UNITED STATES

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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In re:	Docket Nos. 50-247-LR; 50-286-LR
License Renewal Application Submitted by	ASLBP No. 07-858-03-LR-BD01
Entergy Nuclear Indian Point 2, LLC,	DPR-26, DPR-64
Entergy Nuclear Indian Point 3, LLC, and	
Entergy Nuclear Operations, Inc.	June 8, 2015

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PRE-FILED WRITTEN SUPPLEMENTAL TESTIMONY OF

DR. DAVID J. DUQUETTE

REGARDING CONTENTION NYS-38 / RK-TC-5

On behalf of the State of New York ("NYS" or "the State"), the Office of the Attorney General hereby submits the following rebuttal testimony by David J. Duquette, Ph.D. regarding Contention NYS-38/RK-TC-5.

Q. Please state your full name.

A. David J. Duquette.

Q. What is the purpose of this testimony you are now providing?

A. This testimony supplements my initial and rebuttal testimony on Contention NYS-38/RK-TC-5. It has been approximately three years since I provided my initial pre-filed testimony in this matter and two and a half years since I provided rebuttal testimony. The State of New York has asked me to review the record on Contention NYS-38/RK-TC-5 and respond to recent information and events.

Q. What documents did you review in preparation for this supplemental testimony?

A. I reviewed again Entergy's August 20, 2012 Statement of Position Regarding Contention NYS-38/RK-TC-5 (ENT000520), Entergy's Pre-filed Testimony of Entergy witnesses Nelson Azevedo, Robert Dolansky, Alan Cox, Jack Strosnider, Robert Nickel, Ph.D., and Mark Gray regarding Contention NYS-38/RK-TC-5 (ENT000521), and the accompanying exhibits. I also reviewed the NRC Staff's August 20, 2012 Statement of Position on Contention NYS-38/RK-TC-5 (NRC000147), NRC's Pre-filed Testimony of NRC Witnesses Dr. Allen Hiser and Kenneth Karwoski Concerning Portions of Contention NYS-38/RK-TC-5 (NRC000161), which focuses on steam generator issues, and the accompanying exhibits.[1]

[1] NRC Staff also submitted pre-filed testimony on another aspect of Contention NYS-38/RK-TC-5, namely NRC000148. That testimony focused on metal fatigue issues and did not discuss my June 2012 testimony or report on steam generator issues. Accordingly, my testimony here does not discuss NRC000148.

In addition, I also re-reviewed documents previously submitted by the State on this contention including my previous pre-filed testimony and report (NYS000372, NYS000373, NYS000452) and exhibits (including, without limitation, NYS000375 to NYS000394 and NYS000454 to NYS000463, NYS000472, NYS000146, NYS000147, NYS000160). These documents include a presentation from the EPRI Steam Generator Task Force (SGTF) to the NRC entitled "NRC/EPRI Steam Generator Task Force Meeting", dated August 21, 2012 (NYS000463), an NRC chart identifying original and replacement steam generators at U.S. plants prepared in 2009 (NYS000458), a paper numbered ICONE18-29457 entitled "Inspection of the Steam Generator Divider Plate," presented at the 18th International Conference on Nuclear Engineering, authored by D. D'Annucci and E. Lecour of Westinghouse for the May 2010 ICONE meeting (ENT000526), EPRI Report 1025133, "Steam Generator Management Program: Assessment of Channel Head Susceptibility to Primary Water Stress Corrosion Cracking," dated June 2012 (ENT000524), and various summary or demonstrative exhibits prepared by the State (NYS000454 to NYS000456).

In addition, I reviewed a summary chart identifying the materials used in the eight steam generators at Indian Point Unit 2 and Unit 3 (NYS000560), a 2014 EPRI report of cracking in steam generator channel head assemblies (NYS000544A-D), a 2012 Westinghouse Nuclear Safety Advisory Letter (NYS000549); various NRC/EPRI Steam Generator Task Force presentations (NYS000546 and NYS000550); steam generator tube inspection reports (NYS000543 and NYS000537); integrated inspection reports (NYS000536 and NYS541); an in-service inspection summary (NYS000540); steam management program documents (NYS000533, NYS000534, NYS000554, NYS000555); commitment closure and verification forms (NYS000535 and NYS000553); NRC information notices and reports (NYS000551 and NYS000538); license amendment requests and approval letters and related documents (NYS000539, NYS000542, NYS000556, and NYS000547); responses to NRC requests for information (NYS000545); and an NRC report on Lessons learned from San Onofre (NYS000552).

