecessor document, NUREG/CR-6860, were valuable sources used by the industry to prepare MRP-228.

Finally, as previously noted, the NRC Staff has approved the techniques identified in MRP-227-A. *See* Responses to Questions 130 and 131, above.

8. The RVI AMP Includes Appropriate Acceptance Criteria, Corrective Actions, and Preventative Actions

a. Examination Acceptance Criteria

Q189. In general, what are the examination acceptance criteria in the RVI AMP?

A189. (RJD, NFA, TJG, RGL) MRP-227-A contains specific examination acceptance criteria in Section 5 that can be used to determine whether or not a particular examination result must be entered into the plant corrective action program. *See* NL-12-037, Attach. 2 at 52-57 (NYS000496); *see also* MRP-227-A at 5-1 to 5-23 (NRC000114B). The examination acceptance criteria also provide the trigger for expanding the examination program when unacceptable examination results are found. In general, however, MRP-227-A does not define specific engineering evaluations that can be used to justify continued operation in the presence of that examination result, because, it would be impractical to do so for all potential geometries and inspection findings. Section 6 of MRP-227-A describes the corrective action program options and a method to develop engineering evaluations. *See id*. at 6-1 to 6-11.

Specifically, the inspections required in Section 4 of MRP-227-A and relied upon in the IPEC RVI AMP and Inspection Plan are designed to detect all of the pertinent aging effects described above, with conservative examination acceptance criteria. Those acceptance criteria

[redacted]

will be carried forward into the program procedural documents, including the Pre-Inspection Engineering Packages prepared prior to each inspection.

b. Corrective Actions

Q190. Dr. Lahey criticizes the RVI AMP for allegedly failing to include "objective criteria . . . for corrective actions . . . ." Revised Lahey Testimony at 49 (NYS000482). What corrective actions are specified in the event the examination acceptance criteria are not met?

A190. (RJD, NFA, TJG, RGL) If examinations reveal conditions that do not meet the examination acceptance criteria set forth in Table 5-5 of the IPEC RVI Inspection Plan, NL-12-037, Attach. 2 at 52-27, tbl. 5-5 (NYS000496), then the discovery of the condition is entered into the IPEC corrective action program for resolution, *see id*., Attach. 1 at 8 and Attach. 2 at 27. The methodologies described in Section 6 of MRP-227-A (including recommendations for flaw depth sizing and crack growth determinations, and performing applicable limit load, linear elastic and elastic-plastic fracture analyses of relevant flaw indications), or other NRC-approved methods, will be used for corrective action program resolution. *See id*., Attach. 1 at 8 and Attach. 2 at 27; *see also* MRP-227-A at 6-1 to 6-11 (NRC000114B).

The corrective action program considers several potential disposition paths, including: (1) a more detailed examination to further characterize the relevant condition and demonstrate its acceptability for continued operation; (2) an engineering evaluation showing that the relevant condition is acceptable for continued operation, over some defined period up to the end of design life, under the full range of design-basis loadings; (3) repair of the affected component in order to eliminate the relevant condition and render the component acceptable for some defined period up to the end of design life; or (4) replacement of the affected component. *See* MRP-227-A at 6-1 (NRC000114B). Section 6 of MRP-227-A also provides information on methods that can be used for the engineering evaluation of detected conditions that exceed the examination acceptance criteria of Section 5. *See id*. at 6-1 to 6-11.

As MRP-227-A further explains, corrective actions following the detection of unacceptable conditions are provided for in each plant's corrective action program, as required by 10 C.F.R. Part 50, Appendix B, with additional guidance contained in ASME Code, Section XI. *See* NL-12-037, Attach. 1 at 7-8 (NYS000496). This is reflected in Program Element 7 (Corrective Action Program) of the IPEC RVI AMP, which states, in part, that the Entergy (10 C.F.R. Part 50, Appendix B) Quality Assurance Program, including relevant corrective action controls, applies to the RVI AMP. *Id*. It also states that the option of component repair and replacement of PWR RVIs is subject to the long-standing requirements of Entergy's ASME Code Section XI Program. *See id*. at 8.

Contrary to Dr. Lahey's demand, there is no further regulatory requirement for "objective criteria" for corrective actions. On the contrary, it would be impractical to establish pre-defined criteria in advance for all potential unsatisfactory examination results for all components. Instead, such issues are handled on a case-by-case basis through engineering evaluations conducted under Entergy's corrective action and quality assurance programs.

Q191. How are engineering evaluations developed as part of the corrective action process?

A191. (RJD, NFA, TJG, RGL) MRP-227-A Section 6 provides an overview of the methodologies to be used for the development of engineering evaluations. The evaluation process depends upon the loading applied to the component, assembly, or system. *See* MRP-227-A at 6-1 (NRC00114B). Typical loading information to be considered is provided in Section 6.1 and evaluation methodology options—ranging from the satisfaction of limit load

[redacted]

requirements for the internals assembly or component cross section, to the satisfaction of flaw stability requirements using either linear elastic fracture mechanics ("LEFM") or elastic-plastic fracture mechanics ("EPFM")—are also described. *See id* § 6.2.2. This process is based on extensive industry studies—notably MRP-210 (ENT000646)—which have shown that even when using lower bound fracture toughness properties for highly irradiated austenitic stainless steel PWR RVI components, the components are extremely flaw-tolerant.

The IPEC RVI AMP specifies that if an engineering evaluation is used to disposition an examination result that does not meet the examination acceptance criteria, then it shall be conducted in accordance with a NRC-approved evaluation methodology. *See* NL-12-037, Attach. 2 at 27 (NYS000496). As explained above, such engineering evaluations must consider appropriate design-basis loads.

Q192. How does the engineering evaluation process address the potential impacts of such CLB loads?

