August 20, 2012

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 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.") NRC000103.

A1b. [KJK] My name is Mr. Kenneth 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.

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. My responsibilities include 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 and report), 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 to the NRC safety evaluation report for Indian Point Units 2 and 3. A list of the documents follows:

(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. Can you briefly describe the Indian Point SGDPs?

A6. [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.

Q7. Can you briefly describe the TTSWs as they relate to steam generators?

A7. [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.

Q8. Can you describe the structure and purpose of the lower channel head in the Indian Point steam generators?

A8. [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.

Q9. Can you describe the structure and purpose of the tubesheet in the Indian Point steam generators?

A9. [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.

Q10. Can you describe the structure and purpose of the SGDP?

A10. [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.

Q11. Are you familiar with NYS-38/RK-TC-5?

A11. [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 also 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"), and the accompanying exhibits including Exs. NYS000375-NYS000396 and RIV000102-RIV000106.

Q12. What are the "multiple bases" that the Intervenors referred to in Contention NYS-38/RK-TC-5?

A12. [ALH, KJK] As described in the State of New York and Riverkeeper's New Joint Contention NYS-38/RK-TC-5, (September 30, 2011) (ADAMS Accession No. ML11273A196) ("NYS-38/RK-TC-5"), the Intervenors' bases are: Basis (1) that Entergy has deferred defining the process to be used to determine the most limiting locations for environmentally-assisted metal fatigue calculations (CUF_{en} calculations) and selection of those locations; Basis (2) that Entergy has not specified the criteria it will use and assumptions upon which it will rely for modifying the WESTEMS computer model for CUF_{en} calculations; Basis (3) that Entergy has not adequately defined how it will manage primary water stress corrosion cracking (PWSCC) for the steam generator divider plates because it will rely on an industry "report which is not expected to be available until 2013 and, in the meantime to institute an unspecified inspection program to ascertain, long after commencement of the license renewal period, whether stress corrosion cracking is actually occurring in the divider plates of the steam generators;" and, Basis (4) that Entergy "has offered an AMP for reactor vessel internals which it will not actually follow and has promised to follow an AMP the details of which are not disclosed." 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) and Basis (3), as described above.

Q13. Which of the Bases in Contention NYS-38/RK-TC-5 does your testimony address?

A13. [ALH, KJK] Our testimony addresses Basis (3) of Contention NYS-38/RK-TC-5.

Q14. Could you briefly describe the issues raised in NYS-38/RK-TC-5 with respect to the steam generators?

A14. [ALH, KJK] As described in the State of New York and Riverkeeper's New Joint Contention NYS-38/RK-TC-5, (September 30, 2011) (ADAMS Accession No. ML11273A196), 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) 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 TTSW.

Q15. Based on your review, what is your expert opinion regarding NYS-38/RK-TC-5 with respect to aging management for the SGDP?

A15. [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").

Q16. Why do you disagree with the NYS-38/RK-TC-5 contention related to aging management of the SGDP?

A16. [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.

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

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 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.

Q18. Why do you disagree with the NYS-38/RK-TC-5 contention related to aging management for cracking due to PWSCC of the TTSW?

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 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.

Q19. What is the aging effect that is at issue in NYS-38/RK-TC-5 relative to the SGDP and the TTSW?

A19. [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.

Q20. Can you provide a description of PWSCC in relation to the SGDP and the TTSWs?

A20. [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.

Q21. What materials are used at Indian Point for the SGDP and the TTSWs?

A21. [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, would have a chemistry that is 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, and the chemical content cannot at present be proven to be substantially different from that of Alloy 600. As a result, it was conservatively assumed by the Applicant that the TTSWs in IP3 are also susceptible to PWSCC.

Q22. What is an autogenous weld?

A22. [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.

Q23. What has the Applicant proposed for aging management of the SGDP?

A23. [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.

Q24. How does the Applicant's Water Chemistry Control – Primary and Secondary Program manage cracking due to PWSCC?

A24. [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.

Q25. What occurred after issuance of the August 2009 SER (Ex. NRC000006) for Indian Point that led the Applicant to make this commitment?

A25. [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 let 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.

Q26. When did the NRC become aware of the foreign operating experience related to the potential for PWSCC in the SGDP?

A26. [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").

Q27. What is the SGTF?

A27. [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.

Q28. Based on the preliminary information from the foreign operating experience, what were the Staff's initial concerns with the U.S. steam generators?

A28. [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.

Q29. How did the Staff address these concerns?

A29. [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 the investigation into the French operating experience.

Q30. What did the SGTF do to assess the impact of PWSCC on the SGDP?

