


In the Matter of: Entergy Nuclear Operations, Inc.
(Indian Point Nuclear Generating Units 2 and 3)

	ASLBP #: 07-858-03-LR-BD01	Identified: 11/5/2015
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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

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

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

8 A. I reviewed again Entergy's August 20, 2012 Statement
9 of Position Regarding Contention NYS-38/RK-TC-5 (ENT000520),
10 Entergy's Pre-filed Testimony of Entergy witnesses Nelson
11 Azevedo, Robert Dolansky, Alan Cox, Jack Strosnider, Robert
12 Nickel, Ph.D., and Mark Gray regarding Contention NYS-38/RK-TC-5
13 (ENT000521), and the accompanying exhibits. I also reviewed the
14 NRC Staff's August 20, 2012 Statement of Position on Contention
15 NYS-38/RK-TC-5 (NRC000147), NRC's Pre-filed Testimony of NRC
16 Witnesses Dr. Allen Hiser and Kenneth Karwoski Concerning
17 Portions of Contention NYS-38/RK-TC-5 (NRC000161), which focuses
18 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.

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

20 In addition, I reviewed a summary chart identifying the
21 materials used in the eight steam generators at Indian Point
22 Unit 2 and Unit 3 (NYS000560), a 2014 EPRI report of cracking in

1 steam generator channel head assemblies (NYS000544A-D), a 2012
2 Westinghouse Nuclear Safety Advisory Letter (NYS000549); various
3 NRC/EPRI Steam Generator Task Force presentations (NYS000546 and
4 NYS000550); steam generator tube inspection reports (NYS000543
5 and NYS000537); integrated inspection reports (NYS000536 and
6 NYS541); an in-service inspection summary (NYS000540); steam
7 management program documents (NYS000533, NYS000534, NYS000554,
8 NYS000555); commitment closure and verification forms (NYS000535
9 and NYS000553); NRC information notices and reports (NYS000551
10 and NYS000538); license amendment requests and approval letters
11 and related documents (NYS000539, NYS000542, NYS000556, and
12 NYS000547); responses to NRC requests for information
13 (NYS000545); and an NRC report on Lessons learned from San
14 Onofre (NYS000552).

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

17 A. First, I disagree with Entergy and the NRC staff's
18 suggestion set out in their testimony that divider plate
19 cracking is unlikely to occur in the future because it has not
20 been observed to date in United States-based steam generators.
21 I likewise disagree with Entergy and NRC staff's position that
22 Entergy's general approach to aging management issues will

1 effectively provide adequate safety measures if cracking were to
2 occur. Entergy's testimony reflects a "trust us" approach in
3 the absence of real data on the condition of the eight Indian
4 Point steam generators. Second, it is my opinion that in order
5 to adequately address aging degradation in the Indian Point
6 steam generators Entergy must unequivocally commit to and
7 establish a sufficiently detailed aging management program that
8 includes baseline and follow-up inspections of the steam
9 generator channel head and divider plate assemblies, including
10 the tube-to-tubesheet welds.

11 As discussed in my 2012 testimony, as well as that of
12 Entergy and NRC Staff, the EPRI-sponsored Steam Generator Task
13 Force is conducting an extensive research program into the
14 propagation of cracks in the divider plate assembly. I

15 understand that in October 2014, EPRI [REDACTED]

16 [REDACTED]
17 [REDACTED]
18 [REDACTED]. I have

19 reviewed the report, and it does not change my view that Entergy
20 must address potential primary water stress corrosion cracking
21 and fatigue cracking in the eight steam generators at Indian
22 Point before relicensing occurs. Thus, it is still my opinion

1 that inspections of the steam generator channel head and divider
2 plate assemblies and tube-to-tubesheet welds should be conducted
3 before Indian Point Unit 3 begins its period of extended
4 operation, and that such inspections should be conducted
5 promptly at Indian Point Unit 2, since they have not yet been
6 conducted at that facility.