A192. (TJG, RGL, RJD, NFA, JRS) The loads that will be used for RVI engineering evaluation are the established CLB loads for IPEC, as set forth in response to Question 115, above. The MRP-227-A engineering evaluation process also specifies the safety factors required for accident loads. *See* MRP-227-A at 6-3 (NRC000114B); NL-12-037, Attach. 1 at 7-8 (NYS000496). In addition, as previously noted in the context of baffle-former bolts, WCAP-17096 provides standard methodologies for developing engineering evaluations for Westinghouse RVIs, and those methodologies specify the consideration of appropriate loading conditions. *See* WCAP-17096 (ENT000635).

[redacted]

Q193. What are the criteria set forth in the RVI AMP for inspections of "Expansion" components?

A193. (RJD, NFA, TJG, RGL) Consistent with the guidelines in MRP-227-A, inspections of "Expansion" components under the RVI AMP will take place if and when the periodic inspection of its associated "Primary" component detects aging effects that exceed the "Expansion Criteria" specified in the tables of Section 5 of MRP-227-A. *See* NL-12-037, Attach. 2 at 52-57, tbl. 5-5 (NYS000496); *see also* MRP-227-A at 5-15 to 5-20, tbl. 5-3 (NRC00114B). The NRC Staff SE for MRP-227-A states that, following the initial inspection of an "Expansion" component, subsequent examinations of that component shall be on a ten-year interval, and concurred with EPRI's technical justification for that interval. *See* SE for MRP-227-A at 21 (ENT000230); *see also* NL-12-037, Attach. 2 at 43-49 (NYS000496).

c. Inspections of the Lower Support Column Caps

Q194. On the topic of Primary and Expansion components, please explain the basis for Entergy's decision to link inspections of the core barrel girth weld (a "Primary" component under MRP-227-A, and the LSCCs (an "Expansion" component), with the core barrel girth weld as a leading indicator of potential IASCC.

A194. (RGL, NFA, RJD) The core barrel girth weld is a primary inspection component for cracking caused by IASCC because it is a leading indicator for this effect. *See* NL-14-093, Attach. 1 at 2 (NYS000505). During the NRC Staff's review of the IPEC RVI AMP, the NRC Staff asked about potential IASCC in the LSCCs. *See* SSER 2 at 3-43 (NYS000507); NL-13-122, Attach. 1 at 1-2 (NYS000502). In response, Entergy provided information showing that the stresses in the columns are too low to cause IASCC. *See* SSER 2 at 3-43 and -45 (NYS000507); NL-13-122, Attach. 1 at 2-4 (NYS000502).

[redacted]

Although the NRC Staff concurred with Entergy's determination and found that cracking of the LSCCs due to IASCC was unlikely, it nevertheless asked Entergy to modify the RVI AMP to link the LSCCs to an additional Primary component that is an appropriate predictor of IASCC and IE in the LSCCs. *See* SSER 2 at 3-45 (NYS000507); NL-14-093, Attach. 1 at 1 (NYS000505). Entergy identified the core barrel girth weld as an appropriate predictor. *See* SSER 2 at 3-45; NL-14-093, Attach. 1 at 1-4 (NYS000505). The NRC Staff found this linkage appropriate. SSER 2 at 3-47 (NYS000507).

Q195. Why was such a linkage appropriate?

A195. (RGL, NFA, RJD) As Entergy explained, and as the Staff found appropriate, although geometrical configurations are different, the LSCCs are appropriately linked as an expansion component linked to the core barrel girth weld because both are potentially subject to IASCC, but the conditions are expected to be less aggressive in the LSCCs. *See* SSER 2 at 3-47 (NYS000507). First, the LSCCs are fabricated from cast austenitic stainless steel, and they have a metallurgical microstructure similar to the weld. *See* NL-14-093, Attach. 1 at 3 (NYS000505), Attach. 1 at 3; SSER 2 at 3-46 (NYS000507). Second, the operating temperatures and predicted end-of-life fluence for the two components are similar. *See* NL-14-093, Attach. 1 at 3 (NYS000505); SSER 2 at 3-46 (NYS000507). And third, the lower core barrel cylinder girth weld is expected to experience larger tensile stresses than the LSCCs, which are primarily in compression. *See* NL-14-093, Attach. 1 at 3 (NYS000505); SSER 2 at 3-46 (NYS000507).

[redacted]

Q196. Dr. Lahey criticizes this linkage because, he speculates, "these components are very different and they may be exposed to different degradation mechanisms and shock loads." Revised Lahey Testimony at 60 (NYS000482). What is your response?

A196. (RGL, NFA, RJD) Dr. Lahey presents only unsupported speculation when he challenges Entergy's detailed plant-specific technical evaluation of the susceptibility of LSCCs to TE, IE, and IASCC in support of its RVI AMP. *See* SSER 2 at 3-40 to 3-47 (NYS000507). He only asserts that the core barrel girth weld "may" be exposed to different aging mechanisms and shock loads than the LSCCs.

As we have explained, the linkage of these two components is a conservative and fully justified choice. Furthermore, the magnitude or severity of the design basis accident loads does not contribute to the probability of in-service cracking and, therefore, has no impact on the determination of the leading indicators of cracking of the RVI components. We therefore disagree with Dr. Lahey's speculation to the contrary.

Q197. Dr. Lahey also relies on a recent statement by a member of the ACRS: "[t]he relationship between a lower core barrel weld and the tops of these columns is a bit of a stretch . . . [t]hey're totally different types of components, totally different loadings." ***See*** **Revised Lahey Testimony at 60 (NYS000482) (citing ACRS Plant License Renewal Subcommittee Transcript at 209 (Apr. 23, 2015) (NYS000526)).**

A197. (RGL, NFA, RJD) As we have explained in response to the previous question, a detailed consideration of the conditions and loadings on the lower core barrel cylinder girth weld reveals that it is more likely to experience IASCC than the LSCCs, so it is appropriate to link the two, with the weld as the Primary component and the LSCCs as the Expansion component.