Q. What are your overall conclusions having reviewed that information?

A. First, I disagree with Entergy and the NRC staff’s suggestion set out in their testimony that divider plate cracking is unlikely to occur in the future because it has not been observed to date in United States-based steam generators. I likewise disagree with Entergy and NRC staff's position that Entergy’s general approach to aging management issues will effectively provide adequate safety measures if cracking were to occur. Entergy's testimony reflects a "trust us" approach in the absence of real data on the condition of the eight Indian Point steam generators. Second, it is my opinion that in order to adequately address aging degradation in the Indian Point steam generators Entergy must unequivocally commit to and establish a sufficiently detailed aging management program that includes baseline and follow-up inspections of the steam generator channel head and divider plate assemblies, including the tube-to-tubesheet welds.

As discussed in my 2012 testimony, as well as that of Entergy and NRC Staff, the EPRI-sponsored Steam Generator Task Force is conducting an extensive research program into the propagation of cracks in the divider plate assembly. I understand that in October 2014, EPRI [redacted]. I have reviewed the report, and it does not change my view that Entergy must address potential primary water stress corrosion cracking and fatigue cracking in the eight steam generators at Indian Point before relicensing occurs. Thus, it is still my opinion that inspections of the steam generator channel head and divider plate assemblies and tube-to-tubesheet welds should be conducted before Indian Point Unit 3 begins its period of extended operation, and that such inspections should be conducted promptly at Indian Point Unit 2, since they have not yet been conducted at that facility.

In addition, while no industry-qualified technique for inspection of the lower channel head and divider plate assembly currently exists in the United States, any license renewal given to Entergy for the Indian Point facilities should be contingent on the company’s expeditious qualification of an inspection technique capable of identifying and evaluating primary water stress corrosion cracking and fatigue-related cracks. Entergy has identified a remote inspection technique that relies on ultrasonic, visual and liquid penetrant technologies developed by Westinghouse that has been used to successfully inspect divider plates in French steam generators. ICONE Westinghouse Paper (ENT000526). Instead of relying on the current absence of a U.S. industry-qualified inspection technique as an excuse to delay inspections at Indian Point 2 and Indian Point 3, Entergy should conduct the necessary inspections using techniques available now for detecting and evaluating cracks in the lower channel assembly. For example, Entergy can employ the Westinghouse technique pending future industry qualification, or some other similarly effective technique. I note that remote visual and ultrasonic inspections were used to inspect for possible flaws in the tubesheet to channel head transition region in Westinghouse steam generators at Wolf Creek Generating Station and Surry Power Station Unit 2. NRC Information Notice 13-20 (NYS000538).

Q. Why is it important that Entergy inspect the lower assemblies and the tube-to-tubesheet welds of the Indian Point steam generators?

A. Both Entergy and the NRC staff agree that the Indian Point Unit 2 and Indian Point Unit 3 steam generators have divider plates that are constructed from Alloy 600 and that the weld materials are also an Alloy 600 derivative (Alloy 82/182). It is well known that Alloy 600 is susceptible to PWSCC. As of mid-2015, the four steam generators at Indian Point Unit 2 have been in use for approximately 15 years. They were installed following the steam generator accident at Unit 2 in 2000. The four steam generators at Indian Point Unit 3 have been in use for approximately 26 years. They were installed in 1989 to replace the original Unit 3 steam generators, which had been in use for approximately 14 years at that time.

However, the current state of the divider plates, the stub runners, the channel heads, as well as the tube-to-tubesheet welds at Indian Point is largely unknown. Over the past few years, based on reports of cracking in divider plate assemblies in French steam generators, EPRI's SGTF has been examining the susceptibility of divider plate assemblies to PWSCC and investigating the possibility that stress corrosion cracking or fatigue induced cracks could propagate into the pressure boundary components. Entergy has stated that its approach to the divider plate assembly cracking problems is not dependent on the results of EPRI research, but that inspections being committed to by plants with renewed licenses will occur at an "appropriate" time, and that the Indian Point Quality Assurance Program will "drive appropriate safety evaluations." Without specific criteria for determining "appropriateness," Entergy's plan remains a hollow assurance that aging degradation of its steam generators will be adequately managed.