7 In addition, while no industry-qualified technique for
8 inspection of the lower channel head and divider plate assembly
9 currently exists in the United States, any license renewal given
10 to Entergy for the Indian Point facilities should be contingent
11 on the company's expeditious qualification of an inspection
12 technique capable of identifying and evaluating primary water
13 stress corrosion cracking and fatigue-related cracks. Entergy
14 has identified a remote inspection technique that relies on
15 ultrasonic, visual and liquid penetrant technologies developed
16 by Westinghouse that has been used to successfully inspect
17 divider plates in French steam generators. ICONE Westinghouse
18 Paper (ENT000526). Instead of relying on the current absence of
19 a U.S. industry-qualified inspection technique as an excuse to
20 delay inspections at Indian Point 2 and Indian Point 3, Entergy
21 should conduct the necessary inspections using techniques
22 available now for detecting and evaluating cracks in the lower

1 channel assembly. For example, Entergy can employ the
2 Westinghouse technique pending future industry qualification, or
3 some other similarly effective technique. I note that remote
4 visual and ultrasonic inspections were used to inspect for
5 possible flaws in the tubesheet to channel head transition
6 region in Westinghouse steam generators at Wolf Creek Generating
7 Station and Surry Power Station Unit 2. NRC Information Notice
8 13-20 (NYS000538).

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

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

1 replace the original Unit 3 steam generators, which had been in
2 use for approximately 14 years at that time.

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

20 In my June 2012 report, I pointed out that EPRI has
21 generically stated that the divider plates in United States
22 steam generators are thicker than those that have experienced

1 cracking in French steam generators, and that that factor alone
2 may mitigate against PWSCC initiation in United States steam
3 generators. Even if that conclusion proved to be true for some
4 or most United States steam generators, the divider plates at
5 Indian Point Unit 2 and Unit 3 are an exception to this general
6 rule. While the majority of steam generators in the United
7 States have divider plate thicknesses of approximately 1.9
8 inches, the Westinghouse Model 44F steam generators at Indian
9 Point Unit 2 and Unit 3 have plate thicknesses of 1.26 inches,
10 essentially the equivalent of the 1.3 inch thick divider plates
11 used in the French steam generators where PWSCC cracking was
12 first discovered. Thus, barring the possibility of differences
13 in loading or pre-assembly processing of the divider plates and
14 associated assemblies, the steam generators at Indian Point have
15 essentially the same sensitivity to PWSCC as the French steam
16 generators.

17 In my initial June 2012 testimony in this proceeding I
18 referred to cracking that had occurred in the steam generator at
19 Indian Point Unit 2. I agree with Entergy that replacement of
20 mill annealed Alloy 600 tubing with thermally treated Alloy 600
21 tubing may reduce (but not eliminate) the potential for PWSCC in

1 steam generator tubes.² However, no evidence has been presented
2 that the divider plate assemblies are constructed from thermally
3 treated alloys. Even if they are, the geometry of cracking that
4 has been observed in the European steam generators has occurred
5 near the welds joining the divider plates to the stub runners.
6 Welding of these components can be expected to lead to
7 dissolution of the grain boundary precipitates that are believed
8 to provide a degree of PWSCC resistance in thermally treated
9 alloys. Accordingly, the Entergy comments concerning the lack
10 of cracking in the steam generator Alloy 600TT tubes has little
11 or no relevance to the possibility of PWSCC in the divider
12 plates or stub runners - or for that matter in the tube-to-
13 tubesheet welds.

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

16 A. Yes. It is a [REDACTED]
17 [REDACTED]
18 [REDACTED]
19 [REDACTED] [REDACTED] were designated proprietary

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

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

3 Q. What is the purpose of the [REDACTED]
4 [REDACTED]?

5 A. [REDACTED]

6 [REDACTED]

7 [REDACTED]

8 [REDACTED]

9 [REDACTED]

10 [REDACTED]

11 [REDACTED]

12 [REDACTED]

13 [REDACTED]

14 [REDACTED] It was later

15 disclosed that these conditions were identified at a foreign

16 plant. [REDACTED]

17 [REDACTED]

18 [REDACTED]

19 [REDACTED]

20 [REDACTED]

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[REDACTED]

[REDACTED].

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

[REDACTED]

[REDACTED]

A. [REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[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

1 generators (21, 22, 24, 31, 33, 34) following [REDACTED]

2 [REDACTED].

3 Q. Did the NRC take any action to follow up on

4 [REDACTED]

5 A. In October 2013, the NRC issued Information Notice
6 2013-20 entitled, "SG Channel Head and Tubesheet Degradation"
7 (NYS000538) which addressed issues of potential corrosion and
8 degradation in channel heads and tubesheets.