Q198. On the topic of the LSCCs, Dr. Lahey states that Entergy "assumed" that only a limited number of the LSCCs contain flaws of significant size, and that the LSCCs "undergo a range of [unspecified] aging degradation mechanisms" that Entergy has allegedly failed to consider. Revised Lahey Testimony at 59 (NYS000482). Do you agree?

A198. (RGL, NFA, RJD) No. Entergy did not "assume" that only a limited number of the LSCCs contain flaws of significant size. Entergy reached this conclusion based on original component inspections using dye penetrant and radiography. *See* NL-13-122, Attach. 1 at 2 (NYS000502). These inspections concluded that all columns met applicable standards and were considered defect-free, with zero surface-breaking flaws. *See id*. Dr. Lahey does not address this information provided in Entergy's RAI response and referenced in SSER 2.

Q199. Has Westinghouse addressed this issue as it pertains to IPEC?

A199. (RGL, NFA, RJD) Yes. [redacted]

First, the quality controls used during fabrication, such as liquid penetrant inspection and radiography, limited the number and size of the flaws that could be present. This has been confirmed for IP2 and IP3 through a review of the original component inspection results. *See* NL-13-122, Attach. 1 at 2 (NYS000502).

Second, CASS lower support columns have relatively low ferrite content such that susceptibility to TE and combined effects of TE and IE are not expected to be dominant.

[redacted]

Entergy confirmed this to be a valid assumption for IP2 and IP3 as well, through a review of the CMTRs, which showed that the ferrite content in the lower support column caps is below 15%, eliminating concerns about TE. SSER 2 at 3-44 to -45 (NYS000507); *see also* NL-13-122, Attach. 1 at 3-4 (NYS000502).

Third, there are no credible mechanisms for flaw initiation or growth beyond what is permitted by the fabrication controls. [redacted]

[redacted]

Nevertheless, despite the significant margin inherent in the lower support structure and all of the reasons why Dr. Lahey's concerns about combined aging effects on the LSCCs lack foundation, Entergy has committed to link its inspections of the core barrel girth welds to the LSCCs, to address this potential issue.

Q200. Dr. Lahey relies on another quotation from the ACRS meeting, stating that, "Moreover, to have a failure due to a seismic event 'you don't even need to have a crack if these columns are really brittle . . . .'" Revised Lahey Testimony at 60 (quoting ACRS Plant License Renewal Subcommittee Transcript at 211) (NYS000526)). How do you respond?

[redacted]

A200. (RGL, NFA, RJD) There is no basis for an assumption that the LSCCs are "really brittle." The maximum fluence at the top of the most highly irradiated core support column is predicted to be approximately 4.11 x 10^{21} n/cm^2 (E>1 MeV). *See* NL14-093, Attach. 1 at 3 (NYS000505). This is enough fluence to cause some increase in yield stress and loss of fracture toughness. However, failure of CASS materials at this fluence would require significant plastic deformation, and mechanical failure of a support column would require loading beyond the irradiated yield stress.

[redacted]

Q201. Dr. Lahey also suggests that, while the NRC "recognized potential synergy" in aging effects on CASS components, it should have considered such effects on other RVI components.

A201. (TJG, RGL) As we have explained throughout this testimony, the guidelines in MRP-227-A were developed through a systematic evaluation of all RVIs and all potential aging effects on those RVI, including combined effects caused by multiple aging mechanisms. In A/LAI 7, the NRC Staff identified the need for *further* consideration of potential loss of fracture toughness in CASS materials due to the combined effects of IE and TE, and Entergy addressed that issue. *See* SSER 2 at 3-40 to 3-47 (NYS000507). This additional evaluation does not show that either the NRC Staff or Entergy have disregarded potential combined aging effects on other components, as Dr. Lahey suggests.

Moreover, as we explained in response to Question 172, above, CASS materials are known to be susceptible to aging effects from both TE and IE. Wrought stainless steel materials are primarily austenitic and are not susceptible to the same types of IE and TE as the CASS materials.

Q202. Dr. Lahey identifies certain other components that, he asserts, should have been evaluated for potential "synergistic" aging mechanisms: the "core baffles, baffle bolts, and formers." Revised Lahey Testimony at 61 (NYS000482). According to Dr. Lahey, these components are subject to more radiation fluence than the lower support columns. *See id*. Do you agree?

A202. (TJG, RGL, NFA, RJD) No. These components are not made of CASS material so they are not susceptible to TE. MRP-227-A and the RVI AMP identify IASCC (which, as its name implies, is actually the result of multiple underlying mechanisms itself) as the aging mechanism of concern for these components, and these components are all designated for Primary inspections under the RVI AMP and Inspection Plan. The RVI AMP and Inspection Plan also include inspections for the indirect effects of void swelling in former plates. *See* NL-12-037, Attach. 2 at 41 (NYS000496). Therefore, it is not necessary to conduct additional evaluations for potential "synergistic" aging mechanisms.

d. Entergy Is Undertaking Preventive Actions to Address the Effects of Aging on RVIs

Q203. What is Entergy doing to prevent, mitigate or minimize the effects of aging on RVIs?

A203. (TJG, JRS, RJD, ABC, NFA) Though the RVI AMP is focused on condition monitoring, as explained in the RVI AMP, Entergy has undertaken or will implement several types of preventive actions to manage the effects of aging on RVIs at IPEC.

First, the IPEC Water Chemistry Control—Primary and Secondary Program provides for preventive and mitigative action by maintaining primary water chemistry in accordance with EPRI guidelines to minimize the potential for SCC and IASCC. *See* NL-12-037, Attach. 1 at 5 (NYS000496). As stated in NUREG-1801, Revision 1, the purpose of the PWR water chemistry program is to control the chemistry of the primary water, including dissolved oxygen, limit the quantities of impurities and additives, provide for sampling and analysis on a frequent basis, and require corrective actions, as needed. *See* NUREG-1801, Revision 1 at XI M-10 to -13 (NYS00146C); LRA, Appendix B at B-137 to -138 (ENT00015B). This program minimizes contaminant concentrations, thereby mitigating loss of material due to general, crevice, and pitting corrosion and cracking caused by SCC. *See* NUREG-1801, Revision 1 at XI M-11 (NYS00146C). As stated in SSER 2:

> Under "Preventive Actions," LR-ISG-2011-14 states that MRP-227-A relies on PWR water-chemistry control to prevent or

[redacted]

> mitigate aging effects that can be induced by corrosive aging mechanisms (e.g., loss of material induced by general, pitting corrosion, crevice corrosion, or stress corrosion cracking or any of its forms [SCC, PWSCC, or IASCC]), and 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."