In my June 2012 report, I pointed out that EPRI has generically stated that the divider plates in United States steam generators are thicker than those that have experienced cracking in French steam generators, and that that factor alone may mitigate against PWSCC initiation in United States steam generators. Even if that conclusion proved to be true for some or most United States steam generators, the divider plates at Indian Point Unit 2 and Unit 3 are an exception to this general rule. While the majority of steam generators in the United States have divider plate thicknesses of approximately 1.9 inches, the Westinghouse Model 44F steam generators at Indian Point Unit 2 and Unit 3 have plate thicknesses of 1.26 inches, essentially the equivalent of the 1.3 inch thick divider plates used in the French steam generators where PWSCC cracking was first discovered. Thus, barring the possibility of differences in loading or pre-assembly processing of the divider plates and associated assemblies, the steam generators at Indian Point have essentially the same sensitivity to PWSCC as the French steam generators.

In my initial June 2012 testimony in this proceeding I referred to cracking that had occurred in the steam generator at Indian Point Unit 2. I agree with Entergy that replacement of mill annealed Alloy 600 tubing with thermally treated Alloy 600 tubing may reduce (but not eliminate) the potential for PWSCC in steam generator tubes.[2] However, no evidence has been presented that the divider plate assemblies are constructed from thermally treated alloys. Even if they are, the geometry of cracking that has been observed in the European steam generators has occurred near the welds joining the divider plates to the stub runners. Welding of these components can be expected to lead to dissolution of the grain boundary precipitates that are believed to provide a degree of PWSCC resistance in thermally treated alloys. Accordingly, the Entergy comments concerning the lack of cracking in the steam generator Alloy 600TT tubes has little or no relevance to the possibility of PWSCC in the divider plates or stub runners – or for that matter in the tube-to-tubesheet welds.

Q. I show you what has been marked as Exhibit NYS000549. Are you familiar with this document?

A. Yes. It is a [redacted] [redacted] were designated proprietary

[2] Often times the abbreviation "TT" is used to designate thermally treated components, e.g., "Alloy 600 TT" tubes.

by Westinghouse; Entergy subsequently disclosed these documents to the State in this proceeding.

Q. What is the purpose of the [redacted]?

A. [redacted] It was later disclosed that these conditions were identified at a foreign plant. [redacted]

[redacted].

Q. What recommendations, if any, did [redacted]

A. [redacted].

Q. Has Entergy performed the inspections recommended by [redacted]

A. Yes. I understand, based on Entergy's 2013 and 2014 Integrated Inspection Reports for IP2 and IP3 (NYS000536 and NYS000541), that Entergy performed remote video camera inspections of the lower channel head and divider plate to channel head welds for six of the eight Indian Point steam generators (21, 22, 24, 31, 33, 34) following [redacted].

Q. Did the NRC take any action to follow up on [redacted]

A. In October 2013, the NRC issued Information Notice 2013-20 entitled, “SG Channel Head and Tubesheet Degradation” (NYS000538) which addressed issues of potential corrosion and degradation in channel heads and tubesheets.

Q. Directing your attention to Exhibit NYS000545A-D, do you recognize that document?

A. Yes. It is a copy of [redacted] provided the report to Entergy and possibly other reactor operators. As I noted earlier, EPRI designated the document as containing proprietary information.

Q. Does the EPRI report resolve your concerns about the Indian Point steam generators?

A. No, it does not.

Q. Why is that?

A. There are several reasons. To begin with, [redacted]. This is a serious omission, since the steam generators at IP3, installed in 1989, will be operating beyond their 40 year life span towards the end of IP3's period of extended operation. Cracks can experience exponential growth rates in cyclically stressed materials. For example, 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. [redacted] does not provide any assurance whatsoever that this scenario would not occur.

In addition, it appears that [redacted] analysis may be non-conservative because it did not take into account the specific environmental conditions within the Indian Point steam generators, such as high temperatures and corrosivity, which are widely known to accelerate crack growth.

