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

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)

[redacted]

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TABLE OF ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
AMP	aging management program
AMR	aging management review
ASME	American Society of Mechanical Engineers
ANO	Arkansas Nuclear One
B.S.	Bachelor of Science
CLB	current licensing basis
COA	crack-opening area
CUF	cumulative usage factor
CUF_{en}	environmentally corrected CUF
EAF	environmentally-assisted fatigue
EDF	Électricité de France
EPRI	Electric Power Research Institute
FAC	Flow-accelerated corrosion
F_{en}	environmental correction factor
FMP	Fatigue Monitoring Program
GENE	GE Nuclear Energy
GSI	Generic Safety Issue
HWC	Hydrogen water chemistry
IGSCC	Intergranular stress corrosion cracking
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
LOCA	loss-of-coolant accident
LRA	License Renewal Application
LRIP	License Renewal Inspection Program
LWR	light water reactor
M.B.A.	Master of Business Administration
MRP	Materials Reliability Program
M.S.	Master of Science
NACE	National Association of Corrosion Engineers
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NRR	NRC Office of Nuclear Reactor Regulation
NU	Northeast Utilities
NYPA	New York Power Authority
NYS	New York State
OIG	Office of Inspector General
PEO	period of extended operation

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TABLE OF ABBREVIATIONS
(continued)

PTS	pressurized thermal shock
PWR	pressurized water reactor
PNNL	Pacific Northwest National Laboratory
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, Troy, New York
RT_{NDT}	reference temperature for nil-ductility transition
RVI	reactor vessel internals
RVI Program	Reactor Vessel Internals Program
SBO	Station blackout
SER	NUREG-1930, Safety Evaluation Report
SGMP	Steam generator management program
SI	Structural Integrity Associates
SONGS	San Onofre Nuclear Generating Station
SSCs	systems, structures, and components
SSER	Supplement to the Safety Evaluation Report
SRP-LR	NUREG-1800, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants
TLAA	time-limited aging analysis
UFSAR	Updated Final Safety Analysis Report
WESTEMS™	Westinghouse computer program for the evaluation of fatigue for pressurizer and surge line locations
WOG	Westinghouse Owners Group

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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 reactor vessel internals ("RVI") contention (NYS-25). 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-38/RK-TC-5 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 RVIs, steam generator components, and other primary plant components.

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. I became a Manager at NU in 1998, managing 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. Prior to 1998, I was an Engineer for more than ten years and an Engineering Supervisor for another five years at NU.

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Since 2001, I have managed the IPEC engineering section responsible for implementing 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") Management Committee 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-38/RK-TC-5?

A5. (NFA) Yes. I am familiar with the technical issues related to the management of the effects of PWSCC in steam generator components, the effects of fatigue in primary plant components, and the effects of aging on the RVIs. I am also familiar with 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-38/RK-TC-5, I am familiar with Sections 4.3.1 (Class 1 Fatigue); 4.3.2 (Non-Class 1 Fatigue); 4.3.3 (Effects of Reactor Water Environment on Fatigue Life); B.1.12 (the Fatigue Monitoring Program ("FMP")), B.1.18 (ISI AMP), B.1.35 (Steam Generator Integrity Program), B.1.38 (Thermal Aging and Neutron Irradiation Embrittlement of CASS AMP), B.1.41 (Water Chemistry Control Program), and B.1.42 (RVI AMP) of the LRA. As a result, I am familiar with Entergy's plans to manage the potential for PWSCC in steam

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generator components, fatigue in primary plant components, and to manage the effects of aging on RVIs 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-38/RK-TC-5.

A6. (NFA) In my capacity as Supervisor of Code Programs at IPEC, I have been responsible for the IP2 FMP since 2001. I also supervise the IPEC engineering staff responsible for implementing the IP2 and IP3 FMPs. I reviewed draft versions of the Westinghouse Electric Company LLC ("Westinghouse") environmentally-assisted fatigue ("EAF") evaluations for IP2 and IP3 discussed below, and directly interfaced with Westinghouse personnel in resolving technical comments on those drafts before their final approval by Entergy. During my career, I have performed pipe stress analyses, finite element analysis of large components, ASME Code Section XI flaw evaluations, and ASME Code Section III, Class 1 fatigue analyses. Accordingly, I am very familiar with the IPEC FMP, including the description of that program in the LRA; relevant Nuclear Regulatory Commission ("NRC") requirements and guidance; and applicable industry codes. 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.

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, the effects of aging on RVIs, and the management of potential PWSCC in steam generator components. In preparing my testimony, I also reviewed the parties' pleadings on NYS-38/RK-TC-5, the Licensing Board Memorandum and Order (Admitting New Contention NYS-38/RK-

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TC-5) at 11 (Nov. 10, 2011) (unpublished) ("Order Admitting NYS-38/RK-TC-5"), the 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 Amending NYS-38/RK-TC-5"), and the exhibits submitted by NYS and Riverkeeper 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 RVI contention (NYS-25). My role regarding NYS-38/RK-TC-5 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 ("ISI"), 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.

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-38/RK-TC-5?

A11. (RJD) Yes. I am familiar with the technical issues related to the management of the effects of PWSCC in steam generator components and the effects of aging on the RVIs. I am also familiar with 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-38/RK-TC-5, I am familiar with Sections B.1.18 (ISI AMP), B.1.35 (Steam Generator Integrity Program), B.1.38 (Thermal Aging and Neutron Irradiation Embrittlement of CASS AMP), and B.1.42 (RVI AMP) of the LRA. As a result, I am familiar with Entergy's plans to manage the potential for PWSCC in steam generator components and to manage the effects of aging on RVIs 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-38/RK-TC-5.

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 Subcommittee meetings for the IPEC LRA held in 2015.

More specifically, I have been involved in ASME Code-based inspections and evaluations of components and piping at IPEC since 1989. I maintain 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 Electric Power Research Institute ("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.

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, the effects of aging on RVIs, and the management of potential PWSCC in steam generator components. In preparing my testimony, I also reviewed the parties' pleadings on NYS-38/RK-TC-5, the Board's Order Admitting NYS-38/RK-TC-5, the Board's Order Amending NYS-

38/RK-TC-5, and the exhibits submitted by NYS and Riverkeeper 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 RVI contention (NYS-25). 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-38/RK-TC-5 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 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-38/RK-TC-5?

A17. (ABC) Yes. I am familiar with the technical issues related to the management of the effects of PWSCC in steam generator components, the effects of fatigue in primary plant components, and the effects of aging on the RVIs. I am also familiar with the development, and subsequent revision, of the portions of the IPEC LRA that address such issues, including the relevant Entergy AMPs.

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In particular, as relevant to NYS-38/RK-TC-5, I am familiar with Sections 4.3.1 (Class 1 Fatigue); 4.3.2 (Non-Class 1 Fatigue); 4.3.3 (Effects of Reactor Water Environment on Fatigue Life); B.1.12 (the FMP), B.1.18 (ISI AMP), B.1.35 (Steam Generator Integrity Program), B.1.38 (Thermal Aging and Neutron Irradiation Embrittlement of CASS AMP), B.1.41 (Water Chemistry Control Program), and B.1.42 (RVI AMP) of the LRA. As a result, I am familiar with Entergy's plans to manage the potential for PWSCC in steam generator components, fatigue in primary plant components, and to manage the effects of aging on RVIs 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-38/RK-TC-5.

A18. (ABC) As Technical Manager, 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 the ACRS License Renewal Subcommittee and Full Committee meetings for the IPEC LRA held in March 2009, and in September 2009, respectively. I also supported Entergy at the ACRS Subcommittee meeting for the IPEC LRA held in 2009 and 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 metal fatigue and EAF, the effects of aging on RVIs, and the management of potential PWSCC in steam generator components. In preparing my testimony, I also reviewed the parties' pleadings on NYS-38/RK-

TC-5, the Board's Order Admitting NYS-38/RK-TC-5, the Board's Order Amending NYS-38/RK-TC-5, and the exhibits submitted by NYS and Riverkeeper that are relevant to my testimony.

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 Consultant with Talisman International, LLC. Since April 2007, when I retired from the NRC, as discussed below, I have provided consulting services to nuclear utilities, vendors and fuel cycle facilities 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 RVI contention (NYS-25). My role regarding NYS-38/RK-TC-5 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, steam generators, and other primary 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, 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 2001, 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 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.

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-38/RK-TC-5?

A23. (JRS) Yes. I am familiar with the technical issues related to the management of the effects of PWSCC in steam generator components, the effects of fatigue in primary plant components, and the effects of aging on the RVIs. I am also familiar with the portions of the IPEC LRA that address such issues, including the relevant Entergy aging management programs ("AMPs").

In particular, as relevant to NYS-38/RK-TC-5, I am familiar with Sections 4.3.1 (Class 1 Fatigue); 4.3.2 (Non-Class 1 Fatigue); 4.3.3 (Effects of Reactor Water Environment on Fatigue Life); B.1.12 (the Fatigue Monitoring Program ("FMP")), B.1.18 (ISI AMP), B.1.35 (Steam Generator Integrity Program), B.1.38 (Thermal Aging and Neutron Irradiation Embrittlement of CASS AMP), B.1.41 (Water Chemistry Control Program), and B.1.42 (RVI AMP) of the LRA. As a result, I am familiar with Entergy's plans to manage the potential for PWSCC in steam generator components, fatigue in primary plant components, and to manage the effects of aging on RVIs at IPEC.

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Q24. Please further describe the basis for your familiarity with these aspects of the LRA, and the associated technical issues raised in NYS-38/RK-TC-5.

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 metal fatigue and EAF, the effects of aging on RVIs, and the management of potential PWSCC in steam generator components. In preparing my testimony, I also reviewed the parties' pleadings on NYS-38/RK-TC-5, the Board's Order Admitting NYS-38/RK-TC-5, the Board's Order Amending NYS-38/RK-TC-5, and the exhibits submitted by NYS and Riverkeeper that are relevant to my testimony.

E. <u>Mark A. Gray ("MAG")</u>

Q25. Please state your full name.

A25. (MAG) My name is Mark A. Gray.

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

A26. (MAG) I am a Principal Engineer in the Primary Systems Design and Repair group at Westinghouse Electric Company ("Westinghouse") with over 34 years of experience in nuclear component structural analysis.

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

A27. (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 RVI contention (NYS-25). My role regarding NYS-38/RK-TC-5 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, EAF evaluations.

[redacted]

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

A28. (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 employed by 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.

Q29. Please describe your specific mechanical and structural engineering experience, including experience with analysis of fatigue in key reactor components.

A29. (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 of 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 Westinghouse Owners Group 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.

[redacted]

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-38/RK-TC-5?

A30. (MAG) Yes. I am familiar with the technical issues related to the management of the effects of fatigue in primary plant components. I am also familiar with 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-38/RK-TC-5, I am familiar with Sections 4.3.1 (Class 1 Fatigue); 4.3.2 (Non-Class 1 Fatigue); 4.3.3 (Effects of Reactor Water Environment on Fatigue Life); B.1.12 (the FMP) of the LRA. As a result, I am familiar with Entergy's plans to manage the effects of fatigue in primary plant components 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-38/RK-TC-5.

A31. (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 preparing my testimony, I also reviewed the parties' pleadings on NYS-38/RK-TC-5, the Board's Order Admitting NYS-38/RK-TC-5, the Board's Order Amending NYS-38/RK-TC-5, and the exhibits submitted by NYS and Riverkeeper that are relevant to my testimony.

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

Q32. Please state your full name.

A32. (TJG) My name is Timothy J. Griesbach.

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

A33. (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.

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

A34. (TJG) I have been retained by Entergy as an independent technical expert in connection with the adjudication of this contention and the RVI contention (NYS-25). My role regarding NYS-38/RK-TC-5 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 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") (NRC000114A-F).

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

A35. (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 1972. I am a member of the American Nuclear Society and the American Society of Mechanical Engineers ("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 reactor pressure vessels. I also am a member of the ASME Section XI Standards Committee.

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

A36. (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 the Electric Power Research Institute. 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 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 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 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.

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-38/RK-TC-5?

A37. (TJG) Yes. I am familiar with the technical issues related to the management of the effects of PWSCC in steam generator components, the effects of fatigue in primary plant components, and the effects of aging on the RVIs. I am also familiar with 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-38/RK-TC-5, I am familiar with Sections 4.3.1 (Class 1 Fatigue); 4.3.2 (Non-Class 1 Fatigue); 4.3.3 (Effects of Reactor Water Environment on Fatigue Life); B.1.12 (the FMP), B.1.18 (ISI AMP), B.1.35 (Steam Generator Integrity Program), B.1.38 (Thermal Aging and Neutron Irradiation Embrittlement of CASS AMP), B.1.41 (Water Chemistry Control Program), and B.1.42 (RVI AMP) of the LRA. As a result, I am familiar with Entergy's plans to manage the potential for PWSCC in steam generator components, fatigue in primary plant components, and to manage the effects of aging on RVIs 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-38/RK-TC-5.

A38. (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 metal fatigue and EAF, the effects of aging on RVIs, and the management of potential PWSCC in steam generator components. In preparing my testimony, I also reviewed the parties' pleadings on NYS-38/RK-TC-5, the Board's Order Admitting NYS-38/RK-TC-5, the Board's Order Amending NYS-38/RK-TC-5, and the exhibits submitted by NYS and Riverkeeper that are relevant to my testimony.

G. <u>Barry M. Gordon ("BMG")</u>

Q39. Please state your full name.

A39. (BMG) My name is Barry M. Gordon.

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

A40. (BMG) I am an Associate at Structural Integrity Associates, Inc.

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

A41. (BMG) I have been retained by Entergy as an independent technical expert in connection with the adjudication of this contention. My role regarding NYS-38/RK-TC-5 is to provide independent expert testimony based on my experience regarding materials corrosion behavior including PWSCC.

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

A42. (BMG) My professional and educational qualifications are detailed in the attached *curriculum vitae* (ENT000680). Briefly summarized, I received a Master of Science degree in Metallurgy and Material Science from Carnegie Mellon University. I have over 45 years of experience and expertise in materials corrosion behavior in nuclear power plant environments. Upon graduation, I was employed by Westinghouse Bettis as a Materials Engineer, studying the corrosion and hydriding of zirconium fuel cladding followed by mitigation of steam generator corrosion. I was subsequently hired by GE Nuclear Energy ("GENE") to help address intergranular stress corrosion cracking ("IGSCC") of austenitic stainless steels and nickel base alloys in BWR environments. During my 23-year career at GENE, I qualified hydrogen water chemistry ("HWC") and patented zinc injection for water chemistry mitigation of IGSCC. In 1998, I became an Associate with Structural Integrity Associates, Inc. and continue to work on a variety of materials corrosion problems in light water reactors ("LWRs") with continued emphasis on SCC. I am a Corrosion Specialist and Fellow in National Association of Corrosion Engineers ("NACE") International and have been teaching a class on "Corrosion and Corrosion Control in LWRs" at the NRC since 2004.

Q43. Are you familiar with the sections of the IPEC LRA, and its subsequent revisions, that are relevant to the technical issues raised in NYS-38/RK-TC-5?

A43. (BMG) Yes. I am familiar with the technical issues related to the management of the effects of PWSCC in steam generators and other primary plant components. I am also familiar with 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-38/RK-TC-5, I am familiar with Sections B.1.35 (Steam Generator Integrity Program) and B.1.41 (Water Chemistry Control Program) of the LRA. As a result, I am familiar with Entergy's plans to manage the potential for PWSCC in steam generator and other primary plant components.

Q44. Please describe the basis for your familiarity with these aspects of the LRA, and the associated technical issues.

A44. (BMG) I have reviewed various materials in preparing this testimony, including those portions of Entergy's LRA for IP2 and IP3 relating to Entergy's management of potential PWSCC in steam generator components. In preparing my testimony, I also reviewed the parties' pleadings on NYS-38/RK-TC-5, the Board's Order Admitting NYS-38/RK-TC-5, the Board's Order Amending NYS-38/RK-TC-5, and the exhibits submitted by NYS and Riverkeeper that are relevant to my testimony.

H. Randy G. Lott ("RGL")

Q45. Please state your full name.

A45. (RGL) My name is Randy G. Lott.

Q46. By whom are you employed and what is your position?

A46. (RGL) I am a Consulting Engineer at Westinghouse Electric Company ("Westinghouse") with over 35 years of experience in nuclear materials and radiation effects.

Q47. Please describe your role in this license renewal proceeding.

A47. (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 RVI contention (NYS-25). My role regarding NYS-38/RK-TC-5 is to provide independent expert testimony based on my experience developing and implementing aging management strategies for 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.

Q48. Please describe your educational and general professional qualifications, including relevant professional activities.

A48. (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 have been responsible for numerous investigations of materials-related issues in PWRs. I have supervised testing of RPV surveillance capsules and conducted research programs on irradiation embrittlement and annealing of RPV steels. In addition, I have pioneered the application of the Master Curve testing to characterize the ductile-to-brittle fracture toughness transition in RPV steels. My contributions have provided the basis for the reconsideration of Regulatory Guide 1.99, *Radiation Embrittlement of Reactor Vessel Materials, Revision 2* (May 1988), the development of Westinghouse RPV annealing technology, the safety analysis of reactor tanks at Savannah River, the determination of crack growth rates used in alternative plugging criteria for nuclear steam generators, and the evaluation of RVI performance.

Q49. Please describe your specific nuclear materials engineering experience, including experience with analysis of aging effects on RVIs and steam generator components.

A49. (RGL) During my career at Westinghouse I have participated in the evaluation of aging degradation or failure of numerous reactor components including steam generator tubing, BMI flux thimbles, control rod guide tube "split" pins, baffle-former bolts and clevis insert bolts. I have also conducted numerous research programs on highly irradiated stainless steels, including tensile, fracture toughness and IASCC testing. For the past eight years, I have been actively involved in the design and implementation of AMPs for RVIs. As a member of the MRP Reactor Internals Inspection and Evaluation Core Group, I contributed to the EPRI Materials Reliability Program Pressurized Water Reactor Internal Inspection and Evaluation Guidelines (MRP-227-A). My work on aging management strategies for the Westinghouse and Combustion Engineering plants provided the basis for the recommended guidelines. The same recommendations have been adopted in the most recent revision of the NRC's Generic Aging Lessons Leaned (GALL) Report (NUREG-1801).

Q50. Are you familiar with the sections of the IPEC LRA, and its subsequent revisions that are relevant to the technical issues raised in NYS-38/RK-TC-5?

A50. (RGL) Yes. I am familiar with the technical issues related to the management of the effects of PWSCC in steam generator components, the effects of fatigue in primary plant components, and the effects of aging on the RVIs. I am also familiar with 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-38/RK-TC-5, I am familiar with Sections 4.3.1 (Class 1 Fatigue); 4.3.2 (Non-Class 1 Fatigue); 4.3.3 (Effects of Reactor Water Environment on Fatigue Life); B.1.12 (the FMP), B.1.18 (ISI AMP), B.1.38 (Thermal Aging and Neutron Irradiation Embrittlement of CASS AMP), and B.1.42 (RVI AMP) of the LRA. As a result, I am familiar with Entergy's plans to manage the potential for PWSCC in primary plant components, fatigue in primary plant components, and to manage the effects of aging on RVIs at IPEC.

Q51. Please further describe the basis for your familiarity with these aspects of the LRA, and the associated technical issues raised in NYS-38/RK-TC-5.

A51. (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 metal fatigue and EAF, the effects of aging on RVIs, and the management of potential PWSCC in steam generator components. In preparing my testimony, I also reviewed the parties' pleadings on NYS-38/RK-TC-5, the Board's Order Admitting NYS-38/RK-TC-5, the Board's Order Amending

NYS-38/RK-TC-5, and the exhibits submitted by NYS and Riverkeeper that are relevant to my testimony.

II. OVERVIEW OF CONTENTION NYS-38/RK-TC-5

Q52. Are you familiar with contention NYS-38/RK-TC-5, as originally proposed by NYS and Riverkeeper?

A52. (All) Yes. We have reviewed the following: the "State of New York's and Riverkeeper's Joint Motion for Leave to File a New Contention Concerning Entergy's Failure to Demonstrate that It Has All Programs that Are Required to Effectively Manage the Effects of Aging of Critical Components or Systems," dated September 30, 2011; the "State of New York and Riverkeeper's New Joint Contention NYS-38/RK-TC-5," dated September 30, 2011 ("Contention NYS-38/RK-TC-5"); and the associated Declarations of Dr. Richard T. Lahey, Jr., dated September 30, 2011, and Dr. Joram Hopenfeld , dated September 30, 2011.

The contention, as originally pled, alleges that Entergy:

> [I]s not in compliance with the requirements of 10 C.F.R. §§ 54.21(a)(3) and (c)(1)(iii) and the requirements of 42 U.S.C. §§ 2133(b) and (d) and 2232(a) because Entergy does not demonstrate that it has a program that will manage the affects [sic] of aging of several critical components or systems and thus NRC does not have a record and a rational basis upon which it can determine whether to grant a renewed license to Entergy as required by the Administrative Procedure Act.

Contention NYS-38/RK-TC-5 at 1; *see also* State of New York and Riverkeeper, Inc. Revised Statement of Position, Joint Contention NYS-38/RK-TC-5 at 2 (June 9, 2015) ("Intervenors' Revised SOP") (NYS000531).

[redacted]

Q53. Are you familiar with Contention NYS-38/RK-TC-5, as admitted by the Board on November 10, 2011?

A53. (All) Yes. On November 10, 2011, the Board admitted the contention, stating that Intervenors contend that "Entergy's new commitments do not meet NRC regulations for having a program that will adequately manage the effects of aging during the period of extended operations." Order Admitting NYS-38/RK-TC-5 at 10.

Q54. What bases did the Intervenors originally proffer in support of Contention NYS-38/RK-TC-5?

A54. (All) As originally pled, contention NYS-38/RK-TC-5 relies on four specific bases proffered by the Intervenors and identified by the Board. Specifically, Intervenors allege that Entergy:

1. has deferred defining the methods used for determining the most limiting locations for metal fatigue calculations and the selection of those locations;
2. has not specified the criteria it will use and assumptions upon which it will rely for modifying the WESTEMS™ computer model for environmentally adjusted cumulative usage factors (CUF_{en}) calculations;
3. has not adequately defined how it will manage primary water stress corrosion cracking (PWSCC) [in steam generators] because it will not begin inspections until after entering the period of extended operations ("PEO") and Entergy has substituted a document, which will not be released until 2013, for its prior water chemistry program to manage PWSCC of the nickel alloy or nickel-alloy clad steam generator divider plates exposed to reactor coolant; and
4. does not adequately describe the contents of its AMP for reactor vessel internals, based on a revised version of the Materials Reliability Program 227 (MRP-227) guidance document.

Order Admitting NYS-38/RK-TC-5 at 10-11 n.47 (*citing* Contention NYS-38/RK-TC-5 at 1-3); *see also* Licensing Board Order (Denying NRC Staff's Motion for Partial Reconsideration and State of New York/Riverkeeper's Cross-Motion to NRC Staff's Motion for Reconsideration) at 3 n.7 (Apr. 23, 2012) (unpublished) ("April 23 Order").

[redacted]

As discussed further below, Bases (1) and (2) both relate to the issue of metal fatigue and challenge commitments that support Entergy's FMP. Those issues, particularly to the extent they constitute technical challenges to Entergy's fatigue evaluations, are discussed extensively in our metal fatigue testimony, which we incorporate by reference, in full, in this testimony. *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 (Metal Fatigue) (Aug. 10, 2015) ("Entergy's NYS-26B/RK-TC-1B Testimony") (ENT000679). To the extent Bases (1) and (2) challenge Entergy's license renewal commitments or present unique issues not addressed in the metal fatigue contention, we address the Intervenors' challenges in our testimony here.

Q55. What specific Entergy commitments do Intervenors claim are inadequate?

A55. (All) As set forth in the four bases for their contention, Intervenors challenge the adequacy of certain IPEC license renewal commitments (Commitments 30, 41, 42, 43, 44, and 49):

- Basis (1) challenges IPEC license renewal Commitment 43, in which Entergy committed to review its design basis fatigue evaluations to determine whether the previously analyzed locations are limiting for the IP2 and IP3 configurations. *See infra*, Section V.B.1. In Commitment 49, Entergy later clarified that this limiting locations review includes all RVI components with a current licensing basis ("CLB") CUF analysis. *See* NL-13-052, Letter from F. Dacimo, Entergy, to NRC Document Control Desk, "Reply to Request for Additional Information Regarding the License Renewal Application," Attach. 1 at 9 (May 7, 2013) ("NL-13-052") (NYS000501).

- Basis (2) concerns IPEC license renewal Commitment 44, in which Entergy committed to document any "user intervention" in future WESTEMS^{TM} fatigue evaluations for IPEC. *See infra*, Section V.B.2.

- Basis (3) challenges license renewal Commitment 41, and, as the Board has subsequently clarified, Commitment 42, in which Entergy committed to inspect the IPEC steam generator components for indications of PWSCC. *See infra*, Section V.C.

- Basis (4) relates to Commitment 30 in Entergy's original LRA, wherein Entergy committed to manage aging effects on reactor vessel internals by participating in generic industry programs on this issue and to evaluate and implement the results of those programs, as approved by the NRC. *See infra*, Section V.D.

Q56. What is Intervenors' chief complaint with respect to the license renewal commitments identified above?