SSER 2 at 3-66 (NYS000507). The adequacy of the water chemistry program is unchallenged in this contention.

Second, as a further preventive measure, Entergy replaced the split pins at IP2 in 1995 and at IP3 in 2009. *See* NL-12-037, Attach. 2 at 21-22 (NYS000496). It also has committed to replace the IP2 split pins again in 2016. *See id.*; SSER 2 at A-15 (NYS000507).

Third, to address A/LAI 8 in the Safety Evaluation for MRP-227-A, Entergy will use the Fatigue Monitoring Program ("FMP") to manage the effects of fatigue on specified RVI components. *See id.* at 3-51; *see also* Entergy's NYS-26B/RK-TC-1B Testimony at Q119 (ENT000679). The FMP, in turn, tracks plant transients and cycles, thereby assuring that fatigue usage from actual plant transients does not exceed ASME Code design limits. *See* SER at 3-78 to 3-79 (NYS000326B). This tracking of transients is effectively a preventive measure.

Finally, Entergy has implemented neutron flux reduction measures at IPEC as preventive measures to minimize the neutron fluence on the RPV which in turn will minimize radiation-induced aging effects at high fluence locations of the RVIs. *See* NL-12-037, Attach. 1 at 5 (NYS000496).

Q204. When Dr. Lahey states that The IPEC RVI AMP "does not include" any preventive actions, Revised Lahey Testimony at 53 (NYS000482), do you agree?

A204. (ABC, NFA, JRS, RJD, TJG) No. Dr. Lahey highlights a truncated quote from the RVI AMP, but ignores the appropriate context in the very next sentence (explaining the preventive aspect of the water chemistry control program) and the very next paragraph (explaining the preventive aspect of low-leakage core loading). *See* NL-12-037, Attach. 1 at 5 (NYS000496). In addition, as we have just explained, Entergy is undertaking several additional preventive actions to address the effects of aging on RVIs.

9. Environmentally-Assisted Fatigue

Q205. What TLAAs involve the RVIs?

A205. (ABC, JRS, TJG, RJD, NFA) In accordance with 10 C.F.R. § 54.21(c), and as specified in A/LAI 8, CLB TLAAs must be evaluated. *See* 10 C.F.R. § 54.21(c); SE for MRP-227-A at 26, 34 (ENT000230). At IPEC, these TLAAs involve cumulative usage fatigue ("CUF") analyses. Specifically, Section 4.3.1.2 of the LRA lists the CLB cumulative usage factor analyses for the IP2 and IP3 RVI that conform to the definition of a TLAA in 10 C.F.R. § 54.3. *See* LRA at 4.3-11, tbl.4.3-5 and 4.3-12, tbl.4.3-6 (ENT00015B).

Q206. How does Entergy address these TLAAs for RVI components?

A206. (RGL, RJD, JRS, NFA, MAG) To address A/LAI 8 in the SE for MRP-227-A, for RVI components with TLAA CUF analyses, Entergy will manage the effects of aging due to fatigue through the Fatigue Monitoring Program. *See* SSER 2 at 3-51 to 3-53 (NYS000507). This includes a commitment (Commitment 49) under which Entergy recalculated each of the limiting CUFs for RVIs identified in the LRA, including consideration of reactor coolant environmental effects, and will take corrective actions as specified in the Fatigue Monitoring Program prior to the environmentally adjusted CUF ("CUF_{en}") reaching 1.0. *See id.*; *see also id.* at A-15. This commitment and the EAF evaluations of RVIs are further discussed in Entergy's NYS-26B/RK-TC-1B Testimony (ENT000679).

Q207. Does Dr. Lahey critique EAF evaluations for RVIs?

A207. (NFA, ABC, JRS, RJD, RGL, MAG) Yes. Dr. Lahey's critiques of fatigue calculations are raised in this contention and in contention NYS-26B/RK-TC-1B. The State has submitted identical testimony from Dr. Lahey on both contentions, so the concerns are indistinguishable. Entergy addresses all EAF evaluation issues, including EAF evaluations of RVIs, in its testimony on NYS-26B/RK-TC-1B, which is incorporated by reference here. *See* Entergy's NYS-26B/RK-TC-1B Testimony § V.E.2 (ENT000679). In sum, we explain that EAF evaluations have been completed for RVIs and demonstrate that the CUF_{en}s for RVI components are all below 1.0, as required, when projected to the end of the PEO. *Id*. We also explain it is not necessary to apply any additional correction factors to account for the potential effects of IE on fatigue life. *Id*.

Q208. How does the RVI AMP itself address potential fatigue of RVI components, including EAF?

A208. (RGL, RJD, JRS, NFA, MAG) As explained in response to Question 144, above, fatigue is one of the eight age-related degradation mechanisms evaluated during the development of the guidelines in MRP-227-A. As a result, as shown in Table 1, above, the RVI AMP includes inspections intended to identify potential cracking caused by fatigue in susceptible RVI components. These inspection activities are in addition to, not in lieu of, the review of EAF for RVI components under the FMP.

10. Operating Experience

Q209. Please describe the reporting process for inspections conducted pursuant to the IPEC RVI AMP.