Any conclusions in the report based on this analysis would therefore have little to no relevance to the issue of crack growth in the Indian Point steam generators.

Furthermore, [redacted] components made of thermally-treated Alloy 690 (Alloy 690TT), which is more PWSCC resistant than thermally-treated Alloy 600 (Alloy 600TT). Since the IP2 steam generators' tubes, tube-to-tube sheet welds and divider plate assembly components are composed of Alloy 600TT, the report findings are simply inapplicable to IP2. I have concerns about the condition of IP2's steam generators precisely because these components are constructed of materials known to be susceptible to PWSCC.

I am also concerned about PWSCC in Alloy 600TT components and parts in IP3 steam generators. Although the tubes at IP3 steam generators are constructed of Alloy 690TT, the divider plate assemblies are conservatively assumed to be Alloy 600TT. Thus, the [redacted]

Q. [redacted]?

A. [redacted]

Q. [redacted]

A. Yes, it did. [redacted]

Q. Do you wish to comment on that?

A. Yes. First, I want to point out that Entergy has not confirmed that the steam generators at Indian Point do not have a layer of cold-work potentially susceptible to cracking. There is some evidence that the tube-to-tubesheet welds in IP2 have been cold-worked. For example, Westinghouse's Alternative Repair Criteria Analysis (WCAP-17828-NP) at pp. 2-9 and 3-7 (NYS000547) describes the fabrication and material properties of the tube and tubesheet welds and states that “[t]he manufacturing process used to assemble a steam generator creates a strain-hardened condition in the tubes.” These tubes are then inserted into the tubesheet bores and tack-expanded by hydraulic expansion or mechanical hard rolling before being welded to the tubesheet. Therefore, any cold-worked surfaces of the steam generators could be vulnerable to the same conditions experienced by the European reactors.

Moreover, I understand that the French operating experience differs in various ways from the U.S. operating experience which may account for slower crack growth rates observed in these foreign plants. My experience with presentations by Electricite de France (EdF), the operator of the steam generators in which cracking of the divider plate assembly was initially observed, is that, when a reactor in France encounters a limiting problem with a steam generator tube, the French typically “de-rate” the generator, meaning that they reduce the power of the system. In contrast, U.S. nuclear system operators typically “plug” a tube, meaning that the tube is taken out of service by blocking the entry and exit openings, but do not reduce the power rating. This means that, all other things being equal, U.S. pressurized water reactor steam generators may run hotter and be subject to greater stresses than their French counterparts. This difference in operating environments can affect steam generator susceptibility to PWSCC, as well as the growth rate of any cracks that develop. At Indian Point, steam generators with a number of plugged tubes may be more susceptible to PWSCC and fatigue induced cracking than steam generators at French reactors. Thus, while the French experience helped alert industry and government to the potential for divider plate assembly cracking under normal operating conditions in those plants, the lack of significant crack growth observed at the French reactors since the cracks were first reported should not be interpreted to suggest that any cracks found in a U.S. plant today would not propagate.

Q. [redacted]

A. [redacted].

[redacted] I believe any decision to abandon inspection plans [redacted] is misguided. As I stated earlier, regular inspections provide licensees and the NRC an opportunity to gather baseline data for benchmarking objective evidence of degradation and are a critical part of ensuring that systems operate safely. From an engineering perspective, it would be irresponsible to rely exclusively on mathematical modeling data, particularly since we have seen, in both the fracture toughness context (i.e., recently identified non-conservatism of BTP-5-3)(NYS000518-NYS000519) and the San Onofre steam generator tube rupture context (NRC Review of Lessons Learned at San Onofre, March 2014)(NYS000552), that models can be non-conservative, unreliable or just plain wrong.

Q. To your knowledge, has Entergy inspected the divider plate assemblies and tube-to-tubesheet welds of the Indian Point steam generators?