A56. (All) As stated in paragraph 3 of their contention, Intervenors claim that "Entergy impermissibly assumes that a commitment to develop a program in the future whose goal it is to meet the requirements of the regulations and to follow the guidance in GALL is legally sufficient to meet its obligations" under 10 C.F.R. Part 54. They further assert that, "[c]ontrary to Entergy's assumption, GALL is not merely a list of goals that are to be met but a requirement that an AMP be developed and presented that can be tested against those goals to determine if they have been met." Contention NYS-38/RK-TC-5 at 3.

Q57. Have the Intervenors amended contention NYS-38/RK-TC-5 since 2011?

A57. (All) Yes. On November 6, 2014, the NRC Staff issued NUREG-1930, Supp. 2, Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3 (Nov. 2014) ("SSER 2") (NYS000507). The Board provided the Intervenors with an opportunity to file new contentions or amend their existing Track 2 safety contentions following SSER 2. *See* Licensing Board Revised Scheduling Order at 2 (Dec. 9, 2014) (unpublished).

On February 13, 2015, the Intervenors filed the State of New York and Riverkeeper's Joint Motion for Leave to Supplement Previously-Admitted Joint Contention NYS-38/RK-TC-5 (Feb. 13, 2015), along with their proposed supplement ("NYS-38/RK-TC-5 Supplement"). In their motion, the Intervenors proffered an amendment to NYS-38/RK-TC-5 to include a fifth basis alleging that "Entergy's currently proposed AMP for RVI components fails to assure that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation' as required by 10 CFR § 54.21(c)(1)(iii)." NYS-38/RK-TC-5 Supplement at 1. On

March 31, 2015, the Board admitted the amended contention. Order Amending NYS-38/RK-TC-5 at 10.

Basis (5) relates to the IPEC RVI AMP, which is discussed comprehensively in our testimony on Contention NYS-25, which we incorporate by reference, in full, in this testimony. *See generally* 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) (Aug. 10, 2015) ("Entergy's NYS-25 Testimony") (ENT000616).

Q58. Have you reviewed the Intervenors' written statements of position, prefiled testimony, and supporting exhibits concerning NYS-38/RK-TC-5?

A58. (All) Yes. As noted above, the State and Riverkeeper made separate evidentiary submissions in June 2012, November 2010, and June 2015. We have reviewed the following documents to the extent that they are relevant to our testimony:

<u>June 19, 2012 Position Statement and Testimony</u>**:**

- NYS000371, State of New York and Riverkeeper, Inc. Initial Statement of Position in Support of Joint Contention NYS-38/RK-TC-5 ("Intervenors' Initial SOP");
- NYS000372, Pre-Filed Written Testimony of Dr. David J. Duquette Regarding Contention NYS-38/RK-TC-5 ("Duquette Testimony");
- NYS000373, Report of Dr. David J. Duquette in Support of Contention NYS-38/RK-TC-5 ("Duquette Report");
- NYS000374, Pre-Filed Written Testimony of Dr. Richard T. Lahey, Jr. Regarding Contention NYS-38/RK-TC-5 ("Lahey Testimony"); and
- RIV000102, Pre-Filed Written Testimony of Dr. Joram Hopenfeld Regarding Contention NYS-38/RK-TC-5 ("Hopenfeld Testimony").

<u>November 9, 2012 Rebuttal Position Statement and Testimony</u>:

- NYS000451, State of New York and Riverkeeper, Inc. Revised Statement of Position in Support of Joint Contention NYS-38/RK-TC-5;

- NYS000452, Pre-filed Written Rebuttal Testimony of Dr. David J. Duquette Regarding Contention NYS-38/RK-TC-5 ("Duquette Rebuttal");

- NYS000453, Pre-filed Written Rebuttal Testimony of Dr. Richard T. Lahey, Jr. Regarding Contention NYS-38/RK-TC-5 ("Lahey Rebuttal"); and

- RIV000134, Prefiled Rebuttal Testimony of Dr. Joram Hopenfeld Regarding Contention NYS-38/RK-TC-5 ("Hopenfeld Rebuttal").

<u>June 9, 2015 Revised Position Statements and Testimony</u>:

- NYS000531, Intervenors' Revised SOP;

- NYS000532, Pre-Filed Written Supplemental Testimony of Dr. David J. Duquette Regarding Contention NYS-38/RK-TC-5 ("Supplemental Duquette Testimony");

- NYS000562, Revised Pre-Filed Written Testimony of Dr. Richard T. Lahey, Jr. Regarding Joint Contention NYS-38/RK-TC-5 ("Revised Lahey Testimony");

- RIV000143, Supplemental Pre-Filed Written Testimony of Dr. Joram Hopenfeld Regarding Contention NYS-38/RK-TC-5 ("Supplemental Hopenfeld Testimony"); and

- RIV000144, Supplemental Report of Dr. Joram Hopenfeld in Support of Contention NYS-26/RK-TC-1B and Amended Contention NYS-38/RK-TC-5 ("Supplemental Hopenfeld Report").

<u>Other Intervenor Exhibits Filed in 2012 and 2015</u>: We also have reviewed supporting exhibits NYS00146A-C, NYS000147A-D, NYS000150 through NYS000154, NYS000160, NYS000161, NYS000166, NYS000195, NYS000199, NYS000295 through NYS000300, NYS000302 through NYS000369A-B, NYS000375 through NYS000397, NYS000454 through NYS000464, NYS000472, NYS000483 through NYS000528, NYS000533 through NYS000562, RIV000004, RIV000035 through RIV000058, and RIV000102 through RIV000106, RIV000115

[redacted]

through RIV000119, RIV00013, RIV000133, RIV000135 through RIV000141, and RIV000145 through RIV000160.

In reviewing these statements of position, testimony, and reports, we note that the Intervenors have not replaced their 2012 submittals with updated materials in 2015, but merely added new information into the record in 2015. Thus, the claims of the Intervenors, and Drs. Duquette, Lahey and Hopenfeld, are cumulative and overlapping; redundant in some areas and 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 Intervenors' and their experts' claims, as they can best be understood.

Q59. Have you reviewed the Joint Stipulation concerning Contention NYS-38/RK-TC-5 filed by the parties on June 23, 2015?

A59. (All) Yes. In that document, the Intervenors "aver that [Dr. Duquette's] testimony [regarding steam generator wear, plugging and foreign objects] is offered solely for the purpose of presenting Dr. Duquette's opinions on the adequacy of Entergy Commitments 41 and 42 and is not offered as a general challenge to Entergy's Steam Generator Integrity Aging Management Program"). *See* State of New York, Riverkeeper, Inc., Nuclear Regulatory Commission Staff, and Entergy Nuclear Operations, Inc., Joint Stipulation Regarding State of New York Pre-Filed Testimony for Contention NYS-38/RK-TC-5 (Safety Commitments) (June 23, 2015) (ENT000700). Accordingly, we do not address in our testimony matters generally related to the steam generator aging management to the extent they are unrelated to Commitments 41 and 42.

[redacted]

Q60. What other materials have you reviewed in preparing your testimony?

A60. (All) We have reviewed numerous documents in preparing this testimony, including NRC regulations, guidance documents, and technical studies. Key documents include, but are not limited to, the following:

- 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, Rev. 1") (NYS00146A-C);
- NUREG-1801, Generic Aging Lessons Learned Report, Rev. 2 (Dec. 2010) ("NUREG-1801, Rev. 2") (NYS00147A-D);
- NUREG/CR-6260, Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components (Feb. 1995) (NYS000355);
- EPRI, Final Report No. 1020988, Steam Generator Management Program: Phase II Divider Plate Cracking Engineering Study (Nov. 2010) ("EPRI Phase II Study") (ENT000523);
- EPRI, Final Report 1025133, Steam Generator Management Program: Assessment of Channel Head Susceptibility to Primary Water Stress Corrosion Cracking (June 2012) ("EPRI June 2012 Report") (ENT000524); and
- EPRI, Final Report 3002002850, Steam Generator Management Program: Investigation of Crack Initiation and Propagation in the Steam Generator Channel Head Assembly (Oct. 2014) (NYS000544A-D) ("EPRI 2014 Report").

As cited below, we also have reviewed relevant Commission adjudicatory decisions issued 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.

[redacted]

Q61. I show you what has been marked as Exhibits ENTR15001, ENT00015A-B, ENTR00031, ENT000032, ENT000041, ENTR00184 through ENTR00186, ENT000190, ENT000192, ENT000196, ENT000197, ENT000230, ENT000251, ENT000252, ENT000522 through ENT000572, ENT000616 through ENT000618, ENT000641, ENT000657, ENT000679, ENT000680, ENT000683, ENT00686A-C through ENT000688, ENT000692, ENT000695, and ENT000699 through ENT000721. Do you recognize these documents?

A61. (NFA, ABC, JRS, RGL, MAG, BMG) Yes. ENTR15001 is a list of Entergy's exhibits, and includes those documents which we referred to, used, or relied upon in preparing this testimony. We have reviewed those documents, ENTR15001, ENT00015A-B, ENTR00031, ENT000032, ENT000041, ENTR00184 through ENTR00186, ENT000190, ENT000192, ENT000196, ENT000197, ENT000230, ENT000251, ENT000252, ENT000522 through ENT000572, ENT000616 through ENT000618, ENT000641, ENT000657, ENT000679, ENT000680, ENT000683, ENT00686A-C through ENT000688, ENT000692, ENT000695, and ENT000699 through ENT000721, and these are true and accurate copies of the documents that we have referred to and/or relied upon in preparing this 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.

Q62. How do these documents relate to the work that you do as an expert in forming opinions such as those contained in this testimony?

A62. (All) 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 cannot offer legal opinions on the language of the NRC regulations or adjudicatory decisions discussed in our

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testimony. However, reading those regulations and decisions as technical statements, and relying on our expertise and experience, we can interpret the meaning of those documents as they relate to the technical and regulatory issues raised in these contentions. Insofar as our testimony provides opinions on the requirements of NRC regulations, we believe that such opinions will be helpful to the Board, because they provide 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

Q63. Please summarize the purpose of your testimony and the basis for your disagreement with the claims made by the Intervenors and their proffered experts, Drs. Duquette, Lahey, and Hopenfeld, in NYS-38/RK-TC-5.

A63. (All) The purpose of our testimony is to demonstrate that NYS-38/RK-TC-5 lacks merit and, accordingly, should be resolved in Entergy's favor. We show that the particular commitments at issue (Commitments 30, 41, 42, 43, 44, and 49), most of which are now partially or fully complete, support the conclusion that there is reasonable assurance that the effects of aging *will be* adequately managed in accordance with applicable NRC regulations, guidance, and precedent. Specifically, the RVI AMP, FMP and Water Chemistry Control – Primary and Secondary Program ("Water Chemistry Program") described in the IPEC LRA conform to the applicable guidance in NUREG-1801 and comply with NRC license renewal regulations. The LRA provides sufficiently detailed information for the NRC to determine that these AMPs meet the criteria in NUREG-1801, Rev. 1. *See* NUREG-1930, Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3 at 3-79 (Nov. 2009) ("SER") (NYS00326B); *id.* at 3-145 (NYS00326C). Entergy's Commitment 30, in the original

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LRA, and new Commitments 43, 44, and 49 (regarding metal fatigue), and 41 and 42 (regarding PWSCC in steam generator components), made in response to the Staff's RAIs, provide further support for the adequacy of Entergy's planned aging management activities. Entergy's AMPs therefore meet the intent of the NRC Staff's most recent guidance.

We also address the specific technical issues raised and commitments challenged by the Intervenors. In doing so, we demonstrate that Entergy already has accomplished what Intervenors claim is required—to the extent Intervenors' claims are grounded in NRC regulations or guidance. Namely, the LRA contains specific AMPs that are consistent with the ten elements of the NRC-approved AMPs in NUREG-1801, and which contain sufficient detail on the methods, criteria, assumptions, and timing of aging management activities. The AMPs are supplemented by Entergy commitments to undertake specific actions in the future—commitments that are enforceable by the NRC under its ongoing 10 C.F.R. Part 50 oversight and inspection processes. Entergy's AMPs and commitments have been reviewed by the NRC Staff and approved in the first Supplemental SER, 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) and SSER 2.

Q64. The Intervenors claim that Entergy's AMPs and related commitments at issue in NYS-38 are "mere promise[s]" and therefore deficient. *See* Intervenors' Revised SOP at 51 (NYS000531). How do you respond?

A64. (All) Our testimony demonstrates that Intervenors portrayal of Entergy's AMPs and commitments as "mere promise[s]" to provide information after approval of the renewed licenses is incorrect. Such information is required and has been prepared prior to license renewal. Our testimony also demonstrates that Intervenors incorrectly assert that the "necessary factual record is missing because Entergy is not providing *any* of the details required to determine"

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whether its AMPs will be effective or are consistent with NUREG-1801. *Id*. at 50 (emphasis added).

Thus, contrary to the Intervenors' fundamental claim in this contention, Entergy is not relying on vague commitments to implement or develop undefined AMPs and activities for purposes of compliance with Part 54. Rather, the necessary AMPs and activities *already* have been appropriately defined by Entergy and thoroughly reviewed by the NRC Staff in accordance with NUREG-1800 and NUREG-1801—documents that were prepared at the Commission's direction, and which the Commission repeatedly has identified as acceptable to demonstrate that an AMP will effectively manage the effects of aging during the PEO. In sum, Entergy's LRA demonstrates that there is reasonable assurance that the effects of aging on RVIs, the effects of metal fatigue on reactor coolant system ("RCS") components, and the effects of PWSCC on steam generator divider plates and other channel head components will be adequately managed during the PEO, consistent with 10 C.F.R. §§ 54.21(a)(3), 54.21(c)(1)(iii), and 54.29(a).

Q65. The Intervenors also imply that Entergy intends to "relax" its aging management activities whenever the opportunity arises. *See* Intervenors' Initial SOP at 37 (NYS000371); *see also* Intervenors' Revised SOP at 53 (NYS000531). What is your response?

A65. (NFA, RJD, ABC) Entergy strongly disagrees with this implication. Entergy fully recognizes that improper aging management can potentially result in adverse consequences. Indeed, Entergy has legal responsibilities and significant incentives to ensure the safe and reliable continued operation of IPEC. The suggestion that Entergy will somehow seek to relax its stewardship of IPEC in the future is simply wrong.

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Q66. Please summarize how your testimony is organized.

A66. (All) Sections I through III of our testimony provide witness background information, an overview of contention NYS-38/RK-TC-5, and a summary of our testimony and conclusions. Next, in Section IV, we summarize the applicable license renewal regulations and guidance. In Section V, we explain why the commitments at issue fully support the Staff's reasonable assurance findings in the SER, SSER 1, and SSER 2. We specifically refute the various contrary claims made in the Intervenors' testimony and demonstrate that those claims lack merit. As necessary, we discuss key technical concepts, particularly those concerning the potential for PWSCC in steam generator divider plates and other channel head components. Section VI summarizes our testimony and the bases for our conclusion that NYS-38/RK-TC-5 lacks factual and technical merit.

Q67. Given the interrelated nature of the issues raised in NYS 38/RK-TC-5, NYS-26B/RK-TC-5, and NYS-25, do you address the latter in this testimony as well?

A67. (All) Our testimony on contention NYS-38/RK-TC-5 primarily addresses Entergy's safety commitments pertaining to metal fatigue and EAF, and the potential for PWSCC in steam generator components. Because Dr. Lahey's testimony on RVIs in NYS-25 is indistinguishable from his testimony on that topic on this contention, we address RVI-related issues in our testimony on NYS-25. That testimony is incorporated by reference in this document in its entirety. *See generally* Entergy's NYS-25 Testimony (ENT000616). Also, to the extent intervenors raise technical challenges regarding metal fatigue and Entergy's EAF calculations, as opposed to regulatory challenges regarding the adequacy of Entergy's safety commitments, those challenges are addressed in our testimony on NYS-26B/RK-TC-1B, which is incorporated by reference here. *See generally* Entergy's NYS-26B/RK-TC-1B Testimony (ENT000679).

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Q68. As an overarching claim, Intervenors assert that license renewal commitments are unenforceable and cannot be relied upon under Part 54. Do you agree?

A68. (JRS, ABC) No. Licensing commitments (including license renewal commitments) are controlled under an Entergy process that is in accordance with NRC-endorsed guidance. As a practical matter, most of Entergy's commitments have already been completed, as IP2 is in the PEO and IP3 will enter the PEO in December 2015. *See* SSER 2, App. A (NYS000507). Thus, the NRC Staff has already inspected the implementation of Entergy's IP2 commitments required to be implemented prior to the PEO, *see generally* Letter from J. Trapp, NRC, to J. Ventosa, Entergy, "Indian Point Nuclear Generating Unit 2 – NRC License Renewal Team Inspection Report 05000247/2013010" (Sept. 19, 2013) ("IP2 LR Commitment Inspection") (ENT000695), and Entergy understands that the NRC Staff intends to inspect Entergy's IP3 commitments in October 2015. *See* Letter from A. Burritt, NRC, to L. Coyle, Entergy, "Annual Assessment Letter for Indian Point Nuclear Generating Units 2 and 3 (Report 05000247/2014001 and 05000286/2014001)," Encl. at 2 (Mar. 4, 2015) ("IP3 Inspection Plan") (ENT000701). But, aside from the commitments that have already been completed, the Intervenors are wrong that commitments are unenforceable.

As a regulatory matter, Part 54 specifically authorizes licensees to demonstrate compliance with its requirements via prospective actions to be taken after the NRC issues the renewed license. A fundamental aspect of the license renewal process under Part 54 is the requirement for the applicant to identify actions that *will be taken* to provide reasonable assurance of safety during the PEO. *See* 10 C.F.R. § 54.29(a). As such, the license renewal rules fully recognize that new commitments are an essential part of this process.

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Moreover, the review and enforcement of licensee activities to fulfill licensing commitments is a normal part of the NRC Staff's ongoing oversight function. *See* NRR Office Instruction, LIC-105, Revision 5, Managing Regulatory Commitments Made by Licensees to the NRC, at 6 ("LIC-105") (Sept. 16, 2013) (ENT000705).

Intervenors also argue that the implementation details of all aging management activities must be provided in the LRA, and that all commitments should be elevated to license conditions, such that they could only be changed through license amendments. *See* Intervenors' Revised SOP at 42 (NYS000531). However, as we will show, such information and actions are not required for issuance of a renewed license or to support a reasonable assurance finding under 10 C.F.R. Part 54.

Q69. Let's now turn to your testimony and a summary of your response to each of the five bases proffered by the Intervenors in support of NYS-38. First, in Basis (1) of the contention, Intervenors' claim that Entergy has not defined the methods to be used in implementing Commitments 43 and 49 (concerning limiting locations for fatigue). *See* Order Admitting NYS-38/RK-TC-5 at 10-11 n.47 (citing Contention NYS-38/RK-TC-5 at 1-3); NYS-38/RK-TC-5 Supplement at 1. Please summarize the basis for your disagreement with Intervenors' testimony on this basis of the contention.

A69. (MAG, JRS, ABC, NFA, RGL, TJG) In summary, Intervenors claim that these commitments are vague, because they do not explicitly define the locations to be analyzed and the process and timing of the analysis of limiting locations. *See*, *e.g.*, Hopenfeld Testimony at 10-12 (RIV000102). The commitment, however, is clear, and the methodology used to identify limiting locations is well defined. The review, which is now complete, covered all plant components with a CLB cumulative usage factor ("CUF") fatigue analysis. *See* Entergy's NYS-26B/RK-TC-1B

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Testimony at Question Q122 (ENT000679). As Entergy later clarified in Commitment 49, this includes all RVI components with a CLB CUF analysis. *See* SSER 2 at A-15 (NYS000507). Those reviews have been performed consistent with standard ASME Code methods and, as appropriate, using the NRC-approved guidance on EAF in:

- NUREG/CR-5704, Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels (Apr. 1999) (NYS000354);
- NUREG/CR-6583, Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels (Mar. 1998) (NYS000356); and
- NUREG-6909, Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials (Feb. 2007) (NYS000357).

Thus, contrary to Intervenors' claims, the nature and timing of the evaluations under Commitments 43 and 49 are clearly established in the record—because the evaluations are complete. *See generally* Westinghouse, Calculation Note CN-PAFM-12-35, Rev. 1, "Indian Point Unit 2 and Unit 3 EAF Screening Evaluations" (Nov. 12, 2012) ("Westinghouse Calculation Note CN-PAFM-12-35") (NYS000510); Westinghouse, Calculation Note CN-PAFM-13-32, Rev. 3, "Indian Point Unit 2 (IP2) and Unit 3 (IP3) Refined EAF Analyses and EAF Screening Evaluations" (May 28, 2015) ("Westinghouse Calculation Note CN-PAFM-13-32, Rev. 3") (ENT000683). As a result, Commitments 43 and 49 fully support the Staff's finding that there is reasonable assurance that the effects of aging due to fatigue on reactor coolant system components will be adequately managed.

Q70. Turning next to Basis (2) of the contention, Intervenors' claim that Entergy has not specified the criteria and assumptions it will use in "modifying WESTEMS" under Commitment 44 (what the Staff has called "user intervention" in WESTEMS^{TM}). *See* Order Admitting NYS-38/RK-TC-5 at 10-11 n.47 (citing Contention NYS-38/RK-TC-5 at 1-3).

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Please summarize the basis for your disagreement with Intervenors' testimony on this commitment.

A70. (MAG, JRS, ABC, NFA) As explained in Section V.B.2, below, peak editing, or "user intervention," to remove redundant stress peaks and valleys in the context of a WESTEMS™ fatigue evaluation is consistent with the longstanding and established ASME Code methodology for conducting stress and fatigue evaluations. It does not involve "manipulations and interventions," as Dr. Lahey states (Lahey Testimony at 27 (NYS000374)), or the modification of the WESTEMS™ code.

We discuss this issue in detail in our metal fatigue testimony, *see* Entergy's NYS-26B/RK-TC-1B Testimony at § V.B.2 (ENT000679), but respond to the specific claims raised in testimony on this contention here. In summary, WESTEMS™ simply uses an automated approach to assist the analyst in selecting the stress peak and valley times in each transient—a process that, under traditional methods would be accomplished *entirely* by the analyst. *See* Letter from P. Davison, PSEG Nuclear, LLC, to NRC Document Control Desk, "Close-out of the NRC Audit Associated with Use of WESTEMS™ Related to the Salem Nuclear Generating Station, Units 1 and 2 License Renewal Application," Encl. A at 6-8 (Feb. 24. 2011) (ENT000197); *see also generally id*. Encl. C (PVP2010-25891, Method for Selecting Stress States for Use in an NB-3200 Fatigue Analysis); Westinghouse, WESTEMS 4.5.7 User Manual, Volume 2, Revision 6 at 299-301 (Mar. 2015) (ENT000686). Commitment 44 requires that Entergy provide a written explanation and justification of any "user intervention" (*i.e.*, editing and re-analysis of peaks and valleys) in future evaluations using the WESTEMS™ "Design CUF" module. This commitment fully addressed the NRC's initial concerns, *see* SSER 1 at 4-2 (NYS000160), and demonstrates that the Intervenors' concerns, here, lack merit.

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Accordingly, for these reasons, Basis (2), claiming that Entergy has not specified criteria for "user intervention" pertaining to WESTEMS^{TM}, is without merit.

Q71. In Basis (3) of the contention, Intervenors' claim that Entergy has not adequately defined how it will manage PWSCC in steam generator divider plates under Commitments 41 and 42 (steam generator component inspections). *See* Order Admitting NYS-38/RK-TC-5 at 10-11 n.47 (citing Contention NYS-38/RK-TC-5 at 1-3). Please summarize the basis for your disagreement with the Intervenors' testimony on this commitment.

A71. (JRS, ABC, NFA, BMG) By way of background, there is foreign operating experience related to the *potential* for PWSCC to affect steam generator divider plates. *See* SRP-LR, Rev. 2 at 3.1-6 (NYS000161). As we explain in more detail in Section V.C.3 below, the EPRI SGMP Engineering and Regulatory Technical Advisory Group completed a number of studies of divider plate cracking in response to foreign (non-U.S.) operating experience, and issued its final report in October 2014. *See generally* EPRI 2014 Report (NYS000544A-D); EPRI Phase

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II Study (ENT000523). [redacted]

In 2011, in light of then-ongoing EPRI research activities on the long-term potential for crack propagation into steam generator pressure boundary components, Entergy committed to undertake timely inspections of the divider plates to confirm the effectiveness of its Water Chemistry Program in managing the effects of aging due to PWSCC. *See* NL-11-032, Letter from F. Dacimo, Entergy, to NRC Document Control Desk, "Response to Request for Additional Information (RAI), Aging Management Programs," Attach. 2, at 16 (Mar. 28, 2011) ("NL-11-032") (adding Commitment 41) (NYS000151); *see also* Section V.C.2, *infra*. Entergy plans to conduct the first scheduled inspections of the IP3 divider plates during the Spring 2017 refueling outage. Entergy will monitor future technical and regulatory developments on this topic as part of its ongoing license renewal program activities.

Q72. Please describe, in summary fashion, the first scheduled divider plate inspections. Are they adequately defined in Commitment 41?