A209. (TJG, JRS, RJD, ABC, NFA) As explained in section 4.4.5 of the RVI Inspection Plan, "[a] summary report of all inspections and monitoring, items requiring evaluation, and new repairs shall be provided to the MRP Program Manager within 120 days of the completion of an outage during which PWR internals within the scope of MRP-227-A are examined." NL-12-037, Attach. 2 at 27 (NYS000496). As further explained in MRP-227-A:

[redacted]

> This summary of the results will be compiled into an overall industry report which will track industry progress, aid in evaluation of significant issues, identification of fleet trends and determination of any needed revisions to these guidelines. The industry report will be updated biennially for the benefit of the fleet, the regulator, the PWROG and other industry stakeholders. This biennial report will serve to assist in review of operating experience, and required monitoring and trending for aging management programs established by the industry. In order to ensure completeness and consistency of reporting, the MRP will provide a template listing the requested information.

MRP-227-A at 7-3 (NRC000114C). Consistent with this process, the EPRI MRP provides periodic summary reports on PWR RVI inspection operating experience to the NRC, most recently in November 2014. *See generally* EPRI, MRP-219, Rev. 10, Materials Reliability Program: Inspection Data Survey Report (Nov. 2014) (ENT000668). Finally, the results of the IPEC inspections will be available to the NRC Staff for on-site inspection.

Q210. Will the RVI AMP be further refined in the future, based on operating experience and research on the effects of aging on RVIs?

A210. (ABC, NFA, JRS, TJG, RJD, RGL) We expect so. The RVI AMP, like any other AMP, could be refined during the PEO, as necessary, to address new industry operating experience as it becomes available. This process reveals no deficiency in the current AMP—indeed, this process is essential for any AMP to provide the requisite reasonable assurance under the license renewal regulations.

In addition, as previously noted, Appendix A to MRP-227-A contains a summary of PWR RVI operating experience, which we expect to be updated in future revisions to MRP-227-A. The EPRI MRP Program and the PWROG also compile and periodically disseminate updates on recent operating experience. In addition, EPRI research, NRC-funded research, and industry programs designed to obtain data and develop improved modeling and aging prediction methods

[redacted]

for internals is ongoing. The results from these efforts will further inform the RVI AMP at IPEC and other plants throughout the PEO.

In summary, consistent with the operating experience element of the RVI AMP and Commitment 40, Entergy will continue to review domestic and international operating experience during the PEO. *See* NL-12-037, Attach. 1 at 9 (NYS000496); SSER 2 at A-13 (NYS000507). The IPEC RVI AMP will therefore be reinforced by ongoing research activities and the sharing of operating experience.

Q211. You mentioned the operating experience review in Appendix A to MRP-227-A. Does that industry-wide operating experience review contain any key conclusions regarding the ability of PWR RVIs to tolerate the effects of aging?

A211. (ABC, NFA, JRS, TJG, RJD, RGL) Yes. We previously explained that, in general, the design of PWR RVIs is robust and the materials are resilient. Appendix A to MRP-227-A states that "[c]ommercial PWR vessel internals in the United States have experienced safe, relatively trouble-free operation. There have been no instances to date in which PWRs in the U.S. have posed a threat to public safety as a result of PWR internals material aging degradation." MRP-227-A, App. A at A-1 (NRC000114C) (emphasis added). The industry has engaged in a decade-long effort to evaluate aging management of PWR RVIs, implement plant-specific AMPs for aging management of internals, develop a detailed RVI inspection program that has been approved by the NRC, and it continues to collect and share relevant inspection results and operating experience for improved reliability.

[redacted]

B. <u>Entergy's Aging Management Activities for RPVs</u>

Q212. How does the LRA address aging management of the IP2 and IP3 RPVs for the effects of neutron irradiation embrittlement?

A212. (ABC, NFA, JRS) LRA Section 3.1.2.2.3 identifies Entergy's TLAA evaluations and associated AMP for managing the loss of fracture toughness due to neutron irradiation embrittlement of the IPEC RPVs. *See* LRA at 3.1-7 (ENT00015A). This section of the LRA explains how IPEC's Reactor Vessel Surveillance Program manages reduction in fracture toughness due to neutron embrittlement of RPV beltline materials. *Id*. It states that the program evaluates radiation damage based on pre-irradiation and post-irradiation of Charpy V-notch and tensile specimens from the most limiting material used to fabricate the RPV in the reactor core region (*i.e.*, the beltline region). *Id*. Based on the requirements of 10 C.F.R. Part 50, Appendix H, within one year after withdrawal of each capsule under the schedule set forth in the UFSAR and approved by the NRC, Entergy will submit a technical report to the NRC that will include, among other things, the results of the fracture toughness tests conducted. *See* 10 C.F.R. Part 50 Appendix H; LRA, Appendix B at B-111 to -112 (ENT00015B). The RPV surveillance AMP is described in LRA Appendix B, Section B.1.32. *Id.*.

Q213. What TLAAs involve neutron embrittlement of the RPVs?

A213. (JRS, NFA, ABC) The LRA describes the evaluation of four TLAAs related to neutron irradiation embrittlement of the RPVs: (1) the Charpy USE TLAA, described in Section 4.2.2 of the LRA; (2) the pressure-temperature ("P-T") limits TLAA, described in Section 4.2.3; (3) the low temperature overpressure protection ("LTOP") TLAA, described in Section 4.2.4; and (4) the PTS TLAA, described in Section 4.2.5 of the LRA (ENT00015B). But NYS has never mentioned the LTOP TLAA or the P-T limits TLAA in its filings on this contention, so these TLAA are not further discussed in this testimony.

[redacted]

Q214. Do the State or Dr. Lahey challenge any of this information in their Statement of Position or Prefiled Testimony?

A214. (JRS, NFA, ABC) No, not specifically. Nevertheless, as explained above, we are summarizing this information from the LRA and the NRC Staff's review of the LRA, to ensure the record is complete.

1. Reactor Vessel Surveillance Program

Q215. Please provide a summary of IPEC's Reactor Vessel Surveillance Program.

