NRCR000161
Submitted: August 10, 2015

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

NRC STAFF TESTIMONY OF DR. ALLEN L. HISER AND MR. KENNETH J. KARWOSKI
CONCERNING PORTIONS OF STATE OF NEW YORK AND RIVERKEEPER, INC
JOINT CONTENTION NYS-38/RK-TC-5

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

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

A1b. [KJK] My name is Mr. Kenneth J. Karwoski. I am employed as the Senior Level Advisor for Steam Generator Integrity and Materials Inspection, Division of Engineering, Office of Nuclear Reactor Regulation, U.S. NRC, in Washington, DC. A statement of my professional qualifications is attached as Ex. NRC000157.

Q2. Please describe your current responsibilities.

A2a. [ALH] My responsibilities include providing technical advice and assistance to the Division of License Renewal on a variety of technical, regulatory and policy issues related to aging management of nuclear power plant systems, structures, and components. My responsibilities include serving as a

lead technical expert for aging management evaluation and assisting other NRC staff as they implement their review of license renewal applications.

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

Previously, I was the chief of the Steam Generator Tube Integrity and Chemical Engineering Branch of the Division of Engineering, Office of Nuclear Reactor Regulation. In this position I managed technical reviews of related licensing actions and safety issues, including license renewal application sections that pertain to steam generators

A2b. [KJK] My responsibilities include providing technical advice and assistance to the Division of Engineering on a variety of technical, regulatory and policy issues related to steam generators and materials engineering. For the last 14 years, my responsibilities included serving as a lead technical expert for steam generators and assisting other NRC staff as they implement their review of various submittals including license renewal applications.

Q3. Please, explain your duties in connection with the Staff's review of the License Renewal Application ("LRA") submitted by Entergy Nuclear Operations, Inc. ("Entergy," "Applicant" or "Licensee") for the renewal of Indian Point's License Nos. 50-247-LR and 50-286-LR.

A3a. [ALH] As the Senior Technical Advisor in the Division of License Renewal, I assist and guide the Staff in its review of information from applicants. For the review of the Indian Point LRA related to the steam generator divider plate and tube-to-tubesheet welds, I assisted the main reviewer by doing a peer review of her findings. I also was a part of the team that directly contributed to the

development of the Staff's finding that the Applicant's planned aging management of these items provided reasonable assurance that they will continue to perform their intended functions for the period of extended operation.

A3b. [KJK] As the Senior Level Advisor in the Division of Engineering, I assist and guide the Staff in its review of steam generator issues. For the review of the Indian Point LRA related to the steam generator divider plate and tube-to-tubesheet welds, I assisted the Division of License Renewal by providing insights into steam generator designs and operating experience. I also was a part of the team that directly contributed to the development of the Staff's finding that the Applicant's planned aging management of these items provided reasonable assurance that they will continue to perform their intended functions for the period of extended operation.

Q4. Why are you testifying here today?

A4. [ALH, KJK] The purpose of our testimony is to present the Staff's analysis of the Applicant's proposed aging management for the steam generator divider plate assemblies ("SGDPs") and tube-to-tubesheet welds ("TTSWs") and the Staff's views with respect to Contention NYS-38/RK-TC-5 ("NYS-38/RK-TC-5"), filed by the State of New York and Riverkeeper ("Intervenors"). As directed by the Board, we are also providing rebuttal testimony to the portion of NYS-38/RK-TC-5 related to steam generators. Our testimony is being used to support the Staff's Statement of Position concerning NYS-38/RK-TC-5, which the Staff is filing simultaneously with our testimony.

Q5. What did you review in order to prepare your testimony?

A5. [ALH, KJK] In preparation for our testimony we reviewed documents relevant to NYS-38/RK-TC-5, including Board orders, the parties' filings regarding this contention (including pre-filed testimony, report, supplemental testimony, and supporting exhibits), pertinent NRC guidance,

applicable regulations in Part 54, various presentation materials and papers related to SGDP and TTSW cracking, the Applicant's license renewal application (including any applicable applicant responses to requests for additional information and amendments to the application), the NRC safety evaluation report for Indian Point Units 2 and 3, and Supplement 1 and Supplement 2 to the NRC safety evaluation report for Indian Point Units 2 and 3. A list of the documents includes:

(1) "State of New York and Riverkeeper's New Joint Contention NYS-38/RK-TC-5," dated September 30, 2011 (Agency Document Access and Management System ("ADAMS") Accession No. ML11273A196).

(2) "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 (ADAMS Accession No. ML11273A195).

(3) "Declaration of Dr. Richard T. Lahey, Jr.," dated September 30, 2011 (Ex. NYS000302) (ADAMS Accession No. ML11273A192).

(4) "NRC Staff's Answer to State of New York and Riverkeeper's Joint Motion to File a New Contention, and New Joint Contention NYS-38/RK-5," dated October 25, 2011 (ADAMS Accession No. ML11298A379).

(5) "Applicant's Opposition to New York State's and Riverkeeper's Joint Motion to Admit New Contention NY-38/RK-TC-5," dated October 25, 2011 (ADAMS Accession No. ML11298A380).

(6) "State of New York and Riverkeeper's Joint Reply in Support of Admission of Proposed Contention NYS-38/RK-TC-5," November 1, 2011 and its associated documents (ADAMS Accession No. ML11305A265)

(7) Atomic Safety and Licensing Board ("Board") Memorandum and Order (Admitting New Contention NYS-38/RK-TC-5) dated November 10, 2011 (ADAMS Accession No.

ML11314A211).

(8) "Applicant's Motion for Clarification of Licensing Board Memorandum and Order Admitting Contention NYS-38/RK-TC-5" dated November 21, 2011 (ADAMS Accession No. ML11325A433).

(9) "State of New York and Riverkeeper's Joint Response to Entergy's Motion for Clarification about Contention NYS-38/RK-TC-5," dated December 1, 2011 (ADAMS Accession No. ML11335A363).

(10) Board's Order (Granting Entergy's Motion for Clarification of Licensing Board Memorandum and Order Admitting Contention NYS-38/RK-TC-5) dated December 6, 2011 (ADAMS Accession No. ML11340A088).

(11) NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," Revision 2, dated December 2010 (Ex. NRC000009) (ADAMS Accession No. ML103490036).

(12) NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants" (SRP-LR), Revision 2, dated December 2011 (Ex. NYS000161) (ADAMS Accession No. ML103490036).

(13) Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3 (NUREG-1930, Volumes 1 and 2), dated November 2009 (Ex. NRC000005) (ADAMS Accession No. ML093170451 and ADAMS Accession No. ML093170671, respectively).

(14) Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3 (NUREG-1930, Supplement 1), dated August 2011 (Ex. NRC000006) (ADAMS Accession No. ML11242A215).

(15) Request for Additional Information for the Review of the Indian Point Nuclear Generating Unit Numbers 2 and 3, License Renewal Application, dated February 10, 2011 (Ex. NYS000199) (ADAMS Accession No. ML110190809).

(16) Response to Request for Additional Information (RAI), Aging Management Programs, Indian Point Nuclear Generating Units Nos. 2 & 3, Docket Nos. 50-247 and 50-286, License Nos. DPR-26 and DPR-64, dated March 28, 2011 (Ex. NYS000151) (ADAMS Accession No. ML110871459).

(17) Response to Request for Additional Information (RAI), Aging Management Programs, Indian Point Nuclear Generating Units Nos. 2 & 3, Docket Nos. 50-247 and 50-286, License Nos. DPR-26 and DPR-64, dated July 14, 2011 (Ex. NYS000152) (ADAMS Accession No. ML12059A086).

(18) Clarification for Request for Additional Information (RAI), Aging Management Programs, Indian Point Nuclear Generating Units Nos. 2 & 3, Docket Nos. 50-247 and 50-286, License Nos. DPR-26 and DPR-64, dated July 27, 2011 (Ex. NYS000153) (ADAMS Accession No. ML11208C459).

(19) Clarification for Request for Additional Information (RAI), Aging Management Programs Indian Point Nuclear Generating Unit Nos. 2 & 3, Docket Nos. 50-247 and 50-286, License Nos. DPR-26 and DPR-64, dated August 9, 2011 (Ex. NYS000154) (ADAMS Accession No. ML11229A803).

Q6. Have you previously testified in this proceeding?

A6a. [ALH] Yes. I previously provided testimony in support of the Staff's position with respect to the portion of NYS-38/RK-TC-5 concerning the SGDPs and the TTSWs. That testimony was submitted as Exhibit ("Ex.") NRC000161, on August 20, 2012, and resubmitted with revisions August 10, 2015. In addition, I have submitted testimony regarding contentions NYS-25, NYS-26, and RK-TC-2.

A6b. [KJK] Yes. I previously provided testimony in support of the Staff's position with respect to the portion of NYS-38/RK-TC-5 concerning the SGDPs and the TTSWs. That testimony was submitted as Ex. NRC000161 and is being resubmitted with revisions today.

Q7. Could you explain how your testimony today relates to the previous testimony submitted in support of the Staff's position on SGDP portion of the NYS-38/RK-TC-5?

A7. [ALH, KJK] This testimony is meant to amend and supplement the previously submitted testimony contained in Ex. NRC000161 in its entirety.

Q8. Can you briefly describe the Indian Point SGDPs?

A8. [ALH, KJK] The SGDPs are located in the lower bowl of steam generators that have U-shaped steam generator tubes. The major parts in the lower portion of the steam generator design used at Indian Point are shown in Ex. NYS000376 and include the lower channel head (which is not marked), the tubesheet, and the SGDP (marked as "divider plate"). Ex. NYS000376 at OAG10001571_00002.

Q9. Can you briefly describe the TTSWs as they relate to steam generators?

A9. [ALH, KJK] The TTSWs are located on the lower surface of the tubesheet where the tubes exit the tubesheet. The TTSW can provide a pressure boundary separating the primary water from the secondary water, although some plants have amended their license to transfer that function from the TTSW to the tight interference fit between the tubes and the tubesheet. The TTSW also holds the steam generator tubes in their places.

Q10. Can you describe the structure and purpose of the lower channel head in the Indian Point steam generators?

A10. [ALH, KJK] The lower channel head is a thick-walled, hemi-sphere shaped portion of the steam generator pressure boundary, interior to which is primary coolant and exterior to which is air. The lower channel head provides a reactor coolant pressure boundary function.

Q11. Can you describe the structure and purpose of the tubesheet in the Indian Point steam generators?

A11. [ALH, KJK] The tubesheet is a thick plate through which the tubes pass, once on the inlet side of the steam generator and a second time on the outlet side. The tubesheet provides support for the tubes and a pressure boundary between the primary coolant and the secondary coolant.

Q12. Can you describe the structure and purpose of the SGDP?

A12. [ALH, KJK] The SGDP consists of several parts, including the divider plate itself, a stub runner, and welds that connect the two and also connect the stub runner to the tubesheet and the divider plate to the lower channel head. For simplicity we will refer to all parts of this assembly collectively as the SGDP. The SGDP directs the flow of primary coolant from the inlet nozzle into the steam generator tubes and directs flow exiting the steam generator tubes to the outlet nozzle. The SGDP primarily provides a flow direction function. The SGDPs at many plants incorporate a designed through-wall hole between the hot and cold leg portions of the steam generator, at the interface of the SGDP with the channel head at the bottom of the steam generator. This designed hole allows communication of the hot and cold leg flows and/or is used to drain both the inlet and outlet sides of the lower channel head during plant shutdown conditions. The SGDP does not provide a pressure boundary function to separate the primary coolant from the secondary coolant or from the atmosphere.

Q13. Are you familiar with NYS-38/RK-TC-5?