A. While Entergy has performed remote video inspections of the channel heads and divider plate–to-channel head welds for cladding degradation and PWSCC based on Westinghouse's NSAL 12-

1, it appears that inspections were performed for only six of the eight steam generators at Indian Point. Moreover, those inspections were limited in scope and did not employ techniques qualified to detect and measure cracks or flaws due to PWSCC. NRC Integrated Inspection Report, May 9, 2014 at 10 (NYS000541). Indeed, I do not believe Entergy used any magnification as a part of its NSAL 12-1 channel head inspection, as its focus was to identify [redacted]. [redacted]NYS000549). It is my opinion that [redacted] and the operating experiences at Wolf Creek and Surry referenced in the NRC's Information Notice 13-20 (NYS000538) suggest that failure of corrosion-resistant cladding in steam generators like those in use at Indian Point is a potential problem requiring detailed inspection and monitoring. Given the limited information available regarding the current condition of the lower channel head assembly areas of the eight steam generators at Indian Point, Entergy should, as soon as possible, perform an initial baseline inspection of IP2 and IP3 steam generator divider plate and channel head assemblies and tube-to-tubesheet welds as part of the company's "One Time Inspection Program" in order to confirm that its water chemistry program is in fact effective and that primary water stress corrosion cracking is not occurring. Generic Aging

Lessons Learned, Rev. 2 (2010), IV D 1-3,8. Similar to Entergy's In-Service Inspections, subsequent inspections of these steam generator locations should be performed at least once every 10 years. [redacted] underscores the vulnerability of these steam generators to corrosion and cracking, and the need for regular inspections to maintain safe operations.

Finally, recent documents report that Indian Point's steam generators have experienced age-related degradation as a result of wear associated with steam generator tube vibration, and that a number of tubes have been plugged and taken out of service as a result. (IP2 Steam Generator Examination Program Results 2014 Refueling Outage (2R21)(September 8, 2014)(NYS000543). I am concerned about the numerous indications of vibration-induced wear in the steam generator tubes at IP2, as documented in the plant's most recent tube inspection report. During the last outage, Entergy plugged five tubes due to wear. We learned from the San Onofre steam generator tube rupture event that wear, in that case caused by fluid–elastic instability, can quickly progress from flaw or crack initiation to tube failure. Unlike other, longer-acting degradation mechanisms that may be identified before they progress to a critical stage, wear can under certain circumstances rapidly progress between inspection intervals.

I also note that foreign objects were identified during Entergy’s steam generator tube inspections. During the most recent inspection, Entergy plugged at least nine tubes due to foreign objects trapped inside the tubes. Foreign objects in the steam generator can cause dents and dings. For example, in 1990, only one year after Steam Generator 34 was installed at IP3, a fuel alignment pin was found partially lodged in a tube end in the generator. 2007 Indian Point 3 Steam Generator Program (NYS000533) at p. 13, 14. Visual examination revealed that the foreign object made numerous indentations on the channel head surfaces. Follow up inspections indicated that impacts from loose parts resulted in deformities of some tube ends. The presence of foreign objects in the Indian Point steam generators and their potential to cause damage to the reactor coolant pressure boundary is an important concern. According to the NRC’s Information Notice 2013-11 (NYS000551), cracking in dented or dinged regions of Alloy 600TT tubing has been reported, and this operating experience highlights the importance of, and the challenges to, inspecting locations susceptible to degradation and identifying inspection methods capable of detecting that degradation. It is therefore imperative that Entergy remain vigilant in its inspections of the steam generator tubes, tube-to-tubesheet welds, and divider plate and channel head assemblies at IP2 and IP3.

Q. Can you describe Entergy's proposed inspection and aging management program for the lower assembly area and tube-to-tubesheet welds in the steam generators?

A. It is difficult to tell exactly what Entergy has unequivocally committed to do. As I've discussed, in 2011, Entergy presented two commitments regarding the steam generators, Commitment 41 and Commitment 42. These commitments are set out in Appendix A of the NRC Staff's 2011 Supplemental Safety Evaluation Report (NYS000160), at pages A-23 and A-24.

Q. Can you read Commitment 41?

A. Commitment 41 states that, "IPEC will inspect 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 assembly. The IP2 steam generator divider plate inspections will be completed within the first ten years of the period of extended operation (PEO). The IP3 steam generator divider plate inspections will be completed within the first refueling outage following the beginning of the PEO."

Q. What is the implementation schedule for Commitment 41?

A. For IP2, it is "after the beginning of the PEO and prior to September 28, 2023." For IP3, it is "prior to the end of the first refueling outage following the beginning of the PEO," which I understand to be around March or April 2017.