A215. (ABC, NFA, JRS) IPEC's Reactor Vessel Surveillance Program is an established program developed principally upon ASTM E-185, "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels," as required by 10 C.F.R. Part 50, Appendix H. LRA, Appendix B-111 (citing NUREG-1801, Revision 1 (NYS00146C, at XI M-102 to M-104 (AMP XI.M31))).

The Reactor Vessel Surveillance Program, in conjunction with the RPV TLAA evaluations identified above, ensure that reduction in fracture toughness of RPV beltline materials is managed in order to maintain the pressure boundary function of the RPV through the PEO. LRA, Appendix B-111 (ENT00015B). As discussed above, in order to validate calculations of neutron embrittlement, IP2 and IP3 use material surveillance test specimens that are located between the core and the RPV wall directly opposite the center portion of the core. Irradiation of the specimens is higher than irradiation of the RPV because the specimens are located closer to the core than the vessel wall itself. Upon removal, the specimens are tested to determine the shift in the reference temperature of the RPV material. The shift results are then used in conjunction with RG 1.99, Rev. 2 to perform embrittlement projections for the RPV beltline materials. *See* Regulatory Guide 1.99, Radiation Embrittlement of Reactor Vessel Materials, Revision 2 (May 1988) ("RG 1.99, Rev. 2") (ENT000669).

Q216. In the IPEC LRA, did Entergy initially identify a necessary enhancement to the Reactor Vessel Surveillance Program?

A216. (ABC, NFA, JRS) Yes. Upon submittal of the LRA, Entergy committed to implement the following enhancement to IPEC's Reactor Vessel Surveillance Program in order to meet the recommendations of NUREG-1801, Revision 1, Section XI.M31:

> The specimen capsule withdrawal schedules will be revised to draw and test a standby capsule to cover the peak reactor vessel fluence expected through the end of the period of extended operation.
>
> Appropriate procedures will be revised to require that tested and untested specimens from all capsules pulled from the reactor vessel are maintained in storage.

LRA Appendix A-34 (ENT00015B); *see also* SER 3-113 to 3-114 (NYS00326C). Consistent with Commitment 22 in the LRA, Entergy submitted the change to the capsule withdrawal schedule for IP2 and IP3 on August 12, 2013, and November 5, 2014, respectively. *See* NL-13-106, Letter from J. Ventosa, Entergy, to NRC Document Control Desk, "Proposed Revision to Reactor Vessel Surveillance Capsule Withdrawal Schedule Per 10 CFR 50 Appendix H" (Aug. 12, 2013) (ENT000670) (for IP2); NL-14-129, Letter from J. Ventosa, Entergy, to NRC Document Control Desk, "Proposed Revision to Reactor Vessel Surveillance Capsule Withdrawal Schedule Per 10 CFR 50 Appendix H" (Nov. 5, 2014) (ENT000671) (for IP3).

The NRC approved the revisions for IP2 on January 14, 2014, and for IP3 on March 4, 2015. *See* Letter from B. Beasley, NRC, to Vice President, Operations, Entergy, "Indian Point Nuclear Generating Unit No. 2 – Safety Evaluation re: Revision to the Reactor Vessel Surveillance Capsule Withdrawal Schedule per 10 CFR 50 Appendix H (TAC No. MF2558)" (Jan. 14, 2014) (ENT000672) (approving a change to the IP2 surveillance capsule withdrawal schedule to cover the additional twenty year PEO); Letter from B. Beasley, NRC, to Vice

[redacted]

President, Operations, Entergy, "Indian Point Nuclear Generating Unit No. 3 – Safety Evaluation re: Revision to the Reactor Vessel Surveillance Capsule Withdrawal Schedule per 10 CFR 50 Appendix H (TAC No. MF5148)" (Mar. 4, 2015) (ENT000673) (approving a change to the IP3 surveillance capsule withdrawal schedule to cover the additional twenty year PEO).

Q217. What conclusions did the NRC Staff reach with respect to IPEC's Reactor Vessel Surveillance Program?

A217. (ABC, NFA, JRS) In the original SER, the NRC Staff reached the following conclusion:

> On the basis of its review of the applicant's Reactor Vessel Surveillance Program, the staff determined that those program elements, for which the applicant claimed consistency with the GALL Report, are consistent. Also, the staff reviewed the enhancement and confirmed that its implementation would make the existing program consistent with the GALL Report AMP to which it was compared. The staff concluded that the applicant has demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(a)(3). The staff also reviewed the UFSAR supplement for this program and concluded that it provides an adequate summary description of the program, as required by 10 CFR 54.21(d).

SER at 3-114 to 3-115 (NYS000326C).

Q218. Do Dr. Lahey or NYS challenge this conclusion?

A218. (ABC, NFA, JRS) No.

2. Charpy Upper-Shelf Energy TLAA Evaluations

Q219. What LRA section(s) address the TLAA for Charpy USE for the RPVs? Please explain the content of these sections.

A219. (ABC, NFA, JRS) LRA Section 4.2.2 summarizes Entergy's evaluation of the Charpy USE analysis based on a maximum projected 48 effective full-power years ("EFPY")

[redacted]

beltline fluence of IP2 and IP3. Forty-eight EFPY are projected in the LRA for the end of the PEO (60 calendar years) of IP2 and IP3 based on actual capacity factors from the start of commercial operation until 2005, along with a conservatively estimated average capacity factor of 100% from 2005 until the end of the PEO. *See* NL-08-092, Attach. 3 at 1 (ENT000193).