A13. [ALH, KJK] Yes, we are familiar with NYS-38/RK-TC-5. As stated in the Board Memorandum and Order (Admitting New Contention NYS-38/RK-TC-5), (November 10, 2011) (ADAMS Accession No. ML11314A211) ("Order") and Board Order (Granting Entergy's Motion for Clarification of Licensing Board Memorandum and Order Admitting Contention NYS-38/RK-TC-5), (December 6, 2011) (ADAMS Accession No. ML11340A088) ("Memorandum and Order"), NYS-38/RK-TC-5 questions whether Entergy has a program that will manage the effects of aging of several critical components or systems and whether the proffered programs provide an adequate record and rational basis to the NRC upon which it can determine whether to grant a renewed license to Entergy. The Intervenors' contention, which relied on multiple bases, included the claim that there is insufficient information in Entergy's recent commitments that were addressed in the Supplemental Safety Evaluation Report ("SSER"). We are familiar with the Intervenors' statement of position, State of New York and Riverkeeper, Inc. Initial Statement of Position In Support of Joint Contention NYS-38/RK-TC-5 (Ex. NYS000371) ("NYS-38/RK-TC-5 SOP"), the supporting expert testimony from Dr. Richard T. Lahey Jr., Pre-Filed Written Testimony of Dr. Richard T. Lahey, Jr. Regarding Contention NYS-38/RK-TC-5 (Ex. NYS000374) ("Lahey Pre-Filed Testimony"), and Dr. David J. Duquette, Prefiled Written Testimony of Dr. David J. Duquette Regarding Contention NYS-38/RK-TC-5 (Ex. NYS-000372) ("Duquette Pre-Filed Testimony") and Report of Dr. David J. Duquette in Support of Contention NYS-38/RK-TC-5 (Ex. NYS-000373) ("Duquette Report"), the accompanying exhibits including Exs. NYS000375-NYS000396 and RIV000102-RIV000106. We are also familiar with theState of New York and Riverkeeper, Inc. Revised Statement of Position in Support of Joint Contention NYS-38/RK-TC-5 (Ex. NYS000451), Pre-Filed Written Rebuttal Testimony of Dr. David J. Duquette Regarding Contention NYS-38 / RK-TC-5 (Ex. NYS000452), Pre-Filed Written Rebuttal Testimony of Dr. Richard T. Lahey, Jr. Regarding Contention NYS-38 / RK-TC-5 (Ex.

NYS000453), and accompanying exhibits. We are familiar with the State of New York and Riverkeeper, Inc. Revised Statement of Position Joint Contention NYS-38/RK-TC-5, (Ex. NYS000531), the supporting testimony of Dr. Lahey and Dr. Duquette, Revised Pre-Filed Written Testimony of Dr. Richard T. Lahey, Jr., in Support of Joint Contention NYS-38/RK-TC-5 (Ex. NYS000562) and Pre-Filed Written Supplemental Testimony of Dr. David J. Duquette in Support of Joint Contention NYS-38/RK-TC-5 (Ex. NYS000532) and accompanying exhibits.

Q14. What are the "multiple bases" that the Intervenors referred to in Contention NYS-38/RK-TC-5?

A14. [ALH, KJK] As described in the State of New York and Riverkeeper's Joint Contention NYS-38/RK-TC-5, (September 30, 2011) (ADAMS Accession No. ML11273A196) ("NYS-38/RK-TC-5"), the Intervenors' bases are: Basis (1) that Entergy has deferred defining the process to be used to determine the most limiting locations for environmentally-assisted metal fatigue calculations (CUF_{en} calculations) and selection of those locations; Basis (2) that Entergy has not specified the criteria it will use and assumptions upon which it will rely for modifying the WESTEMS computer model for CUF_{en} calculations; Basis (3) that Entergy has not adequately defined how it will manage primary water stress corrosion cracking (PWSCC) for the steam generator divider plates because it will rely on an industry "report which is not expected to be available until 2013 and, in the meantime to institute an unspecified inspection program to ascertain, long after commencement of the license renewal period, whether stress corrosion cracking is actually occurring in the divider plates of the steam generators;" and, Basis (4) that Entergy "has offered an AMP for reactor vessel internals which it will not actually follow and has promised to follow an AMP the details of which are not disclosed." See NYS-38/RK-TC-5 at 1-3. The Intervenors' expert testimony and statement of position submitted on June 19, 2012 are associated with Basis (1), Basis (2), Basis (3), and Basis (4), as described above. The

Intervenors' expert testimony and statement of position submitted on June 9, 2015, are associated with all four bases described above.

Q15. Which of the Bases in Contention NYS-38/RK-TC-5 does your testimony address?

A15. [ALH, KJK] Our testimony addresses Basis (3) of Contention NYS-38/RK-TC-5.

Q16. Could you briefly describe the issues raised in NYS-38/RK-TC-5 with respect to the steam generators?

A16. [ALH, KJK] As described in the State of New York and Riverkeeper's Joint Contention NYS-38/RK-TC-5, (September 30, 2011) (ADAMS Accession No. ML11273A196), the Intervenors contend that the Applicant's commitments for aging management of the SGDP are not sufficiently defined for the Applicant to have demonstrated that its "AMP [aging management program] in those areas are consistent with the GALL [Generic Aging Lessons Learned report] even though it has committed to comply with GALL." "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," at 2-3, dated September 30, 2011 (ADAMS Accession No. ML11273A195). The pre-filed testimony of the Intervenors' experts includes the Applicant's planned inspections of the TTSW as similarly deficient. (*See* Ex. NYS000374, Lahey Pre-Filed Testimony, at 21-22 and Ex. NYS000372, Duquette Pre-Filed Testimony, at 28) Finally, the Intervenors' experts raise concerns regarding impact of accident loads on aged components. (See Ex. NYS000532, Lahey Prefiled Supplemental Testimony, at 92). The aging effect of concern in the steam generator portion of NYS-38/RK-TC-5 is cracking due to primary water stress corrosion cracking ("PWSCC") in the SGDP and the TTSW.

Q17. Based on your review, what is your expert opinion regarding NYS-38/RK-TC-5 with respect to aging management for the SGDP?

A17. [ALH, KJK] We disagree with NYS-38/RK-TC-5 as it relates to the contended inadequacy of the aging management planned by the Applicant to address the SGDP. Based on our review, we have concluded that the Applicant's plans to manage aging in the SGDP are adequate. The Applicant chose to verify the effectiveness of the Water Chemistry Control – Primary and Secondary Program, which is intended to preclude or minimize PWSCC in the SGDP, using a one-time inspection to assess if cracking is present in the SGDP. If the Applicant detects cracking in the SGDP during the inspection, the issue would be resolved through the corrective action program ("CAP").

Q18. Why do you disagree with the NYS-38/RK-TC-5 contention related to aging management of the SGDP?

A18. [ALH, KJK] We believe that the Applicant's use of the Water Chemistry Control – Primary and Secondary Program, with a one-time inspection to validate the effectiveness of this program by verifying the absence of cracking in each of the SGDPs, adequately demonstrates that the requirements of 10 CFR 54.29(a)(1) are met. With the use of the Water Chemistry Control – Primary and Secondary Program and the one-time inspections, the Staff determined that there is reasonable assurance that the effects of aging will be managed during the period of extended operation, such that the functionality of the Indian Point steam generators will be maintained consistent with the current licensing basis.

Q19. Based on your review, what is your expert opinion regarding NYS-38/RK-TC-5 with respect to aging management in the TTSW?

A19. [ALH, KJK] We disagree with NYS-38/RK-TC-5 as it relates to the contended inadequacy of the aging management planned by the Applicant to address the TTSW. Based on our

review, we have concluded that the Applicant's plans to manage aging in the TTSW are adequate. The Applicant chose to verify the effectiveness of the Water Chemistry Control – Primary and Secondary Program, which is intended to preclude or minimize PWSCC in the TTSW, using a one-time inspection of a sample of the TTSWs to assess if cracking is present in the TTSWs (or by demonstrating that the TTSWs are not a part of the reactor coolant pressure boundary or are not susceptible to PWSCC). If the Applicant detects cracking in the TTSW during the inspection, the issue would be appropriately resolved through the CAP.

Q20. Why do you disagree with the NYS-38/RK-TC-5 contention related to aging management for cracking due to PWSCC of the TTSW?

A20. [ALH, KJK] We believe that the Applicant's use of the Water Chemistry Control – Primary and Secondary Program, with a one-time inspection to validate the effectiveness of this program by verifying the absence of cracking in a sample of the TTSW (or by demonstrating that the TTSW either are not a part of the reactor coolant pressure boundary, or are not susceptible to PWSCC), adequately demonstrates that the requirements of 10 CFR 54.29(a)(1) are met. With the use of the Water Chemistry Control – Primary and Secondary Program and the one-time inspections, the Staff determined that there is reasonable assurance that the effects of aging will be managed during the period of extended operation, such that the functionality of the Indian Point steam generators will be maintained consistent with the current licensing basis.

Q21. What is the aging effect that is at issue in NYS-38/RK-TC-5 relative to the SGDP and the TTSW?

A21. [ALH, KJK] The aging identified in NYS-38/RK-TC-5 for the SGDP and TTSW is cracking due to PWSCC. The issues with PWSCC of the SGDP are the possibility that the cracking could turn and migrate upwards to the tubesheet and then propagate to the TTSW, or the cracks could migrate

parallel to the tubesheet to the "triple point," where the SGDP and the tubesheet intersect with the steam generator channel head. In the latter case, the possibility of these potential cracks propagating through the channel head material could result in a breach of the reactor coolant pressure boundary.

The issue with PWSCC of the TTSWs, either from cracks initiated in the TTSWs or from cracks initiated in the SGDP that propagate to the TTSWs, is the possibility that such cracks, if they were to grow through the TTSW, could result in a breach of the reactor coolant pressure boundary.

Q22. Can you provide a description of PWSCC in relation to the SGDP and the TTSWs?

A22. [ALH, KJK] In general, PWSCC occurs in the primary water of pressurized water reactors when susceptible materials are subject to a sufficiently high tensile stress. For nickel alloys that are used to fabricate various components in nuclear power plants, operating experience has identified that components fabricated from nickel alloy base material, such as Alloy 600, and associated welds, such as Alloy 82 and 182 materials, can be susceptible to PWSCC, as evidenced by cracking in numerous components. Cracking due to PWSCC may occur in the SGDP and the TTSWs fabricated from Alloy 600/82/182 materials.

Alloy 690, another nickel alloy base material, and associated welds such as Alloys 52 and 152 have a higher chromium content that appears to make them much more resistant to PWSCC than Alloy 600/82/182 materials, based on laboratory testing and service experience. The Alloy 690/52/152 materials have been used in many replacement component applications, such as replacement steam generators.

Q23. What materials are used at Indian Point for the SGDP and the TTSWs?

A23. [ALH, KJK] For both IP2 and IP3, the SGDP base material is Alloy 600 and the Applicant has conservatively assumed that all the welds are composed of the weld metals associated with Alloy 600. Regarding the TTSWs, both units have tubesheets that are clad with weld materials associated

with Alloy 600. Because IP2 has tubes that are fabricated from Alloy 600, the TTSWs, which are autogenous welds, have a chemistry similar to that of Alloy 600. Because IP3 has tubes that are fabricated from Alloy 690, the autogenous TTSWs represent a mixture of Alloys 600 and 690. As a result, it was conservatively assumed by the Applicant that the TTSWs in IP3 are also susceptible to PWSCC.

Q24. What is an autogenous weld?

A24. [ALH, KJK] In general, "autogenous welds" are welds that are made by melting the steam generator tube and the tubesheet cladding material to form a fused weld. This is different from a weld made using a filler material to bridge the gap between two components. The resultant autogenous weld metal is a mixture of the two joined materials. In the case of the TTSWs, the weld is a mixture of the tube material and the cladding, such that, in general, the composition of the weld metal is intermediate between that of the tube material and that of the cladding material.

Q25. What has the Applicant proposed for aging management of the SGDP?

A25. [ALH, KJK] The Applicant has stated that aging of the SGDP will be managed using the Water Chemistry Control – Primary and Secondary Program. In addition, the Applicant committed to perform a one-time inspection to provide additional verification on the effectiveness of the Water Chemistry Control – Primary and Secondary Program. The one-time inspection program will check whether PWSCC is present in the SGDP after a sufficient amount of operation has occurred to provide a meaningful verification. Specifically, Commitment 41 states:

> Indian Point 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), i.e. prior to September 28, 2023. The IP3 steam generator divider plate inspections will be completed within the first refueling outage following the beginning of the PEO.

See Ex. NRC0000006 at A-23.

Q26. How does the Applicant's Water Chemistry Control – Primary and Secondary Program manage cracking due to PWSCC?

A26. [ALH, KJK] The Applicant's Water Chemistry Control – Primary and Secondary Program monitors and controls reactor water chemistry to minimize the environmental effect of cracking due to PWSCC in the components exposed to reactor coolant, including the SGDP and TTSWs. The Applicant uses the Water Chemistry Control – Primary and Secondary Program in conjunction with Commitments 41 (inspection) and 42 (analysis or inspection) to manage cracking due to PWSCC in the SGDP and the TTSWs, respectively. The program specifies control ranges for various chemical species, including dissolved oxygen, hydrogen, sulfate, chloride, and fluoride concentrations. Monitoring and control of these chemistry parameters mitigate the environmental effects on corrosion and stress corrosion cracking (including PWSCC) of the components in the reactor coolant system.

For example, coolant oxygen concentrations are minimized to mitigate general corrosion and stress corrosion cracking. If the dissolved oxygen concentration in the reactor coolant exceeds specified limits, actions are initiated to reduce the dissolved oxygen concentration.

Monitoring and control of sulfate concentrations helps to mitigate intergranular attack (i.e., cracking along grain boundaries) that may contribute to initiation and propagation of an intergranular crack in conjunction with PWSCC. The Water Chemistry Control – Primary and Secondary Program specifies control ranges for sulfate in the primary coolant and corresponding actions.