A30. [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.

Q31. Would you explain what is meant by a "fully degraded SGDP"?

A31. [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.

Q32. Did the SGTF provide any additional assessments with respect to the investigation on French steam generators?

A32. [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. The French are now considering eliminating or reducing the frequency of the SGDP inspections.

Q33. Has the Staff verified the SGTF assessments of SGDP and the French operating experience?

A33. [ALH, KJK] Yes.

Q34. How did the Staff verify the SGTF representation regarding the French operating experience?

A34. [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.

Q35. What has the Staff determined regarding the potential impact of PWSCC in the SGDP?

A35. [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 French 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.

Q36. 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?

A36. [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.

Q37. What does Commitment 41 state regarding the examination technique to be used for the SGDP?

A37. [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.

Q38. Does Commitment 41 require the Applicant to use a qualified examination technique for these inspections?

A38. [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 indicates, 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.

Q39. Will the technique that Indian Point uses for its SGDP inspections be qualified?

A39. [ALH, KJK] Yes. As stated in A38, the Applicant would need to use a qualified technique for compliance with 10 CFR Part 50, Appendix B.

Q40. Are you familiar with Dr. Duquette's and Dr. Lahey's testimony that there is no qualified technique to inspect the SGDP?

A40. [ALH, KJK] Yes.

Q41. Can you briefly explain what a qualified technique means?

A41. [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.

Q42. Why doesn't the Industry have any inspection techniques for the SGDP that are qualified?

A42. [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.

Q43. Has the Industry been developing inspection techniques for the SGDP that are qualified?

A43. [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.

Q44. Has the Staff ever accepted a similar approach where the inspection technique has not been specified?

A44. [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.

Q45. When will the Applicant perform SGDP inspections for IP2 and IP3?

A45. [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 before December 13, 2017.

Q46. Why are the last allowable dates for inspection for IP2 steam generators and the IP3 steam generators so far apart?

A46. [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 currently have 11 additional calendar years of operation beyond the IP2 steam generators.

Q47. Why is it acceptable to wait so long before inspecting the steam generators?

A47. [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.

Q48. Which steam generators will be examined under Commitment 41?

A48. [ALH, KJK] Each steam generator will be inspected at both IP2 and IP3.

Q49. How often will the steam generators at IP2 and IP3 be inspected?

A49. [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.

Q50. What are the acceptance criteria for the Commitment 41 inspections?

A50. [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.

Q51. If cracking is identified, what actions will the Applicant be required to take?

A51. [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).

Q52. Why does the NRC believe that the Applicant's planned aging management is adequate for the SGDP?

A52. [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.

Q53. What has the Applicant proposed for aging management of the TTSWs?

A53. [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 seek a license amendment based on an analysis showing that the

TTSWs are not susceptible to PWSCC or are not properly classified as part of the reactor coolant pressure boundary. 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 tube-to-tubesheet welds 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.

Q54. What occurred after issuance of the SER for Indian Point that led the Applicant to make this commitment?

Q54. [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.

Q55. How are Option 1 and Option 2 of Commitment 42 interrelated?

A55. [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.

Q56. What is the Applicant required to do under Option 1 of Commitment 42?

A56. [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.

Q57. Can you provide an example of one method that might show that "the tubesheet cladding and welds are not susceptible to PWSCC?"

A57. [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.

Q58. Can you describe how Indian Point might be able to redefine "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?"

A58. [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.

Q59. 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?

Q59. [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.

Q60. Has the NRC granted this type of license amendment in the past?

A60. [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))

Q61. If Indian Point does not complete the actions provided for under Option 1, what kind of inspection will be conducted under Option 2?

A61. [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.

Q62. How are the TTSW inspections under Option 2 different from the steam generator tube inspections that Indian Point has been doing for many years?

A62. [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.

Q63. When will TTSW inspections for IP2 and IP3 have to be completed?

A63. [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.

Q64. 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?

A64. [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.

Q65. Which steam generator will be inspected under Option 2 of Commitment 42?

A65. [ALH, KJK] A representative sampling of the TTSWs in each steam generator at each unit will be inspected.

Q66. What is the frequency of the TTSW inspections?

A66. [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.

Q67. What will happen if these one-time TTSW inspections identify cracking?

A67. [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.

Q68. Are the actions identified in A67 the only additional actions that will be taken should cracking be identified in the TTSW at Indian Point?

A68. [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.

Q69. In an overall sense, why does the NRC believe that the Applicant's planned aging management is adequate for the TTSWs?

A69. [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.