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

11 A. Yes. It is a copy of [REDACTED]

12 [REDACTED]

13 [REDACTED]

14 [REDACTED]

15 [REDACTED]

16 [REDACTED] provided the report
17 to Entergy and possibly other reactor operators. As I noted
18 earlier, EPRI designated the document as containing proprietary
19 information.

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

22 A. No, it does not.

1 Q. Why is that?

2 A. There are several reasons. To begin with, [REDACTED]

3 [REDACTED]

4 [REDACTED]

5 [REDACTED]

6 [REDACTED]

7 [REDACTED]. This is a serious omission, since the steam
8 generators at IP3, installed in 1989, will be operating beyond
9 their 40 year life span towards the end of IP3's period of
10 extended operation. Cracks can experience exponential growth
11 rates in cyclically stressed materials. For example, a small
12 crack that develops during the first 25 years of an IP3 steam
13 generator's life may rapidly develop into a crack that
14 compromises the integrity of a reactor pressure boundary or
15 other safety related component before the renewed licensing
16 period ends. [REDACTED] does not provide any assurance
17 whatsoever that this scenario would not occur.

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

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

4 Furthermore, [REDACTED] components made
5 of thermally-treated Alloy 690 (Alloy 690TT), which is more
6 PWSCC resistant than thermally-treated Alloy 600 (Alloy 600TT).
7 Since the IP2 steam generators' tubes, tube-to-tube sheet welds
8 and divider plate assembly components are composed of Alloy
9 600TT, the report findings are simply inapplicable to IP2. I
10 have concerns about the condition of IP2's steam generators
11 precisely because these components are constructed of materials
12 known to be susceptible to PWSCC.

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

17 Thus, the [REDACTED]
18 [REDACTED]
19 [REDACTED]
20 [REDACTED]

21 Q. [REDACTED] ?

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A. [REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

Q. [REDACTED]

[REDACTED]

[REDACTED]

A. Yes, it did. [REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[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

1 the tube and tubesheet welds and states that "[t]he
2 manufacturing process used to assemble a steam generator creates
3 a strain-hardened condition in the tubes." These tubes are then
4 inserted into the tubesheet bores and tack-expanded by hydraulic
5 expansion or mechanical hard rolling before being welded to the
6 tubesheet. Therefore, any cold-worked surfaces of the steam
7 generators could be vulnerable to the same conditions
8 experienced by the European reactors.

9 Moreover, I understand that the French operating experience
10 differs in various ways from the U.S. operating experience which
11 may account for slower crack growth rates observed in these
12 foreign plants. My experience with presentations by Electricite
13 de France (EdF), the operator of the steam generators in which
14 cracking of the divider plate assembly was initially observed,
15 is that, when a reactor in France encounters a limiting problem
16 with a steam generator tube, the French typically "de-rate" the
17 generator, meaning that they reduce the power of the system. In
18 contrast, U.S. nuclear system operators typically "plug" a tube,
19 meaning that the tube is taken out of service by blocking the
20 entry and exit openings, but do not reduce the power rating.
21 This means that, all other things being equal, U.S. pressurized
22 water reactor steam generators may run hotter and be subject to

1 greater stresses than their French counterparts. This
2 difference in operating environments can affect steam generator
3 susceptibility to PWSCC, as well as the growth rate of any
4 cracks that develop. At Indian Point, steam generators with a
5 number of plugged tubes may be more susceptible to PWSCC and
6 fatigue induced cracking than steam generators at French
7 reactors. Thus, while the French experience helped alert
8 industry and government to the potential for divider plate
9 assembly cracking under normal operating conditions in those
10 plants, the lack of significant crack growth observed at the
11 French reactors since the cracks were first reported should not
12 be interpreted to suggest that any cracks found in a U.S. plant
13 today would not propagate.

14 Q. [REDACTED]

15 [REDACTED]

16 [REDACTED]

17 A. [REDACTED]

18 [REDACTED]

19 [REDACTED]

20 [REDACTED]

21 [REDACTED]

22 [REDACTED].