A72. (RJD, NFA, ABC, JRS) As explained in Section V.C.2.d, the inspections under Commitment 41 are adequately defined, as the commitment itself clearly confirms that "[t]he examination technique used will be capable of detecting PWSCC in the steam generator divider plate assembly." Entergy plans to use EVT-1 inspections using a robot-mounted camera, similar

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to methods used for inspections of other steam generator components, as we explain further in response to Question 153. Such methods would be consistent with the standards in the ASME Code. *See* ASME Code, Section XI, Article IWA-2000, "Examination and Inspection" § 2210 (2001) ("ASME Code, IWA-2000") (ENT000531). As such, Intervenors' criticisms of the tailored, performance-based standard specified in Commitment 41 as inadequately defined identify no material deficiency in Entergy's LRA. Commitment 41 further supports the Staff's finding of reasonable assurance, and is consistent with the Staff's acceptance of similar commitments from other applicants.

Q73. What is the status of Commitment 42?

A73. (RJD, NFA, ABC) For IP2, Entergy has already implemented Commitment 42 by seeking and obtaining a license amendment, under "Option 1" of the commitment, to redefine the RCS pressure boundary. *See* Letter from D. Pickett, NRC, to Vice President, Operations, Entergy, "Indian Point Nuclear Generating Unit No. 2 – Issuance of Amendment re: H* Alternate Repair Criteria for Steam Generator Tube Inspection and Repair (TAC No. MF3369)" (Sept. 5, 2014) ("H* Amendment Issuance") (NYS000542). Thus, inspections of the tube-to-tubesheet welds at IP2, under "Option 2" of the commitment, are not necessary. With respect to IP3, Entergy is evaluating the EPRI 2014 Report (NYS000544A-D) to determine whether it supports implementation of the analysis option of Commitment 42. Another alternative available to Entergy is to inspect the IP3 tube-to-tubesheet welds under Option 2 of Commitment 42.

Q74. In Bases (4) and (5) of the contention, Intervenors' claim that the LRA does not adequately describe the contents of its AMP for RVIs, based on MRP-227-A, *see* Order Admitting NYS-38/RK-TC-5 at 10-11 n.47 (citing Contention NYS-38/RK-TC-5 at 1-3), and that Entergy's RVI AMP fails to assure that the effects of aging on the RVIs' intended

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functions will be adequately managed through the PEO, as required by 10 CFR § 54.21(c)(1)(iii). NYS-38/RK-TC-5 Supplement at 1. Please summarize the basis for your disagreement with the Intervenors' claims.

A74. (JRS, ABC, NFA, TJG) As explained in detail in our testimony on contention NYS-25, the LRA complies with 10 C.F.R. Parts 50 and 54 and is fully consistent with the guidance for acceptable AMPs for RVIs in NUREG-1801, Revisions 1 and 2, as updated in LR-ISG-2011-04. *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). Our testimony on contention NYS-25 also explains that MRP-227-A provides an NRC-accepted approach for managing the effects of aging on RVIs and Entergy's RVI AMP is consistent with EPRI's MRP-227-A, and 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). *See generally* Entergy's NYS-25 Testimony (ENT000616).

IV. <u>LICENSE RENEWAL REGULATORY STANDARDS AND GUIDANCE</u>

Q75. In general, what regulatory standards govern the NRC's review of Entergy's LRA and the issuance of a renewed operating license?

A75. (ABC, JRS) Our testimony on NYS-26B/RK-TC-1B discusses in detail the regulatory standards that govern the review of Entergy's LRA and the issuance of a renewed license. *See* Entergy's NYS-26B/RK-TC-1B Testimony § IV.B (ENT000679). To summarize, the NRC standards governing the issuance of a renewed operating license are set forth in 10 C.F.R. §§ 54.21 and 54.29(a). An applicant must demonstrate that, during the PEO, it will manage the effects of aging on the functionality of structures and components that have been identified to require an aging management review ("AMR") under Section 54.21(a)(1). *See* 10 C.F.R. §§ 54.21(a)(3), 54.29(a)(1). In addition, an applicant must evaluate time-limited aging analyses ("TLAAs") in accordance with 10 C.F.R. § 54.21(c)(1). *Id.* § 54.29(a)(2). As defined in 10 C.F.R. § 54.3(a), TLAAs are time-limited calculations or analyses that are part of the CLB. *See also* Entergy's NYS-26B/RK-TC-1B Testimony at Q81 (ENT000679).

Pursuant to Section 54.29(a), the NRC will issue a renewed license if it finds that actions have been identified, and have been or will be taken by the applicant, such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the CLB for the PEO. *See id.* § 54.29(a).

Q76. How does the NRC make its reasonable assurance determination for purposes of license renewal?

A76. (ABC, REN, JRS) We understand that the Commission and its Staff make a determination of whether there is reasonable assurance on a case-by-case basis, using sound technical judgment and verifying the applicant's compliance with NRC regulations. In the license renewal context, Branch Technical Position RLSB-1 in the SRP-LR explains that the license renewal process "is not intended to demonstrate absolute assurance that structures and components will not fail, but rather that there is reasonable assurance" that they will continue to perform their intended functions consistent with the CLB during the PEO. SRP-LR, Rev. 2 at A.1-1 (NYS000161). Also, as previously noted, based upon a recent Commission decision interpreting the license renewal regulations, we understand that if an applicant's AMP is consistent with an AMP identified in NUREG-1801, then a commitment to implement that AMP demonstrates reasonable assurance under 10 C.F.R. § 54.29. *Seabrook*, CLI-12-05, slip op. at 4.

[redacted]

Q77. Do the regulations in 10 C.F.R. Part 54 require an applicant to implement all actions necessary to establish reasonable assurance of safety throughout the PEO *prior* to issuance of the renewed license?

A77. (ABC, JRS) No. The regulations discussed in the two previous responses require applicants to identify actions that "have been or will be taken" such that there is reasonable assurance that the effects of aging will be adequately managed during the PEO. *See* 10 C.F.R. § 54.29(a).

Q78. What guidance documents has the NRC issued to assist in implementing the requirements of 10 C.F.R. Part 54?

A78. (ABC, JRS) The two primary guidance documents issued by the NRC Staff are NUREG-1801 (also referred to as the "GALL Report") and the SRP-LR. *See generally* SRP-LR (NYS000195); SRP-LR, Rev. 2 (NYS000161); NUREG-1801, Rev. 1 (NYS00146A-C); NUREG-1801, Rev. 2 (NYS00147A-D). Our testimony on contention NYS-26B/RK-TC-1B, incorporated by reference here, explains the function, format, and general content of these guidance documents. *See* Entergy's NYS-26B/RK-TC-1B Testimony § IV.B (ENT000679).

Q79. Please summarize the purposes and roles of the SRP-LR and NUREG-1801.

A79. (ABC, JRS) The SRP-LR provides guidance to the NRC Staff in conducting its review of LRAs. It provides acceptance criteria for determining whether the applicant has met the requirements of 10 C.F.R. § 54.21. *See* SRP-LR § 3.1.2 (NYS000195). For each of the systems, structures, and components ("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. NUREG-1801, in turn, provides generic aging management review results for SSCs in the scope of license renewal and describes generic AMPs that the NRC Staff has

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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, and show that the programs proposed for the applicant's facility satisfy the ten elements of a valid AMP that the Staff has previously reviewed and approved as documented in NUREG-1801. *See id.* at 2-3.

Q80. What are the ten elements of an AMP?

A80. (ABC, JRS) As set forth in the SRP-LR, the ten elements used to define an AMP include the following: (1) scope of the program; (2) preventive actions; (3) parameters monitored or inspected; (4) detection of aging effects; (5) monitoring and trending; (6) acceptance criteria; (7) corrective actions; (8) confirmation process; (9) administrative controls; and (10) operating experience. Each of the ten elements is described in further detail in SRP-LR, Rev. 2 § A.1.2.3 (NYS000161).

Q81. Did the NRC Staff issue any revisions to NUREG-1801 and the SRP-LR issued following Entergy's preparation and the Staff's review of the IPEC LRA?

A81. (ABC, JRS) Yes. In December 2010, the NRC Staff issued NUREG-1801, Rev. 2, and the SRP-LR, Rev. 2. These revisions were issued more than three years after Entergy submitted the IPEC LRA, 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, Rev. 1.

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Q82. With respect to the issues raised in this contention, what are the relevant changes in NUREG-1801, Rev. 2 and the SRP-LR, Rev. 2?

A82. (ABC, JRS, MAG, RGL) The revisions to NUREG-1801 and the SRP-LR include several changes that relate to the issues raised in this contention. The first is a change to the NRC Staff's discussion of the components to be evaluated for the effects of EAF identified in NUREG/CR-6260. Our testimony on contention NYS-26B/RK-TC-1B, incorporated by reference here, explains this change to NUREG-1801. *See* Entergy's NYS-26B/RK-TC-1B Testimony at § IV.B (ENT000679). Briefly, unlike the previous revision, NUREG-1801, Rev. 2 states that applicants should evaluate additional plant-specific component locations for EAF if they may be more limiting than those considered in NUREG/CR-6260. *See* NUREG-1801, Rev. 2, at X M1-2 (NYS00147C).

Q83. Did NUREG-1801, Rev. 2 and the SRP-LR, Rev. 2 include changes related to the potential for PWSCC?

A83. (ABC, JRS, BMG) Yes. The second change relates to the potential for PWSCC to affect PWR steam generator nickel alloy divider plate assemblies. Specifically, NUREG-1801, Rev. 2 states that for managing potential cracking due to PWSCC in nickel alloy steam generator divider plate assemblies and associated welds, "effectiveness of the chemistry control program should be verified to ensure that cracking due to PWSCC is not occurring." NUREG-1801, Rev. 2, at IV D1-3 (NYS00147B). This change was made because, as explained in the SRP-LR, Rev. 2 at 3.1-6 (NYS000161), there is operating experience showing cracking due to PWSCC in the nickel alloy ("Alloy 600") divider plate assemblies in foreign recirculating steam generators similar to domestic Westinghouse steam generators—primarily in French plants operated by Électricité de France ("EDF").

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The NRC Staff explained that, although divider plate cracks may not have a significant safety impact in and of themselves, cracks could propagate to adjacent steam generator components, which, unlike the divider plates, are part of the reactor coolant pressure boundary. *See* SRP-LR, Rev. 2 at 3.1-6 (NYS000161). Thus, the Staff now recommends that, if steam generator materials are potentially susceptible to cracking and crack propagation is possible, then the water chemistry program's effectiveness should be verified through inspections to ensure PWSCC is not occurring. *See id.* at 3.1-13.

Q84. Did the NRC Staff make any other changes to NUREG-1801, Rev. 2 and the SRP-LR, Rev. 2 regarding PWSCC?

A84. (ABC, JRS, BMG) Yes. In addition, NUREG-1801, Rev. 2 states that to manage potential PWSCC in nickel alloy steam generator tube-to-tube sheet welds, "the effectiveness of the water chemistry program should be verified [through a one-time inspection] to ensure cracking is not occurring." NUREG-1801, Rev. 2, at IV D1-8 (NYS00147B). As explained in the SRP-LR, Rev. 2, the Staff made this change because PWSCC could occur in steam generator nickel alloy tube-to-tubesheet welds exposed to reactor coolant. *See* SRP-LR, Rev. 2 at 3.1-6 (NYS000161).

Q85. Has the NRC explained the basis for its concern with respect to the need for inspections of steam generator tube-to-tubesheet welds?

A85. (ABC, JRS, BMG) Yes. Regulatory Issue Summary ("RIS") 2011-05 explains that the changes discussed in the preceding questions were added to "ensure adequate aging management of divider plate assemblies and to provide consistency between once-through steam generators and recirculating steam generators for tube to tubesheet welds." NRC Regulatory Issue

Summary 2011-05, Information on Revision 2 to the Generic Aging Lessons Learned Report for License Renewal of Nuclear Power Plants at 5 (July 1, 2011) (ENT000192).

Furthermore, we understand that tube-to-tubesheet weld cracking is an issue of longstanding interest to the NRC Staff and the U.S. nuclear industry, based primarily on operating experience with such effects in U.S. plants. *See*, *e.g.*, NRC Information Notice 2005-09, Indications in Thermally Treated Alloy 600 Steam Generator Tubes and Tube-to Tubesheet Welds (Apr. 7, 2005) ("IN 2005-09") (ENT000527) (discussing NRC concerns regarding tube-to-tubesheet weld cracking induced by residual stresses at a U.S. plant in 2004). Thus, although the propagation of PWSCC-induced cracks from the divider plate into the tube-to-tubesheet welds is a postulated concern, we understand that this is not the primary driver for the change to line item IV.D1.RP-385 in NUREG-1801, Rev. 2 (tube-to-tubesheet weld inspections or analysis).

Q86. Are there any other changes to NUREG-1801, Rev. 2 and the SRP-LR, Rev. 2 that are relevant to NYS-38/RK-TC-5 that you have not already mentioned?

A86. (ABC, JRS, TJG, RJD, RGL) Yes. With respect to managing the effects of aging on RVIs, NUREG-1801, Rev. 2 contains a new AMP (XI.M16A) that addresses PWR RVIs. *See* NUREG-1801, Rev. 2 at XI M16A-1 (NYS00147D). This new AMP relies on the implementation of EPRI report MRP-227. This new AMP has now been updated through interim staff guidance. *See* LR-ISG-2011-04 at 3 (ENT000641) (incorporating MRP-227-A into SRP-LR, Rev. 2 and NUREG-1801, Rev. 2); *see also* Entergy's NYS-25 Testimony at Q118 (ENT000616).

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Q87. Do Entergy Commitments 41, 42, 43, and 44 address these changes to NUREG-1801 and the SRP-LR?

A87. (ABC, JRS) As explained in Sections V.B.1 and V.C herein, Entergy's Commitments 41, 42, and 43 address these recent changes to NRC guidance. Commitment 44 is not related to any specific change to the NRC Staff's guidance in NUREG-1801 or the SRP-LR.

Q88. How does the NRC evaluate AMPs in light of NUREG-1801?

A88. (ABC, JRS) If an applicant has committed to implement an AMP approved in NUREG-1801, then a showing of consistency between the applicant's AMP and NUREG-1801 is equivalent to a demonstration of the ten required elements of an AMP, *see* response to Question 80, and a demonstration of reasonable assurance under the license renewal regulations. *See, e.g.*, NUREG-1801, Rev. 2 at XI M16A-1 (NYS00147D) (noting that the XI.M16A AMP "provide[s] reasonable assurance that the effects of age-related degradation mechanisms will be managed during the period of extended operation"); *see also* LR-ISG-2011-04 at A-2 (ENT000641) (incorporating MRP-227-A into AMP XI.M16A). The Commission has endorsed this process as the guidance provided in NUREG-1801 is based on extensive research and evaluation of operating experience derived from a comprehensive set of sources. *See* NUREG-1801, Rev. 2, at 2 (NYS00147A).

Q89. Do you agree with Intervenors' claim that Entergy "assumes that a commitment to develop a program in the future whose goal is to meet the requirements of the regulations and to follow the guidance in GALL is legally sufficient"? Intervenors' Revised SOP at 46 (NYS000531).

A89. (JRS, ABC) No. As we will show with respect to each of the commitments challenged in this contention, the record does not support a conclusion that Entergy somehow

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assumes or has deferred development of an AMP for purposes of compliance with license renewal regulations. All required AMPs for IP2 and IP3 at issue in this contention are fully developed, documented and made available to Intervenors.

V. ENTERGY'S LICENSE RENEWAL APPLICATION, INCLUDING THE COMMITMENTS THEREIN, PROVIDES SUFFICIENT INFORMATION TO SUPPORT A FINDING OF REASONABLE ASSURANCE THAT ENTERGY'S AMPs ARE CONSISTENT WITH NUREG-1801 AND THAT THE EFFECTS OF AGING WILL BE ADEQUATELY MANAGED

A. The NRC's Reliance on Applicant Commitments to Show that the Effects of Aging Will Be Adequately Managed Is Well Established and Fully Consistent with the Governing Regulations in 10 C.F.R. Part 54

Q90. What are Intervenors' general criticisms of Entergy's LRA commitments, in light of their interpretation of the requirements of 10 C.F.R. Part 54?

A90. (JRS, ABC) As it relates to NYS-38/RK-TC-5, Intervenors' general criticism of the LRA is that it "does not contain (1) sufficient information, (2) adequate programs, and (3) enforceable, binding commitments concerning the aging of certain components." *See* Intervenors' Revised SOP at 1 (NYS000531). In particular, Intervenors assert that commitments made in docketed licensing correspondence are not necessarily binding and that the NRC Staff routinely fails to monitor and track licensee commitments. *See id.* at 51-57.

Q91. Please summarize your response to Intervenors on the issue of the enforceability of regulatory commitments.

A91. (JRS, ABC) Contrary to Intervenors' assertions and as discussed further below, commitments are tracked by licensees and monitored and inspected by the NRC Staff. This applies equally to commitments made during operation under Part 50 or to those made for license renewal under Part 54. In fact, once a renewed license is issued, license renewal commitments become part of the CLB, which is enforced by the NRC under its ongoing Part 50 oversight process. *See* 10 C.F.R. § 54.33.

Regardless of how a licensee's commitment is documented—in a license condition, in the Updated Final Safety Analysis Report ("UFSAR"), or in docketed correspondence—the NRC Staff can take enforcement action if a licensee fails to meet its commitments. Depending on the circumstances, the NRC can take enforcement action through a notice of violation or a notice of deviation. Further, licensees may only alter commitments through formal commitment management processes that, when appropriate, require prior approval from or notice to the NRC. The commitment change process is well-established. Prior NRC permission for commitment changes may be required depending on the safety significance and timing of the change. This process is governed by the clear criteria established in 10 C.F.R. § 50.59 and in the NRC-approved guidance in NEI 99-04, Guidelines for Managing NRC Commitment Changes (July 1999) ("NEI 99-04") (ENT000534). Thus, we disagree with Intervenors' argument that applicant commitments are unenforced or unenforceable.

Further, to the extent the Intervenors imply that Entergy will intentionally seek to avoid or evade its commitments by, for example, suggesting that Entergy will "relax[]" commitments without following required processes or notifying the NRC, *see* Intervenors' Revised SOP at 53 (NYS000531), such arguments are pure unsupported speculation.

Q92. Can licensee commitments provide an adequate basis for the NRC to make its required findings in evaluating an LRA?

A92. (JRS, ABC) Yes. Licensee commitments are a well-established and essential mechanism for ensuring that licensees *implement* their AMPs in a timely and effective manner. *See, e.g.*, Final Rule: Nuclear Power Plant License Renewal, 56 Fed. Reg. at 64,946. They are fully authorized and contemplated by 10 C.F.R. 54.29(a). *See* Response to Question 68. In fact, from our experience, all license renewal applicants rely on commitments to demonstrate reasonable assurance.

Q93. Are there different categories of commitments, and when is each category used?

A93. (JRS, ABC) Yes. Licensee commitments—whether made in the license renewal context or otherwise—are categorized by the NRC in one of three different ways. First, some commitments are captured in license conditions that are written into the facility operating license. *See* SER at 1-21 to 1-22 (NYS00326A). License conditions are typically reserved for items of high regulatory or safety significance. *See* LIC-105 at 4 (ENT000705).

Second, commitments can be included in the UFSAR. *See*, *e.g.*, NUREG-2101, Safety Evaluation Report Related to the License Renewal of Salem Nuclear Generating Station at 3-276 (June 2011) ("NUREG-2101") (ENT000536) ("The staff finds the applicant's proposal acceptable because the applicant provided the appropriate commitment in the UFSAR supplement").

Third, some commitments, such as those contained in a licensee response to an NRC generic communication, in a letter supporting a license amendment request, or in response to an RAI, may be made to NRC in writing but may not be captured in either a license condition or the UFSAR—these are referred to as "regulatory commitments." *See generally* LIC-105 at 1 (ENT000705); *see also id*. at 13 (noting that safety concerns or regulatory nonconformances, such as a failure to comply with a commitment, can lead to enforcement action). This regulatory hierarchy for handling commitments is described in SECY-00-45, Acceptance of NEI 99-04, "Guidelines for Managing NRC Commitments" (Feb. 22, 2000) (ENT000538).

As discussed below, the NRC has the authority and regulatory mechanisms to enforce commitments in any of these three categories.

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Q94. Is there any basis to elevate the metal fatigue commitments (43, 44, and 49) to license conditions?

A94. (ABC, NFA, JRS) No. The technical reviews in Commitments 43 and 49 are complete, as we mentioned in our response to Question 69, and therefore the issue is moot for these two commitments. Commitment 44 is a documentation issue that has been resolved on a generic basis by the NRC Staff. *See* NRC, Safety Evaluation Report, "Topical Report on ASME Section III Piping and Component Fatigue Analysis Utilizing the WESTEMS™ Computer Code" at 16 (WCAP-17577, Revision 2) (undated) ("SER for WCAP-17577") (ENT000687). In any event, EAF, in general, is not an issue of high regulatory or safety significance. *See* Memorandum from A. Thadani, NRR, to W. Travers, EDO, "Closeout of Generic Safety Issue 190, 'Fatigue Evaluation of Metal Components for 60-year Plant Life,'" at 1 (Dec. 26, 1999) ("GSI-190 Closeout Memorandum") (ENT000190). Therefore, a license condition in lieu of these regulatory commitments would not be appropriate.

Q95. Is there any basis to elevate the steam generator commitments (41 and 42) to license conditions?

A95. (ABC, NFA, JRS, BMG, TJG) No. As further explained in Section V.C.3, below, EPRI has conducted extensive studies on the potential for PWSCC in steam generator divider plates and tube-to-tubesheet welds and concluded that these issues are "*not a safety concern*" for the U.S. fleet. *See* EPRI 2014 Report at 1-2 (NYS000544A) (emphasis in original). Since this is not an issue of high regulatory or safety significance, a license condition in lieu of Commitments 41 and 42 would be inappropriate.

Q96. How are License Conditions enforced?

A96. (JRS, ABC) A licensee commitment that is captured as a condition to the renewed license is no longer a commitment; it becomes part of the license. If a licensee fails to meet this obligation, then the NRC can take enforcement action for a direct violation of the license. *See* 10 C.F.R. § 2.201(a) ("In response to an alleged violation of . . . the conditions of a license . . . , the Commission may serve on the licensee or other person subject to the jurisdiction of the Commission a written notice of violation"). The terms of a license condition can only be changed through a license amendment. *See* 10 C.F.R. § 50.90. License conditions are often described as "legally" binding, but this label does not mean that other types of commitments are not binding or that the NRC Staff cannot enforce them.

Q97. How are commitments in the UFSAR enforced?

A97. (JRS, ABC) If there is a failure to comply with a commitment that is written into the UFSAR, including license renewal commitments, then the NRC can take enforcement action in the form of either a notice of violation or notice of deviation. A notice of violation would be issued if failure to implement the commitment resulted in non-compliance with a NRC regulation, such as 10 C.F.R. Part 50, Appendix B, Criteria XVI (corrective action requirements). *See* 10 C.F.R. § 2.201(a) (authorizing the issuance of notices of violation for failure to comply with the provisions of "this chapter," *i.e.*, the NRC's regulations in 10 C.F.R.). If no violation of an NRC regulation is identified, then the NRC can issue a notice of deviation, which is processed as a non-escalated enforcement action. *See* NRC Enforcement Manual, Rev. 9, at 111-112 (Sept. 9, 2013) (ENT000706).

Changes to commitments that have been incorporated into the UFSAR can be made by a licensee through the well-established process in 10 C.F.R. § 50.59. Briefly, licensees can only

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change commitments in the UFSAR without prior NRC approval if they meet the criteria in Section 50.59. If not, the licensee must obtain NRC approval of the change via a license amendment request. *See* 10 C.F.R. §§ 50.90, 50.91, 50.92. Changes that licensees make to the UFSAR under Section 50.59 must be reported to the NRC and are subject to NRC inspection. *See* 10 C.F.R. § 50.59(d)(2). If changes made by the licensee fail to meet the criteria in Section 50.59, then the NRC can take enforcement action in the form of a notice of violation.

Q98. How are Regulatory Commitments enforced?

A98. (JRS, ABC) If a licensee fails to comply with a written regulatory commitment to the NRC that has not been captured in a license condition or incorporated into the UFSAR, then the NRC can still take enforcement action by issuing the licensee a notice of deviation. *See* LIC-105 at 13 (ENT000705). Regulatory commitments can be changed by the licensee under the licensee's commitment management procedures. Entergy manages these types of commitments consistent with NRC-approved guidance in NEI 99-04. *See* EN-LI-110, Rev. 5, Entergy Nuclear Management Manual, Commitment Management Program at 3 (Jan. 2012) (ENT000541); *see also* NRC Regulatory Issue Summary 2000-17, Managing Regulatory Commitments Made by Power Reactor Licensees to the NRC Staff at 2 (Sept. 21, 2000) (ENT000542); LIC-105 at 3-14 (ENT000705). Changes made in commitments under the licensee's administrative process consistent with NEI 99-04 are generally reported to the NRC annually or along with FSAR updates required by 10 CFR § 50.71(e). *See* NEI 99-04 §§ 4-5 (ENT000534). Changes to license renewal commitments that are credited in the SER as the basis for NRC Staff safety conclusions are reported to the NRC. *See id.* at 9-10.

Q99. The Intervenors claim that the established methods for changing commitments in the FSAR and regulatory commitments (as opposed to license conditions) thwart the

hearing process and make commitments meaningless and unenforceable. *See* Intervenors' Revised SOP at 51-54, 56-57 (NYS000531). Please respond to these claims.