Similarly, the program has limits for other chemical species to limit the potential for corrosion.

Q27. What occurred after issuance of the August 2009 SER (Ex. NRC000006) for Indian Point that led the Applicant to make this commitment?

A27. [ALH, KJK] After the Indian Point SER was issued in August 2009, the NRC published

Revision 2 of the Generic Aging Lessons Learned Report ("GALL") in December 2010 (Ex. NRC000009). Based on foreign operating experience, Item IV.D1.RP-367 in Revision 2 of GALL was modified to state that "effectiveness of the chemistry control program should be verified to ensure that cracking due to PWSCC is not occurring" in Alloy 600 divider plates (id. at IV D1-3). In response to the foreign operating experience that led to this modification in Revision 2 of the GALL report, the NRC Staff requested additional information from the Applicant. After a series of Applicant responses and additional requests for information ("RAI"), the NRC Staff concluded that Commitment 41 is sufficient to verify the effectiveness of the Applicant's Water Chemistry Control – Primary and Secondary Program.

Q28. When did the NRC become aware of the foreign operating experience related to the potential for PWSCC in the SGDP?

A28. [ALH, KJK] In the 2002 to 2006 timeframe, the NRC Staff became aware of crack-like indications that were detected in the SGDPs in French steam generators. In light of the similarities between the design of the French and U.S. steam generators, the Staff began investigating the possible implications to U.S. steam generators, which included discussions with the U.S. nuclear industry and the industry's Steam Generator Task Force ("SGTF").

Q29. What is the SGTF?

A29. [ALH, KJK] The SGTF is an industry group that meets with the NRC Staff in a public forum, typically on a semi-annual basis. These meetings are interface meetings where the SGTF provides updates on recent operating experience and their plans to address that operating experience, and the status of their activities to address various steam generator issues. The Intervenors have used slides from several of the NRC-SGTF public meetings as exhibits in their testimony.

Q30. Based on the preliminary information from the foreign operating experience, what were the Staff's initial concerns with the U.S. steam generators?

A30. [ALH, KJK] The Staff's initial concerns with the potential for cracks to exist in the SGDP were (1) whether the tubesheet (a thick plate which separates the radioactive side of the steam generator from the non-radioactive side) would be adversely affected by the cracks in the SGDP; and (2) whether the cracks in the SGDP would affect previous analyses used to justify limiting the extent of inspection of a portion of the steam generator tube within the tubesheet.

Q31. How did the Staff address these concerns?

A31. [ALH, KJK] After the preliminary discussions on PWSCC in the SGDP, the Staff held numerous public discussions. During these public meetings, the SGTF provided assessments of the impact from cracks in the SGDP and updates on their investigation into the foreign operating experience.

Q32. What did the SGTF do to assess the impact of PWSCC on the SGDP?

A32. [ALH, KJK] The SGTF assessed the effects of cracks in the SGDP on safety analyses for design basis events, steam generator tube repair criteria, steam generator tube repair methods, and the structural analyses of the steam generator. The results of these analyses indicated that a fully degraded SGDP does not adversely affect steam generator performance, is not a safety concern during plant operations, and does not necessitate any changes in current steam generator analyses. "NEI Steam Generator Task Force NRC/Industry Update" (ADAMS Accession No. ML102300418) (Ex. NRC000158) dated August 12, 2010, at slides 45 and 46.

Q33. Would you explain what is meant by a "fully degraded SGDP"?

A33. [ALH, KJK] For the purposes of this testimony, a "fully degraded SGDP" is assumed to have a crack that is entirely through the thickness and the entire length of the SGDP.

Q34. Did the SGTF provide any additional assessments with respect to the investigation on French steam generators?

A34. [ALH, KJK] Yes, the SGTF indicated more recent information from the French showed that the SGDP cracks do not appear to be growing deeper although some cracks have linked up with other cracks. They also indicated that the French have not taken any actions to repair SGDPs with cracks.

Q35. Has the Staff verified the SGTF assessments of SGDP and the French operating experience?

A35. [ALH, KJK] Yes.

Q36. How did the Staff verify the SGTF representation regarding the French operating experience?

A36. [ALH, KJK] The NRC Staff conducted technical exchanges with the French nuclear regulator. The operating experience provided by the SGTF is consistent with the information provided by the French nuclear regulator.

Q37. What has the Staff determined regarding the potential impact of PWSCC in the SGDP?

A37. [ALH, KJK] Given the assessments performed by the SGTF and the information the Staff has obtained regarding the operating experience with the cracks in the SGDPs in the foreign steam generators, the Staff determined that no regulatory action is warranted at this time for plants operating under their original license. The Staff's determination relied, in part, on the limited severity of the cracks that have been observed in the SGDPs and the limited number of units affected. The NRC Staff will continue to monitor operating experience as additional information may develop. If the operating

experience indicates an adverse trend, the NRC Staff will re-evaluate the need for regulatory action for current operating reactors.

Q38. Why did the Staff seek additional actions from license renewal applicants, when it had previously determined that no action was warranted in light of the available information?

A38. [ALH, KJK] In the 2009-2010 timeframe, the Staff began to consider the potential for cracks in the SGDP to grow into the reactor coolant pressure boundary. "EPRI, Implementation Status of Industry Change Management Plan for Materials Related R&D Committees Under NEI 03-08" (ADAMS Accession No. ML101600470) (Ex. NRC000159 dated June 2, 2010, at slide 50). Specifically, the Staff questioned whether a crack in the SGDP could grow into the steam generator channel head (a pressure boundary component) or into a TTSW (also a pressure boundary component in many plants). If significant cracking were to occur in the channel head or TTSW, it could compromise the integrity of the reactor coolant pressure boundary. This issue was raised during the review of license renewal applications given the potential for longer steam generator operating times combined with the possibility that inspections of the SGDP might not be performed during the original licensed period.

Q39. What does Commitment 41 state regarding the examination technique to be used for the SGDP?

A39. [ALH, KJK] Commitment 41 states that the examination technique used will be capable of detecting PWSCC in the steam generator divider plate assemblies. See Ex. NRC000006 at A-23.

Q40. Does Commitment 41 require the Applicant to use a qualified examination technique for these inspections?

A40. [ALH, KJK] No, Commitment 41 does not specify use of a qualified examination technique. Nonetheless, the performance of non-destructive testing is governed by 10 CFR Part 50, Appendix B which states, in part, that the testing should be controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria and other special requirements. As a result, the inspection method used must be demonstrated that it can perform its intended purpose/function. In this case, detect PWSCC in the SGDP.

Q41. Will the technique that Indian Point uses for its SGDP inspections be qualified?

A41. [ALH, KJK] Yes. As stated in response to Q40, the Applicant would need to use a qualified technique for compliance with 10 CFR Part 50, Appendix B.

Q42. Are you familiar with Dr. Duquette's and Dr. Lahey's testimony that there is no qualified technique to inspect the SGDP?

A42. [ALH, KJK] Yes.

Q43. Can you briefly explain what a qualified technique means?

A43. [ALH, KJK] A qualified technique is a technique that has been demonstrated through testing to perform its intended function. This demonstration is in accordance with industry standards or Codes. The rigor of the demonstration is based on the significance of the issue at hand, where issues with a high likelihood of occurrence would likely be qualified for crack detection and sizing, and issues with a low likelihood of occurrence would likely be qualified for crack detection only. For the SGDP, the

demonstration would emphasize the ability of the inspection technique to detect PWSCC but not necessarily characterize the depth and size of the flaw.

Q44. Why doesn't the Industry have any inspection techniques for the SGDP that are qualified?

A44. [ALH, KJK] The Industry develops and qualifies inspection techniques on an as-needed basis. Since there has not been a need to inspect the SGDP previously, the Industry has not qualified a technique for detecting or sizing PWSCC in the SGDP.

Q45. Has the Industry been developing inspection techniques for the SGDP that are qualified?

A45. [ALH, KJK] With the commitments made by the Applicant and other applicants who have also received renewed licenses, the Industry is working to either show that cracking is not a concern on the integrity of the reactor coolant pressure boundary or will develop qualified inspection technology for the SGDP.

Q46. Has the Staff ever accepted a similar approach where the inspection technique has not been specified?

A46. [ALH, KJK]. Yes. Steam generator tube inspections are governed by plant technical specifications. These specifications do not specify the inspection method even though the steam generator tubes are a significant fraction of the reactor coolant pressure boundary. The technical specifications indicate, in part, that the methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube.

Q47. Does the NRC have requirements that are similar in detail to what Entergy has committed?

A47. [ALH, KJK] Yes. For example, there are inspection requirements that are performance based.

Q48. Can you provide an example?

A48. [ALH, KJK] Yes. From TSTF-510, which the NRC has accepted and Indian Point has adopted, "The number and portions of the tubes inspected and the methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging [or repair] criteria." TSTF-510, Rev. 2, Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection (Mar. 1, 2011) Ex. NRC000189.

Q49. Does this requirement specify the method of inspection?

A49. [ALH, KJK] No.

Q50. Has this approach been successful for steam generator tube inspections?

A50. [ALH, KJK] Yes. Although there have been some instances where tube integrity has not been maintained, these instances tend to be isolated and there are specific circumstances surrounding these events.

Q51. Has the Commission taken any position on risk informed and performance based regulation?

A51. [ALH, KJK] Yes, the Commission has issued a white paper that defines the terms and Commission expectations regarding risk-informed and performance-based regulation. Staff Requirements - SECY-98-144 - White Paper On Risk-Informed And Performance-Based Regulation, Ex. NRC000190. The Commission has indicated that the definitions and concepts in the paper have proven suitable for application to nuclear power plants and certain non-reactor activities.

Q52. When will the Applicant perform SGDP inspections for IP2 and IP3?

A52. [ALH, KJK] Commitment 41 states that the SGDP inspections at IP2 will occur after the beginning of the period of extended operation on September 29, 2013, and prior to September 28, 2023. The inspections at IP3 will occur prior to the end of the first refueling outage following the beginning of the period of extended operation. Since the IP3 license expires on December 13, 2015, the first refueling outage after the beginning of the period of extended operation is expected to begin on or before December 13, 2017.

Q53. Why are the last allowable dates for inspection for IP2 steam generators and the IP3 steam generators so far apart?

A53. [ALH, KJK] The dates are so far apart because the age of the steam generators at IP2 and IP3 are significantly different. The current IP2 and IP3 steam generators were placed into operation in 2000 and 1989, respectively. The IP3 steam generators have 11 additional calendar years of operation beyond the IP2 steam generators.

Q54. Why is it acceptable to wait so long before inspecting the steam generators?

A54. [ALH, KJK] The inspections are scheduled after a sufficient amount of operating time has accumulated on each steam generator to allow for PWSCC to develop to detectable levels, if it develops at all. The timing of these inspections is acceptable because it provides for an increased likelihood that the degradation will be identifiable, if it were to occur in these steam generators, and it allows for detection of any degradation in the SGDP prior to it propagating to locations that could compromise the reactor coolant pressure boundary.

Q55. Which steam generators will be examined under Commitment 41?

A55. [ALH, KJK] Each steam generator will be inspected at both IP2 and IP3.

Q56. How often will the steam generators at IP2 and IP3 be inspected?

A56. [ALH, KJK] The inspections are one-time inspections and they are scheduled after sufficient operating time for each steam generator to allow for PWSCC to develop to detectable levels, if it develops at all. Based on operating experience to date, the inspections are also early enough in the steam generator operating life to detect any cracking prior to the cracks jeopardizing the safety functions of the reactor coolant pressure boundary components.

Q57. What are the acceptance criteria for the Commitment 41 inspections?

A57. [ALH, KJK] Because these examinations are to verify effectiveness of the Water Chemistry Control – Primary and Secondary Program, the acceptance criterion is the absence of cracking in the SGDP. Should cracking be identified, then the Applicant would enter the findings into its CAP. Through the CAP, the Applicant would take actions to address any cracking identified as a result of the inspections.

Q58. If cracking is identified, what actions will the Applicant be required to take?

A58. [ALH, KJK] Should cracking be identified during any of the examinations at Indian Point, the Applicant will assess the finding within its CAP to determine what actions are necessary to address the specific finding, what additional actions may be necessary for the steam generators at the unit, what additional actions may be necessary for the other unit, and what effect the finding has on the need to implement follow-up or periodic inspections. Of course the appropriate corrective action will depend on the actual inspection findings, but could include more detailed evaluation of the inspection results, repair of the affected component(s), or replacement of the affected component(s).

Q59. Why does the NRC believe that the Applicant's planned aging management is adequate for the SGDP?