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[REDACTED]

[REDACTED]

[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 1, it appears that inspections were performed for only six of
2 the eight steam generators at Indian Point. Moreover, those
3 inspections were limited in scope and did not employ techniques
4 qualified to detect and measure cracks or flaws due to PWSCC.
5 NRC Integrated Inspection Report, May 9, 2014 at 10 (NYS000541).
6 Indeed, I do not believe Entergy used any magnification as a
7 part of its NSAL 12-1 channel head inspection, as its focus was
8 to identify [REDACTED]. [REDACTED] (NYS000549). It
9 is my opinion that [REDACTED] and the operating experiences at
10 Wolf Creek and Surry referenced in the NRC's Information Notice
11 13-20 (NYS000538) suggest that failure of corrosion-resistant
12 cladding in steam generators like those in use at Indian Point
13 is a potential problem requiring detailed inspection and
14 monitoring. Given the limited information available regarding
15 the current condition of the lower channel head assembly areas
16 of the eight steam generators at Indian Point, Entergy should,
17 as soon as possible, perform an initial baseline inspection of
18 IP2 and IP3 steam generator divider plate and channel head
19 assemblies and tube-to-tubesheet welds as part of the company's
20 "One Time Inspection Program" in order to confirm that its water
21 chemistry program is in fact effective and that primary water
22 stress corrosion cracking is not occurring. Generic Aging

1 Lessons Learned, Rev. 2 (2010), IV D 1-3,8. Similar to Entergy's
2 In-Service Inspections, subsequent inspections of these steam
3 generator locations should be performed at least once every 10
4 years. [REDACTED]

5 [REDACTED]
6 [REDACTED] underscores the vulnerability of these
7 steam generators to corrosion and cracking, and the need for
8 regular inspections to maintain safe operations.

9 Finally, recent documents report that Indian Point's steam
10 generators have experienced age-related degradation as a result
11 of wear associated with steam generator tube vibration, and that
12 a number of tubes have been plugged and taken out of service as
13 a result. (IP2 Steam Generator Examination Program Results 2014
14 Refueling Outage (2R21)(September 8, 2014)(NYS000543). I am
15 concerned about the numerous indications of vibration-induced
16 wear in the steam generator tubes at IP2, as documented in the
17 plant's most recent tube inspection report. During the last
18 outage, Entergy plugged five tubes due to wear. We learned from
19 the San Onofre steam generator tube rupture event that wear, in
20 that case caused by fluid-elastic instability, can quickly
21 progress from flaw or crack initiation to tube failure. Unlike
22 other, longer-acting degradation mechanisms that may be

1 identified before they progress to a critical stage, wear can
2 under certain circumstances rapidly progress between inspection
3 intervals.

4 I also note that foreign objects were identified during
5 Entergy's steam generator tube inspections. During the most
6 recent inspection, Entergy plugged at least nine tubes due to
7 foreign objects trapped inside the tubes. Foreign objects in
8 the steam generator can cause dents and dings. For example, in
9 1990, only one year after Steam Generator 34 was installed at
10 IP3, a fuel alignment pin was found partially lodged in a tube
11 end in the generator. 2007 Indian Point 3 Steam Generator
12 Program (NYS000533) at p. 13, 14. Visual examination revealed
13 that the foreign object made numerous indentations on the
14 channel head surfaces. Follow up inspections indicated that
15 impacts from loose parts resulted in deformities of some tube
16 ends. The presence of foreign objects in the Indian Point steam
17 generators and their potential to cause damage to the reactor
18 coolant pressure boundary is an important concern. According to
19 the NRC's Information Notice 2013-11 (NYS000551), cracking in
20 dented or dinged regions of Alloy 600TT tubing has been
21 reported, and this operating experience highlights the
22 importance of, and the challenges to, inspecting locations

1 susceptible to degradation and identifying inspection methods
2 capable of detecting that degradation. It is therefore
3 imperative that Entergy remain vigilant in its inspections of
4 the steam generator tubes, tube-to-tubesheet welds, and divider
5 plate and channel head assemblies at IP2 and IP3.

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

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

15 Q. Can you read Commitment 41?

16 A. Commitment 41 states that, "IPEC will inspect steam
17 generators for both units to assess the condition of the divider
18 plate assembly. The examination technique used will be capable
19 of detecting PWSCC in the steam generator divider plate
20 assembly. The IP2 steam generator divider plate inspections
21 will be completed within the first ten years of the period of
22 extended operation (PEO). The IP3 steam generator divider plate

1 inspections will be completed within the first refueling outage
2 following the beginning of the PEO."

3 Q. What is the implementation schedule for Commitment 41?

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

8 Q. Can you please read Commitment 42?

9 A. Commitment 42 provides that "IPEC will develop a plan
10 for each unit to address the potential for cracking of the
11 primary to secondary pressure boundary due to PWSCC of tube-to-
12 tubesheet welds using one of the following two options."