Specific to IP2, two RPV plates had projected USE levels below 50 ft-lbs at 48 EFPY. LRA at 4.2-4 (showing one intermediate shell plate (B2002-3) with projected 48 EFPY USE of 47.4 ft-lbs and one lower shell plate (B2003-1) with 49.8 ft-lbs). For IP3, one lower shell plate fell below 50 ft-lbs. *See* NL-08-014, Letter from Fred R. Dacimo, Entergy, to NRC Document Control Desk, "Clarifications to Reactor Vessel Surveillance Program and Neutron Embrittlement Time-Limited Aging Analyses and Audit Item #105; and Revision to License Renewal Regulatory Commitment List," Attach. 1 at 7-8 (Jan. 17, 2008) ("NL-08-014") (ENT000674) (plate B2803-3, projected to be 49.8 ft-lbs at the end of the PEO). The projected USE for all remaining plate and weld beltline materials for IP2 and IP3 is above 50 ft-lbs at 48 EFPY. LRA at 4.2-4 to 4.2-5 For the plate and weld beltline materials that exceed 50 ft-lbs, according to 10 C.F.R. Part 50, Appendix G, Section IV.A.1, no equivalent margins analysis ("EMA") is required.

For those materials with USE values below 50 ft-lbs, Entergy credited an NRC-approved EMA. WCAP-13587, Rev. 1 demonstrated that the minimum acceptable USE for RV plate material in 4-loop Westinghouse PWR plants, such as IP2, is 43 ft-lbs. *See* LRA at 4.2-3, Appendix A-39. As discussed below, the analysis in WCAP-13587, Rev. 1 applies to IP2 and IP3.

LRA Section 4.2.2 further stated that in the NRC safety assessment for WCAP-13587, Rev. 1, the NRC Staff concluded that WCAP-13587 demonstrated margins of safety equivalent to those of the relevant ASME code. *See* LRA at 4.2-3; *see also* Letter from M. Hodges, NRC, to W. Rasin, Nuclear Management and Resources Council, "Safety Assessment of Report WCAP-13587, Revision 1, 'Reactor Vessel Upper Shelf Energy Bounding Evaluation for Westinghouse Pressurized Water Reactors,' September 1993," Encl. at 9-10 (Apr. 21, 1994) (ENT000675). Consistent with this conclusion, the NRC, in an FSER for a similar PWR, found the use of the WCAP-13587 EMA acceptable for beltline and forging materials with USE values below 50 ft-lbs. *See* LRA at 4.2-3 (citing NUREG-1785, Safety Evaluation Report Related to the License Renewal of H.B. Robinson Steam Electric Plant, Unit 2 at 4-4, 4-8, 4-12 (Mar. 2004) (ENT000676). As B2002-3, B2003-1, and B2803-3 are above 43 ft-lbs, Entergy determined that the plates are acceptable, based on the EMA approved by the NRC. *See* LRA at 4.2-3 Accordingly, the LRA concluded that the TLAA for Charpy USE is projected through the PEO in accordance with 10 C.F.R. § 54.21(c)(1)(ii). *See id*.

Entergy also demonstrated in a June 11, 2008 letter to the NRC, that its EMA for the IP2 and IP3 RPVs is equivalent to the methodology in RG 1.161 and Appendix K to Section XI of the ASME Code. *See* NL-08-092, encl. 3 (ENT000193). The NRC Staff confirmed that the methods and formulas cited in RG 1.161 and Appendix K to Section XI of the ASME Code are the same as those in WCAP-13587, Rev. 1, except that the formulas for calculating the stress intensity factors from radial thermal gradients in WCAP-13587, Rev. 1 result in higher values, and are therefore more conservative. *See* SER at 4-11 (NYS00326E).

Q220. What did the NRC Staff conclude, concerning the applicability of WCAP-13587, Rev. 1 to IP2 and IP3 RPVs?

A220. (ABC, NFA, JRS) Consistent with Entergy's LRA and subsequent RAI responses, the NRC Staff concluded that the minimum acceptable value of 43 ft-lbs in WCAP-

[redacted]

13587, Rev. 1 is applicable to the IP2 and IP3 RPVs. *See* SER at 4-11 to 4-12 (NYS00326E). Because the USE values satisfy the minimum acceptable values in WCAP-13587, Rev. 1, they also satisfy RG 1.161 and Appendix K to Section XI of the ASME Code, along with the margins of safety against fracture equivalent to those required by Appendix G to Section XI of the ASME Code. *See id*. Consequently, the NRC Staff concluded that Entergy's USE values satisfy the requirements of 10 C.F.R. Part 50, Appendix G, Section IV.A.1 through the PEO. *See id*. at 4-12.

Q221. Do Dr. Lahey or NYS challenge this conclusion?

A221. (ABC, NFA, JRS) No.

3. Pressurized Thermal Shock TLAA

Q222. What LRA section(s) address the TLAA for PTS limits?

A222. (ABC, NFA, JRS) Section 4.2.5 of the LRA discusses the adjusted reference temperatures calculated pursuant to 10 C.F.R. § 50.61(c) and compares these values to screening criteria in 10 C.F.R. § 50.61(b). All projected 48 EFPY RT_{PTS} values for IP2 are within the screening criteria (270°F), and thus the LRA concludes that the TLAA for PTS is projected through the PEO in accordance with 10 C.F.R. § 54.21(c)(1)(ii).

The LRA indicated that one lower shell plate (B2803-3) in IP3 would exceed the screening criteria (270°F) by 9.9 °F approximately 9 years after entering the PEO. *See* LRA at 4.2-9; *see also* NL-07-140, Letter from F. Dacimo, Entergy, to NRC Document Control Desk, "Reply to Request for Additional Information Regarding License Renewal Application," Attach. 1 at 8 (Nov. 28, 2007) (ENT000677). The PTS evaluation tables were revised in 2008 in response to an RAI, but Entergy came to the same conclusions for IP2 and IP3, except plate B2803-3 now exceeds the screening criteria by 9.5°F (not 9.9°F). *See* NL-08-014, Attach. 1 at 8 (ENT000674).