Q70. Do either Commitment 41 or Commitment 42 specify the inspection technique that will be used to detect PWSCC?

A70. [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.

Q71. 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?

A71. [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 A38, and are subject to review under the NRC reactor oversight process by experienced inservice inspection inspectors from NRC's Region I.

Q72. Why would the NRC accept commitments from the Applicant that do not specify the inspection technique to be used?

Q72. [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.

Q73. 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?

A73. [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.

Q74. 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?

A74. [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 are 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.

Q75. 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?

A75. [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.

Q76. Is the Applicant’s proposed aging management of the SGDP and the TTSWs consistent with the GALL Report?

A76. [ALH, KJK] 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.

Q77. 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?

A77. [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.

<u>Scope of Program</u>

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.

Q78. 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?

A78. [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.

<u>Scope of Program</u>

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.

<u>Preventive Actions</u>

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.

<u>Parameters Monitored or Inspected</u>

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.

<u>Detection of Aging Effects</u>

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.

Q79. Do you agree with Intervenors' expert that PWSCC in the SGDP might compromise the pressure boundary in the steam generator?

A79. [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 prior to being able to compromise the pressure boundary. Such cracking has not been observed in any of the reported operating experience.

Q80. How does the aging management proposed by the Applicant for the SGDP and the TTSWs prevent the reactor coolant pressure boundary from being compromised?

A80. [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 the 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 provide 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 will be identified and remediated prior to the development of any safety issue, such as compromises to the reactor coolant pressure boundary.

Q81. 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?

A81. [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.

Q82. 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?

A82. [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 and they are now considering eliminating or reducing the frequency of SGDP inspections.

Q83. 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?

A83. [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.

Q84. Can you briefly explain what is meant by a "barrier to crack propagation"?

A84. [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.

Q85. 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?

A85. [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.

Q86. Has the NRC concluded that the Water Chemistry Control – Primary and Secondary Program is not sufficient to prevent PWSCC in the SGDP?

A86. [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.

Q87. Do you agree with Dr. Duquette's conclusions that cracks from the SGDP are likely to propagate to the TTSWs?

A87. [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.

Q88. Do you agree that the exact stress states of the SGDP or TTSWs are not currently well understood by the industry?

A88. [ALH, KJK] We agree with Dr. Duquette that the precise stress states of the SGDP and the TTSWs, including weld residual stresses, are not well understood. The U.S. industry has research underway to evaluate the stress states for the SGDP and the TTSW. Ex. NRC000160 (NEI Steam Generator Task Force NRC/Industry Update dated August 4, 2011) (ML11216A050). Finite element modeling will be used to determine the stresses in the channel head and in the tubesheet regions. The results from the modeling will be used to determine if the stresses are high enough to drive a crack from the SGDP to the triple point and through the channel head, and to determine if the stresses are high enough to propagate cracks from the SGDP into the tubesheet cladding to the TTSW.

Q89. Is it important to know the precise stress states for the SGDP, tubesheet cladding, and TTSW in order to have an adequate AMP?

A89. [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. It isn't clear how critical such information is, because 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.

Q90. Do you agree with Dr. Duquette's observations on the location and size of cracking identified overseas?

A90. [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.

Q91. Do you agree with Dr. Duquette's conclusions regarding the rapid propagation of cracks into the TTSWs?

A91. [ALH, KJK] No, we do not agree with Dr. Duquette's conclusions. Dr. Duquette's posits that PWSCC initiation in the TTSW may lead to a rapid compromise of the pressure boundary, and "[a]lloys that exhibit resistance to crack initiation often show marked susceptibility to rapid crack propagation rates." (Ex. NYS000390 at 17) The context of Dr. Duquette's statements isn't clear. He provides no evidence demonstrating that TTSW cracking will propagate rapidly within the weld itself, and similarly there is no evidence that TTSW cracking will propagate into the steam generator tube material, let alone propagate “rapidly” in the tube material. If cracks in the TTSWs were to propagate into the tube material, these cracks would be detected by the inservice inspections of the tubes. In U.S. plants, cracks have been detected near the tube ends in some plants but, there is no evidence that they are rapidly propagating.

We note that Dr. Duquette cites no sources to justify his statement that resistant alloys might exhibit rapid crack propagation rates. As such, we cannot agree with Dr. Duquette’s conclusions.

Q92. 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?

A92. [ALH, KJK] Yes, the Applicant has properly addressed this newly identified foreign operating experience with PWSCC in the SGDP and 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 represent new information. 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.

Q93. Does this conclude your testimony?

A93. [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 20, 2012

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

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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