A99. (ABC, JRS) As we just explained, there are long-established NRC-approved methods—under Section 50.59 or NEI 99-04—that allow licensees to change commitments when necessary and technically justified, but this does not render those commitments "unenforceable." Instead, the NRC Staff is fully authorized to take enforcement action against licensees who violate commitments or who change their commitments without following the NRC's regulatory requirements. *See* Response to Question 98. Further, as we explain in the same response, contrary to Intervenors' statements, *see* Intervenors' Revised SOP at 53 (NYS000531), licensees must provide periodic notice to the NRC of changes to commitments and maintain a list of commitment changes that do not meet the criteria for reporting to the NRC. This list is subject to NRC inspection. This process applies not just to IPEC or license renewal, but to all operating plants regulated by the NRC under 10 C.F.R. Part 50.

Q100. Does the NRC inspect license renewal commitments?

A100. (JRS, ABC) Yes. NRC Inspection Manual, Manual Chapter 2516, Policy Guidance for the License Renewal Inspection Program (Aug. 13, 2013) ("MC 2516") (ENT000657) provides policy and guidance for review and inspection activities associated with the NRC License Renewal Inspection Program ("LRIP"). Under MC 2516, the LRIP provides "guidance for the inspection of license renewal programs, documentation, and other activities necessary for the staff to assess whether an applicant's LRA, AMPs, implementation activities, and on-site documentation provide reasonable assurance that the effects of aging will be adequately managed consistent with the CLB during the period of extended operation." MC 2516 at 2 (ENT000657). MC 2516 identifies inspections to be performed prior to the PEO. Also, the

NRC conducts triennial audits of all licensee commitments and commitment management process. *See* LIC-105 at 6 (ENT000705). As a practical matter, a hearing is not a substitute for the NRC Staff's enforcement and inspection activities.

Q101. Please summarize the license renewal inspections specified under MC 2516.

A101. (JRS, ABC) During the Part 54 license renewal review process, the NRC Staff conducts inspections using Inspection Procedure 71002, *see* NRC Inspection Manual, Inspection Procedure 71002, License Renewal Inspection (Nov. 23, 2011) ("IP 71002") (ENT000543), and MC 2516 (ENT000657). IP 71002, which recognizes that the applicant may make changes to the plant or the current licensing basis while the NRC is reviewing the LRA, specifies inspections to ensure that committed tasks are being tracked both prior to and during the PEO. *See* IP 71002 at 4 (ENT000543).

After approval of a renewed license and prior to the PEO, MC 2516 specifies that inspections be performed in accordance with Inspection Procedure 71003 to verify that license renewal commitments and aging management programs are being implemented in accordance with 10 C.F.R. Part 54, the NRC's SER, and the UFSAR Supplement. *See* MC 2516 at 6-8 (ENT000657); *see also* NRC Inspection Manual, Inspection Procedure 71003, Post-Approval Site Inspection for License Renewal (Feb. 25, 2013) ("IP 71003") (ENT000703). MC 2516 also specifies that, for applicants with timely renewal applications, inspections be performed in accordance with IP 71013. *See* MC 2516 at 8 (ENT000657); *see also* NRC Inspection Manual, Inspection Procedure 71013, Site Inspection for Plants with a Timely Renewal Application (Sept. 25, 2013) ("IP 71013") (ENT000704).

Commitments made in the LRA are also subject to NRC oversight, inspection, and enforcement. *See* LIC-105 at 6 (ENT000705) (specifying triennial audits of licensee commitment management programs). The NRC makes no exception to this process for IPEC. In fact, these inspections have already been completed for IP2. *See generally* IP2 LR Commitment Inspection (ENT000695).

Q102. Do the inspections you previously discussed include NRC inspections of commitment tracking and implementation?

A102. (ABC, JRS) Yes. A stated objective of IP 71003 is to "verify [that] . . . regulatory commitments . . . are implemented and/or completed in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 54, 'Requirements for the Renewal of Operating Licenses for Nuclear Power Plants.'" IP 71003 at 1 (ENT000703). IP 71003 specifies that post-renewal inspections will verify that "the licensee adequately evaluated, and reported when necessary, changes to regulatory commitments from the SER for license renewal in accordance with NEI 99-04 [and] changes to AMPs, TLAAs and other license renewal activities incorporated as part of the UFSAR supplement in accordance with 10 CFR 50.59." *Id*. at 2. IP 71003 also specifies that considerations for the selection of commitments to be inspected should include, among other things, risk significance and results of one-time inspections. *See id*. at 3. The procedure further states that:

> The [inspection] sample should include a review of selected regulatory commitments which were accepted by the staff during the course of the license renewal application review and which describe a modification or enhancement to a program or future actions necessary for compliance with 10 CFR Part 50 or 10 CFR Part 54. . . . The inspection team should determine there is reasonable assurance the commitment tracking program is effective.

Id. at 4-5.

This inspection regime shows that the NRC clearly recognizes the importance of such commitments in the license renewal process and effectively managing the effects of aging consistent with 10 C.F.R. Part 54. If these inspections show that license renewal commitments are not being met, then the NRC can take appropriate enforcement action as described above. The NRC also publishes its inspection reports, providing the public the opportunity to review those reports, and, if appropriate, file petitions under 10 C.F.R. § 2.206.

Q103. Has a former NRC Chairman summarized the NRC's process for inspecting the fulfillment of license renewal commitments?

A103. (ABC, JRS) Yes. In 2012, then-Chairman Macfarlane responded to a letter from Congressman Markey raising various questions about license renewal-related commitments. *See* Letter from A. Macfarlane, Chairman, NRC, to E. Markey, U.S. Congressman, Encl. at 2-14 (July 12, 2012) ("Macfarlane Letter") (ENT000544). The Chairman noted that the NRC inspects the fulfillment of license conditions and commitments associated with license renewal under IP 71003. *See id.*, Encl. at 3. She also described the enforcement process, consistent with our testimony above:

> If inspectors identify inadequacies in the implementation of commitments before they are required to be in place, the findings are identified to the licensee, documented in their inspection report, and are subject to re-inspection. The licensee is obligated to enter the findings in their corrective action program to be tracked and corrected. If inspectors identify instances where the licensee has not satisfactorily completed commitments after they are required to be in place, the licensee is subject to a violation against 10 CFR Part 50.

Id.

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Q104. Intervenors claim that an audit report by the NRC's Office of Inspector General ("OIG") indicates that the NRC Staff routinely fails to monitor licensee commitments and, as a result, Entergy's commitments cannot support a reasonable assurance finding. *See* Intervenors' Revised SOP at 55-57 (NYS000531) (citing OIG-A-17, Audit of NRC's Management of Licensee Commitments (Sept. 19, 2011) ("2011 OIG Audit Report") (NYS000181)). Do you agree?

A104. (JRS, ABC) No. The OIG Audit Report, which is now four years old, contains no statements that support Intervenors' claim. The report merely recommends that the Staff strive for greater consistency in implementing commitment management audits, achieve a better institutional understanding of the definition and use of commitments, and improve its tracking of commitments. *See* 2011 OIG Audit Report at iii, 5, 22-23 (NYS000181). The report did not conclude, as Intervenors suggest, that licensee or applicant commitments are not binding or enforceable, or that all license renewal commitments must be elevated to license conditions. *Compare* Intervenors' Revised SOP at 56 (NYS000531) (listing the report's actual conclusions) *with id.* at 56-57 (leaping to the conclusion that license renewal commitments "are *not* binding or enforceable" (emphasis in original)). On the contrary, the OIG concluded that "NRC commitments are a valuable regulatory tool," play a "key role" in facilitating the agency's safety decision-making process, and provide "additional assurance to the agency that a licensee action will not adversely affect the safe operation of the plant." *See* 2011 OIG Audit Report at 22 (NYS000181).

Q105. Have the OIG findings been resolved and, if so, has the OIG reviewed the Staff's actions?

A105. (ABC, JRS) Yes, to both. Since issuance of the 2011 OIG Audit Report, the NRC Staff has resolved the OIG's issues. *See* Memorandum from S. Dingbaum, Assistant Inspector Gen. for Audits, NRC, to M. Satorius, Exec. Dir. for Operations, NRC, "Status of Recommendations: Audit of NRC's Management of Licensee Commitments (OIG-11-A-17)" at 1 (Nov. 25, 2013) (ENT000707) ("[a]ll recommendations related to this report are now closed."); *see also* Macfarlane Letter, Encl. at 4-7 (ENT000544) (providing the status of Staff action on each of the recommendations in the OIG Report). The former Chairman's letter also noted that "the OIG audit report did not address commitments in the license renewal context. Thus, the OIG audit report's recommendations are not based upon OIG observations about the use of commitments in the license renewal process." *Id.*, Encl. at 4 (ENT000544).

Q106. Citing an even earlier OIG report on license renewal, Dr. Hopenfeld has argued that a commitment to implement an AMP consistent with GALL is insufficient to demonstrate reasonable assurance because the Staff has been found not to conduct in-depth technical review of LRAs. *See* Hopenfeld Rebuttal at 8 (citing OIG-07-A-15, Audit of NRC's Licensee Renewal Program (Sept. 6, 2007) ("2007 OIG Report") (RIV000116). Is this assertion correct?

A106. (ABC, JRS) No. First, Dr. Hopenfeld's claim attacks NUREG-1801 and the NRC Staff's license renewal review process, as established and endorsed by the Commission. Second, Dr. Hopenfeld has mischaracterized the 2007 OIG Report. *See id.* As the Commission itself explained shortly after the OIG issued its 2007 report on license renewal, "[T]he OIG did not determine, and we do not otherwise find, that past license renewal safety reviews were inadequate

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or that the license renewal review process requires a comprehensive revision. The OIG's recommendations do not undermine our general confidence in the Staff's safety review, and consequently we see no threat to the public health and safety or the common defense and security." *Oyster Creek*, CLI-08-23, 68 NRC at 465. In this proceeding, the NRC Staff's issuance of hundreds of RAIs, conduct of numerous audits and inspections, and preparation of the SER and two supplements thereto over the eight years that the IP2 and IP3 LRA has been under review are compelling evidence of the rigor and thoroughness of its safety review in this proceeding.

Q107. Intervenors claim that Entergy's commitments amount to mere promises that fall short of the establishing a record sufficient to support the Staff's reasonable assurance determination. *See* Duquette Testimony at 5 (NYS000372). Do you agree?

A107. (ABC, JRS) No, we do not agree. The commitments cited by Intervenors are not "mere promise[s]" and do not defer "definitive" safety findings by the Staff for post-hearing resolution. *See* Intervenors' Revised SOP at 33, 51 (NYS000531). Entergy has fully described the AMPs associated with the commitments cited in NYS-38/RK-TC-5, including the RVI AMP, the FMP, and the Water Chemistry Program in its LRA and the subsequent revisions thereto. Based upon its review of that information, the Staff found those AMPs to be complete and fully consistent with the corresponding NUREG-1801 programs. *See* SSER 1 at 6-1 (NYS000160) ("The staff concludes that the additional information provided by Entergy Nuclear Operations, Inc., does not alter the conclusions stated in the SER and that the requirements of 10 CFR 54.29(a) have been met."); SSER 2 at 3-26 (NYS000507) ("On the basis of its review of the applicant's RVI AMP, the staff concludes that the applicant has demonstrated that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(a)(3).").

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In Section V of this testimony, we demonstrate that the commitments challenged in this contention fully support the technical and regulatory adequacy of the IPEC LRA and the relevant AMPs. In fact, several of the commitments at issue have already been fully implemented by Entergy; specifically, Commitment 30 (for both units) and 42, 43, and 49 for IP2. In addition, Entergy has completed the technical reviews for Commitments 43 and 49 for IP3, which will be fully implemented prior to the IP3 PEO.

Overall, contrary to Intervenors' claims, these commitments are not mere promises to develop aging management activities at a later date. The commitments in question define specific aging management activities that demonstrate that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the IPEC CLB.

Q108. Has the Staff reviewed Entergy's AMPs as part of its review of the LRA?

A108. (JRS, ABC) Yes. The NRC Staff has verified the adequacy of Entergy's AMPs during its LRA review, which has included extensive RAIs and on-site audits. *See e.g.*, SER at 3-4 to 3-10 (NYS00326B); Audit Report For Plant Aging Management Programs and Reviews, Indian Point Nuclear Generating Unit Nos. 2 and 3 (Jan. 13, 2009) (ENT000041). Namely, the NRC Staff audited, reviewed, and evaluated the IPEC AMPs against corresponding AMPs in NUREG-1801. *See* SER at 3-4 to 3-10 (NYS00326B). The Staff also evaluated and assessed those AMRs or AMPs related to emergent issues, and AMPs that vary somewhat from NUREG-1801 or an NRC-approved precedent. *See id*. at 3-149 to 3-220 (NYS00326C); *id.* at 3-291 to 294 (NYS00326D).

Moreover, as discussed above, SSER 1, in particular, explains *why* Entergy's AMP revisions and commitments provide reasonable assurance that the effects of aging of the subject structures and components will be adequately managed throughout the PEO. *See, e.g.*, SSER 1 at 3-20 to 3-23 (NYS000160); *id.* at 4-1 to 4.3. In addition, as we will further discuss, SSER 2 shows that Entergy has completed Commitment 30 at IP2 and IP3, and Commitments 43 and 44 at IP2. *See* SSER 2 at A-11, A-14 (NYS000507). As to Commitment 30 (regarding RVIs), the Staff reviewed Entergy's RVI AMP and Inspection Plan in detail in SSER 2. *Id*. at 3-13 to 3-59.

Q109. Has the Staff reviewed Entergy's license renewal commitment implementation activities at IPEC?

A109. (JRS, ABC) Yes. Prior to the PEO, the NRC Staff reviews an applicant's or licensees' implementation of its AMPs, license conditions, and commitments associated with license renewal as part of its IP 71003 inspection process, and has already done so for IP2. *See* IP2 LR Commitment Inspection (ENT000695). As we have previously noted, Entergy understands that the NRC Staff will do so for IP3 as well, before it enters the PEO. *See* IP3 Inspection Plan, Encl. at 2 (ENT000701).

B. <u>Entergy's Commitments Related to Environmentally-Assisted Fatigue Support the Finding that the IPEC FMP Provides Reasonable Assurance that the Effects of Fatigue Will be Adequately Managed</u>

1. Commitments 43 and 49: Limiting Locations Review

Q110. Please describe Commitment 43.

A110. (ABC, NFA, JRS, RGL, MAG) In Commitment 43, Entergy committed to review the IPEC "design basis ASME Code Class 1 fatigue evaluations to determine whether the NUREG/CR-6260 locations that have been evaluated for the effects of the reactor coolant environment on fatigue usage are the limiting locations" for IPEC. *See* NL-11-032, Attach. 1, at 26 (NYS000151). If more limiting locations were identified, then Entergy also committed to evaluate the "most" limiting location "for the effects of the reactor coolant environment on fatigue usage." *Id*. Entergy agreed to implement Commitment 43 prior to the PEO. *Id*.

[redacted]

Q111. Did Entergy later submit a related commitment, regarding RVIs?

A111. (ABC, NFA, JRS) Yes. On May 7, 2013, Entergy submitted Commitment 49, which clarified that the limiting locations review would include RVI Components. *See* NL-13-052, Attach. 1 at 9 (NYS000501); SSER 2 at 3-52, A-15 (NYS000507). Specifically, Entergy committed to "[r]ecalculate each of the limiting CUFs provided in Section 4.3 of the LRA for the reactor vessel internals" prior to entering the PEO. NL-13-052, Attach. 2 at 20 (NYS000501).

Q112. Have you provided testimony on the technical and regulatory issues related to the implementation of Commitments 43 and 49?

A112. (ABC, NFA, JRS, RGL, MAG) Yes. Our testimony on Contention NYS-26B/RK-TC-1B addresses these matters in detail and is incorporated by reference here. *See* Entergy's NYS-26B/RK-TC-1B Testimony §§ IV, V.E (ENT000679). In summary, we explain that Westinghouse has conducted comprehensive new evaluations of all non-NUREG/CR-6260 IP2 and IP3 components with CLB CUF evaluations, including RVIs, and confirmed that CUF_{en} values for all limiting locations at IPEC are not projected to exceed 1.0 during the PEO, thereby demonstrating that Entergy will adequately manage the effects of aging as required by 10 C.F.R. §§ 54.21(a)(3) and (c)(1)(iii). *See id.*

Q113. Has Entergy completed the limiting locations review described in Commitments 43 and 49?

A113. (ABC, NFA, JRS, RGL, MAG) Yes. *See* Entergy's NYS-26B/RK-TC-1B Testimony § V.E (ENT000679). Briefly, Entergy completed an initial screening review to determine whether the NUREG/CR-6260 locations are the limiting locations for IPEC on November 9, 2012. *See* Westinghouse Calculation Note CN-PAFM-12-35 (NYS000510). This screening review included all ASME Class 1 design basis fatigue evaluations, and all RVI

[redacted]

components with CLB CUF fatigue evaluations, consistent with Commitment 43, as clarified in Commitment 49. *See id*. at 9-11. The screening review identified several locations that were potentially more limiting than those identified in NUREG/CR-6260. *See id*. at 9-10.

Westinghouse then completed a refined EAF evaluation for the IP2 locations identified in Westinghouse Calculation Note CN-PAFM-12-35 as potentially more limiting, including reactor coolant pressure boundary and RVI locations. *See* Westinghouse Calculation Note CN-PAFM-13-32, Rev. 1, Indian Point Unit 2 (IP2) and Unit 3 (IP3) Refined EAF Analyses and EAF Screening Evaluations (Aug. 19, 2013) ("Westinghouse Calculation Note CN-PAFM-13-32, Rev. 1") (NYS000511). Entergy has therefore fully implemented Commitment 43 and 49 for IP2. *See* Entergy, Commitment Closure Verification Form, LRC # 43 (Aug. 27, 2013) (ENT000708); Entergy, Commitment Closure Verification Form, LRC # 49 (Aug. 27, 2013) (ENT000709).

Westinghouse has also completed the underlying technical fatigue analysis for Commitments 43 and 49 at IP3. *See generally* Westinghouse Calculation Note CN-PAFM-13-32, Rev. 3 (ENT000683). Entergy will formally close Commitments 43 and 49 for IP3 before the PEO.

Q114. Do Drs. Lahey and Hopenfeld provide testimony regarding Commitments 43 and 49 on the record for this contention?

A114. (ABC, NFA, JRS, MAG, RGL) Yes. They have provided testimony on this issue in both 2012 and 2015, and generally contend that Entergy's limiting locations review did not fulfill Commitments 43 and 49 because it was not properly scoped, used non-conservative inputs and methods, and did not account for shock loads or combinations of aging mechanisms such as metal fatigue and irradiation embrittlement. *See generally* Lahey Testimony (NYS000374); Lahey Rebuttal (NYS000453); Revised Lahey Testimony (NYS000562); Hopenfeld Testimony

(RIV000102); Hopenfeld Rebuttal (RIV000134); Supplemental Hopenfeld Testimony (RIV000143); Supplemental Hopenfeld Report (RIV000144).

Q115. Does their 2015 testimony on these issues for this contention differ from their testimony for NYS-26B/RK-TC-1B?

A115. (ABC, NFA, JRS, RGL, MAG) Not substantively. In 2015, Dr. Lahey and Dr. Hopenfeld submitted substantively identical testimony in the two contentions on these issues. *See* Supplemental Hopenfeld Report (providing a single combined report that does not distinguish between NYS-26B/RK-TC-1B and NYS-38/RK-TC-5); *compare* Revised Lahey Testimony (NYS000562) *with* Revised Pre-filed Written Testimony of Dr. Richard T. Lahey, Jr. Regarding Consolidated Contention NYS-26B/RK-TC-1B (June 9, 2015) ("Revised Lahey Testimony on NYS-26B/RK-TC-1B") (NYS000530). Therefore, we have responded to most of Dr. Lahey's and Dr. Hopenfeld's claims regarding Commitments 43 and 49 in our testimony on NYS-26B/TK-TC-1B, which is incorporated by reference here. *See* Entergy's NYS-26B/RK-TC-1B Testimony §§ IV, V.E (ENT000679).

Q116. Did Dr. Lahey or Dr. Hopenfeld raise any particular claims in their testimony on this contention that are not addressed in your testimony on NYS-26B/RK-TC-1B? If so, please describe any such claims.

A116. (ABC, NFA, JRS, RGL, MAG) Yes, there are a few unique claims that the Intervenors' witnesses made in 2012 on this contention. Generally, these claims challenge the scope of components included in the limiting locations review. Thus, we address these unique claims, and the revised testimony submitted by Intervenors in 2015 on these same issues, in our testimony here.

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Q117. Turning to those claims, in his 2012 direct testimony on NYS-38/RK-TC-5, Dr. Hopenfeld described his opinions on the appropriate scope of the limiting locations review (which Entergy committed to perform through Commitments 43 and 49), including his view that the first step should be "selecting and listing all components that are susceptible to fatigue." Hopenfeld Testimony at 12 (RIV000102); *see also* Supplemental Hopenfeld Report at 25-27 (RIV000144). Please respond to Dr. Hopenfeld's statements regarding the scope of Commitment 43.

A117. (ABC, NFA, JRS, RGL, MAG) The design basis fatigue review envisioned by Dr. Hopenfeld is not required. The CUFs that are in the CLB for IPEC are all listed in the LRA. *See* LRA at 4.3-9 to 4.3-17, tbls.4.3-3 (IP2 RPV), 4.3-4 (IP3 RPV), 4.3-5 (IP2 RVIs), 4.3-6 (IP3 RVIs), 4.3-7 (IP2 pressurizer), 4.3-8 (IP3 pressurizer), 4.3-9 (IP2 steam generators), 4.3-10 (IP3 steam generators), 4.3-11 (IP2 control rod drive mechanisms), 4.3-12 (IP3 control rod drive mechanisms). These locations, selected at the time of the SPU or earlier, are now part of the IP2 and IP3 CLB. Therefore, there is no need, for purposes of this license renewal proceeding, to reconsider all plant components and identify a new set of CUF locations as called for by Dr. Hopenfeld.

To the extent Dr. Hopenfeld is requesting EAF analyses of primary plant components beyond those with CLB CUF evaluations, such claims are a challenge to the CLB and are, accordingly, beyond the scope of this proceeding. Under 10 C.F.R. § 54.21(c)(1)(iii), the FMP is intended to manage the effects of aging addressed by fatigue TLAAs that are part of the CLB for IP2 and IP3. *See* NUREG-1801, Rev. 1 at X M-1 (NYS00146C) ("In order not to exceed the *design limit* on fatigue usage . . . .") (emphasis added); *see also id*. at X-iii (showing the FMP as an AMP intended to manage the effects of aging associated with a TLAA under Section

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54.21(c)(1)(iii)). As previously noted, the CUFs resulting from CLB fatigue TLAAs for IPEC are all listed in the LRA.

We also note that the ASME Code Section XI inservice inspection program provides further assurance of the continued structural integrity of RCS components, including inspections to manage cracking due to fatigue, regardless of whether a component has a CLB fatigue analysis or not. *See* LRA at B-63 to B-68 (ENT00015B). Intervenors do not challenge the adequacy of that program in this contention.

Q118. Is there a technical reason why the limiting locations for fatigue need not change for license renewal?

A118. (ABC, NFA, JRS, RGL, MAG) Yes. The fatigue life of a component is influenced by: (1) the component geometry, (2) the component material, or (3) the method of operation, which could affect the applied loads. *See* ASME Boiler & Pressure Vessel Code, Section III, Article NB-3000, "Design" §§ 3200, 3650 (1989) ("ASME Code, NB-3000") (NYS000349). None of these parameters is affected by simply increasing the service time of the component from 40 to 60 years—rather, a component's CUF is affected by the number cycles, and cycles are specifically addressed in the fatigue calculations. *See id*. § 3222.4(e). Thus, the locations identified as limiting CUF locations during the initial 40 years of service do not change simply as a result of increased service time (*i.e.*, from 40 to 60 years). One or more of the aforementioned CUF-related parameters must change.

Q119. Dr. Hopenfeld argues that Entergy's testimony, *i.e.*, that the existing CLB CUF analysis locations were selected because they were the most limiting, is unsupported. *See* Hopenfeld Rebuttal at 14, 17-18 (RIV000134). More specifically, Dr. Hopenfeld makes four assertions: (1) that a component with a limiting CUF may not be limiting under the

[redacted]

CUF_{en} analysis; (2) that some components were designed to ANSI B31.1, with no CUF; (3) that some components may be subject to the combined effects of fatigue and PWSCC, in which case the F_{en} methodology is inapplicable; and (4) that the original CLB CUF calculations assumed nominal wall thickness, but in fact, there are large local variations in wall thicknesses. *See id.*; *see also* Supplemental Hopenfeld Report at 25-27 (RIV000144). How do you respond?

A119. (ABC, NFA, JRS, RGL, MAG) Dr. Hopenfeld's Item (1) is mere speculation, as the limiting locations review considered all CLB CUF locations and considered reactor coolant environmental effects. *See* Westinghouse Calculation Note CN-PAFM-12-35 at 8 (NYS000510) ("Westinghouse has performed EAF screening evaluations for IP2/IP3 that consider all components with a fatigue usage factor listed in the IP2/IP3 LRA."); *id*. at 20 (explaining the F_{en} application methodology).