A59. [ALH, KJK] We believe that the Applicant's planned aging management is adequate for the SGDP for several reasons. First, the primary aging management is provided by the Applicant's Water Chemistry Control – Primary and Secondary Program, which has proven to be effective in managing cracking. At this time, there is no evidence that PWSCC in the SGDP has compromised the safe operation of steam generators in the U.S. or in foreign plants.

Second, the planned examinations to verify the effectiveness of the Water Chemistry Control – Primary and Secondary Program will confirm the condition of the SGDP and the effectiveness of the Water Chemistry Control – Primary and Secondary Program. Third, the timing, scope, acceptance criteria and corrective actions for the examinations will be sufficient to ensure the continued safe condition of the steam generators.

Q60. What has the Applicant proposed for aging management of the TTSWs?

A60. [ALH, KJK] The Applicant has stated that aging of the TTSWs will be managed using the Water Chemistry Control – Primary and Secondary Program. In addition, the Applicant committed to

additional actions. Commitment 42 requires the Applicant to conduct an inspection of the TTSWs or show by analysis that the TTSWs are not susceptible to PWSCC or seek a license amendment redefining the reactor coolant pressure boundary such that the welds are no longer part of it. Specifically, Commitment 42 states:

Indian Point will develop a plan for each unit to address the potential for cracking of the primary to secondary pressure boundary due to PWSCC of TTSWs using one of the following two options.

> <u>Option 1 (Analysis)</u>
>
> Indian Point will perform an analytical evaluation of the steam generator tube-to-tubesheet welds in order to establish a technical basis for either determining that the tubesheet cladding and welds are not susceptible to PWSCC, or redefining the pressure boundary in which the tube-to tubesheet weld is no longer included and, therefore, is not required for reactor coolant pressure boundary function. The redefinition of the reactor coolant pressure boundary must be approved by the NRC as part of a license amendment request.
>
> <u>Option 2 (Inspection)</u>
>
> Indian Point will perform a one-time inspection of a representative number of tube-to-tubesheet welds in each steam generator to determine if PWSCC cracking is present. If weld cracking is identified:
>
> a. The condition will be resolved through repair or engineering evaluation to justify continued service, as appropriate, and
> b. An ongoing monitoring program will be established to perform routine tube-to tubesheet weld inspections for the remaining life of the steam generators.

See Ex. NRC0000006, at A-23 and A-24.

The commitment states that Option 1 will be implemented prior to March 2024 for IP2, and prior to the end of the first refueling outage following the beginning of the period of extended operation for IP3. The commitment states that Option 2 will be implemented between March 2020 and March 2024 for IP2, and prior to the end of the first refueling outage following the beginning of the period of extended operation for IP3.

Q61. What occurred after issuance of the SER for Indian Point that led the Applicant to make this commitment?

Q61. [ALH, KJK] As we explained previously, after the Staff issued the Indian Point SER, the

NRC subsequently published Revision 2 of the Generic Aging Lessons Learned Report ("GALL"), which, in part, modified Item IV.D1.RP-385 for the TTSW (Ex. NRC000009 at IV D1-8). GALL, Revision 2, states that "A plant-specific program is to be evaluated; the effectiveness of the water chemistry program should be verified to ensure cracking is not occurring." Similar to the RAIs with respect to the SGDP, the Staff and Applicant exchanged a series of RAIs and responses resulting in the Applicant's adoption, and the Staff's acceptance, of Commitment 42.

Q62. How are Option 1 and Option 2 of Commitment 42 interrelated?

A62. [ALH, KJK] The Applicant has a choice when fulfilling its obligations under Commitment 42. If the Applicant has not sought and obtained the license amendment under Option 1 or has not established that the TTSWs are not susceptible to PWSCC by the time the inspections are scheduled to be completed under Option 2, the Applicant will be required to implement the inspections of Option 2. If cracking is identified, the Applicant would need to take any necessary corrective actions, which could include seeking a license amendment similar to the one contemplated under Option 1 of Commitment 42.

Q63. What is the Applicant required to do under Option 1 of Commitment 42?

A63. [ALH, KJK] Under Option 1, the Applicant would develop the technical basis to demonstrate that the TTSWs are not susceptible to PWSCC or that the TTSWs are not needed to ensure the integrity of the pressure boundary between the primary and secondary systems. The latter would involve development of a technical basis to redefine the reactor coolant pressure boundary to exclude the TTSWs and submittal of a license amendment to that effect for NRC approval. If the Applicant is unable to complete either of these actions or elects not to complete them, then the inspections contemplated under Option 2 would be implemented by the Applicant.

Q64. Can you provide an example of one method that might show that "the tubesheet cladding and welds are not susceptible to PWSCC?"

A64. [ALH, KJK] The Applicant would need to develop data or analyses that justifies a conclusion that the TTSWs are not susceptible to PWSCC. For example, for IP3 with Alloy 690 steam generator tubes, the Applicant might be able to perform an experimental study to demonstrate that the autogenous TTSWs have sufficiently high chromium content that there is a low likelihood of initiating PWSCC in the TTSW for IP3. This would apply to IP3 only, which uses Alloy 690 steam generator tubes. Alloy 690 has been demonstrated to be more resistant to PWSCC than Alloy 600 and is used in most replacement steam generators.

Q65. Can you describe how Indian Point might be able to redefine "the pressure boundary in which the TTSW is no longer included and, therefore, is not required for reactor coolant pressure boundary function?"

A65. [ALH, KJK] As designed, the TTSWs provide a reactor coolant pressure boundary function because they prevent primary coolant from leaking between the steam generator tube and the tubesheet into the secondary water and they hold the tubes in place. During fabrication of the steam generator, the steam generator tubes are inserted into the tubesheet, and then the tube is expanded into the tubesheet to hold the tube in place while the TTSW is formed for each tube. Because this expansion process creates a tight interference fit, it restricts the amount of leakage that can occur and limits the possibility that the tube will come out of the tubesheet.

These features enable the function of reactor coolant pressure boundary to be "redefined" from the TTSW to a portion of the steam generator tube that has been expanded into the tubesheet. In other words, the TTSWs may not be necessary to maintain the reactor coolant pressure boundary between the primary and secondary systems. The NRC has approved similar redefinitions of the reactor coolant pressure boundary in the past.

Q66. Could you briefly describe the process for the Applicant to seek to transfer the pressure boundary function for IP2 and IP3 from the TTSWs to the interference fit between the tubes and the tubesheet?

Q66. [ALH, KJK] Redefinition of the reactor coolant pressure boundary to eliminate the TTSWs as a part of the reactor coolant pressure boundary can be approved by the NRC using a license amendment request in accordance with 10 CFR 50.90. As part of the process of seeking a license amendment, the Atomic Energy Act affords an opportunity for the public to request a hearing on the license amendment, if the Applicant chooses to proceed with an amendment instead of the inspections.

Q67. Has the NRC granted this type of license amendment in the past?

A67. [ALH, KJK] The NRC has approved this type of license amendment in the past for many units including, but not limited to, Catawba Nuclear Station, Unit 2; Surry Power Station, Unit Nos. 1 and 2; Joseph M. Farley Nuclear Plant, Unit 2; and St. Lucie Plant, Unit No. 2. *See* (Catawba Nuclear Station, Units 1 and 2, Issuance of Amendments Regarding Technical Specification Amendments for Permanent Alternate Repair Criteria for Steam Generator Tubes (TAC Nos. ME6670 and ME6671), dated March 12, 2012 (ML12054A692); Surry Power Station, Unit Nos. 1 and 2, Issuance of Amendments Regarding Virginia Electric and Power Company License Amendment Request for Permanent Alternate Repair Criteria for Steam Generator Tube Inspection and Repair (TAC Nos. ME6803 and ME6804), dated April 17, 2012 (ML120109A270); Issuance of Amendment No. 64 to Facility Operating License No. NPF-8 –Joseph M. Farley Nuclear Plant, Unit 2, Regarding Tests of Steam Generator Tubes in the Tubesheet Region (TAC No. 65287), September 21, 1987 (8709250240); St. Lucie Plant, Unit No. 2 - Issuance of Amendment Regarding Depth of Required Tube Inspections and Plugging Criteria Within the Tubesheet Region of the Original Steam Generators

(TAC No. MC5084), dated April 11, 2006 (ML060790352). As noted by New York, the Staff approved this license amendment for IP2.

Q68. If Indian Point does not complete the actions provided for under Option 1, what kind of inspection will be conducted under Option 2?

A68. [ALH, KJK] If Option 2 of Commitment 42 is implemented, Indian Point will perform a one-time inspection of the TTSWs in each steam generator to determine if PWSCC is present.

Q69. How are the TTSW inspections under Option 2 different from the steam generator tube inspections that Indian Point has been doing for many years?

A69. [ALH, KJK] Current steam generator tube inspections examine the entire length of the tube using an eddy current probe that is inserted into the tube and travels the full length of the tube from the TTSW at the tube inlet to the TTSW at the tube outlet, but inspections of the weld are not required. The TTSW inspections may involve various techniques such as eddy current or visual inspections. The technique used will have to be capable of detecting PWSCC in the TTSWs.

Q70. When will TTSW inspections for IP2 and IP3 have to be completed?

A70. [ALH, KJK] In accordance with Commitment 42, the Applicant would need to complete the TTSW inspections at IP2 between March 2020 and March 2024. Similarly, IP3 would complete its inspections before the end of the first refueling outage following the beginning of the period of extended operation. The first refueling outage after the beginning of the period of extended operation is expected to begin on or before December 13, 2017.

Q71. Why is it acceptable for the Applicant to wait until as late as 2024 to inspect IP2 and only until the first refueling outage after the period of extended operation for IP3?

A71. [ALH, KJK] The timing of the inspections is related to the previous discussion explaining the timing of the inspections for the SGDP. The inspections are scheduled after sufficient operating time for each steam generator to allow for PWSCC to develop to detectable levels, if it develops at all. The timing of these inspections is acceptable because it provides for an increased likelihood that the degradation will be identifiable, if it were to occur in these steam generators, and it allows for detection of any degradation in the TTSW prior to it compromising the function served by the TTSW.

Q72. Which steam generator will be inspected under Option 2 of Commitment 42?

A72. [ALH, KJK] A representative sampling of the TTSWs in each steam generator at each unit will be inspected.

Q73. What is the frequency of the TTSW inspections?

A73. [ALH, KJK] The inspections are one-time inspections and are scheduled after sufficient operating time for each steam generator to allow for PWSCC to develop to detectable levels, if it develops at all. Based on operating experience to date, the inspections are also early enough in the steam generators operating lifetime to detect any cracking prior to the cracks jeopardizing the safety functions of the components.

Q74. What will happen if these one-time TTSW inspections identify cracking?

A74. [ALH, KJK] Because these examinations are to verify effectiveness of the Water Chemistry Control – Primary and Secondary Program, Commitment 42 states that (1) the condition identified by the inspections will be resolved through repair or engineering evaluation to justify continued service, as appropriate, and (2) an ongoing monitoring program will be established to perform routine TTSW inspections for the remaining life of the steam generators. Therefore, the Applicant will, within its CAP, assess the conditions that it finds and disposition them to ensure safe

plant operation. In addition, it will continue to inspect these locations to ensure that any new cracks that may develop are promptly identified and dispositioned.

Q75. Are the actions identified in response to Q74 the only additional actions that will be taken should cracking be identified in the TTSW at Indian Point?

A75. [ALH, KJK] No. Should cracking be identified during any of the TTSW examinations at Indian Point, the Applicant will assess the finding within its CAP. The evaluation initiated under the CAP would (1) determine what actions are necessary to address the specific cracking identified, (2) what additional actions may be necessary for the other steam generators at the same unit, (3) what additional actions may be necessary for the other unit, and (4) what effect the finding has on the need to implement follow-up or periodic inspections. Although items (1) and (4) are specified in Commitment 42, item (3) could result in additional or accelerated inspections at the other unit, and item (4) could result in additional inspections beyond the initial sample inspected.

Q76. In an overall sense, why does the NRC believe that the Applicant's planned aging management is adequate for the TTSWs?

A76. [ALH, KJK] We believe that the Applicant's planned aging management is adequate for the TTSWs for several reasons. First, the primary aging management of the TTSWs is provided by the Applicant's Water Chemistry Control – Primary and Secondary Program, which has proven to be effective in managing cracking. Second, the planned examinations will verify the effectiveness of the Water Chemistry Control – Primary and Secondary Program and will provide additional confirmation of the condition of the TTSWs. Third, the timing, scope, acceptance criteria and corrective actions for the examinations will be sufficient to ensure the continued safe condition of the steam generators. Lastly, the interference fit between the tube and the tubesheet as a result of the tube expansion process during

fabrication provides some assurance that the reactor coolant pressure boundary integrity will be maintained, limiting the importance of any cracking if it were to occur.