13 Q. What is Option 1?

14 A. Option 1, which is also referred to as the "analysis"
15 option, states that "IPEC will perform an analytical evaluation
16 of the steam generator tube-to-tubesheet welds in order to
17 establish a technical basis for either determining that the
18 tubesheet cladding and welds are not susceptible to PWSCC, or
19 redefining the pressure boundary in which the tube-to-tubesheet
20 weld is no longer included and, therefore, is not required for
21 reactor coolant pressure boundary function. The redefinition of

1 the reactor coolant pressure boundary must be approved by the
2 NRC as a license amendment request."

3 Q. What is the implementation schedule for Option 1?

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

7 Q. What is Option 2?

8 A. Option 2, which is also referred to as the
9 "inspection" option, provides that "IPEC will perform a one-time
10 inspection of a representative number of tube-to-tubesheet welds
11 in each steam generator to determine if PWSCC cracking is
12 present. If weld cracking is identified:

13 a. The condition will be resolved through repair or
14 engineering evaluation to justify continued service, as
15 appropriate, and

16 b. An ongoing monitoring program will be established to
17 perform routine tube-to-tubesheet weld inspections for the
18 remaining life of the steam generators."

19 Q. What is the implementation schedule for Option 2?

20 A. For IP2, the implementation schedule is "between March
21 2020 and March 2024", and for IP3, "prior to the end of the

1 first refueling outages following the beginning of the PEO,"
2 which again, I understand to be around March or April of 2017.

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

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

10 Q. What is the status of those Commitments today?

11 A. I understand that on September 5, 2014, NRC staff
12 approved an amendment to Entergy's operating license for Indian
13 Point Unit 2 so as to "redefine" the reactor coolant pressure
14 boundary to exclude tube-to-tubesheet welds (Amendment 277) and
15 thereby relieved Entergy of the obligation to inspect the tube-
16 to-tubesheet welds. (Technical Specification Amendment
17 277)(NYS000542). As a result of that license amendment, on
18 September 17, 2014 Entergy "deemed" its Commitment 42 "complete
19 for IP2." Commitment Closure Verification Form/ Corrective
20 Action (LR-LAR-2011-00174)(NYS000553). Based on the data
21 available today, I believe the NRC Staff was premature in
22 granting Amendment 277. The NRC and the nuclear industry's

1 understanding of PWSCC in the steam generator environment
2 continues to evolve. In fact, the NRC recently committed over
3 \$2.3 million to fund research at Pacific Northwest National
4 Laboratories for the purpose of evaluating PWSCC in nickel-based
5 alloys used in steam generator and reactor components. NRC
6 Weekly Information Report, May 15 2015 (NYS000557).

7 For now, it appears that it is Entergy's position that
8 Commitment 41 relating to the divider plate assembly inspections
9 is still open for IP2 and IP3 (Commitment 41 Closure
10 Verification Form (NYS000535), but that Commitment 42 relating
11 to tube-to-tubesheet welds is open for IP3 only (NYS000553).

12 However, it is unclear what impact [REDACTED] will have
13 on these remaining open commitments. As I noted earlier, the

14 [REDACTED]

15 [REDACTED]

16 [REDACTED]

17 [REDACTED]

18 [REDACTED] Although Entergy disclosed [REDACTED] it is not
19 clear what use, if any, Entergy has made, or will make, of the
20 document. In my opinion, Entergy should not -- and cannot --
21 rely on the [REDACTED] to avoid inspecting the channel head and
22 divider plate assemblies, including the tube-to-tubesheet welds

1 in the eight Indian Point steam generators. To the extent that
2 Entergy remains committed to performing inspections after
3 license renewal, potentially well into the plants' periods of
4 extended operation, that is an inadequate assurance for managing
5 aging steam generators at Indian Point. Rather, Entergy should
6 affirmatively and clearly commit to performing inspections as
7 soon as possible for IP2, and certainly before the period of
8 extended operation for IP3. Additionally, Entergy must identify
9 the inspection techniques it intends to use, develop acceptance
10 criteria, and provide a detailed plan for addressing any flaws
11 or indications that it may encounter. I also recommend that
12 Entergy conduct follow-up inspections at least every 10 years,
13 given the primarily Alloy 600TT construction of IP2 steam
14 generator components and assemblies and the age of the IP3 steam
15 generators.

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

22 Q. Does this conclude your supplemental testimony?

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

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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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