Dr. Hopenfeld's Item (2) challenges the adequacy of the CLB, not Entergy's evaluation of the CUF TLAA in the LRA. As we have explained, Entergy's review under Commitment 43 (and 49) includes all components at IP2 and IP3 with a CLB CUF calculation, so a screening CUF_{en} was prepared and evaluated for all relevant locations. *See id.* at 8, 20. As we have explained in response to Question 117, above, Dr. Hopenfeld's demand that Entergy review components without a CLB CUF is a challenge to the CLB. Moreover, the CUFs included in the limiting locations review did include components originally designed to ANSI B31.1 standards, to identify the limiting Class 1 piping locations. *See* Westinghouse Calculation Note CN-PAFM-12-35 § 5.2 (NYS000510).

Item (3) also is misplaced speculation. A fatigue analysis is not intended to address PWSCC—it is intended to provide reasonable assurance that a component will not experience fatigue cracking. *See* Entergy's NYS-26B/RK-TC-1B Testimony at Q66 (ENT000679). Instead, the effects of aging due to PWSCC on susceptible primary plant components, including dissimilar metal welds, are monitored through several inspection programs which address potential cracking (regardless of the underlying aging mechanism), including the ISI Program, the Nickel Alloy Inspection Program, the Reactor Vessel Head Penetration Inspection Program, the Steam Generator Integrity Program, and the RVI AMP. *See* LRA at B-63 to B-68, B-74 to B-77, B-109 to B-110, B-118 to B-120 (ENT00015B); NL-12-037, Letter from F. Dacimo, Entergy, to NRC Document Control Desk, "License Renewal Application – Revised Reactor Vessel Internals Program and Inspection Plan Compliant with MRP-227-A," Attach. 1 (Feb. 17, 2012) ("NL-12-037") (NYS000496).

Finally, Item (4) is likewise unfounded speculation and premised on faulty logic. Dr. Hopenfeld relies on flow-accelerated corrosion ("FAC") program carbon steel component inspection data to allege deficiencies in EAF evaluation processes for primary plant components that are stainless steel or clad with stainless steel. Hopenfeld Rebuttal at 18 (RIV000134) (citing Hearing Transcript at 1877-1879 (October 17, 2012)). However, EAF (*environmentally-assisted* fatigue) evaluations are relevant to components subject to the reactor coolant environment. *See* GSI-190 Closeout Memorandum at 2 (ENT000190). But such components are not subject to FAC (as Dr. Hopenfeld admits). *See* Hopenfeld Rebuttal at 13 ("stainless steel is not susceptible to wall thinning by FAC")).

In any event, the use of design geometry in the Indian Point fatigue analyses for primary plant components is acceptable for large-bore piping because, at the time of installation, those components were inspected to confirm they meet design requirements. Moreover, for all components, potential variations in wall thicknesses are accounted for in the stress indices and

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design factors in the ASME Code. *See* ASME Code, NB-3000 §§ 3100, 3680 (NYS000349). Deviations in dimensions from the ASME-required wall thicknesses for primary equipment or non-standard piping (like the RCL piping) would have been recorded and evaluated in the CLB evaluations per ASME requirements. *See id*. §§ 3100, 3680.

Q120. In 2012, Dr. Hopenfeld and Dr. Lahey both criticized Entergy's Commitment 43 as failing to provide results in time to be tested at a hearing. Dr. Lahey asserts that the results of that review must be "tested and resolved in these ASLB hearings." Lahey Testimony at 30 (NYS000374). Similarly, Dr. Hopenfeld states that "it was not appropriate for the NRC Staff to accept Entergy's vague commitment to determine at some point in the future what additional locations must be analyzed." Hopenfeld Testimony at 11 (RIV000102); *see also id.* ("[a]n actual analysis to determine the most limiting locations must be performed *before* a determination is made about license renewal."). How do you respond?

A120. (ABC, NFA, JRS, RGL, MAG) Dr. Lahey and Dr. Hopenfeld's concerns are moot. As previously explained, Entergy has completed its limiting locations review for IP2 and IP3, in accordance with Commitments 43 and 49. *See* Westinghouse Calculation Note CN-PAFM-13-32, Rev. 1 (NYS000511) (documenting completion of IP2 evaluations); Westinghouse Calculation Note CN-PAFM-13-32, Rev. 3 (ENT000683) (documenting completion of IP3 evaluations); Westinghouse Calculation Note CN-PFAM-12-35 (NYS000510) (documenting completion of screening evaluations).

2. Commitment 44: Documenting Peak Editing in WESTEMS^{TM} Fatigue Evaluations

Q121. Please describe Commitment 44?

A121. (ABC, NFA, JRS, RGL, MAG) In Commitment 44, Entergy committed to "include written explanation and justification of any user intervention in future evaluations using the WESTEMS 'Design CUF' module." NL-11-032, Attach. 2 at 18 (NYS000151). Entergy originally agreed to implement Commitment 44 within 60 days of issuance of the renewed operating license, *see id.*, but later amended that commitment and agreed to implement Commitment 44 prior to the PEO for consistency with other similar commitments. *See* NL-11-101, Letter from F. Dacimo, Entergy, to NRC Document Control Desk, "Clarification for Request for Additional Information (RAI), Aging Management Programs," Attach. 1, at 2 (Aug. 22, 2011) (NRC000156).

Q122. Have you provided testimony on the technical and regulatory issues related to the implementation of Commitment 44?

A122. (ABC, NFA, RJD, JRS, RGL, TJG) Yes. Our testimony on Contention NYS-26B/RK-TC-1B addresses these matters in detail and is incorporated by reference here. *See* Entergy's NYS-26B/RK-TC-1B Testimony § V.B.2 (ENT000679). In summary, we explain that the elimination of redundant peaks and valleys in ASME Code analyses, whether prepared by hand or using computer software, is conducted according to ASME Code Rules; that the NRC has generically resolved its preliminary concerns with "user intervention."

Q123. Has Entergy implemented Commitment 44?

A123. (ABC, NFA) Entergy has implemented this commitment at IP2 by changing its FMP to incorporate this requirement. *See* Entergy, Commitment Closure Verification Form, LRC

[redacted]

44 (June 19, 2013) (ENT000710). The commitment will be formally implemented at IP3 prior to the PEO.

(MAG, NFA) [redacted]

Q124. Do Drs. Lahey and Hopenfeld provide testimony regarding Commitment 44 on the record for this contention?

A124. (ABC, NFA, JRS, MAG, RGL) Yes. They have provided testimony on this issue in both 2012 and 2015. *See generally* Lahey Testimony (NYS000374); Lahey Rebuttal (NYS000453); Revised Lahey Testimony (NYS000562); Hopenfeld Testimony (RIV000102); Hopenfeld Rebuttal (RIV000134); Supplemental Hopenfeld Testimony (RIV000143); Supplemental Hopenfeld Report (RIV000144). In general, they characterize the selection of inputs for EAF evaluations (such as heat transfer coefficients and loads) as "user intervention," and allege that Entergy has not disclosed all "user intervention" in its EAF evaluations to date.

Q125. Does their 2015 testimony on Commitment 44 in this contention differ from their testimony on NYS-26B/RK-TC-1B?

A125. (ABC, NFA, RJD, JRS, RGL, TJG) Not substantively. In 2015, Dr. Lahey and Dr. Hopenfeld submitted substantively identical testimony in the two contentions on these issues. *See* Supplemental Hopenfeld Report (providing a single combined report that does not distinguish between NYS-26B/RK-TC-1B and NYS-38/RK-TC-5); *compare* Revised Lahey Testimony (NYS000562), *with* Revised Lahey Testimony on NYS-26B/RK-TC-1B (NYS000530).

Therefore, we have responded to the vast majority Dr. Lahey's and Dr. Hopenfeld's claims regarding Commitment 44 in our testimony on NYS-26B/TK-TC-1B, which is incorporated by reference here. *See* Entergy's NYS-26B/RK-TC-1B Testimony § V (ENT000679).

Q126. Did Dr. Lahey or Dr. Hopenfeld raise any claims in their 2012 testimony on Commitment 44 that are not addressed in your testimony on NYS-26B/RK-TC-1B?

A126. (ABC, NFA, JRS, RGL, MAG) Yes, Dr. Lahey raises one claim—regarding the scope of Commitment 44—which has only been raised in this contention.

Q127. Turning to that unique claim, Dr. Lahey alleges that, while Entergy agreed to disclose user intervention for "future" evaluations, "nothing was said about the previous WESTEMS evaluations that were done for IP-2 & IP-3 and the affect [sic] that user interventions had on those CUF_{en} results." Lahey Testimony at 26 (NYS000374); *see also* Revised Lahey Testimony at 77 (NYS000562). How do you respond?

A127. (MAG) As previously explained in response to Question 123, above, [redacted]

Q128. In their 2015 testimony, do the Intervenors' witnesses raise any new challenges regarding peak editing in WESTEMS™?

A128. (NFA, ABC, JRS, RGL, MAG) Not specifically. But to the extent they raise more general challenges to the use of engineering judgment in fatigue evaluations performed for IPEC license renewal, we address those additional claims in our testimony on NYS-26B/RK-TC-1B, which is incorporated by reference here. *See generally* Entergy's NYS-26B/RK-TC-1B Testimony §§ V.B.2, V.C, and V.D.8 (ENT000679).

[redacted]

C. Entergy's Commitments 41 and 42 Support the Finding that the Effects of PWSCC in Steam Generator Components Will be Adequately Managed

1. Technical Background on PWSCC in Steam Generator Components

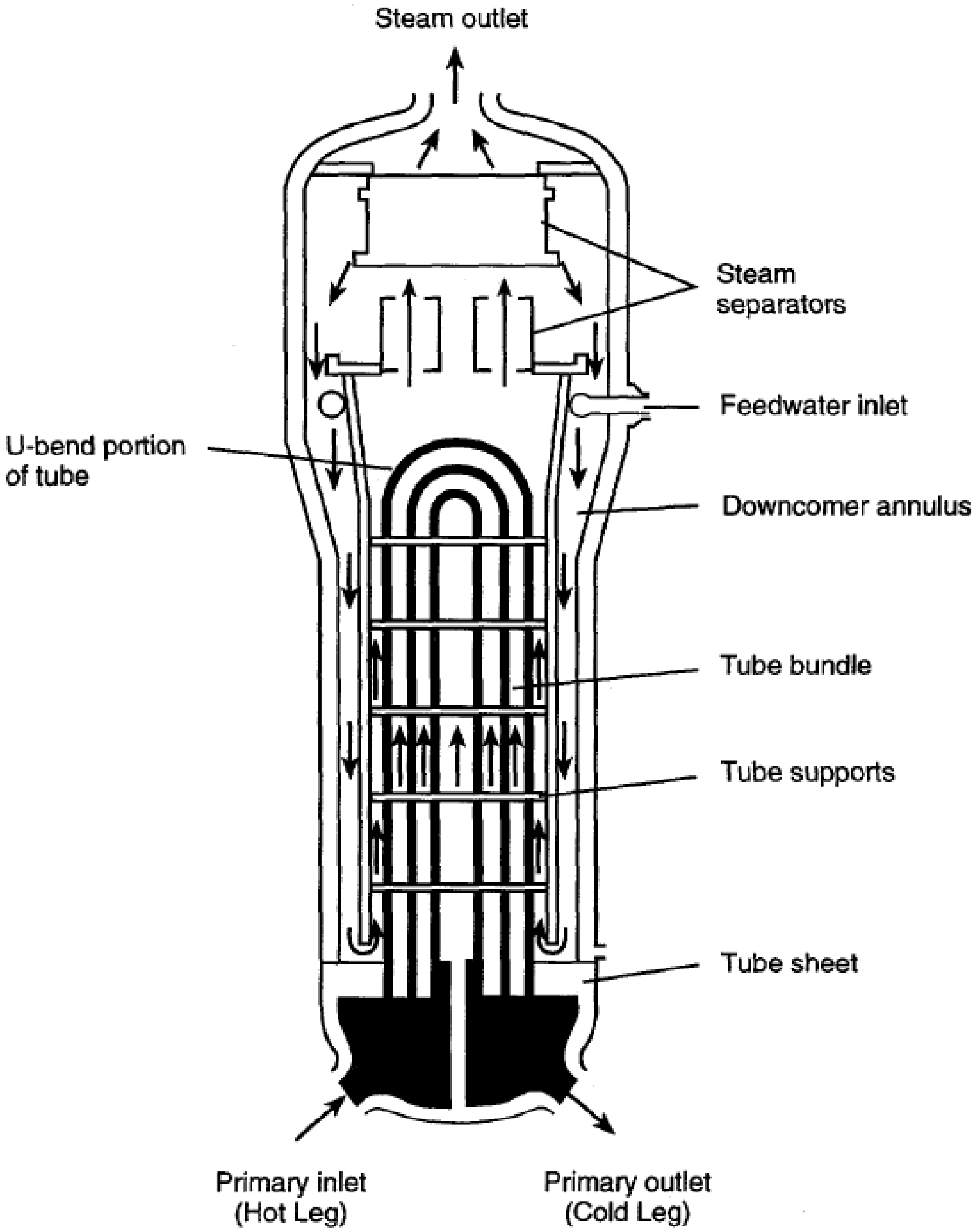
a. *Overview of the IPEC Steam Generators*

Q129. To provide some background and better understand the technical issues regarding steam generators raised in NYS-38/RK-TC-5, please describe the general design of the IPEC steam generators, including the divider plate assembly.

A129. (NFA, RJD, BMG) IP2 and IP3 each have four Westinghouse model 44F steam generators, which are vertical shell and U-tube steam generators with a recirculating design on the secondary side. Figures 2 and 3 below are cross-sectional and cut-away views showing the design of a typical PWR recirculating steam generator, like the ones installed at IPEC.

[redacted]

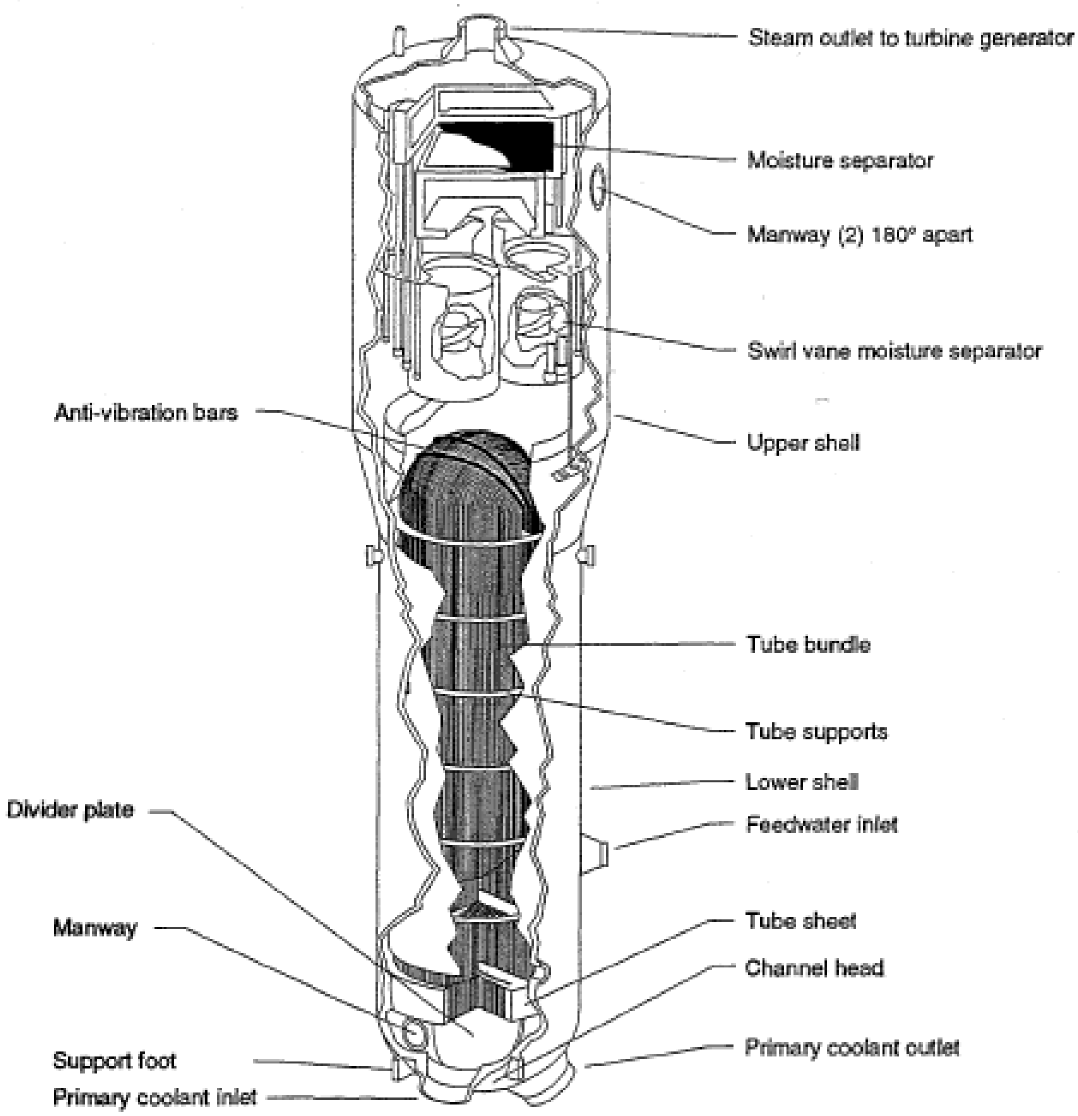
Figure 2. PWR Recirculating Steam Generator (Cross-Section)



Source: IN 2005-09, Attach. 1 (ENT000527).

[redacted]

Figure 3. PWR Recirculating Steam Generator (Cut-Away View)



Source: NUREG/CR-6365, Steam Generator Tube Failures, at 5 (Fig. 2) (Apr. 1996).

On the primary side, reactor coolant enters the inlet side of the channel head at the bottom of the steam generator through the inlet nozzle, flows through the U-tubes to an outlet channel, and leaves the generator through another bottom nozzle. The inlet and outlet channels are separated by a partition known as the divider plate. *See* LRA at 2.3-4 (ENT00015A).

[redacted]

Q130. Please further describe the divider plate and tubesheet, as part of the overall steam generator design.

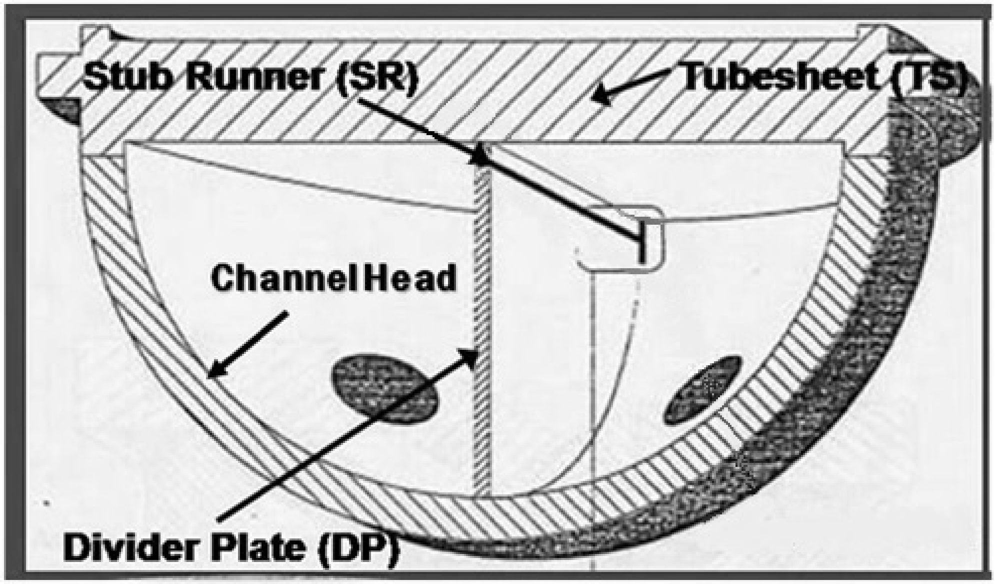
A130. (NFA, RJD, BMG) [redacted]

[redacted]

[redacted] Figure 4 provides a representative sketch of the divider plate geometry, showing the channel head, the divider plate, the tubesheet, and the stub runner. Figure 5 shows a tube installed in the tubesheet. Finally, Figure 6 provides another view of what is referred to as the "triple point" of the tubesheet-channel head complex—*i.e.*, the junction between the channel head, divider plate and tubesheet.

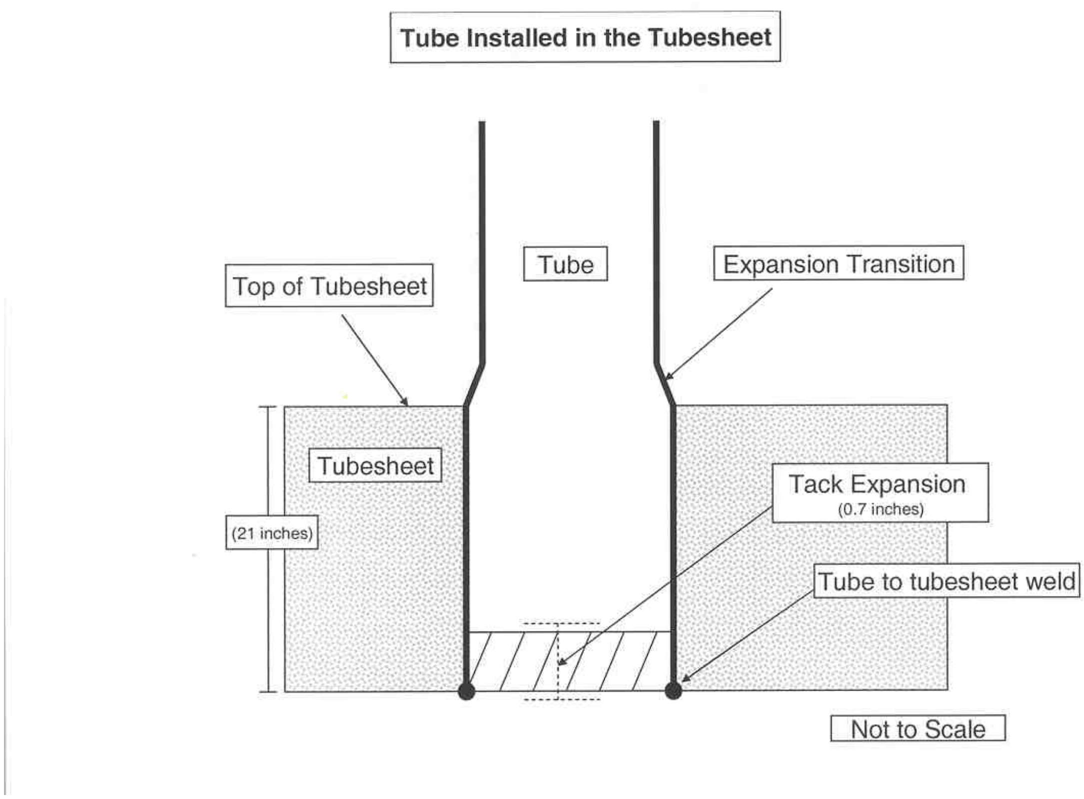
[redacted]

Figure 4. Sketch of Divider Plate Geometry



Source: Adapted from EPRI 2014 Report at 1-1, fig.1-1 (NYS000544A).

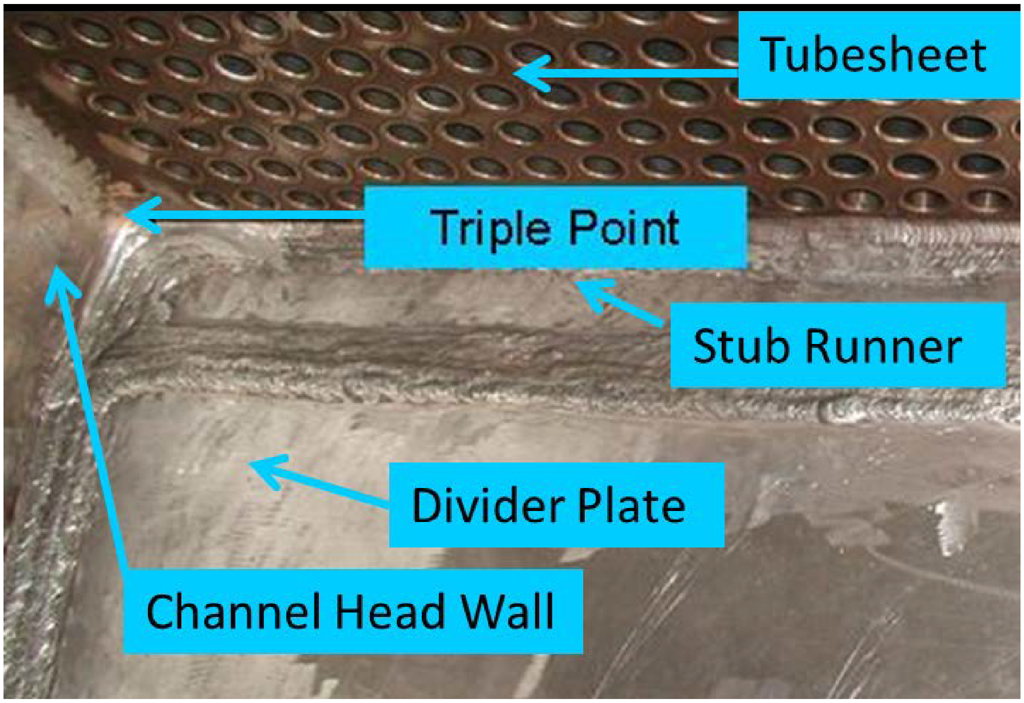
Figure 5. Sketch of Tube Installed in the Tubesheet



Source: IN 2005-09, Attach. 2 (ENT000527) (representative example; measurements are not IPEC-specific).