Q77. Do either Commitment 41 or Commitment 42 specify the inspection technique that will be used to detect PWSCC?

A77. [ALH, KJK] No, neither commitment requires the selection of a specific technique. It is quite possible that IP2 and IP3 may utilize different techniques due to the time difference between the inspections.

Q78. If these commitments do not specify an inspection technique, how does the NRC know that the inspection technique used to detect PWSCC in the SGDP and the TTSW will be adequate?

A78. [ALH, KJK] Several inspection techniques are routinely used to detect PWSCC in nickel alloys in nuclear power plants, including visual, surface, and volumetric examination methods. Some examples for the examination methods that are routinely used are described below:

- Visual examination: visual VT-1 and enhanced VT-1
- Surface examination: eddy current testing (ECT) and penetrant testing (PT)
- Volumetric examination: ECT or ultrasonic testing (UT)

For both TTSW and SGDP examinations, the Applicant will need to develop an adequate justification for the technique selected. The technique, necessarily, will need to be capable of detecting PWSCC under the conditions within the steam generators, such as deposits on the surface to be inspected, among others. These inspections must meet the requirements of 10 CFR Part 50, Appendix B, as discussed in response to Q40, and are subject to review under the NRC reactor oversight process by experienced inservice inspection inspectors from NRC's Region I.

Q79. Why would the NRC accept commitments from the Applicant that do not specify the inspection technique to be used?

Q79. [ALH, KJK] The NRC accepted the Applicant's commitments for two main reasons. First, the Staff is aware that there are inspection techniques that are capable of detecting PWSCC available now; so the ability to detect PWSCC is not an issue. Second, it is unnecessary to have the Applicant identify a specific technique when there are likely to be improved techniques and innovations that will occur prior to the need for inspections. It would be counterproductive to require a specific technique now that may be overcome by better techniques in the future. The specific circumstances for each steam generator may provide additional reasons to select one technique over another that is not related to its ability to detect cracking, as appropriate. For example, Dr. Duquette has expressed concern regarding the dose that might result from these inspections. (Ex. NYS000375 at 9, 27.) It would be reasonable for the Applicant to try to select a technique that minimizes the dose when the inspections are performed. The general approach of not specifying the inspection technique is consistent with the requirements for inspecting the steam generator tubing at all pressurized water reactors.

Q80. Is there anything profoundly challenging regarding the inspection techniques that would be needed for the TTSW and SGDP inspections that the Applicant is committed to implement?

A80. [ALH, KJK] No, the inspection techniques that are currently anticipated to be utilized have long track records of identifying cracking in nuclear power plants. Although no specific inspections of the SGDP or the TTSWs have been implemented in the U.S., inspections overseas have successfully identified PWSCC in the SGDP. To prepare for these inspections, the Applicant would need to (1) demonstrate the crack detection capability of the specific technique they will implement, and (2)

develop “delivery hardware” that will enable the inspection hardware to access the critical surfaces for examination such that the inspections can be completed in an efficient and effective manner with a minimum of dose to the inspection personnel.

Q81. Has Entergy taken any actions with respect to selecting how it intends to complete Commitment 42 for either IP2 or IP3?

A81. [ALH, KJK] Yes. Indian Point 2 exercised the analysis option in Commitment 42 to redefine the pressure boundary, thereby completing Commitment 42 and eliminating the need for performance of the one-time inspection of a representative number of TTSWs in each steam generator.

Q82. Could you explain the changes related to Commitment 42 for IP2?

A82. [ALH, KJK] Entergy has redefined the reactor coolant pressure boundary in a license amendment that was approved by the NRC staff. Exs. NYS000539 and NYS000542. As a result, the TTSW no longer serves as the pressure boundary and a one-time inspection of a representative sample of TTSW is no longer required to fulfill this commitment (NYS000553).

Q83. Why is the tube-to-tubesheet weld no longer needed to be part of the pressure boundary?

A83. [ALH, KJK] The tube-to-tubesheet joint consists of both the TTSW and an interference fit between the tube and the tubesheet for the full depth of the tubesheet. When the steam generator was designed, only the TTSW was credited for ensuring the tube would be held in place. There was no credit for the interference fit to hold the tube in the tubesheet when the steam generators were originally designed. With the license amendment, the licensee determined that the interference fit is sufficient to hold the tube into the tubesheet with no credit being taken for the TTSW holding the tube in place.

Q84. How does this interference fit work?

A84. [ALH, KJK] The tube is expanded into one of the holes in the tubesheet creating close contact between the tube and the tubesheet. This expansion of the tube along with other factors hold the tube in place. It is similar to a clamp on a piece of pipe.

Q85. Can you give an example of these "other" factors?

A85. [ALH, KJK] The tube and the tubesheet are made from different materials. As the steam generator heats up, the materials expand differently. The tube expands more than the "tubesheet hole". This further holds the tube in place since the tube is being expanded further into the tubesheet. For the clamp analogy, the clamp on the pipe is becoming tighter.

Q86. Wouldn't the weld also prevent leakage? Does the interference fit prevent leakage?

A86. [ALH, KJK] The weld prevents leakage. The interference fit does not necessarily prevent all leakage; however, for approval of the license amendment, the licensee demonstrated that the interference fit is sufficient such that leakage from normal operations and accident conditions can be managed and kept within acceptable levels. These acceptable levels are consistent with the design and licensing basis for the facility.

Q87. Since the reactor coolant pressure boundary has been redefined in this amendment, is the TTSW required for ensuring structural and leakage integrity of the steam generator tube?

A87. [ALH, KJK] No. The TTSW is no longer needed to ensure the structural or leakage integrity of the tube. Thus, the TTSW no longer performs a safety or safety-related function.

Q88. Was the public offered the opportunity for a hearing on the amendment that redefined the reactor coolant pressure boundary for IP2?

A88. [ALH, KJK] Yes. The NRC published a notice and an opportunity for hearing in Federal Register. Biweekly Notice; Applications and Amendments to Facility Operating Licenses and Combined Licenses Involving No Significant Hazards Considerations, 79 Fed. Reg. 15144 (Mar. 18, 2014) Ex. NRC000191.

Q89. To your knowledge, did anyone request a hearing?

A89. [ALH, KJK] No.

Q90. Do you agree with Dr. Duquette's assessment that approving redefinition of the pressure boundary was premature?

A90. [ALH, KJK] No.

Q91. Why do you not agree with his testimony on the redefinition of the pressure boundary?

A91. [ALH, KJK] Dr. Duquette implies that recently committed research funds for the purpose of evaluating PWSCC in nickel-based alloys demonstrates, in part, that we were premature in our approval of the redefinition of the reactor coolant pressure boundary at Indian Point 2. The research that Dr. Duquette references is for determining crack growth rates in Alloy 690 material. The steam generator tubes at Indian Point 2 are made from thermally treated Alloy 600. Even if the research was on Alloy 600 material, the research referenced would not have any bearing on our review of the amendment.

Q92. Do you agree with the Intervenors' experts' concerns that the Applicant's aging management plan is too vague or conceptual to be a meaningful and enforceable standard?

A92. [ALH, KJK] No. The Intervenors' experts inappropriately focus on one small aspect of the Aging Management Programs used for the SGDP and the TTSWs, namely the one-time verification inspections in Commitments 41 and 42. Specifically, aging management of the SGDP and the TTSW are accomplished through implementation of the Water Chemistry Control – Primary and Secondary Program, as verified by the one-time inspections in Commitments 41 and 42. The concerns of the Intervenors' experts incorrectly narrows the scope of the aging management for the SGDP and the TTSWs to exclude additional actions the Applicant is required to take.

Even assuming that the inappropriate narrowing of the program by Intervenors' expert was true, the one-time inspections are well defined as to the timing, extent, and acceptability. Further, the Applicant is required to select a testing technique that is capable of detecting PWSCC in the SGDP and TTSWs, consistent with the requirements of 10 CFR Part 50, Appendix B. The selection of the testing technique is appropriately left to the Applicant's discretion and technical justification at the time of the inspection due to the multiple variables that may weigh in favor of one type of testing technique over another. A specification of the inspection technique that seems the best fit based on information available now may be considered outdated at the time of the inspections that the Applicant has committed to implement. Several techniques have been demonstrated to be successful in detecting PWSCC in the materials used to fabricate the SGDP and the TTSWs. As such, we do not consider the one-time verification inspections vague or undefined. The other aspects of the aging management of the SGDP and the TTSWs and the inspections are well defined and easily identified.

Q93. Do you agree with the Intervenors' assessments that there is insufficient detail to determine the adequacy of the aging management of the SGDP and the TTSWs?

A93. [ALH, KJK] No, we do not agree with the Intervenors' assessments. We believe that there is sufficient information provided by the Applicant to determine the sufficiency of its proposed aging management for the SGDP and the TTSWs. First, the Intervenors seem confused as to the applicable requirements for adequate aging management. GALL is not the only way to satisfy the regulatory requirements. It represents an acceptable way. The regulations set forth the requirements that must be satisfied. As applicable here, the regulations state that the Applicant must demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation. 10 C.F.R. § 54.29(a)(1). The term "aging management program" that seems central to the Intervenors' assertions is only provided for in NRC guidance documents, such as the GALL report and the Standard Review Plan for License Renewal. See Exs. NRC000008 and NRC000010. These documents do not provide requirements that applicants must meet, but rather provide NRC Staff guidance for one way to demonstrate compliance with the regulations. For example, an applicant's proposed aging management program that is consistent with the GALL Report and meets the criteria of the SRP-LR is presumed to meet the regulations. Conversely, the failure for an applicant's program to meet the guidance in the GALL or the Standard Review Plan for License Renewal does not mean that the applicant cannot or did not demonstrate compliance with the applicable regulations.

Q94. Is the Applicant's proposed aging management of the SGDP and the TTSWs consistent with the GALL Report?

A94. [ALH, KJK] Yes, the Applicant's proposed aging management of the SGDP and the TTSWs is based on its Water Chemistry Control – Primary and Secondary Program, which is defined in the LRA as a ten-element aging management program consistent with the GALL Report and has been found acceptable as evaluated in the Staff's SER. (Ex. NRC000005 at 3-147 to 3-149.) From that perspective both the SGDP and the TTSWs will be managed consistent with the GALL Report.

The one-time verification inspections of the SGDP and the TTSW have not been proposed in the format of ten elements, and were not evaluated in the SSER for their consistency with GALL aging management programs.

Q95. Does the NRC Staff have an opinion on whether the actions proposed by the Applicant for the SGDP, including Commitment 41, are consistent with GALL?

A95. [ALH, KJK] Yes. The Staff believes the actions proposed by the Applicant are consistent with the aging management provisions of GALL. Although the one-time verification inspections have not been proposed by the Applicant in a ten-element format, the Staff has evaluated, for the purpose of this testimony, the Applicant's proposed actions consistent with the GALL ten-element evaluation for aging management programs. As discussed below, each element is satisfied by the Applicant's proposed actions.

Scope of Program

The scope of the one-time verification inspection of the SGDP is to identify any indications of PWSCC in the SGDP, with a focus on those areas that have been identified as cracked from prior inspections world-wide. This element is acceptable because it identifies the areas to be inspected and the aging effect to be managed.

Preventive Actions

Actions to prevent PWSCC in the SGDP are provided by the Water Chemistry Control – Primary and Secondary Program. This element is acceptable because it identifies the appropriate preventive actions.

Parameters Monitored or Inspected

This one-time verification inspection will detect the presence of PWSCC in the SGDP. The specific technique will have been demonstrated to be capable of detecting PWSCC. This element is acceptable because it provides the performance criteria for the inspection technique.

Detection of Aging Effects

The SGDP of each steam generator will be inspected on a schedule consistent with Commitment 41. This is a one-time verification inspection using a technique which has been demonstrated to be capable of detecting PWSCC. This element is acceptable because it specifies the schedule and frequency (one time) of the inspection.

Monitoring and Trending

This is a one-time inspection program and thus there is no monitoring and trending. This element is acceptable because there is no monitoring and trending for a one-time inspection program.

Acceptance Criteria

The acceptance criterion for these one-time verification inspections is the absence of indications of PWSCC. Identification of PWSCC will require corrective actions. This element is acceptable because it specifies that any indications of PWSCC will trigger corrective actions.

Corrective Actions

Corrective actions will be in accordance with the CAP. Corrective actions will consider disposition of the observed condition, the need to expand the inspection scope for the unit, the need to assess the impact of the findings on the other unit, and plans to initiate a periodic inspection program. This element is acceptable because it relies on the applicant's CAP and it addresses the necessary corrective actions.

Confirmation Process

The confirmation process is consistent with that for the other aging management programs. This element is acceptable because it is consistent with the applicant's other programs.