Q. Can you please read Commitment 42?

A. Commitment 42 provides that "IPEC 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."

Q. What is Option 1?

A. Option 1, which is also referred to as the "analysis" option, states that "IPEC 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 a license amendment request."

Q. What is the implementation schedule for Option 1?

A. For IP2, implementation is "prior to March 2024," and for IP3, "prior to the end of the first refueling outage following the beginning of the PEO."

Q. What is Option 2?

A. Option 2, which is also referred to as the "inspection" option, provides that "IPEC 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."

Q. What is the implementation schedule for Option 2?

A. For IP2, the implementation schedule is "between March 2020 and March 2024", and for IP3, "prior to the end of the first refueling outages following the beginning of the PEO," which again, I understand to be around March or April of 2017.

Q. Can you summarize what Entergy has agreed to do under those Commitments?

Q. Under Commitment 41, Entergy committed to inspect and assess the condition of the divider plate assemblies in the IP2 and IP3 steam generators. Under Commitment 42, Entergy committed to either perform an analytical evaluation or an inspection of the tube-to-tubesheet welds.

Q. What is the status of those Commitments today?

A. I understand that on September 5, 2014, NRC staff approved an amendment to Entergy's operating license for Indian Point Unit 2 so as to "redefine" the reactor coolant pressure boundary to exclude tube-to-tubesheet welds (Amendment 277) and thereby relieved Entergy of the obligation to inspect the tube-to-tubesheet welds. (Technical Specification Amendment 277)(NYS000542). As a result of that license amendment, on September 17, 2014 Entergy "deemed" its Commitment 42 "complete for IP2." Commitment Closure Verification Form/ Corrective Action (LR-LAR-2011-00174)(NYS000553). Based on the data available today, I believe the NRC Staff was premature in granting Amendment 277. The NRC and the nuclear industry's understanding of PWSCC in the steam generator environment continues to evolve. In fact, the NRC recently committed over $2.3 million to fund research at Pacific Northwest National Laboratories for the purpose of evaluating PWSCC in nickel-based alloys used in steam generator and reactor components. NRC Weekly Information Report, May 15 2015 (NYS000557).

For now, it appears that it is Entergy's position that Commitment 41 relating to the divider plate assembly inspections is still open for IP2 and IP3 (Commitment 41 Closure Verification Form (NYS000535), but that Commitment 42 relating to tube-to-tubesheet welds is open for IP3 only (NYS000553). However, it is unclear what impact [redacted] will have on these remaining open commitments. As I noted earlier, the [redacted] Although Entergy disclosed [redacted] it is not clear what use, if any, Entergy has made, or will make, of the document. In my opinion, Entergy should not -- and cannot -- rely on the [redacted] to avoid inspecting the channel head and divider plate assemblies, including the tube-to-tubesheet welds in the eight Indian Point steam generators. To the extent that Entergy remains committed to performing inspections after license renewal, potentially well into the plants' periods of extended operation, that is an inadequate assurance for managing aging steam generators at Indian Point. Rather, 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. Additionally, Entergy must identify the inspection techniques it intends to use, develop acceptance criteria, and provide a detailed plan for addressing any flaws or indications that it may encounter. I also recommend that Entergy conduct follow-up inspections at least every 10 years, given the primarily Alloy 600TT construction of IP2 steam generator components and assemblies and the age of the IP3 steam generators.

In conclusion, from my perspective in 2011 and 2012 there was substantial uncertainty about what pathway Entergy would pursue with respect to steam generators; moreover, essential details were lacking in the various optional pathways Entergy identified. The recent EPRI Report and the operating license amendment have not resolved these uncertainties and unknowns.

Q. Does this conclude your supplemental testimony?

A. Yes. However, I reserve the right to offer further opinions if new information is presented.

UNITED STATES

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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In re:	Docket Nos. 50-247-LR; 50-286-LR
License Renewal Application Submitted by	ASLBP No. 07-858-03-LR-BD01
Entergy Nuclear Indian Point 2, LLC,	DPR-26, DPR-64
Entergy Nuclear Indian Point 3, LLC, and	
Entergy Nuclear Operations, Inc.	June 8, 2015

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DECLARATION OF DAVID J. DUQUETTE

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

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

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