[redacted]

Figure 6. Triple Point of the Tubesheet-Channel Head Complex



Source: EPRI 2014 Report at 2-8, fig.2-5 (NYS000544A) (photograph is representative; not IPEC-specific).

Q131. Please describe the materials of construction for the IP2 and IP3 steam generator divider plates, tubes, and related weld materials.

A131. (NFA, RJD, BMG) The steam generator divider plates at IP2 and IP3 are Alloy 600 ("Alloy 600TT"). *See* NL-11-032, Attach. 1, at 20 (NYS000151). The divider plate weld materials at both plants are conservatively assumed to be Alloy 600TT as well, as that material is generally more susceptible to PWSCC than thermally-treated Alloy 690 ("Alloy 690TT"). *See id*. The steam generator tubes at IP2 are made of Alloy 600TT; the tubes at IP3 are made of Alloy 690TT. *See* LRA at 2.3-21 (ENT00015A).

b. Background on Potential PWSCC in Nickel Alloy Materials

Q132. Entergy Commitments 41 and 42 both relate to the aging management of potential cracking caused by PWSCC in the steam generator divider plates and tube-to-tubesheet welds, respectively? What is PWSCC?

A132. (NFA, JRS, BMG) PWSCC is an intergranular cracking corrosion mechanism that requires: (1) the presence of high applied and/or residual tensile stress; (2) susceptible alloy microstructures (*e.g.*, few intergranular carbides); and (3) high temperature water. In a PWR environment, this cracking mechanism is most likely for *nickel alloys*—specifically, Alloy 600 components and their compatible weld materials, Alloys 82/182. *See* NUREG-1801, Rev. 1 at IX-34 (NYS00146C); NUREG-1801, Rev. 2, at IX-36 (NYS00147C).

For example, nickel alloy components subject to relatively high operating or residual tensile stresses are potentially vulnerable to PWSCC, especially welded or work-hardened components that were not annealed, or where the annealing temperatures were insufficient to limit chromium depletion and carbide precipitation on the grain boundaries. *See* EPRI, MRP-175, Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values § 2.1 (Dec. 2005) ("MRP-175") (NYS000319).

Q133. How does water chemistry affect PWSSC?

A133. (NFA, RGL JRS, BMG) Zinc additions have been effective in mitigating PWSCC initiation and propagation in Alloy 600. *See generally* H. Kawamura et al., Paper No. 141, The Effect of Zinc Addition to Simulated PWR Primary Water on the PWSCC Resistance, Crack Growth Rate and Surface Oxide Film Characteristics of Prefilmed Alloy 600 (Corrosion 98, 1998) (ENT000711). Also, maintaining optimal hydrogen concentrations delays crack initiation and slows crack growth. *See generally* P.L. Andresen et al., Effects of Hydrogen on Stress Corrosion

[redacted]

Crack Growth Rate of Nickel Alloys in High-Temperature Water, 64 CORROSION 707 (Sept. 2008) (ENT000712).

Q134. Does IPEC use zinc injection to mitigate the potential for PWSCC?

A134. (NFA, RJD, BMG) Yes. Although zinc is injected primarily for radiological protection reasons (*i.e.*, to reduce the source term and its resultant worker radiation exposures), at IP2 zinc concentrations are maintained at 10 to 20 parts per billion (ppb). *See* Entergy, 0-CY-2310, Rev. 24, Reactor Coolant System Specifications and Frequencies at 15 (Jan. 16, 2015) (ENT000692) ("IPEC RCS Specs"). Zinc concentrations in this range serve to significantly reduce PWSCC initiation and crack growth. At present, Entergy does not inject zinc at IP3 due to the lower radiological source term at that unit.

Q135. One of New York's witnesses, Dr. Duquette, has described the industry and regulator efforts over the past three decades in research and development to address PWSCC as having "limited success." Duquette Testimony at 12 (NYS000372). Do you agree with that characterization?

A135. (NFA, RJD, JRS, BMG) No. The industry has made substantial progress in recent years in understanding the effects of water chemistry on PWSCC and various other means to mitigate and control those effects, and also the nickel alloy chemical and metallurgical characteristics that provide the major contributions to material susceptibility. *See generally*, *e.g.*, B. Gordon, Corrosion and Corrosion Control in Light Water Reactors, 65 JOURNAL OF METALS 1043 (Aug. 2013) (ENT000713). EPRI has spearheaded these efforts through its Steam Generator Management Program ("SGMP").

[redacted]

2. Entergy's Programs and Commitments for Managing PWSCC-Related Aging Effects on the IPEC Steam Generator Divider Plates and Tube-to-Tubesheet Welds During PEO

a. Relevant Sections of the Original IPEC LRA

Q136. What section(s) of the original IPEC LRA address the management of aging effects on the steam generator divider plates and tube-to-tubesheet welds?

A136. (ABC, NFA, RJD, JRS, BMG) Chapter 2 of the IPEC LRA summarizes Entergy's detailed assessment of structures and components that require aging management review. Chapter 3 identifies cracking due to PWSCC as an applicable aging effect for nickel alloy plant components. *See* LRA at 3.1-9 (ENT00015A). For the steam generator divider plates, the Water Chemistry Program manages the aging effect of cracking (whether caused by PWSCC or other aging mechanisms). *See id*. at 3.1-144, 3.1-162. For the steam generator tubesheets, the Water Chemistry and Steam Generator Integrity Programs manage cracking. *See id*. at 3.1-10. The appendices to the LRA contain descriptions of Entergy's Water Chemistry and Steam Generator Integrity Programs.

Q137. Please summarize the description of the Water Chemistry Program in the Appendices to the LRA.

A137. (ABC, NFA, RJD, JRS, BMG) Appendix A to the LRA presents information required by 10 C.F.R. § 54.21(d) relating to the Water Chemistry Program that supplements the UFSAR for IPEC. *See* LRA, App. A (ENT00015B). Specifically, the supplement to the UFSAR, presented in sections A.2 and A.3 of Appendix A, contains summary descriptions of the Water Chemistry Program. *See id.* at A-38, A-65.

Appendix B to the LRA describes AMPs credited for managing aging effects during the PEO. *See id*., App. B. Section B.1.41 describes the IPEC Water Chemistry Program and indicates that it is consistent with the program described in Section XI.M2 of NUREG-1801, Revision 1, with one enhancement (which is not relevant to this contention). *See id.* at B-138; NUREG-1801, Rev. 1, at XI M-10 (NYS00146C). When Entergy submitted the original LRA, the IPEC program was based on EPRI guidelines for an effective water chemistry program contained in TR-105714, Rev. 5, Pressurized Water Reactor Primary Water Chemistry Guidelines. *See* LRA at B-137 (ENT00015B). The IPEC Water Chemistry Program now relies upon revision 6 to the EPRI Water Chemistry Guidelines. *See* LRA at B-137 to B-139 (ENT00015B) (stating that "[f]uture revisions of the EPRI primary and secondary water chemistry guidelines will be adopted as required, commensurate with industry standards"); SER at 3-148 (NYS00326C) (stating the same); *see also generally* EPRI, Final Report 1014986, Pressurized Water Reactor Primary Water Chemistry Guidelines, Vol. 1, Rev. 6 (Dec. 2007) (ENT000557).

Q138. What are the key aspects of the Water Chemistry Program that manage the effects of aging due to PWSCC in nickel alloy steam generator components?

A138. (NFA, ABC, BMG) Under the NRC-approved EPRI Water Chemistry Guidelines, Entergy maintains optimal hydrogen concentrations to mitigate PWSCC. *See* IPEC RCS Specs at 11 (ENT000692). In addition, as explained in response to Question 134, zinc injections used at IP2 have beneficial effects with respect to PWSCC. Another key supplement to the Water Chemistry Program is the One-Time Inspection Program, which verifies through inspections that the program is effectively managing the effects of aging. *See* LRA at B-137 (ENT00015B); *see also id*. at B-90 to B-93.

Q139. Please summarize the description of the Steam Generator Integrity Program in the appendices to the LRA.

A139. (ABC, NFA, RJD, JRS, BMG) Appendix A to the LRA presents information required by 10 C.F.R. § 54.21(d) relating to the Steam Generator Integrity Program that

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supplements the UFSAR for IPEC. *See* LRA, App. A (ENT00015B). Specifically, the supplement to the UFSAR, presented in sections A.2 and A.3 of Appendix A, contain summary descriptions of the Steam Generator Integrity Program. *See id.* at A-34 to A-35, A-62.

Section B.1.35 of the LRA describes the IPEC Steam Generator Program and indicates that it is consistent with the program described in Section XI.M19 of NUREG-1801, Revision 1, with one enhancement. *See* LRA at B-118 (ENT00015B); NUREG-1801, Rev. 1, at XI M-68 (NYS00146C). The enhancement specifies a revision to procedures regarding certain monitoring and trending activities and is not directly relevant to the issues in this contention. LRA Section B.1.41 further states that the IPEC Steam Generator Integrity Program is implemented in accordance with NEI 97-06, "Steam Generator Program Guidelines." *See* LRA at B-137 (ENT00015B); *id.* at B-118.

Q140. Did the NRC review and approve these aspects of Entergy's LRA?

A140. (ABC, NFA, RJD, JRS, BMG) Yes. The NRC Staff reviewed the IPEC Water Chemistry Program and, in its November 2009 SER, concluded that the program elements are acceptable and consistent with the ten program elements in NUREG-1801, Revision 1, Section XI.M2. *See* SER at 3-148 (NYS00326C); *see also id*. at 3-241 (noting that cracking due to PWSCC in steam generator divider plates is managed through the Water Chemistry Program, which is consistent with NUREG-1801). With respect to the Steam Generator Integrity Program, the NRC Staff concluded that the program elements are consistent with the ten program elements in NUREG-1801, Revision 1, Section XI.M19. *See* SER at 3-115 (NYS00326C).

b. *Overview of License Renewal Commitments 41 and 42*

Q141. Following issuance of the SER, did the NRC Staff issue additional RAIs on the topic of steam generator divider plates?

A141. (NFA, RJD, ABC, JRS) Yes. On February 10, 2011, the Staff issued additional RAIs on Entergy's LRA, including questions related to potential PWSCC of steam generator divider plates. *See* NL-11-032, Attach. 1, at 20-21 (NYS000151). Similar RAIs were issued to a number of other license renewal applicants. *See*, *e.g.*, Letter from J. Daily, NRC, to D. Heacock, Dominion Energy Kewaunee, "Request for Additional Information for the Review of the Kewaunee Power Station License Renewal Application (TAC No. MD9408)," Encl. at 5 (Mar. 11, 2010) (ENT000558).

Q142. What were the NRC Staff's questions?

A142. (NFA, RJD, ABC, JRS) Based on the foreign operating experience discussed above in response to Question 83, the Staff asked Entergy to describe the materials of construction of the IP2 and IP3 steam generator divider plate assemblies and associated welds. *See* NL-11-032, Attach. 1, at 20 (NYS000151). The Staff further requested that, if any of the material was susceptible to cracking (*i.e.*, Alloy 600 and associated weld materials), then Entergy should explain how it plans to manage PWSCC to prevent the potential propagation of cracks to items that are part of the reactor coolant pressure boundary. *See id* at 21.

Q143. Please describe Entergy's response to the Staff's February 2011 steam generator divider plate RAI.

A143. (NFA, RJD, ABC, JRS, BMG) Entergy responded to the NRC Staff's RAIs in NL-11-032 (March 28, 2011) (NYS000151), and subsequently amended its response. *See* NL-11-074, Letter from F. Dacimo, Entergy, to NRC Document Control Desk, "Response to Request for Additional Information (RAI), Aging Management Programs" (July 14, 2011) ("NL-11-074")

(NYS000152); NL-11-090, Letter from F. Dacimo, Entergy, to NRC Document Control Desk, "Clarification for Request for Additional Information (RAI), Aging Management Programs" (July 27, 2011) (NYS000153). In response to the first question, Entergy explained that the IP2 and IP3 divider plates are Alloy 600, and that Entergy conservatively assumed that the weld materials are also Alloy 600. *See* NL-11-032, Attach. 1, at 20 (NYS000151).

In response to the second question, Entergy explained that the industry was commencing an effort to study divider plate crack growth and develop a resolution to the issue through the EPRI SGMP Engineering and Regulatory Technical Advisory Group. *See id*. at 21. At that time, EPRI already had concluded that a cracked divider plate in a Westinghouse Model 44F steam generator (such as those at IPEC) was not a safety concern (*see* EPRI Phase II Study at v, 9-1 (ENT000523)), [redacted] NL-11-032, Attach. 1, at 21 (NYS000151). Nonetheless, recognizing that EPRI's final resolution of this generic issue still was in progress, Entergy committed to inspect the IP2 and IP3 steam generators to assess the condition of the divider plate assemblies using an examination technique that is capable of detecting PWSCC. *See id*. Specifically, Commitment 41 states:

> IPEC will perform an inspection of steam generators for both units to assess the condition of the divider plate assembly. The examination technique used will be capable of detecting PWSCC in the steam generator divider plate assemblies. The IP2 steam generator divider plate inspections will be completed within the first ten years of the period of extended operation (PEO). The IP3 steam generator divider plate inspections will be completed within the first refueling outage following the beginning of the PEO.

NL-11-074, Attach. 1 at 14 (NYS000152); *see also* SSER 1, App. A at A-23 (NYS000160).

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Q144. Did the NRC Staff also issue RAIs concerning the IPEC steam generator tube-to-tubesheet welds?

A144. (ABC, NFA, RJD, JRS, BMG) Yes. On February 10, 2011, the NRC also issued to IPEC, and separately to other plants, a series of questions on steam generator tube-to-tubesheet welds. Those questions were not tied to the foreign operating experience on divider plates, but instead were based on the Staff's concern that certain tube-to-tubesheet welds and cladding materials may have insufficient chromium content to prevent initiation of PWSCC, and that cracks in the divider plate could subsequently propagate to the tubesheet cladding and potentially affect the tube-to-tubesheet welds. *See* NL-11-032, Attach. 1 at 21 NYS000151). The Staff noted that unless there is an NRC-approved redefinition of the reactor coolant system pressure boundary, the effectiveness of the Water Chemistry Program should be verified through a one-time inspection to ensure PWSCC is not occurring in the tube-to-tubesheet welds. *See id.* at 22.

Based on this background, the NRC Staff asked Entergy to justify how the Steam Generator Integrity Program (or another program) is capable of managing PWSCC in such welds. *See id*. at 22-23.

Q145. Please describe Entergy's response to the tube-to-tubesheet weld RAI.

A145. (ABC, NFA, RJD, JRS, BMG) Entergy responded to the NRC's RAIs in NL-11-032 (NYS000151), and subsequently amended its response in NL-11-074 (July 14, 2011) (NYS000152). First, Entergy stated that the IP2 tube-to-tubesheet welds are part of the reactor coolant system pressure boundary and no alternate repair criteria have been approved (*i.e.*, the pressure boundary had not been redefined via license amendment). *See* NL-11-032, Attach. 1, at 22 (NYS000151). Second, Entergy stated that it would address the NRC Staff's concern through one of two options, either through an analysis or inspection. Under the analysis option, Entergy would evaluate the tube-to-tubesheet welds in order to establish a technical basis for either determining that these welds are not susceptible to PWSCC, or redefining the reactor coolant pressure boundary, such that the welds would not be required for the pressure boundary function. *See id*. at 22-23. The latter option requires NRC approval of the analysis through a license amendment. *See id*. at 22.

Under the inspection option (*i.e.*, if the analysis results are not acceptable), Entergy would perform a one-time inspection of a representative number of welds in each steam generator and, if cracking is identified, the condition will be resolved through a repair or engineering evaluation, and an ongoing monitoring program will be established for the life of the steam generators. *See id*. at 23-24. At IP2, the analyses or inspections would take place between March 2020 and March 2024, or between 20 and 24 years of service. *See id*. at 23. At IP3, the analyses or inspections would take place within the first two refueling outages in the PEO. *See id*. at 24. Entergy later amended this aspect of the commitment to specify that IP3 inspections will take place by the end of the first refueling outage during the PEO. *See* NL-11-074, Attach. 2, at 15 (NYS000152).

Q146. Did the NRC review and approve Entergy's responses to the RAIs discussed in the foregoing questions?

A146. (NFA, RJD, ABC, JRS, BMG) Yes. The NRC Staff addressed these issues in the first supplement to its SER issued on August 30, 2011. *See generally* SSER 1 (NYS000160). SSER 1 explained that Commitments 41 and 42 are acceptable. Specifically, Commitment 41 is acceptable because Entergy made a commitment to inspect the divider plate assembly in each steam generator at both IPEC units during the PEO, in a time period consistent with the detection of potential PWSCC cracks, with appropriate examination techniques. *See id.* at 3-19.

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SSER 1 also explained that Commitment 42 is acceptable because Entergy will manage the aging effect of cracking due to PWSCC in the tube-to-tubesheet welds either by: (1) demonstrating that those welds are no longer included in the reactor coolant pressure boundary function (or are not susceptible to PWSCC); or (2) implementing a one-time inspection on a representative number of welds. *See* SSER 1 at 3-23 (NYS000160). The Staff also concluded that any inspection would take place in a time period consistent with the detection of PWSCC, because it is unlikely that significant detrimental PWSCC will have initiated before the identified time periods. *See id*. The Staff further noted that if aging effects are identified by the inspections, then Entergy will take corrective actions, including evaluating degradation and then implementing routine inspections of the tube-to-tubesheet welds for the remaining life of the steam generators. *See id*.

Q147. Dr. Duquette asserts that Entergy "accept[s] the premise that there is a high probability that PWSCC will occur in the divider plates in Westinghouse steam generators including those at Indian Point, and is likely to progress into the channel head assembly . . . ." Duquette Report at 21 (NYS000373). How do you respond?

A147. (ABC, NFA, RJD) We strongly disagree with Dr. Duquette's suggestion that Entergy (along with the industry and the NRC) accepts the premise that there is a high probability that PWSCC will occur in Westinghouse steam generator divider plate, including those at IPEC. As noted above, Entergy has implemented a water chemistry program intended to minimize PWSCC and conservatively committed to perform inspections of the IPEC steam generator divider plate assemblies to *confirm the absence* of PWSCC indications during the PEO. *See* SSER 1 at 3-18 to 3-19 (NYS000160) (discussing Commitment 41). Simultaneously, as discussed below in Section V.C.3, EPRI's extensive research on this subject refutes Dr. Duquette's "premise" that

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there is a "high probability" of PWSCC in steam generators at U.S. plants. In fact, EPRI concludes the opposite.

c. Timing of Inspections

Q148. With respect to the timing of Entergy's inspections under Commitment 41, Dr. Duquette asserts that without "specific criteria" for determining the "appropriateness" of the timing of inspections and safety evaluations performed under the Quality Assurance Program, "Entergy's plan remains a hollow assurance that aging degradation of its steam generators will be adequately managed." Supplemental Duquette Testimony at 8 (NYS000532). What is the basis for the timing of steam generator divider plate inspections, as set forth in Commitment 41?

A148. (JRS, NFA, RJD, BMG) The NRC Staff has found it acceptable when license renewal applicants commit to conduct steam generator divider plate inspections after the commencement of the PEO and after the components have seen more than approximately 20 years of service. *See* NUREG-1961, Safety Evaluation Report Related to the License Renewal of Palo Verde Nuclear Generating Station, Units 1, 2, and 3 at 3-158 (Apr. 2011) ("NUREG-1961") (ENT000537) ("The staff finds that the timing of this inspection for each unit is acceptable because the proposed implementation schedule allows operation of the SGs for between 20 and 25 years, and it is unlikely that significant detrimental PWSCC cracking will have initiated before this time.").

The IPEC inspections to be conducted under Commitment 41 are consistent with this time frame: at the time of inspections, the steam generators will have approximately 23 years of service for IP2 and 28 for IP3. For IP3, the inspections will take place during the first refueling outage following the beginning of the PEO. The difference in timing is because the IP2 steam generators were replaced in 2000 and the IP3 steam generators in 1989.

Q149. What is the basis for the timing of the tube-to-tubesheet weld analysis or inspection options in Commitment 42?

A149. (ABC, NFA, RJD, JRS, BMG) As we previously mentioned, for IP2, Entergy has already implemented Commitment 42 using the analysis option to seek and obtain a license amendment. *See* H* Amendment Issuance (NYS000542). Thus, inspections of the tube-to-tubesheet welds at IP2 are not necessary. For IP3, the timing of any inspections under Commitment 42 would be similar to those for Commitment 41, and are intended to allow sufficient service time to develop indications in the steam generators, if they are in fact susceptible to PWSCC, while also performing the inspections sufficiently early to detect potential flaws before they can become structurally significant. Allowing approximately 28-years of operation before the inspection for IP3 also reflects the fact that the IP3 tubing material is made from Alloy 690TT material, which is more resistant to PWSCC due to its higher chromium content. *See* Responses to Questions 181, 188, and 189 (discussing resistance of high-chromium materials to PWSCC).

Q150. Dr. Lahey criticizes the timing of Commitment 42 as well, noting that inspections of these components at IP3 for PWSCC "will not be made until after the first refueling outage after the reactor enters the period of extended operation." Lahey Testimony at 11 (NYS000374). Please respond to Dr. Lahey.

A150. (ABC, JRS, RJD) There is no requirement that actual inspections or other aging management activities be completed before the PEO begins, and Dr. Lahey cites none. From a technical perspective, Entergy's Commitment 42 provides reasonable assurance because, as noted in response to the previous question, the time frame selected by Entergy allows sufficient service time to develop detectable indications in the steam generators, if they are in fact susceptible to

PWSCC, while also performing the inspections sufficiently early to ensure that potential flaws will not develop into structurally significant cracking.

Q151. Dr. Duquette states that the NRC expects the LRAs it reviews using GALL Revision 2 to include a commitment to inspect steam generator divider plates (and tube-tubesheet welds) once the PEO commences and they have been in service for 20 years. According to Dr. Duquette, good engineering practice dictates that GALL Revision 2 should be applied to the IPEC LRA, rather than revision 1. *See* Duquette Report at 17-18, 21-22 (NYS000373). Please comment on this statement from Dr. Duquette.

A151. (NFA, RJD, JRS) To the extent Dr. Duquette wants Entergy to commit to inspect steam generator divider plates and tube-to-tubesheet welds once the plant enters the PEO and the components are in service for over 20 years, Entergy has done so in Commitment 41 (for IP2 and IP3) and Commitment 42 (for IP3). And as explained in response to Question 160, Entergy completed Commitment 42 for IP2 in 2014.

d. Methodology for Inspections

Q152. What methods is Entergy considering to inspect for PWSCC in the steam generator divider plate assemblies?

A152. (NFA, RJD, JRS, BMG) Commitment 41 states that the method used will be capable of detecting PWSCC in the steam generator divider plate assembly. *See* NL-11-032, Attach. 1, at 21 (NYS000151).

Q153. What inspection technique is Entergy evaluating?

A153. (NFA, RJD) Entergy plans to conduct EVT-1 inspections using a robot-mounted camera, similar to methods used for inspections of other steam generator components. Such methods would be consistent with the standards in the ASME Code. *See* ASME Code, IWA-2000 § 2210 (ENT000531). EVT-1 is an enhanced visual technique capable of detecting tight cracks, *see* MRP-227-A at 5-21 to 5-22 (NRC000114B), and is used in the RVI AMP to detect SCC. *See* NL-12-037, Attach. 2 at 37-50, tbls. 5-2, 5-3 (NYS000496).

Q154. Are there other methods available for divider plate inspections?

A154. (NFA, RJD, JRS, BMG) Yes. Divider plates can also be inspected for cracking using the surface examination methods provided in the ASME Code. *See* ASME Code, IWA-2000 §§ 2220, 2222 (ENT000531). However, manually performing liquid penetrant examinations inside the steam generator bowls would be a relatively dose-intensive activity, so a remote method of performing the exam is preferable.

Q155. Dr. Duquette states that considering the potential for high radiation doses to personnel, Entergy has failed to present a reliable, defined program for remote inspection. *See* Duquette Rebuttal at 6 (NYS000452). He further states that while techniques may be in use in other countries, they have not been qualified for service here. *See* Duquette Rebuttal at 3-4, 5-6. What is your response to Dr. Duquette?

A155. (JRS, NFA, RJD) We disagree for the reasons stated above. Entergy has several options that would provide for adequate inspections without high radiation dose, including the robotic inspections it is evaluating now. Further, in his November 2012 rebuttal testimony, Dr. Duquette expressly conceded that "conventional inspection techniques may be available to detect cracking." Duquette Rebuttal at 6 (NYS000452). Nevertheless, although manual inspections are not preferred and they do result in higher radiation exposures, they are still an effective and available option for Entergy.