Administrative Controls

The administrative controls are consistent with those for the other aging management programs. This element is acceptable because it is consistent with the applicant's other programs.

Operating Experience

Prior to implementation of the one-time verification inspections of the SGDP, industry operating experience and research results will be considered to focus and optimize the inspections, in terms of the areas to be inspected and the inspection technique. This element is acceptable because the program will utilize the latest industry operating experience to provide the most informative inspections.

Q96. Does the NRC Staff have an opinion on whether the actions proposed by the Applicant for the TTSWs, including Commitment 42, is consistent with GALL?

A96. [ALH, KJK] Yes, the Staff believes actions proposed by the Applicant are consistent with the aging management provisions of GALL.

If the Applicant is able to utilize Option 1 of Commitment 42 to, in effect, make a determination that the TTSW does not require aging management, then the Applicant would not need to perform any inspection of the TTSW as previously discussed.

If the Applicant utilizes Option 2 of Commitment 42, then an evaluation of the proposed actions can be shown to satisfy each of the GALL ten elements for aging management programs, as discussed below.

Scope of Program

The scope of the one-time verification inspection of the TTSW is to identify any indications of PWSCC in the TTSWs. This element is acceptable because it identifies the areas to be inspected and the aging effect to be managed.

Preventive Actions

Actions to prevent PWSCC in the TTSWs are provided by the Water Chemistry Control – Primary and Secondary Program. This element is acceptable because it identifies the appropriate preventive actions.

Parameters Monitored or Inspected

This one-time verification inspection will detect the presence of PWSCC in the TTSWs. The specific technique will have been demonstrated to be capable of detecting PWSCC. This element is acceptable because it provides the performance criteria for the inspection technique.

Detection of Aging Effects

A representative sample of the TTSWs in each steam generator will be inspected on a schedule consistent with Commitment 42. This is a one-time verification inspection using a technique which has been demonstrated to be capable of detecting PWSCC. This element is acceptable because it specifies the schedule and frequency (one time) of the inspection.

Monitoring and Trending

This is a one-time inspection program and thus there is no monitoring and trending. This element is acceptable because there is no monitoring and trending for a one-time inspection program.

Acceptance Criteria

The acceptance criterion for these one-time verification inspections is the absence of indications of PWSCC. Identification of PWSCC will require corrective actions. This element is acceptable because it specifies that any indications of PWSCC will trigger corrective actions.

Corrective Actions

Corrective actions will be in accordance with the CAP. Corrective actions will consider disposition of the observed condition, the need to expand the inspection scope for the unit, the need to assess the impact of the findings on the other unit, and plans to initiate a periodic inspection program. This element is acceptable because it relies on the applicant's CAP and it addresses the necessary corrective actions.

Confirmation Process

The confirmation process is consistent with that for the other aging management programs. This element is acceptable because it is consistent with the applicant's other programs.

Administrative Controls

The administrative controls are consistent with those for the other aging management programs. This element is acceptable because it is consistent with the applicant's other programs.

Operating Experience

Prior to implementation of the one-time verification inspections of the TTSWs, industry operating experience and research results will be considered to focus and optimize the inspections, in terms of the areas to be inspected and the inspection technique. This element is acceptable because the program will utilize the latest industry operating experience to provide the most informative inspections.

Q97. Do you agree with Intervenors' expert that PWSCC in the SGDP might compromise the pressure boundary in the steam generator?

A97. [ALH, KJK] We do not agree that cracks in the SGDP will compromise the reactor coolant pressure boundary. Cracks in the SGDP do not compromise the reactor coolant pressure boundary integrity at all. These cracks would need to grow significantly and grow out of the SGDP into reactor coolant pressure boundary components in order to compromise the pressure boundary. Such cracking has not been observed in any of the reported operating experience.

Q98. How does the aging management proposed by the Applicant for the SGDP and the TTSWs prevent the reactor coolant pressure boundary from being compromised?

A98. [ALH, KJK] First, the principal aging management of the SGDP and the TTSWs, through implementation of the Water Chemistry Control – Primary and Secondary Program, is a preventive measure that minimizes the likelihood of PWSCC in the SGDP and the TTSWs. The additional measures provided in Commitments 41 and 42, including the one-time verification inspections, provide confirmation that the Water Chemistry Control – Primary and Secondary Program is effective. The

inspections will assess the condition of the SGDP and the TTSWs such that the Applicant will be able to determine and implement any appropriate corrective actions.

The Intervenors' concern regarding the potential for the cracks to compromise the reactor coolant pressure boundary is not supported by the current U.S. or foreign operating experience. Specifically, the types of cracks necessary to compromise the reactor coolant pressure boundary far exceed the cracks that have been identified in the foreign units. One purpose of the one-time verification inspections is to ensure that any degradation in the SGDPs and TTSWs will be identified and remediated prior to the development of any safety issue, such as compromises to the reactor coolant pressure boundary.

Q99. Do you agree with Intervenors' expert that the potentially high doses to personnel as experienced in the examination of French steam generators makes the inspection unacceptable or unqualified?

A99. [ALH, KJK] Dr. Duquette cites a number of documents regarding the high doses that are incurred by the current inspection techniques in France. None of these references indicate any concern that the inspection techniques are unable to detect PWSCC. The references do not indicate that doses were unacceptably high. Even the Intervenors do not seem to say that the doses are so high that the inspections could not be completed. Based on the Intervenors' documents, the only supportable conclusion is that inspections were completed successfully.

When Indian Point implements its inspections for the SGDP and TTSWs, they will necessarily need to take into consideration many things, including ensuring the radiation dose is as low as reasonably achievable. However, this contention does not address worker dose and the Applicant does have other regulatory considerations that it must meet in this area.

Q100. Do you agree with Dr. Duquette's assertion that PWSCC initiated in the SGDP will preferentially turn toward the triple point of the tubesheet-channel head complex?

A100. [ALH, KJK] The report cited by Dr. Duquette represented understanding of the French operating experience in 2007. Our current understanding is that more recent reviews of experience by the French have indicated that the cracking generally lies parallel to the stub runner and not in an orientation that would indicate growth towards the tubesheet, as described in the Intervenor's exhibit. (Ex. NYS000390 at 17) As stated previously, our understanding is that the French have not taken any actions to repair SGDPs with cracks.

Q101. Do you agree with the Intervenors' expert's assertion that there is no barrier to crack propagation from the tubesheet cladding material into the TTSW?

A101. [ALH, KJK] No, we do not agree that such a crack would have no barrier to propagating into the TTSW. Propagation of possible SGDP cracks to the TTSW would involve multiple barriers. First the potential SGDP crack would need to propagate from the interface of the tubesheet cladding with the SGDP to the TTSW, and then the crack would need to propagate into the TTSW and grow to a size that would breach the reactor coolant pressure boundary. Each of these steps would take time, depending on the local material, geometry, and specific environmental and loading parameters within the tubesheet cladding and the TTSW. These same factors would also affect the crack propagation trajectory, such that the cracks may not even propagate to the TTSW but may find another preferred path or may stop growing.

Q102. Can you briefly explain what is meant by a "barrier to crack propagation"?

A102. [ALH, KJK] We think that the Intervenors' expert means that the cracking would propagate in the tubesheet cladding material to the TTSW without any impediment, such that one would assume that cracking in the TTSW would start as soon as the cracking progresses to the

tubesheet cladding material. For the reasons stated in the prior answer, there clearly are barriers to crack propagation from the SGDP to the TTSW.

Q103. Do you agree with the concerns of Dr. Lahey and Dr. Duquette that a through-wall divider plate crack would compromise the intended heat transfer function of the steam generator?

A103. [ALH, KJK] Based on the analyses done by the SGTF, we do not believe that this is a significant issue that requires immediate attention. When we previously investigated this issue, the SGTF's analyses indicated that even a fully degraded SGDP does not adversely affect steam generator performance, is not a safety concern during plant operations, and does not necessitate any changes in current steam generator analyses. ("NEI Steam Generator Task Force NRC/Industry Update" (ADAMS Accession No. ML102300220)) (Ex. NRC000158) dated August 12, 2010, at slides 45 and 46. As we noted previously, the SGDPs at many plants incorporate a designed through-wall hole between the hot and cold leg portions of the steam generator, at the interface of the SGDP with the channel head at the bottom of the steam generator, which allows communication of the hot and cold leg flows and/or is used to drain both the inlet and outlet sides of the lower channel head during plant shutdown conditions.

Q104. Has the NRC concluded that the Water Chemistry Control – Primary and Secondary Program is not sufficient to prevent PWSCC in the SGDP?

A104. [ALH, KJK] No. It appears that the Intervenors may have misinterpreted from this statement in the SER, "[t]he Staff determined that the Water Chemistry – Primary and Secondary Program *might not be effective* in managing cracking due to PWSCC in SG divider plate assemblies." (Ex NRC000006 at 3-18)." This is an important distinction because the Staff believes that water chemistry control is effective for limiting or eliminating the likelihood of PWSCC in the SGDP, and thus one-time inspections as proposed in Commitments 41 and 42 are reasonable to verify effectiveness of

the Water Chemistry Control – Primary and Secondary Program. If the Staff were to make a determination that water chemistry control was not sufficient, then applicants would need to take additional measures, possibly including periodic inspections, to be able to demonstrate effective aging management.

Q105. Do you agree with Dr. Duquette's conclusions that cracks from the SGDP are likely to propagate to the TTSWs?

A105. [ALH, KJK] No, we do not agree that SGDP cracks "will likely" propagate into the TTSWs. Although we understand that SGDP cracks could potentially propagate from the SGDP into the tubesheet cladding, and then cracks in the tubesheet cladding could potentially propagate into the TTSWs, we see no basis for concluding that cracks in the SGDP are going to preferentially propagate from the SGDP into the TTSWs. The exact path that such hypothetical cracks would take is dependent on a variety of characteristics, including the residual stresses from the cladding and the fabrication of the TTSW, geometry and environmental factors, and any material cracking preferences that may promote cracking towards or possibly away from the TTSW. Even if these hypothetical cracks do propagate to the TTSW or the channel head, thereby having the potential to affect the reactor coolant pressure boundary, propagation from the SGDP to the TTSW or the channel head would take additional time. The foreign operating experience has not identified instances of SGDP cracks propagating into the TTSWs or the channel head. Therefore, we concluded that the one-time inspections proposed by the Applicant will be performed in a time frame that will preclude any possible SGDP cracks from adversely affecting the functions served by the TTSWs or the channel head.

Q106. Do you agree that the exact stress states of the SGDP or TTSWs are not currently well understood by the industry?

A106. [ALH, KJK] When Dr. Duquette originally asserted that exact stress states of the SGDP and TTSW was not well understood, we agreed because the industry had not completed its testing, analysis, and modeling for these components. Ex. NRC000160 (NEI Steam Generator Task Force NRC/Industry Update dated August 4, 2011) (ML11216A050). Since that time, the industry through EPRI has completed its research and provided its results to the NRC. Ex. NYS000544A-D. Finite element modeling was used to determine the stresses in the channel head and in the tubesheet regions. The analyses in the report assess whether cracks in the SGDP can propagate into the TTSWs and the channel head and the severity of any cracks that may propagate in these reactor coolant pressure boundary components and materials. The staff is currently reviewing the information developed by EPRI to address these issues.

Q107. Is it important to know the precise stress states for the SGDP, tubesheet cladding, and TTSW in order to have an adequate AMP?

A107. [ALH, KJK] No, it is not important to know the precise stress states for the SGDP, tubesheet cladding and TTSWs to adequately manage aging for these locations. There are other factors that will serve to impact the likelihood of initiating and propagating PWSCC in the SGDP, including environmental, material, and loading characteristics. The one-time inspections of the SGDP and the TTSWs that the Applicant has addressed in Commitments 41 and 42 will identify the state of condition of the SGDP and the TTSWs and will confirm adequate aging management of the SGDP and the TTSWs.

Q108. Do you agree with Dr. Duquette's analysis of foreign objects in the SG?

A108. [ALH, KJK] No.

Q109. Please explain why you disagree with Dr. Duquette's foreign object analysis?