[redacted]

Although the use of flux reduction methods utilizing a low-low leakage loading plan and the installation of flux suppressors used since 1986 have been successful in reducing neutron fluence, Entergy concluded that these methods would not prevent the plate from reaching the screening criterion during the PEO. LRA at 4.2-7 (ENT00015B). Accordingly, Entergy provided the following commitment:

> As required by 10 CFR 50.61(b)(4), IP3 will submit a plant-specific safety analysis for plate B2803-3 to the NRC three years prior to reaching the RT_{PTS} screening criterion. Alternatively, the site may choose to implement the revised [PTS rule] when approved.

SER at 4.2-17 (NYS000326E). Therefore, LRA Section 4.2.5 concluded that the TLAA for PTS for the IPEC RPVs will, pursuant to 10 C.F.R. § 54.21(c)(1)(iii), be adequately managed for the PEO.

The submittal of a plant-specific analysis, along with the 10 C.F.R. § 50.61 requirements or the alternative PTS requirements of 10 C.F.R. § 50.61a, ensures that the RPV is not operated in an unsafe condition during the PEO. During this time, the RPV can be operated only for that service period within which the RT_{PTS} values are below the screening criteria or, alternatively, if the NRC specifically approves other corrective actions (*e.g.*, modifications to equipment, systems, plant operation or thermal annealing treatment) for IPEC to take in order to demonstrate that RPV integrity will be maintained. *See* 10 C.F.R. § 50.61(b)(4)-(7). Moreover, submittal of the plant-specific analysis at least three years prior to exceeding the screening criteria—as explicitly specified in 10 C.F.R. § 50.61(b)(4)—allows for consideration of additional surveillance data, if any, collected before reaching the screening criteria.

[redacted]

Q223. What conclusions did the NRC Staff reach related to IPEC's PTS screening criteria values?

A223. (ABC, NFA, JRS) The NRC Staff accepted Entergy's commitment to submit plant-specific safety analysis for plate B2803-3 to the NRC three years prior to reaching the RT_{PTS} screening criterion, concluding that it will ensure that the PTS-related aging effects will be properly managed during the period of extended operation, pursuant to 10 C.F.R. § 54.21(c)(1)(iii). *See* SER at 4-17 (NYS000326E).

Q224. Do Dr. Lahey or NYS challenge this conclusion?

A224. (ABC, NFA, JRS) No. Dr. Lahey discusses this conclusion on page 43 of his revised testimony (NYS000482), but he does not allege any deficiency in the IPEC LRA or its associated analyses.

VIII. CONCLUSION

Q225. Please summarize your testimony and the bases for your conclusion regarding NYS-25.

A225. (NFA, ABC, JRS, RJD, TJG, RGL, MAG) NYS-25 lacks merit because IPEC's RVI AMP adequately addresses the aging effects caused by neutron irradiation embrittlement and other aging mechanisms, alone or in combination, on the RVIs consistent with NRC regulations and guidance. The AMP is based on the extensive industry program that culminated in the NRC-approved MRP-227-A. Contrary to Dr. Lahey's opinions, the IPEC RVI AMP is based on a state-of-the-art, systematic evaluation of known and potential degradation mechanisms, the resulting aging effects, and consequences of those effects for RVIs. The conditions addressed in the development of the MRP-227-A guidelines include the full range of design basis and CLB loads. The IPEC RVI AMP includes appropriate preventive and corrective actions, covers an appropriate scope of components, provides appropriate acceptance criteria, and appropriately inspects RVI components in a timely manner. And finally, as demonstrated in Entergy's testimony on the metal fatigue contention, the EAF evaluations prepared in support of the IPEC LRA, including EAF evaluations of RVI components, are fully documented, conservative engineering analyses that support a finding that the effects of fatigue, including the effects of reactor water environment, will be adequately managed. For all these reasons, the RVI AMP provides reasonable assurance that the intended functions of IPEC RVIs will be maintained consistent with the CLB throughout the PEO.

Q226. Does this conclude your testimony?

A226. (NFA, ABC, JRS, RJD, TJG, RGL, MAG) Yes.

Q227. In accordance with 28 U.S.C. § 1746, do you state under penalty of perjury that the foregoing testimony is true and correct?

A227. (NFA, ABC, JRS, RJD, TJG, RGL, MAG) Yes.

Executed in accord with 10 C.F.R. § 2.304(d)
Nelson F. Azevedo
Supervisor of Code Programs
Entergy Nuclear Generation Co.
295 Broadway, Suite 1
Buchanan, NY 10511
914-254-6775
nazeved@entergy.com

Executed in accord with 10 C.F.R. § 2.304(d)
Robert J. Dolansky
Code Programs Engineer
Entergy Nuclear Generation Co.
295 Broadway, Suite 1
Buchanan, NY 10511
914-254-6737
rdolans@entergy.com

[redacted]

Executed in accord with 10 C.F.R. § 2.304(d)
Alan B. Cox
Independent Consultant
Entergy License Renewal Services
1448 SR 333
N-GSB-45
Russellville, AR 72802
479-858-3173
acox@entergy.com

Executed in accord with 10 C.F.R. § 2.304(d)
Jack R. Strosnider
Senior Nuclear Safety Consultant
Talisman International, LLC
9712 Breckenridge Pl.
Montgomery Village, MD 20886
202-471-4244
jstrosnider@talisman-intl.com

Executed in accord with 10 C.F.R. § 2.304(d)
Randy G. Lott
Consulting Engineer
Westinghouse Electric Company LLC
Nuclear Services
1000 Westinghouse Drive
Cranberry Township, PA 16066
(412) 374-4157
LottRG@westinghouse.com

Executed in accord with 10 C.F.R. § 2.304(d)
Mark A. Gray
Principal Engineer
Westinghouse Electric Company LLC
Nuclear Services
1000 Westinghouse Drive
Cranberry Township, PA 16066
(412) 374-4602
GrayMA@westinghouse.com

[redacted]

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

Timothy J. Griesbach
Senior Associate
Structural Integrity Associates, Inc.
5215 Hellyer Ave., Suite 210
San Jose, CA 95138
(408) 833-7350
TGriesbach@structint.com

August 10, 2015

DB1/ 84085605