Q156. Dr. Lahey also criticizes Commitment 41 as vague, claiming that it does not describe: (1) the inspection methodology; (2) the number of steam generators to be inspected; (3) the acceptance criteria; or (4) corrective action criteria. *See* Revised Lahey

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Testimony at 94 (NYS000562). Dr. Duquette makes similar criticisms, and also cites a lack of "monitoring and trending protocols." ***See*** **Duquette Testimony at 28 (NYS000372). Please respond to these criticisms.**

A156. (ABC, JRS, NFA, RJD) We disagree with the State's experts. First, Commitment 41 is quite specific, as it states that the technique used will be "capable of detecting PWSCC in the divider plate assembly." *See* SSER 2 at A-13 (NYS000507). Thus, the commitment is tailored and performance based. As we have also explained in response to Question 154, techniques capable of detecting PWSCC exist, including the remote visual technique Entergy is evaluating. Dr. Duquette states that EPRI has "admitted" that there are "still" no qualified techniques in the U.S. (*see* Duquette Report at 16 (NYS000373)), but this merely reflects the fact that the need to specifically qualify such techniques in the U.S. has not yet arisen.

Second, the commitment states that steam generators at both units will be inspected. *See* SSER 2 at A-13 (NYS000507). There is no ambiguity here either—all eight steam generators at the two units will be inspected.

Third, the examination acceptance criteria are implicit in the commitment. The purpose of the inspections is to detect cracking, and any detected flaws will be evaluated to determine the appropriate corrective action, consistent with the IPEC 10 C.F.R. Part 50, Appendix B corrective action program. *See* SRP-LR, Rev. 2 at A.1-6 to A.1-7 (NYS000161).

Finally, if a flaw is detected, then it must be properly evaluated under the Entergy Quality Assurance Program. This program includes the quality assurance elements for all AMPs in the IPEC LRA, including the corrective action, confirmation process, and administrative controls elements. *See* LRA at A-17, A-44, B-2 to B-3 (ENT00015B); SER at 3-220 to 3-222

[redacted]

(NYS00326C). As to an alleged lack of monitoring and trending protocols, such protocols are not necessary for one-time inspections.

Q157. Drs. Lahey and Duquette state that the "details of the inspections for [PWSCC] . . . will apparently not be available until after extended operations are expected to begin." Lahey Testimony at 10-11 (NYS000374); *see also* Duquette Testimony at 10, 26 (NYS000372); Duquette Report at 19, 21, 22 (NYS000373).

A157. (ABC, NFA, RJD, BMG) As explained in the previous answer, sufficient details to reach a reasonable assurance finding are available now. There is no requirement that additional implementation details be provided during the LRA review, because the reasonable assurance finding required by 10 C.F.R. § 54.29(a) rests on Staff review and acceptance of actions that have been or will be taken by the applicant. Indeed, renewed operating licenses already have been issued in other proceedings where similar commitments regarding PWSCC inspections have been made. *See, e.g.*, Letter from L. Hartz, Dominion Energy Kewaunee., to, NRC Document Control Desk, "Response to Request for Additional Information for the Review of the Kewaunee Power Station License Renewal Application," Attach. 1, at 6-8 (Sept. 23, 2010) (ENT000565) (committing to perform similar inspections before the midway point of the PEO) (renewed license issued in February 2011). Similar commitments have been made by licensees in other pending license renewal proceedings. *See, e.g.*, Letter from P. Freeman, NextEra Energy Seabrook, to NRC Document Control Desk, "Response to Request for Additional Information, NextEra Energy Seabrook License Renewal Application, Request for Additional Information – Set 10," Encl. 3, at 2-4 (Mar. 22, 2011) (ENT000566). The implementation of these commitments will be subject to the NRC's oversight and inspection authority. Thus, sufficient details are available now to make the requisite findings under the regulations.

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Q158. Dr. Duquette makes the same criticisms of Commitment 42 as he makes for Commitment 41; *i.e.,* that there are no details on inspection methods or techniques, acceptance criteria, monitoring and trending protocols, and corrective actions. *See* Duquette Testimony at 28 (NYS000372). Dr. Lahey similarly states that there are unanswered questions about the methodology to be used for the one-time inspections of the tube-tubesheet welds under Commitment 42. *See* Lahey Testimony at 22 (NYS000374). Please respond to these claims.

A158. (JRS, NFA, RJD) As we have previously noted, no inspections are necessary under the analysis option for Commitment 42. For IP2, the NRC has approved Entergy's analysis. *See generally* H* Amendment Issuance (NYS000542). For IP3, we disagree with Dr. Duquette's criticisms of the inspections—sufficient information is available in the record on inspection methods and techniques, acceptance criteria, monitoring and trending, and corrective actions.

As to inspection techniques, as we have previously explained, there is no requirement to specify the particular techniques to be used at this time. In addition, it would not be prudent to commit now to a specific technique years before the inspections are undertaken. In any event, as we have shown in response to Question 154, capable inspection techniques are available.

Commitment 42 is also specific as to acceptance criteria, monitoring and trending protocols, and corrective actions. If any weld cracking is identified, then the condition must be resolved through a repair or engineering evaluation and an ongoing monitoring program will be established. *See* SSER 1 at A-24 (NYS000160).

Thus, contrary to Dr. Lahey's statement, Commitment 42 is not simply a proposal to develop a plan. Lahey Testimony at 21-22 (NYS000374). It is a specific commitment that binds Entergy to one of two acceptable options to adequately manage aging for a newly identified issue for this type of steam generator tube-to-tubesheet welds—a process that the NRC has found acceptable in other recent license renewal applications. *See* NUREG-1958, Safety Evaluation Report Related to the License Renewal of Kewaunee Power Station at 3-230 to 3-232 (Jan. 2011) (ENT000546); NUREG-2101 at 3-279 to 3-282 (ENT000536); NUREG-1961 at 3-173 to 3-177 (ENT000537).

Q159. Similarly, Dr. Duquette suggests that there is no qualified inspection procedure to detect the propagation of cracking from the divider plate to the tube-tubesheet weld. *See* Duquette Testimony at 9 (NYS000372); *see also id.* at 27 (stating that EPRI has admitted that there are no qualified inspection techniques to inspect the steam generator channel head); Duquette Report at 16 (NYS000373). How do you respond to Dr. Duquette?

A159. (JRS, NFA, RJD) We disagree with Dr. Duquette. As we have previously explained in response to Question 156, above, EPRI's "admission" does not mean that it will be a challenge to develop qualified techniques, but merely that the need to qualify inspection techniques for the divider plate has not yet arisen in the United States. In short, we have no concerns about the ability to qualify inspection techniques.

e. Option to Analyze Tube-to-Tubesheet Welds under Commitment 42

Q160. You previously mentioned that Entergy implemented the analysis option of Commitment 42 for IP2. Please explain.

A160. (ABC, NFA, RJD, JRS) On January 16, 2014, Entergy filed a license amendment request to redefine the reactor coolant pressure boundary, such that the welds would not be required for the pressure boundary function. *See* NL-14-001, Letter from J. Ventosa, Entergy, to NRC Document Control Desk, "Proposed License Amendment for Alternate Repair Criteria for Steam Generator Tube Inspection and Repair" (Jan. 16, 2014) ("H* LAR") (NYS000539). As explained further below, this is a well-established process that has led to license amendments for numerous other plants. On September 5, 2014, the NRC granted that license amendment request. *See* H* Amendment Issuance (NYS000542). Accordingly, Entergy implemented Commitment 42 for IP2 through the approved analysis option on October 15, 2014. *See* License Renewal Commitment Closure Verification Form – Commitment #42, Rev. 1 (Oct. 15, 2014) (NYS000553).

Q161. Please briefly describe the technical basis for the H* methodology.

A161. (ABC, NFA, RJD, JRS) This H* methodology is described in Westinghouse, WCAP-17091-NP, Rev. 0, H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model 44F) (June 2009) ("WCAP-17091-NP") (ENT000570), which provides the technical bases for redefining the steam generator tube reactor coolant pressure boundary. The steam generator tubes are mechanically expanded to form a tight seal between the tube and tubesheet. *See id*. at 10-1. The seal created by the mechanical expansion of the tube, combined with the tube-to-tubesheet welds, forms a part of the reactor coolant pressure boundary. Under the H* method, analyses are performed to redefine the primary coolant pressure boundary to the height above the bottom of the tubesheet, below which degradation would not affect the primary coolant pressure boundary function. *See id*.

The applicant for an H* license amendment must provide plant-specific analyses similar to those demonstrated in WCAP-17091-NP. The H* analyses are divided into two parts. The first is a structural evaluation that calculates the value of H*, defined as the length of the steam generator tube below which the structural integrity of the primary-to-secondary pressure boundary is unaffected by any level of tube degradation. *See* WCAP-17091-NP at 1-2, 10-3 to -4 (ENT000570). The second is a leakage evaluation for the tube expansion region, based on the use of the structural evaluation results to conservatively estimate mean residual contact pressures and pull-out forces for that region. *See id*. at 1-2, 10-4 to -5. The required technical demonstration is that "the primary-to-secondary leak rate during a postulated [steam line break] is not exceeded." Biweekly Notice; Applications and Amendments to Facility Operating Licenses and Combined Licenses Involving No Significant Hazards Considerations, 79 Fed. Reg. 15,144, 15,148 (Mar. 18, 2014).

Q162. Dr. Lahey argues that the redefinition of the reactor coolant pressure boundary, while legally plausible, "does not resolve NYS' concerns or eliminate the physical pathway through which radiation can be released to the environment." Revised Lahey Testimony at 98 (NYS000562). What is your response?

A162. (JRS, RJD, NFA) Dr. Lahey's testimony suggests that he has not sufficiently reviewed the available documentation to understand the technical and physical bases that support this approach. As explained in response to Question 161, contrary to his claim, the H* approach explicitly acknowledges and evaluates potential leakage through the seal welds under normal and postulated design basis accidents. The approach includes the performance of a leakage evaluation for the tube expansion region, based on the use of the structural evaluation results to conservatively estimate mean residual contact pressures and pull-out forces for that region. *See* WCAP-17091-NP (ENT000570) at 1-2 ("The leakage analysis is based on a first principles application of the Darcy model for leakage through a porous medium, supported by empirical test results that show that there is no correlation between loss coefficient and contact pressure for the conditions of interest.").

The technical bases provided in the IP2 LAR specifically considered and demonstrated that leakage through the steam generators and structural integrity criteria will continue to be met consistent with the CLB. *See* H* LAR, Attach. 1 at 14 (NYS000539). The NRC Staff conducted an extensive, detailed review of the technical bases for the change, as documented in its Safety Evaluation Report for the requested IPEC license amendment. *See* H* Amendment Issuance, Encl. 2 (NYS000542). Furthermore, under the revised technical specifications, Entergy will perform periodic inspections of the portion of the tubes in the tubesheet that now constitute the reactor coolant pressure boundary in accordance with the Steam Generator Integrity Program. *See id*., Encl. 2 at 6. These changes are now part of the IPEC licensing basis and will be maintained during the PEO in accordance with the NRC-issued license amendment.

Q163. Dr. Duquette asserts that the NRC was "premature" in granting the IP2 H* license amendment because the understanding of PWSCC in the steam generator environment continues to evolve. Supplemental Duquette Testimony at 26-27 (NYS000532). He notes that the NRC recently committed over $2.3 million to fund research at Pacific Northwest National Laboratories ("PNNL") to evaluate PWSCC in nickel-based alloys used in steam generator and reactor components. *Id.* at 27. Do you agree?

A163. (ABC, JRS) No. As a threshold matter, the H* license amendment is now part of the IP2 CLB, so the question of whether the NRC's granting of it was "premature" is moot.

As to the PNNL research on PWSCC crack growth rates in nickel alloy materials, neither Dr. Duquette, nor the source he cites—an NRC Weekly Information Report containing a brief administrative description of the agency's procurement action—provide any explanation for why that research was a necessary prerequisite for the NRC's granting of the H* license amendment. *See* SECY-15-0073, Weekly Information Report – Week Ending May 15, 2015, Encl. A (May 19, 2015) (NYS000557). Indeed, the H* amendment, as we have explained, is based on a structural and leakage evaluation for the steam generator tubes, not any assessment of PWSCC in nickel

[redacted]

alloys. Dr. Duquette's comments identify no errors in the H* methodology in general or as applied to IPEC.

Q164. Which option under Commitment 42 has Entergy selected for IP3?

A164. (ABC, NFA, RJD, JRS, BMG) As noted above, Commitment 42 permits IPEC to perform "an analytical evaluation of the steam generator tube-to-tubesheet welds in order to establish a technical basis for . . . determining that the tubesheet cladding and welds are not susceptible to PWSCC." *See* SSER 1 at 3-22 (NYS000160). With respect to IP3, Entergy is evaluating the EPRI 2014 Report (NYS000544A-D) to determine whether it supports implementation of the analysis option of Commitment 42. Another alternative available to Entergy is to inspect the IP3 tube-to-tubesheet welds under Option 2 of Commitment 42.

Q165. Dr. Duquette claims that Entergy has chosen to analyze the possibility of crack propagation from the divider plate into the tube-to-tubesheet welds, but it does not state how the analyses would be performed, or what would trigger the need for inspections. *See* Duquette Rebuttal at 6. How do you respond?

A165. (RJD, NFA, JRS, BMG) As we just explained, the EPRI 2014 Report evaluates the possibility of crack propagation from the divider plate into the tube-to-tubesheet welds. *See* EPRI 2014 Report § 4 (NYS000544B-C). Thus, it is now clear how EPRI conducted this evaluation. The NRC Staff is reviewing this report. *See generally* H. Cothron et al., EPRI, Slides: NRC/Industry Meeting Regarding Tube-to-Tubesheet Weld and Divider Plate Cracking Report (July 30, 2015) ("EPRI July 30, 2015 NRC Meeting Slides") (ENT000714); NRC, Notice of Meeting with Industry on Divider Plate and Tube-to-Tubesheet Weld (July 1, 2015) ("NRC Divider Plate and Tube-to-Tubesheet Weld Meeting Notice") (ENT000715). Entergy will continue to work with EPRI on this issue to determine if the EPRI 2014 Report supports the closure of Commitment 42 for IP3.

Q166. If inspections are necessary, then what methods will be used to inspect the tube-to-tubesheet welds?

A166. (NFA, RJD) Entergy's preferred approach is to use Option 1 (the analysis approach) rather than the Option 2 inspections. If, in the event the inspections become necessary under Commitment 42, then they could be done using a robot-mounted camera, similar to the planned divider plate inspections. Other options include using a liquid penetrant surface examination, which is an approved method in the ASME Code. *See* ASME Code, IWA-2000 § 2222 (ENT000531). This inspection method has the potential for relatively high personnel radiation exposure. Another alternative would be to use an eddy current surface examination from the inside of the steam generator tubes, in accordance with ASME Code. *See id*. § 2223. Entergy will select the appropriate method closer to the inspection time, based on the methods that are then available.

Q167. Is it uncommon for license renewal applicants to rely on industry programs or experience with regard to future actions performed under AMPs?

A167. (ABC, JRS, BMG) No. The NRC Staff has issued interim staff guidance on the ongoing review of operating experience as part of license renewal. This shows that AMPs are not static, and are always subject to improvement based on operating experience—even after issuance of a renewed license. *See* Final License Renewal Interim Staff Guidance, LR-ISG-2011-05, "Ongoing Review of Operating Experience" at 1 (Mar. 9, 2012) (ENT000564). In fact, the Operating Experience ISG states that acceptable license renewal AMPs "should be informed, and enhanced when necessary, based on the ongoing review of both plant-specific and industry operating experience. . . . This LR-ISG also provides a framework to ensure that license renewal applicants' operating experience review activities will adequately address operating experience . . . *during the term of the renewed license*." *Id*. (emphasis added).

f. Response to Intervenor Critiques of Commitments 41 and 42 Based on Other Steam-Generator-Related Information

Q168. [redacted]

A168. (BMG, NFA, RJD, JRS) [redacted]

[redacted]

Q169. [redacted]

A169. (JRS, BMG, RJD, NFA) [redacted] Subsequently, in August 2013, EPRI issued technical report 3002000473. *See generally* EPRI, Final Report No. 3002000473, Steam Generator Channel Head Degradation Failure Modes and Effects Analysis (Apr. 2013) (ENT000716). The EPRI report endorsed the inspection recommendations in NSAL 12-1. *See id*. at vi. The NRC later issued Information Notice 2013-20 on October 3, 2013, in which it reviewed channel head inspection findings from the foreign plant referenced in NSAL-12-1 and domestic inspection results from Wolf Creek and Surry Unit 2. *See generally* NRC Information Notice 2013-20, "Steam Generator Channel Head and Tubesheet Degradation" (Oct. 3, 2013) ("IN 2013-20") (NYS000538).

[redacted]

Q170. Did Entergy perform the inspections Westinghouse recommended?

A170. (RJD, NFA) Yes. As shown by the Steam Generator Examination Program Results (*see* NYS000537 and NYS000543), Entergy performed remote video camera inspections of the channel head on all eight steam generators (not only six, as Dr. Duquette incorrectly suggests on pages 12-13 and 19-20 of his Supplemental Testimony (NYS000532)) at IPEC in accordance with the NSAL-12-1 recommendations. *See* NL-13-032, Letter from R. Walpole, Entergy, to NRC Document Control Desk, "Technical Specification 5.6.8 - IP3 Steam Generator Tube Inspection Report – Spring 2013 Refueling Outage," at 1 (Aug. 15, 2013) ("NL-13-032") (NYS000537) ("The scope of the inspection included all four steam generators . . . ."); *id*., Encl 1. at § 3.0; NL-14-13, Letter from J. Ventosa, Entergy, to NRC Document Control Desk, "Steam Generator Examination Program Results 2014 Refueling Outage (2R21)," Attach. 1 at 2 (describing inspections of "all four steam generators" including the "primary bowl drain area") (Sept. 8, 2014) ("NL-14-113") (NYS000543).

Entergy's actions (and those of other licensees) illustrate how the industry proactively addresses emerging issues, and how the IPEC Corrective Action Program effectively performs evaluations and initiates appropriate actions based on industry operating experience.

Q171. Dr. Duquette refers to the steam generator tube issues at San Onofre Nuclear Generating Station ("SONGS"). Based on those events, he claims to be "concerned about the numerous indications of vibration-induced wear in the steam generator tubes at IP2, as documented in the plant's most recent tube inspection report." Supplemental Duquette Testimony at 21 (NYS000532). Is his concern founded?

A171. (RJD, NFA, JRS, BMG) No. Dr. Duquette does not provide any technical basis for his concerns or explain why those concerns would have any relevance to the adequacy of

Entergy's Commitments 41 or 42. Vibration-induced tube wear is a separate issue from PWSCC in the divider plate (or tube-to-tubesheet welds). Dr. Duquette articulates no connection between the two technical issues, nor can we discern any. Ultimately, Dr. Duquette's reference to the SONGS tube rupture event is baseless and misleading. At SONGS, a significant design error caused the replacement steam generators to be subjected to substantially more severe thermal-hydraulic conditions than expected, which, in concert with other factors, contributed to rapid steam generator tube wall degradation shortly after installation. *See* Memorandum from M. Johnson, DEDO, to M. Satorius, EDO, "Review of Lessons Learned from the San Onofre Steam Generator Tube Degradation Event" at 2 (Mar. 6, 2015) ("SONGS Lessons Learned Memo") (NYS000552).

Q172. According to Dr. Duquette, the presence of foreign objects "trapped inside the tubes" of the Indian Point steam generators and their potential to cause damage to the reactor coolant pressure boundary is an "important concern." Supplemental Duquette Testimony at 22 (NYS000532). Do you share this same concern?

A172. (RJD, NFA, JRS, BMG) No. Dr. Duquette states that during the most recent IP2 inspection (during the 2R21 refueling outage), "Entergy plugged at least nine tubes due to foreign objects trapped inside the tubes." Supplemental Duquette Testimony at 22 (NYS000532). But, his testimony fails to point out that the objects in question were located on the *secondary* side of the steam generator (*i.e.*, not inside the tubes), where there is no possibility that the foreign objects could influence the progress of any postulated PWSCC in the divider plate, channel head, or tube-to-tubesheet welds. *See* NL-14-113, Attach. 1-B at 2 (NYS000543).

Q173. Dr. Duquette also cites the discovery of a fuel alignment pin in a steam generator tube end in 1990. Supplemental Duquette Testimony at 22 (NYS000532) (citing

2007 Indian Point 3 Steam Generator Program, Engineering Report No. IP3-RPT-SG-01796, Rev. 8 at 13-14 (NYS000533)). Does the cited event provide any support for Intervenors' contention?

A173. (RJD, NFA, BMG) No. As documented in Section 6.3.1 of the IP3 Steam Generator Integrity Program procedure, during the 1990 refueling outage, a foreign object was found partially lodged in a tube end at location Row 1 - Column 34 in the hot leg of IP3 steam generator 34. *See* Entergy Procedure SEP-SG-IP3-001, IP3 Steam Generator Program, Revision 1 at 16 (May 21, 2014) (ENT000717). The object was removed and determined to be a fuel assembly alignment pin from the upper reactor vessel internals. *See id*. Visual examinations revealed that the alignment pin had made indentations on the channel head surfaces. *See id*. The plant inspected all 3212 open tube ends, the tubesheet, tube-to-tubesheet welds, the divider plate, and the cladding. *See id*. Plant personnel and Westinghouse evaluated the thermal-hydraulic and structural integrity of the tube ends, and found them acceptable without repairs. *See id*. They also evaluated the channel head condition and determined that the structural integrity of the dented components was not degraded and that no repairs were necessary. *See id*.

During the 1992 refueling outage, plant personnel performed a follow-up visual inspection on the hot leg channel head of steam generator 34, using a high-resolution video camera and a qualified inspector. *See id*. The follow-up inspection and comparative analysis results showed no change in the channel head since the first inspection. *See id*.

Dr. Duquette articulates no connection between this event from 25 years ago and the potential for PWSCC in the IP3 channel head, nor can we discern any.

Q174. Dr. Duquette states that NRC Information Notice 2013-11, "Crack-Like Indications at Dents/Dings in the Freespan Region of Thermally Treated Alloy 600 Steam

Generator Tubes" (July 3, 2013) (NYS000551), indicates that cracking in dented or dinged regions of Alloy 600TT tubing has been reported, and that this operating experience highlights the importance of, and the challenges to, inspecting locations susceptible to degradation and identifying inspection methods capable of detecting that degradation. Please state your view on this matter.

A174. (BMG, RJD, JRS) Again, Dr. Duquette's statements appear to constitute a general challenge to the adequacy of the Steam Generator Integrity AMP, which NYS has acknowledged is not at issue here. Regardless, Entergy does not disagree that it must "remain vigilant in its inspections of the steam generator tubes, tube-to-tubesheet welds, and divider plate and channel head assemblies at IP2 and IP3." Entergy is doing precisely that through its implementation of the Steam Generator Integrity Program, and will continue to do so during the PEO. As discussed above, that program includes inspections for foreign objects as well as for dings and dents in the steam generator tubes. *See* LRA at B-118 to B-120 (ENT00015B); *see also generally* NL-14-113 (NYS000543); NL-13-032 (NYS000537). Thus, Information Notice 2013-11 does not identify any issues or concerns that Entergy is not already addressing through its program.

Q175. Dr. Duquette claims that Entergy is relying on "a 'trust us' approach in the absence of real data on the condition of the eight Indian Point steam generators." Supplemental Duquette Testimony at 5 (NYS000532). He further states that "the current state of the divider plates, the stub runners, the channel heads, as well as the tube-to-tubesheet welds at Indian Point is largely unknown." ***Id.*** **at 8. Do you agree?**

A175. (ABC, RJD, NFA, JRS) No. The condition of all IPEC steam generator components is known and understood as a result of the Steam Generator Integrity Program. As part of the Steam Generator Integrity Program, visual inspections of the steam generator bowls, tubesheets, and tube plugs are performed periodically—as documented in the 2013 inspection report for IP3 and the 2014 report for IP2. *See* NL-14-113, Attach. 1 at § 3.0 (NYS000543); NL-13-032, Encl. 1 § 3.0 (NYS000537). Entergy also inspected the primary bowl drain area of the channel heads, to address the operating experience discussed in NSAL-12-1 (NYS000549). *See* NL-14-113, Attach. 1 § 3.0 (NYS000543); NL-13-032, Encl. 1 § 3.0 (NYS000537).

Q176. Citing the operating experience described in NSAL-12-1 (NYS000549) and IN 2013-20 (NYS000538), Dr. Duquette proposes that Entergy should, as soon as possible, perform an initial baseline inspection of IP2 and IP3 steam generator divider plate and channel head assemblies and tube-to-tubesheet welds as part of the company's "One Time Inspection Program." Supplemental Duquette Testimony at 20 (NYS000532). He later recommends that Entergy conduct follow-up inspections at least every 10 years. *Id.* at 21. Do you agree with his proposal?

A176. (ABC, RJD, NFA, JRS) No. Dr. Duquette is relying on invalid comparisons. As we explained previously, the operating experience discussed in NSAL-12-1 (NYS000549) involved [redacted], not PWSCC in divider plates or tube-to-tubesheet welds. There also is no basis to consider the issues in NSAL-12-1 as a generic safety concern. In any case, Entergy has performed inspections of all eight IPEC steam generators based on NSAL-12-1 recommendations. *See generally* NL-13-032 (NYS000537); NL-14-113 (NYS000543). Thus, NSAL-12-1 provides no reason to question the adequacy of Commitments 41 or 42.