A109. [ALH, KJK] Dr. Duquette's analysis appears to confuse primary and secondary side foreign objects (also referred to as loose parts). Foreign objects/loose parts have been observed both on the primary side of steam generators (in the channel head and inside the tubes) and on the secondary side of the steam generator (outside the tubes). Foreign objects/loose parts on the primary side of the steam generator are much less frequent than loose parts on the secondary side of the steam generator. Finding foreign objects/loose parts in the secondary side of a steam generator is fairly common. As a result, licensees routinely inspect for loose parts and the damage that these loose parts may have caused. In addition, plugging/repairing of tubes as a result of loose parts on the secondary side is fairly common. Entergy did plug 9 tubes as a result of loose parts in their last Indian Point 2 steam generator inspection outage (Ref NYS000543); however, these loose parts were not inside the tubes as Dr. Duquette indicates but rather they were located "outside" the tubes on the secondary side of the steam generator. Letter NL-15-001, Letter from Lawrence Coyle, to NRC, Response to Request for Additional Information (RAI) for the Steam Generator Examination Program Results 2014 Refueling Outage (2R21) Indian Point Unit No. 2 Docket No. 50-247 License No. DPR-26 (Jan. 15, 2015) Ex. NRC000192. These particular loose parts did not cause any tube damage. Loose parts on the secondary side of the steam generator have resulted in denting of the tubes. The NRC staff is not aware of any denting of the pressure boundary portion of the tube caused by loose parts on the primary side of the steam generator; however, such loose parts have caused damage to tube ends as Dr. Duquette cites for Indian Point 3.

Q110. Do you agree with Dr. Duquette's analysis of tube wear in the IP2 SGs?

A110. [ALH, KJK] No.

Q111. Please explain why you disagree with Dr. Duquette's analysis of tube wear?

A111. [ALH, KJK] Dr. Duquette compares the wear observed at San Onofre with the wear observed at IP2. The wear which led to the tube leak at San Onofre was caused by a phenomenon referred to as fluidelastic instability. This phenomenon results in large amplitude tube motions such that the tubes may contact each other and the tube supports. When steam generators are designed, the designers try to prevent this phenomenon from occurring. On the other hand, the wear at Indian Point 2 is considered typical wear. It is a result of normal tube vibration. This type of wear is fairly common in steam generators and has been effectively managed for decades at pressurized water reactors. The amount and severity of wear observed at Indian Point 2 is comparable to that at other units. Comparing the wear observed at Indian Point 2 to the wear attributed to fluidelastic instability is inaccurate and inappropriate.

Q112. Do you agree with Dr. Duquette's observations on the location and size of cracking identified overseas in the SGDP?

A112. [ALH, KJK] Dr. Duquette's observations on the location of PWSCC is consistent with the most recent information available from an industry presentation, which states that all indications are in or close to the heat affected zones of the tubesheet to stub runner weld or the stub runner to divider plate weld. (Ex. NYS000390 at 14) This report also indicates that successive inspections at some plants identified no change in the number of cracks and few variations in crack depth. (Ex. NYS000390 at 14) Further, cracks that were previously sized as 7 to 8 mm in depth (out of a 34-mm divider plate thickness) were sized in laboratory testing as <2 mm deep using techniques qualified for depth sizing. (Ex. NYS000390 at 14) This presentation also stated that destructive examinations identified that the heat affected zone of the welds was found to have carbide dissolution in about a 2-mm thick layer. (Ex. NYS000390 at 15) The correlation of this dissolution layer thickness with the maximum depth of the service cracks examined in a laboratory may indicate that the cracking, if it is found at all, would be limited to a depth of no more than 2 mm.

Q113. Based on the French analysis to date, what are the contributing factors that led to the French steam generator divider plate cracking?

A113. [ALH, KJK] In addition to possible differences in loading and pre-assembly processing which Dr. Duquette cited, another contributing factor to the PWSCC observed in the French reactors was a stub runner of low mechanical properties (low yield strength) and a divider plate with high mechanical properties (high yield strength). These contributing factors including the mix of low- and high-yield strength materials have resulted in relatively superficial flaws of low safety significance.

Q114. What is the NRC Staff's current assessment of the French operating experience with the SGDP cracking?

A114. [ALH, KJK] Even if we assumed that the conditions for this cracking exists in one or more of the IP2/3 steam generators, the degradation does not appear to be growing through the thickness of the divider plate (i.e., the degradation remains shallow). Miloudi, S., Firmin, E., Deforge, D., Vaillant, F. and Lemaire, E. (2012) *Destructive Examinations on Divider Plates from Decommissioned Steam Generators Affected by Superficial Stress Corrosion Cracks, in 15th International Conference on Environmental Degradation of Materials in Nuclear Power Systems-Water Reactors (eds J. T. Busby, G. Ilevbare and P. L. Andresen), John Wiley & Sons, Inc., Hoboken, New Jersey, Canada. doi: 10.1002/9781118456835.ch225 Ex. NRC000194. As a result, the safety significance is very limited. The French have not repaired any of the divider plates. To summarize, our assessment is that there is a low likelihood that cracking would occur in the divider plate, and if it were to occur, its safety significance would be limited.

Q115. Are you familiar with Dr. Duquette's recent testimony identifying issues related to the channel head degradation and inspections?

A115. [ALH, KJK] Yes.

Q116. Is there any relationship between the channel head degradation and SG divider plate cracking and Commitments 41 and 42?

A116. [ALH, KJK] No. The commitment to inspect the divider plate was driven by French operating experience where cracks were observed in the stub runner near the divider plate to stub runner weld. This weld runs parallel to the tubesheet and is near the bottom of the tubesheet. The channel head degradation that was cited in NYS000549 (Nuclear Safety Advisory Letter) and NYS000538 (NRC Information Notice) was in different locations and is not related to the foreign operating experience with SGDP.

Q117. What is a Nuclear Safety Advisory Letter (NSAL)?

A117. [ALH, KJK] An NSAL is a document that Westinghouse issues to notify users of its components of recently identified potential safety issues.

Q118. What is the recipient expected to do with this information?

A118. [ALH, KJK] The information is provided so that the recipient can review the issue and determine if any action is required for its plant.

Q119. How does this differ from the purpose of an NRC Information Notice?

A119. [ALH, KJK] The two documents have similar purposes in that they provide information, normally on operating experience at a plant, for the recipient to assess and determine if any corrective

actions are needed for its plant. Although an NSAL may also recommend that recipients take specific actions, an NRC Information Notice does not recommend specific actions to be taken

Q120. Have licensees typically taken action to address issues highlighted in NRC Information Notices despite the Information Notice not having specific requirements or specific actions that must be followed?

A120. [ALH, KJK] In my experience from the steam generator area, licensees review the circumstances cited in NRC Information Notices and take actions as necessary to address those issues.

Q121. Besides an Information Notice, does the NRC have other options to request that licensees take actions, if it determines such actions are necessary, and can you describe these other options?

A121. [ALH, KJK] We have several options, depending on the significance of the potential issue and the timeliness of any needed actions. In the case that the issue represents an immediate and significant safety issue, we can issue Orders that would require plants to take specific actions, with the possibility of immediate shut down to facilitate the actions. If the issue has the potential for a high safety significance but the immediacy for action isn't clear, then we could issue an NRC Bulletin. A Bulletin requests that licensees provide plant-specific information. When submitting this information, licensees may commit to actions or indicate they took action to address the issue. The staff would then review the responses and determine if the response sufficiently addresses the issue. If the response is not sufficient, then the staff would engage the licensee to ensure their actions will be effective and timely, with the possibility that the staff could issue Orders to require the plant to take a different or a more immediate action.

Q122. Does it appear that Entergy has evaluated the information contained in the NSAL and in the NRC Information Notice?

A122. [ALH, KJK] Yes.

Q123. Has Entergy performed the channel head inspections discussed in the NSAL?

A123. [ALH, KJK] Yes, it is my understanding that Entergy inspected all 8 of their steam generators.

Q124. Dr. Duquette indicated that only 6 of the 8 steam generators were inspected. Why do you believe that all 8 were examined?

A124. [ALH, KJK] Dr. Duquette cites NRC inspection reports for his conclusion that only 6 steam generators were inspected. The NRC inspection program is a sampling program. It is not uncommon for an NRC inspector to only review a subset of all inspections performed by licensees. Since the NSAL indicates that all steam generators are to be inspected, I would conclude based on the general statements made by IP2 and IP3 in reports describing their inspections that all steam generators were inspected. Exs. NYS000537 and NYS000543.

Q125. Is there anything else you would like to comment on regarding his discussion of these inspections and findings?

A125. [ALH, KJK] Dr. Duquette notes that the techniques employed by Entergy as a part of its NSAL 12-1 channel head inspections did not appear to use any magnification as its focus was to identify gross defects. This may be true; however, it is important to note that some of the defects that licensees found during these inspections were very small. For example, the flaw observed at Wolf Creek was only 0.1 inches deep and 2 inches long. Ex. NRC000538. This would not be considered a

"gross" defect. In addition, it was in an area not "required" to be inspected by NSAL 12-1 (NSAL 12-1 inspections are focused on the bottom of the channel head).

Q126. What did Entergy find during those inspections for IP2 and IP3?

A126. [ALH, KJK] No anomalies or degradation were detected during those inspections. Exs. NYS000543; Letter NL-13-143, Letter from Robert Walpole, to NRC, Response to Request for Additional Information Regarding Steam Generator Tube Inspection Performed During the Spring 2013 Refueling Outage Indian Point Unit No. 3 Docket No. 50-286 License No. DPR-64 (Oct. 29, 2013) Ex. NRC000193.

Q127. Can you draw any conclusions regarding the results of channel head inspections to the "aging management program the licensee plans to implement for the DP and TTSW" to manage cracking?

A127. [ALH, KJK] Yes. If there had been significant cracking or deformation of the divider plate assembly, it would have been detected. The actions taken by licensees in response to the foreign operating experience cited in the NSAL demonstrates that the industry can perform inspections, assess the findings, and take corrective action, as appropriate. This operating experience supports that licensees can effectively implement commitments similar to those made by IP2 and IP3 to address potential degradation of TTSW and SGDPs.

Q128. Are you familiar with Dr. Duquette's testimony regarding the exponential growth of cracks in cyclically stressed materials?

A128. [ALH, KJK] Yes.

Q129. Do you have any issues with any portions of that testimony?

A129. [ALH, KJK] Yes.

Q130. Please explain your issues with his exponential crack growth and this portion of the testimony?

A130. [ALH, KJK] The management of cracking in the divider plate and the TTSW relies primarily on the water chemistry program and a one-time inspection to verify the absence of cracking, as discussed in Commitments 41 and 42. If cracking were to occur in either the SGDP or TTSW, the licensee would put this issue into their corrective action program and determine what additional actions, if any, are needed. In this assessment, the licensee would assess whether additional inspections are needed and the frequency for these inspections. The crack growth used in these assessments may depend on the driving force for the cracking. These assessments may result in applying an exponential crack growth if it is warranted in the specific situation that is encountered.

Q131. Dr. Duquette describes an EPRI report and states that it indicates that renewal licensees could reference the report to revoke previous commitments to inspect the lower channel head area of steam generators, including the divider plate assembly and tube-to-tubesheet welds." Do you fully agree with this characterization?

A131. [ALH, KJK] The EPRI report focuses on inspection of the SGDP to address the cracking observed in foreign plants and the TTSW, with the latter related to Commitment 42. Any relaxation of commitments would have to follow existing NRC processes. Since the commitments on the SGDP and the TTSW will be in the FSAR as a result of an expected license condition, it is our expectation that any revision to the commitments would be in accordance with the process outlined in 10 CFR 50.59. This would include an assessment of the applicability of the report to the specific conditions in the unit.

Q132. In Dr. Duquette's testimony he indicates that he believes the fatigue crack growth analysis was non-conservative, do you agree with Dr. Duquette?

A132. [ALH, KJK] The NRC staff is still reviewing the EPRI report, Ex. NYS000544A-D, but we do not agree the results are non-conservative because it appears that EPRI did take into account environmental effects such as may be observed at Indian Point, and they analyzed for 40 years of fatigue crack growth.

Q133. Please explain why you disagree with Dr. Duquette's analysis?

A133. [ALH, KJK] The fatigue crack growth analysis performed by EPRI used the crack growth rate curves for water environments contained within the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. These curves represent conditions present in pressurized water reactor environments. In addition, it appears that EPRI assumed a crack pre-existing in the channel head and grew the crack for 40 years. This appears conservative since a flaw in the SGDP would need to initiate, and then grow into the channel head material. This would take time. Based on operating experience to date, it appears that more than 20 years of steam generator operation are needed for cracks to initiate in the SGDP. As a result, growing the flaw that pre-exists in the channel head for 40 years would appear to be conservative since it would extend beyond the period of extended operation contemplated by the license renewal application.

Q134. What is your experience with crack growth in steam generator tubes?

A134. [ALH, KJK] Crack growth in steam generator tubes can vary. In general, the growth rate is low and tube integrity can be effectively managed. Occasionally, a crack may grow faster than the typical crack growth rate. This has occurred a few times in the industry; however, in most instances the cracking has been managed through normal inservice inspections.

Q135. Are you aware of any data that suggests cracks in the divider plate will grow exponentially?

A135. [ALH, KJK] No, we are not aware of any data that suggests cracks in the divider plate will grow exponentially.

Q136. Has Dr. Duquette provided any information to support his contention that cracks in the divider plate will grow exponentially?

A136. [ALH, KJK] No, outside of his opinion, Dr. Duquette has not pointed to any relevant data, calculations, operating experience, or other similar information regarding exponential growth of cracks in the divider plate.

Q137. What is your perspective on the operating experience with cracking of the divider plate and TTSW?

A137. [ALH, KJK] Worldwide, there have only been a small number of cracks reported in divider plate welds. This has occurred in a very limited number of foreign steam generators. The cracks appear to be shallow and slow growing, if growing at all. In addition, there appears to be extenuating circumstances leading to this cracking as discussed in response to Q113. Some steam generators have been in operation for over 30 years. If significant cracking were occurring, we would expect it to have manifested itself already. Based on operating experience to date, we believe there is a low likelihood of cracking in the divider plate and TTSW and when (and if) it occurs it is likely to be of low safety significance.

Q138. Are you familiar with Dr. Duquette's testimony regarding the frequency and timing of inspections of the SGDP and TTSWs?

A138. [ALH, KJK] Yes.

Q139. Please describe the frequency and timing at which Dr. Duquette would like Entergy to perform the inspections of the SGDP and TTSWs?

A139. [ALH, KJK] It is our understanding that Dr. Duquette would like Entergy to inspect the SGDP and TTSWs as soon as possible at IP2 and before the period of extended operation for IP3. He also recommends that Entergy conduct follow-up inspections at least every 10 years. Ex. NRC000532, Duquette PFT at page 28.

Q140. Are these inspections necessary for meeting the regulatory requirements for managing aging of the SGDP and TTSWs?

A140. [ALH, KJK] No.

Q141. In your expert opinion, why are inspections proposed by Dr. Duquette not necessary for meeting the Commission's regulatory requirements?

A141. [ALH, KJK] The aging effect of PWSCC of the SGDP and the TTSW is managed through the Water Chemistry Control – Primary and Secondary Program. Therefore, the primary means of managing the aging effects is water chemistry not inspections. The one-time inspection for the SGDP and TTSWs is a verification inspection to provide additional assurance that the aging effect is being effectively managed through the water chemistry control program. Even if one were to assume the water chemistry control program does not manage the aging effect, given the low likelihood of cracking in either the SGDP or the TTSWs and the low safety significance of the cracking, performing inspections as soon as possible is not necessary. Similarly, routine inspections are not needed given the low likelihood of cracking and the low safety significance. In addition, the time frame for Entergy's plans to inspect IP2 and IP3 are appropriate to ensure that sufficient operating time for each steam generator has occurred to allow for PWSCC to develop to detectable levels, if it develops at all. Also,

based on operating experience to date, the inspections are also early enough in the steam generator operating life to detect any cracking prior to the cracks jeopardizing the safety functions of the reactor coolant pressure boundary components.

Q142. Has Dr. Duquette provided support for his assertion that "a small crack that develops during the first 25 years of an IP3 steam generator's life may rapidly develop into a crack that compromises the integrity of a reactor pressure boundary or other safety related component before the renewed licensing period ends"?

A142. [ALH, KJK] Outside of his opinion, Dr. Duquette has not pointed to any relevant data, calculations, operating experience, or other similar information that these cracks would rapidly grow and result in compromising the reactor pressure boundary or a safety-related component. He does not even identify in his testimony what the other safety-related component might be.

Q143. Has the Applicant properly addressed this newly identified foreign operating experience with PWSCC in the SGDP and the possibility of cracking in the TTSW using good engineering practice?

A143. [ALH, KJK] Yes, the Applicant has properly addressed this newly identified foreign operating experience with PWSCC in the SGDP and the possibility of cracking in the TTSW using good engineering practice. We would point out that most, if not all, of the items addressed in the SER Supp. 1, such as PWSCC of SGDP and TTSW, were pursued by the Staff because they represented new information at the time. Based on this, the Applicant has addressed these items, and in this case, the NRC Staff has found that the Applicant has provided an effective means to provide adequate aging management for the SGDP and the TTSW.

Good engineering practice does not normally lend itself to precise definitions. "Good engineering practice" for aging degradation issues could use the following four general concepts: (1)

safety significance of the finding (how far progressed was the finding - were the safety margins reduced?), (2) safety significance of the issue (what would be the safety impact if the finding was progressed to the point that the safety margin was eliminated?), (3) applicability to one's plant (how similar are the materials of fabrication, environment, operating conditions, etc.), and (4) development of reasonable measures to assess the condition of one's plant (including inspection methodology, timing, frequency, acceptance criteria, corrective actions, etc.) based on the evaluations of items (1), (2), and (3).

Here, the Applicant has used good engineering practice through its involvement in the SGTF activities that have addressed items (1) and (2). In addition, the Applicant's evaluations to respond to requests for additional information address item (3). Finally, the commitments that the Applicant has made to verify effectiveness of its Water Chemistry Control – Primary and Secondary Program through one-time inspections of the SGDP and the TTSWs address item (4). Therefore, we would conclude that the Applicant has used good engineering practice in proposing adequate aging management to address potential issues with PWSCC of the SGDP and the TTSWs.

Q144. From your review of the New York's expert's testimony, does he seem to favor the use of analysis to make decisions regarding these steam generator aging management issues or does he seem to favor the use of inspections?

A144. [ALH, KJK] The testimony from New York's expert clearly favors inspections, and he clearly wants the inspections to occur sooner than planned by the applicant and accepted by the staff. He states that "[F]rom an engineering perspective, it would be irresponsible to rely exclusively on mathematical modeling data, particularly since we have seen . . . that models can be non-conservative, unreliable or just plain wrong." [NYS00532 at page 19] And also "Entergy should affirmatively and clearly commit to performing inspections as soon as possible for IP2, and certainly before the period of extended operation for IP3." [NYS00532 at page 29].

[ALH] Some of New York's experts on other contentions take decidedly different views often rejecting inspections in favor of additional analysis or essentially replacing components on some set schedule prior to possibility of aging but without regard to whether they could continue to perform their intended function.

Q145. In your expert opinion, is it ever appropriate to dismiss a particular aging management method out of hand whether it is an analysis method or a condition monitoring program?

A145. [ALH, KJK] No.

Q146. Please explain why a method for performing aging management should not be dismissed out of hand?

A146. [ALH, KJK] There are many ways to address aging of structures, systems and components. Depending on the component, one or more methods may be acceptable (or unacceptable). An evaluation of an aging management program needs to be performed for each structure, system and component to ensure its acceptability (e.g., an inspection program for an inaccessible component would not be acceptable).

Q147. Dr. Duquette raises several issues regarding the fundamental operating conditions of French reactors as compared to U.S. reactors, do you agree with his analysis?

A147. [ALH, KJK] No.

Q148. Could you please explain your concerns regarding his analysis of the operating conditions between French and U.S. reactors?

A148. [ALH, KJK] In the United States, nuclear power plants typically operate continuously at 100 percent power. This type of operation is termed "base-loading." The French nuclear power plants

tend to vary their power level during operation in response to electricity demand. This type of operation is referred to as "load following." Operation in these modes is not a result of steam generator tube degradation as Dr. Duquette implies. Nonetheless, there have been instances in the U.S. and presumably in France where plants have operated at lower power levels to address steam generator issues.

Q149. In terms of stresses, is one of these modes of operation, base-loading or load following, clearly better?

A149. [ALH, KJK] No. There are steady state and cyclic stresses. Components will most likely experience more cyclic stresses in a plant that load follows than a plant that is base loading. It all depends on the specific conditions at the plant and the power levels at which they are operating, and the frequency and magnitude of the power changes.

Q150. Is there anything that has occurred since your previous testimony or raised by New York's expert most recent testimony that changes your opinion regarding the adequacy of Entergy's proposed program for monitoring the SGDP and TTSWs?

A150. [ALH, KJK] No.

Q151. In your expert opinion, does Entergy's proposal to use the water chemistry program and a one-time inspection of the SGDP and TTSW satisfy the regulatory requirements for issuing a renewed license?

A151. [ALH, KJK] Yes.

Q152. Are Commitments 41 and 42 sufficiently clear on what actions need to be taken by Entergy?

A152. [ALH, KJK] Yes.

Q153. Are you familiar with New York's assertions regarding the unenforceability of commitments?

A153. [ALH] Yes.

Q154. Do you agree with New York's assertions regarding whether these license renewal commitments would be enforceable by the Staff, New York, or the public?

A154. [ALH] No. The assertions fail to account for the license conditions that are imposed as a condition of issuing the renewed license that would incorporate the license renewal commitments, including Commitments 41 and 42, in the IP2 and IP3 FSAR.

Q155. Can you explain why New York is mistaken regarding the enforceability of Commitments 41 and 42?

A155. [ALH] New York's analysis is flawed for several reasons, including the failure to consider: (1) the impact of license conditions imposed as part of issuing the renewed license, and (2) the Commission's regulations that impose controls regarding the method for making changes to the FSAR, including license amendments under 10 C.F.R. § 50.90.

Q156. Could you explain the license conditions that are imposed as a condition of license renewal?

A156. [ALH] Yes. The license conditions vary from plant-to-plant depending on the circumstances of each plant. There are, however, several license conditions that are imposed for each

renewed license. The staff indicated in its SER that it intended to impose these license conditions as a condition of license renewal. For example in the recently issued license renewal for Callaway, the Staff imposed:

> (17) <u>License Renewal License Conditions</u>
>
> (a) The information in the Final Safety Analysis Report (FSAR) supplement, submitted pursuant to 10 CFR 54.21(d), is henceforth part of the FSAR which will be updated in accordance with 10 CFR 50.71(e). As such, the licensee may make changes to the programs and activities described in the FSAR supplement, without prior Commission approval, provided the licensee evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.
>
> (b) The licensee's FSAR supplement submitted pursuant to 10 CFR 54.21(d), as revised during the license renewal application review process, and as revised in accordance with license condition 2.C.(17)(a), describes certain programs to be implemented and activities to be completed prior to the period of extended operation.
>
> 1. [Licensee] shall implement those new programs and enhancements to existing programs no later than April 18, 2024.
>
> 2. [Licensee] shall complete those designated inspection and testing activities, as noted in Appendix A of the [Safety Evaluation Report for License Renewal], no later than April 18, 2024, or the end of the last refueling outage prior to the period of extended operation, whichever occurs later.
>
> 3. [Licensee] shall notify the NRC in writing within 30 days after having accomplished item (b)1 above and include the status of those activities that have been or remain to be completed in item (b)2 above.

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

Q157. Can you explain what the license condition that the Staff would impose on the renewed license does?

A157. [ALH] The license condition accomplishes several related issues. First, the condition will require Entergy to update the FSAR with the proposed AMPs and other commitments made during the license renewal review process. It requires the programs to be implemented by a date certain, or end

of the last refueling outage prior to the period of extended operations. Second, it also establishes the dates for certain inspections and testing activities to be completed, which can vary from “prior to the period of extended operation” to specific dates for inspections scheduled during the license renewal period. Third, it requires the licensee to notify the Staff when it completes the implementation of the AMP and testing that are required prior to period of extended operation. This notification allows the Staff to schedule the IP71003 inspection for the plant, which is implemented to verify completion of commitments. Finally, the requirement to incorporate these programs as part of the licensee’s FSAR places any changes to the programs under the Commission’s regulatory control and change process. The licensee would need to evaluate any proposed changes to the FSAR under 10 C.F.R. § 50.59 and, depending on the proposed change and evaluation, make changes administratively or through a formal license amendment.

Q158. Can you explain how the Commission’s regulations act to control changes to the license renewal commitments?

A158. [ALH] As discussed above, the license condition requires the licensee to place the programs described in its license renewal commitments in its FSAR. Once part of the FSAR, changes are controlled under the Commission regulations at 10 C.F.R. § 50.59. While the full evaluation process for determining whether a particular change can be made by the licensee alone or whether it would require a license amendment is complex, the licensee is required to document any determination regarding decisions to modify the FSAR without a licensee amendment. Changes to the FSAR are submitted to Staff on a regular basis for review. Should members of public or New York conclude that the licensee has made changes to its FSAR improperly, they could petition the Director of NRR to take action under 10 C.F.R. § 2.206. The Staff, similarly, could take enforcement action if changes were made to the FSAR by the licensee but required a license amendment. Alternatively, members of the

public or New York would have an opportunity to challenge any change to the FSAR submitted as a license amendment under 10 C.F.R. § 50.90 through the normal hearing process.

Q159. Does this conclude your testimony?

A159. [ALH, KJK] Yes.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF ALLEN L. HISER, JR.

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

/Executed in Accord with 10 CFR 2.304(d)/

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August 10, 2015

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AFFIDAVIT OF KENNETH J. KARWOSKI

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

/Executed in Accord with 10 CFR 2.304(d)/

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August 10, 2015