With regard to the different event at the Surry Power Station, Unit 2, discussed in Information Notice 2013-20, the NRC explains that observed corrosion of tubesheet material was the result of a damaged tube being inadvertently placed into service from 1991 until 2006, following a faulty repair. *See* IN 2013-20 at 4-5 (NYS000538). Such a maintenance error is unrelated to the effects of aging. In any event, as noted in response to Question 170, above, Entergy has inspected all eight IPEC steam generators at IP2 and IP3 for the type of degradation discussed in NRC Information Notice 2013-20. Overall, the events at Surry are once again unrelated to the adequacy of Commitment 41 (and 42).

Q177. Dr. Duquette suggests that because IPEC has, in the past, experienced PWSCC with Alloy 600 materials, there are already corrosion risks at the facility and appropriate measures must be taken. *See* Duquette Testimony at 27 (NYS000372). Please respond to this claim.

A177. (NFA, RJD, BMG) Dr. Duquette's testimony discusses the steam generator tube leakage event which resulted in the replacement of the steam generators at IP2 in 2000. That event was a result of cracking in the mill-annealed Alloy 600 tubing material which was used in the original IP2 and IP3 steam generators. *See* NL-00-043, Letter from J. Baumstark, Consol. Edison Co. of N.Y., to NRC Document Control Desk, "Root Cause Evaluation For Steam Generator Tube Rupture Event of February 15, 2000," Attach. at 2, 4 (Apr. 14, 2000) ("NL-00-043") (ENT000567).

As previously explained, the original steam generators with mill-annealed tubing have now been replaced with steam generators using Alloy 600TT tubing at IP2 and Alloy 690TT tubing at IP3. [redacted] corrosion or cracking degradation has been noted to date in the replaced steam generators at IP2. *See* Letter from E. Shields, Westinghouse, to W. Wittich, Entergy, "Steam Generator Operational Assessment," Encl.

at 3 (June 29, 2010) (ENT000569). The historical issues Dr. Duquette identifies do not reveal any deficiency in Commitment 41 (or 42).

Q178. Dr. Lahey concludes his revised testimony on Commitment 41 by referring to certain recent failures of an IP3 "steam generator feedwater line" and a transformer as evidence that "age-related degradation concerns are not hypothetical." Revised Lahey Testimony at 103 (NYS000562). Therefore, he claims, IPEC lacks an adequate AMP for steam generators. *Id.* What is your response?

A178. (NFA, RJD, JRS) As a general matter, we agree that age-related degradation concerns are not hypothetical. But, potential degradation in a feedwater line, on the secondary side of the steam generator, is a separate issue from PWSCC in the divider plate (or tube-to-tubesheet welds). Dr. Lahey articulates no connection between the two technical issues (or between the transformer failure and potential PWSCC), nor can we discern any. Per the parties' June 23, 2015 Joint Stipulation, general challenges to the IPEC Steam Generator Integrity AMP are not part of this contention.

3. EPRI's Studies of PWSCC in Steam Generator Components

a. Overview of EPRI Steam Generator Studies

Q179. Please summarize the technical bases for EPRI's determination that divider plate cracking is not a significant safety issue.

A179. (NFA, RJD, JRS, BMG) Following the discovery of PWSCC in steam generators in Europe, EPRI studied the issue of steam generator divider plate and channel head degradation for over seven years.

The basis for EPRI's determination that divider plate cracking is not a significant safety issue is set forth in several EPRI reports, which culminated with the EPRI 2014 Report issued in October 2014. To summarize, EPRI issued its first report on this issue in June 2007. *See*

[redacted]

generally EPRI, Final Report 1014982, Divider Plate Cracking in Steam Generators – Results of Phase 1: Analysis of Primary Water Stress Corrosion Cracking and Mechanical Fatigue in the Alloy 600 Stub Runner to Divider Plate Weld Material (June 2007) (ENT000530). [redacted] At IPEC, as previously noted, both units use the Westinghouse Model 44F steam generator. LRA at 2.3-21 (ENT00015A). [redacted]

The EPRI Phase II Study, published in November 2010, [redacted]

[redacted]

Q180. [redacted]

[redacted]

[redacted]

[redacted]

[redacted] EPRI has published follow-up studies on this topic, as discussed below. In addition to this effort, EPRI conducted further research to assess the behavior of PWSCC cracks when they encounter material that is not susceptible to PWSCC (such as the low-alloy steel channel head), and to develop crack growth models and flaw acceptance criteria. *See* EPRI, Nuclear Sector Roadmaps at 36-37 (Jan. 2012) (NYS000393); *see also generally* EPRI 2014 Report (NYS000544A-D).

[redacted]

Q181. [redacted]

A181. (JRS, BMG, NFA, RJD, ABC) [redacted]

[redacted]

Q182. Dr. Duquette further speculates that U.S. PWR steam generators "may run hotter and be subject to greater stresses than their French counterparts." Supplemental Duquette Testimony at 17-18 (NYS000532). Is that correct?

A182. (RJD, NFA, JRS, BMG) No. Dr. Duquette provides no technical basis for his speculation that U.S. steam generators "may" run hotter than French steam generators. In fact, French plants typically operate with hot leg temperatures of approximately 325°C. *See* T. Couvant et al., Paper Reference No. A004T04, PWSCC of Steam Generator Divider Plates in Alloy 600: Coupling Field Characterizations with R&D Studies at 1 (Fontevraud 7, Sept. 26-30, 2010) (ENT000719). This is higher than the nominal hot leg temperatures of the IPEC units. The fact that the French plants generally operate at a slightly higher temperature 617ºF (325ºC) than the IPEC plants—which have hot leg temperatures of approximately 600ºF (316ºC) at both units—is significant. As Dr. Duquette acknowledges, *see* Supplemental Duquette Report at 18 (NYS000532), lower operating temperatures are advantageous from a PWSSC initiation and crack

[redacted]

growth perspective. Furthermore, if PWSCC did initiate, then the lower temperature would result in crack growth significantly less than crack growth rates at the French plants' crack growth rates. *See generally* EPRI, MRP-55, Rev. 1, Materials Reliability Program: Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Thick-Wall Alloy 600 Materials (Nov. 2002) (ENT000721).

Q183. Dr. Duquette next asserts "steam generators with a number of plugged tubes may be more susceptible to PWSCC and fatigue induced cracking than steam generators at French reactors." Supplemental Duquette Testimony at 18 (NYS000532). What is your response to his claims?

A183. (RJD, NFA, JRS, BMG) We disagree with Dr. Duquette. As a threshold matter, insofar as Dr. Duquette implies that French plants do not plug steam generator tubes, he is incorrect. *See* M. Boccanfuso, et al., Paper Reference No. A165-T06, Steam Generator Mechanical Plug Failure: A Tribologic Problem at 1 (Fontevraud 7, Sept. 26-30, 2010) (ENT000720) ("On 13 May 2008, after a primary circuit pressure testing on Saint-Alban reactor 2, a tube on the hot leg of the steam generator that had been plugged was found without a plug.").

With regard to the issue of plugged tubes, the steam generators at IPEC were designed with enough margin to allow 10% of the tubes to be plugged before there is any effect on plant operation. *See*, *e.g.*, Entergy, IP3 UFSAR, Revision 20, Chapter 14 at 4 (§ 14.0.1) (NYSR0013I). In comparison, there are very few tubes plugged at IPEC. Each unit has 12,856 tubes (3,214 tubes per steam generator x 4 steam generators per unit). *See* NL-13-032, Encl. 1 § 3.0(f) (NYS000537); NL-14-13, Attach. 1 at 5 (NYS000543). At IP2, only 48 tubes are plugged, and at IP3 only 16 tubes are plugged. NL-14-113, Attach. 1 at 4 (NYS000543); NL-13-032, Encl. 1 § 3.0(f) (NYS000537). These numbers are far less than the design assumption of more than 10%

(1,285) of the tubes plugged. Therefore, the small number of plugged tubes at IP2 and IP3 has no impact on plant operation, and does not cause the IPEC steam generator tubes or any other components to run hotter than the French plants. In fact, as noted in response to Question 182, the IPEC hot leg temperatures are lower than those reported for the French plants.

Q184. Are there other EPRI studies addressing divider plate cracking?

A184. (NFA, RJD, JRS, BMG) Yes. [redacted]

[redacted]

[redacted]

Q185. In the previous response, you referred to "highly cold worked material." What is cold working?

A185. (BMG) Cold working is the result of mechanical working (*e.g.*, rolling, drilling, shearing, bending, cutting, stamping, machining, etc.) of a metal at usually ambient temperatures. It is common for a fabrication process to produce cold work that only affects a thin superficial surface layer and does not affect the bulk material underneath. *See*, *e.g.*, *id.* at 5-18 (NYS000544B); EPRI July 30, 2015 NRC Meeting Slides at 48, 54 (ENT000714).

Q186. Dr. Duquette asserts that Entergy has not confirmed that the IPEC steam generators do not have a layer of cold-work potentially susceptible to cracking, and that "any cold-worked surfaces of the steam generators could be vulnerable to the same conditions experienced by the European reactors." He also claims that "[t]here is some evidence that the tube-to-tubesheet welds in IP2 have been cold-worked." Duquette Supplemental Testimony at 16. Does he raise a valid concern?

A186. (RJD, NFA, BMG) No. First, the tube-to-tubesheet welds at IP2 are no longer considered part of the RCS pressure boundary due to NRC approval of Entergy's Technical Specification amendment to implement the H* methodology at IPEC. As a result, cold work or cracking in the tube-to-tubesheet weld area is not a concern. [redacted]

[redacted]

Q187. [redacted]

A187. (BMG) [redacted]

Q188. You have made several references to the chromium content of a metal. Can you explain further how chromium content affects a material's susceptibility to PWSCC?

A188. (BMG) [redacted]

[redacted]

[redacted]

[redacted]

Q189. Please summarize the primary technical findings and conclusions of the EPRI 2014 Report, particularly as they are relevant to the contention.

A189. (NFA, BMG, TJG) [redacted].

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

Q190. Given these conclusions, what are Entergy's plans with regard to inspections of the steam generator divider plates at IP2 and IP3, and the tube-to-tubesheet welds at IP3?

A190. (NFA, RJD, ABC) As noted previously, Entergy is planning to undertake inspections of the IP3 divider plates during the Spring 2017 refueling outage, which is when the first inspections are scheduled. The specific inspection techniques to be used have not yet been finalized, but Entergy is evaluating the use of EVT-1 inspections using a robot-mounted camera, similar to methods used for inspections of other steam generator components.

As a member of EPRI and per its commitment to monitor operating experience, Entergy intends to continue to monitor the NRC's review of the EPRI 2014 Report (NYS000544A-D). Should the NRC conclude that it produces an acceptable basis for not conducting steam generator divider plate inspections, then Entergy will further consider options related to Commitment 41. If Entergy determines to make any revisions to that commitment, it will do so in full compliance with the commitment change procedures described in Section V.A of this testimony.

As to the tube-to-tubesheet welds, Entergy is evaluating the 2014 EPRI Report to determine whether it supports the closure of Commitment 42 for IP3 under the analysis option.

b. The EPRI 2014 Report Is Not Limited to 40 Years

Q191. Dr. Duquette states that the EPRI 2014 Report does not resolve his concerns about the IPEC steam generators. *See* Supplemental Duquette Testimony at 13 (NYS000532). [redacted]

A191. (BMG, ABC) [redacted]

[redacted]

[redacted]

> [redacted]

[redacted]

[redacted] Thus, the entering conditions for the crack growth evaluation in the EPRI 2014 Report are highly conservative and, to our knowledge, have never been observed in the U.S. or elsewhere, in 20 years or more of operation. For this reason, and contrary to Dr. Duquette's speculation, the report covers up to 60 years of steam generator operation. *See* EPRI July 30, 2015 NRC Meeting Slides at 57 (ENT000714).

[redacted]

Q192. Is there an additional reason why the EPRI 2014 Report covers a steam generator life span longer than 40-years?

A192. (TJG) Yes. [redacted]

[redacted] [redacted]

[redacted]

[redacted]

[redacted]

(ABC) [redacted]

As indicated in the LRA, the projected numbers of cycles for 60 years of plant operation are generally less that the numbers assumed for 40 years of operation at the time of initial plant design. *See* LRA at 4.3-4 to -7, tbls. 4.3-1, 4.3-2 (ENT00015B). [redacted]

Q193. [redacted]

A193. (BMG) [redacted]

[redacted]

Q194. Dr. Duquette asserts that the EPRI 2014 Report is inapplicable to IP2 because EPRI's research was based on components made of Alloy 690, which is more PWSCC-resistant than the Alloy 600 found in certain IP2 steam generator components. *See* Supplemental Duquette Testimony at 15 (NYS000532). What is your response to that claim?

A194. (RJD, NFA, JRS, BMG, TJG, ABC) With respect to Commitment 42 (regarding the tube-to-tubesheet welds), Dr. Duquette's criticism is irrelevant because IP2 has already been granted an H* license amendment that redefines the RCS pressure boundary. [redacted]

Q195. Dr. Duquette opines that the 2014 EPRI Report should not provide the basis for licensees' retraction of inspection commitments, because "it would be irresponsible to rely exclusively on mathematical modeling data" to support that action. Supplemental Duquette Testimony at 19 (NYS000532). As support, he cites an alleged nonconservatism in NRC Branch Technical Position (BTP) 5-3 and the San Onofre Nuclear Generating Station ("SONGS") steam generator tube rupture event. *See id.* at 18-19 (citing NYS000518, NYS000519, NYS000552). What is your response?

A195. (RFD, JRS, BMG) Dr. Duquette's position is flawed in multiple respects. Dr. Duquette has identified no actual, specific deficiencies in the EPRI 2014 Report. Further, his references to the steam generator degradation issues at SONGS and to claims of potential non-conservatism in BTP-5-3 are not relevant to the steam generator inspection plans established by Entergy. SONGS experienced tube failures as a result of a major design flaw that became evident during the first cycle of operation following steam generator replacement. *See* SONGS Lessons Learned Memo at 3 (NYS000552). BTP 5-3 relates to the fracture toughness of ferritic materials used for pressure-retaining components of the RCS boundary and certain RPV embrittlement calculations. *See generally* M. Kirk and S. Sheng, Assessment of BTP 5-3 Protocols to Estimate $RT_{NDT(u)}$ and USE (NRC/EPRI Annual Materials Issue Program Information Exchange Meeting, June 4, 2014) (NYS000518). The discovery of potential non-conservatisms in BTP 5-3 has no relevance to PWSCC in steam generator components.

c. Shock Loads and Maintenance of Intended Functions

Q196. Dr. Lahey claims that while the EPRI reports indicate that cracked steam generator divider plates can withstand the transient pressure differentials expected during various postulated accidents, the effects of "shock loads" have apparently not been considered. Revised Lahey Testimony at 91-92 (NYS000562). He claims that this is a significant safety issue because a gross failure of the divider plate under such loads could compromise core cooling. *Id.* at 92-93. According to Dr. Lahey, the November 2010 EPRI Phase II Study (ENT000523) and June 2012 EPRI Report (ENT000524) do not resolve his concerns. Revised Lahey Testimony at 101 (NYS000562). Please respond to his statements.

A196. (RGL, NFA, RJD, JRS) Dr. Lahey's assertions lack a technical basis. While Dr. Lahey does not specifically define what "shock loads" he is concerned about, it appears that his principal concern is the potential effect of a thermal or pressure shock load on a divider plate that has been "seriously age-weakened" by "thermal fatigue and PWSCC-induced embrittlement." Revised Lahey Testimony at 93 (NYS000562). As a threshold matter, PWSCC (as its name implies) causes cracking, not embrittlement as Dr. Lahey states. *See* MRP-175 at A-1 ("General Description of Stress Corrosion Cracking") (NYS000319).

[redacted]

[redacted]

Q197. Does the EPRI Phase II Study address Dr. Lahey's concerns about potential compromise of the steam generator's heat transfer function? *See* Revised Lahey Testimony at 91-93 (NYS000562).

A197. (JRS) Yes. [redacted]

Q198. Do you agree with Dr. Lahey that the "quasi-static analysis that EPRI has done for LOCA loads does not address my concerns at all" and that "special computer codes that can accurately track the thermal and pressure transients on the divider plate" are necessary? *See* Revised Lahey Testimony at 102 (NYS000562).

A198. (RGL, JRS, NFA, RJD) No. It is not clear why Dr. Lahey believes that the analysis completed by EPRI for LOCA loads was inappropriate. [redacted] Therefore, the current licensing basis analysis of the divider plate defines the most limiting condition. Dr. Lahey identifies no reason to believe that these loads were incorrectly calculated.

Q199. Dr. Lahey expresses a concern about "cracks spreading from tubesheet cladding to tube-to-tubesheet welds." Revised Lahey Testimony at 94 (NYS000562). How do you view that concern from a technical perspective?

A199. (JRS, BMG) [redacted] Dr. Lahey's concern thus lacks a technical basis.

Q200. In Rebuttal, Dr. Hopenfeld claims that EPRI's studies of steam generator PWSCC are incomplete and inadequate. He asserts that EPRI failed to consider the impact of a failed divider plate during a station blackout event, "also called the high dry sequence," where the core is uncovered, all of the steam generators are dry, and steam is flowing by natural convection through the primary system. *See* Hopenfeld Rebuttal at 10.

A200. (ABC, NFA, JRS, RGL) Dr. Hopenfeld's concern regarding the potential effect of a "breached" SG divider plate on a station blackout ("SBO") event is misplaced. [redacted]

[redacted]

[redacted] The "high-dry" accident sequence to which he refers is a beyond-design basis, severe accident. *See generally* Entergy, IP2 UFSAR, Revision 25, Ch. 14 (ENT000634) (not including high-dry scenario); IP3 UFSAR, Ch. 14 (NYSR0013I-J) (same). It is also unrealistic, because an SBO is mitigated by steam-driven auxiliary feedwater pumps feeding the steam generators. The steam generators, therefore, would not be "dry." Furthermore, if the steam generators were dry there would be no driving force for natural circulation and no steam flow "by natural convection from the core to the steam generators and back to the core" as Dr. Hopenfeld posits. Hopenfeld Rebuttal at 10.

D. <u>Entergy's Commitment 30 Regarding Reactor Vessel Internals Has Been Fully Implemented and Supports the Finding that the Effects of Aging on RVI Components Will be Adequately Managed</u>

Q201. What is Commitment 30?

A201. (ABC, NFA, RJD, JRS, RGL, TJG) In Commitment 30, 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 PEO. *See* SSER 2 at A-11.

Q202. Have you provided testimony on the technical and regulatory issues related to the implementation of Commitment 30?

A202. (ABC, NFA, RJD, JRS, RGL, TJG) Yes. Our testimony on Contention NYS-25 addresses these matters in detail and is incorporated by reference here. *See generally* Entergy's NYS-25 Testimony (ENT000616).

Q203. Has Entergy satisfied Commitment 30?

A203. (ABC, NFA, RJD, JRS, RGL, TJG) Yes. As discussed in further detail in our Testimony on NYS-25, Entergy has fulfilled Commitment 30. *See* Entergy's NYS-25 Testimony at Q52 (ENT000616); *see also* NL-11-107, Letter from F. Dacimo, Entergy, to NRC Document

[redacted]

Control Desk, "License Renewal Application – Completion of Commitment # 30 Regarding the Reactor Vessel Internals Inspection Plan" (Sept. 28, 2011) (NYS000314); 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 14 (Sept. 27, 2013) (NYS000502). More generally, Entergy's NYS-25 Testimony explains the adequacy of the IPEC RVI AMP. *See generally* Entergy's NYS-25 Testimony (ENT000616). That testimony is incorporated by reference here.

Q204. Does Dr. Lahey provide testimony regarding Entergy's RVI AMP on the record for this contention?

A204. (ABC, NFA, RJD, JRS, RGL, TJG) Yes.

Q205. Does Dr. Lahey's 2015 testimony regarding Entergy's RVI AMP in this contention (NYS000562), differ from his testimony on NYS-25 (NYS000482), or NYS-26B/RK-TC-1B (NYS000530)?

A205. (ABC, NFA, RJD, JRS, RGL, TJG) No. We have reviewed the three documents and have identified no substantive differences between them on the topic of aging management of RVI components. We respond to Dr. Lahey's claims regarding the aging management of RVIs in our testimony on NYS-25, which, as we have noted, is incorporated by reference here. *See generally* Entergy's NYS-25 Testimony (ENT000616). With regard to Dr. Lahey's claims regarding EAF evaluations, including EAF evaluation of RVI components, we respond to those claims in our testimony on NYS-26B/RK-TC-1B, which is incorporated by reference here. *See generally* Entergy's NYS-26B/RK-TC-1B Testimony (ENT000679).

[redacted]

Q206. Does Dr. Hopenfeld provide different testimony regarding Entergy's RVI AMP on the record for this contention?

A206. (ABC, NFA, RJD, JRS, RGL, TJG) Not directly. His testimony focuses on EAF issues, but it does include claims regarding EAF evaluations for RVI components. *See, e.g.*, Supplemental Hopenfeld Report at 14 (RIV000144) (alleging Entergy's EAF evaluations did not account for RVI radiation exposure).

Q207. Does Dr. Hopenfeld's testimony regarding EAF evaluations for RVIs in this contention differ from his testimony on NYS-26B/RK-TC-1B?

A207. (ABC, NFA, RJD, JRS, RGL, TJG) No. Dr. Hopenfeld provided a single, combined report that does not distinguish between NYS-26B/RK-TC-1B and NYS-38/RK-TC-5. *See generally* Supplemental Hopenfeld Report (RIV000144). Therefore, we have responded to Dr. Hopenfeld's claims regarding EAF evaluations for RVI components our testimony on NYS-26B/TK-TC-1B, which is incorporated by reference here. *See* Entergy's NYS-26B/RK-TC-1B §§ V.E.2 (ENT000679).

VI. <u>CONCLUSIONS</u>

Q208. Please summarize your testimony and the bases for your conclusion that NYS-38/RK-TC-5 lacks factual and technical merit.

A208. (NFA, ABC, JRS, MAG, RJD, BMG) NYS-38/RK-TC-5 lacks merit for the following principal reasons:

- Entergy has provided sufficient information to demonstrate that its AMPs at issue in this contention are consistent with NUREG-1801, Revision 1, and meet the intent of NUREG-1801, Revision 2. There is no further requirement to complete all AMP implementation activities prior to the issuance of a renewed license, as Intervenors claim. Nevertheless, all AMP implementation activities required to be completed prior to the PEO have already been completed for IP2, and will be completed for IP3 by the time it enters the PEO.
- As explained in detail in Entergy's testimony on NYS-26B/RK-TC-1B, consistent with accepted NRC guidance and industry methods of analysis, Entergy has demonstrated that

[redacted]

the effects of fatigue, including the effects of the reactor water environment, will be adequately managed such that affected components will remain capable of performing their intended function throughout the PEO, consistent with 10 C.F.R. §§ 54.21(a)(3), (c)(1)(iii), and 54.29(a).

- Entergy analyzed the effects of EAF for the NUREG/CR-6260 locations at IPEC, consistent with the guidance in NUREG-1801, Revision 1 and NUREG/CR-6583 and 5704. Entergy's Commitment 43 is to review its design basis ASME Code fatigue evaluations to determine whether the NUREG/CR-6260 locations that have been evaluated for the effects of the reactor coolant environment on fatigue usage are the limiting locations for IPEC. This commitment is sufficiently specific to satisfy 10 C.F.R. § 54.21(a)(3) and (c)(1)(iii), and was found acceptable by the NRC Staff, and has now been completed. This commitment further supports the adequacy of the Entergy FMP, by providing additional assurance that the CLB will be maintained throughout the PEO.

- Entergy's Commitment 44 to provide written justification for future user interventions in analyses conducted under the WESTEMS™ software has been implemented with respect to the completed EAF calculations and further supports the adequacy of future analyses that may be conducted under Entergy's FMP.

- The recent foreign operating experience identified by the NRC Staff in the SRP-LR does not provide any basis for an alleged deficiency in Entergy's Water Chemistry Program on the issue of potential PWSCC of steam generator divider plates. On the contrary, EPRI research to date supports the conclusion that this issue is not a safety concern for the steam generators at IPEC. Nevertheless, Entergy has committed to undertake timely inspections to confirm the effectiveness of its AMP. In addition, Entergy has completed Commitment 42 for IP2 through its NRC-approved H* license amendment. Commitments 41 and 42 reinforce the conclusion that the effects of PWSCC will be adequately managed such that affected components will remain capable of performing their intended function throughout the PEO, consistent with 10 C.F.R. §§ 54.21(a)(3) and 54.29(a).

- 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 because MRP-227-A provides an NRC-accepted approach for managing the effects of aging on RVIs, and Entergy's RVI AMP is consistent with MRP-227-A. Thus, there is reasonable assurance that the effects of aging on the IPEC RVIs will be adequately managed throughout the PEO, as required by 10 C.F.R. Part 54.

Q209. Does this conclude your testimony?

A209. (NFA, ABC, JRS, MAG, RJD, RGL, TJG, BMG) Yes.

Q210. In accordance with 28 U.S.C. § 1746, do you state under penalty of perjury that the foregoing testimony is true and correct?

A210. (NFA, ABC, JRS, MAG, RJD, RGL, TJG, BMG) Yes.

[redacted]

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