

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 14, 2012

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PRE-FILED WRITTEN 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 testimony by Dr. David J. Duquette regarding Contention NYS-38/RK-TC-5.

Q. Please state your name and address.

A. David J. Duquette, Materials Engineering Consulting Services, 4 North Lane, Loudonville, New York 12211.

**Experience**

Q. What is your educational background?

A. My educational and professional experience are

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1 detailed in the attached curriculum vitae (CV)(Exhibit  
2 NYS000166) and report (Exhibit NYS000372); also attached is a  
3 list of my publications, awards, and other professional  
4 activities. I am a graduate of the United States Coast Guard  
5 Academy and the Massachusetts Institute of Technology (MIT). I  
6 performed my graduate work at the Corrosion Laboratory at the  
7 Massachusetts Institute of Technology, spent two years as a  
8 Research Associate at the Advanced Materials Research and  
9 Development Laboratory at Pratt and Whitney Aircraft prior to  
10 joining the faculty at Rensselaer Polytechnic Institute.

11 Q. What is your professional experience, particularly as  
12 it relates to corrosion prevention?

13 A. My research is primarily in the area of corrosion  
14 science and engineering. I have supervised more than 50  
15 graduate research dissertations in corrosion and related  
16 sciences. I am the author or co-author of more than 230  
17 publications and 20 book chapters. I present invited lectures  
18 internationally 20 to 25 times per year. Last year, I completed  
19 nine years of service on the United States Nuclear Waste  
20 Technical Review Board, having been appointed to the Board by  
21 President Bush in 2002. The Nuclear Waste Technical Review  
22 Board was created by Congressional legislation to provide  
23 scientific oversight and advice on spent nuclear fuel and high

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1 level nuclear waste and reports to the U.S. Congress and U.S.  
2 Secretary of Energy. I also maintain an active consulting  
3 practice, primarily in the area of corrosion and mechanical  
4 failures. A list of my publications is attached at Exhibit  
5 NYS000166.

6 Q. Can you cite specific examples of recognition by the  
7 scientific community?

8 A. I have been elected a Fellow of three learned  
9 societies, ASMI (formerly the American Society of Metals), NACE  
10 (formerly known as the National Association of Corrosion  
11 Engineers) and ECS (the Electrochemical Society). I have  
12 received the Whitney Award from NACE for outstanding corrosion  
13 research, an A.V. Humboldt Senior Scientist Award from the  
14 German government, as well as other awards from the scientific  
15 community.

16 Q. Do you have experience with respect to nuclear power  
17 plants or systems?

18 A. Yes.

19 Q. Please describe that experience.

20 A. I have served on Electric Power Research Institute  
21 (EPRI) panels for corrosion control in nuclear power systems,  
22 and was funded by EPRI for 5 years and by the Department of  
23 Energy for 11 years for corrosion research in nuclear systems.

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1 I have supervised Ph.D. students performing research on nuclear  
2 systems for U.S. Navy applications at the Knolls Atomic Power  
3 Laboratory located in upstate New York. As part of my work, I  
4 have also had personal tours of numerous reactors and related  
5 waste facilities because of my service on the Nuclear Waste  
6 Technical Review Board. These reactors included Dresden,  
7 Savannah River, Hanford, several French plants, as well as  
8 plants in England, Germany, Spain, and Argentina. In each of  
9 those tours high level aspects of technical management of the  
10 facilities, including aging and maintenance of the  
11 infrastructures were discussed in detail.

12 I have experience with materials degradation and corrosion  
13 issues in nuclear plants including consultation for Three Mile  
14 Island Unit 1, the closure of Three Mile Island Unit 2, Diablo  
15 Canyon Unit 1 and Unit 2 (MIC corrosion of stainless steel  
16 piping), Seabrook, and the plants formerly operated by  
17 Commonwealth Edison at Braidwood, Byron, Clinton, Dresden,  
18 LaSalle, Quad Cities, and Zion.

19 Q. Do you have experience with respect to steam  
20 generators at nuclear power plants?

21 A. Yes.

22 Q. Please describe that experience.

23 A. I have examined the issue of corrosion and materials

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1 degradation of steam generators at various U.S. nuclear plants  
2 including Three Mile Island Unit 1 (stress corrosion cracking of  
3 steam generator), Seabrook (corrosion of steam generator), and  
4 the Commonwealth Edison plants including Braidwood, Byron,  
5 Clinton, Dresden, LaSalle, Quad Cities, and Zion Unit 1 and Unit  
6 2 (stress corrosion cracking in steam generators). Those  
7 facilities include both pressurized water reactor and boiling  
8 water reactor designs and utilize steam generators manufactured  
9 by Westinghouse, General Electric, and Babcock & Wilcox.

#### 10 **Overview**

11 Q. What is the purpose of your testimony?

12 A. The purpose of my testimony is to provide support for,  
13 and my views on, an aspect of New York's Contention 38 ("NYS-  
14 38"), which was admitted for litigation by the Atomic Safety  
15 Licensing Board ("ASLB"). Contention NYS-38 asserts, among  
16 other things, that Entergy has not demonstrated that it has a  
17 program that will manage the effects of aging of critical  
18 components or systems at the Indian Point nuclear power  
19 facilities and that therefore the NRC does not have a record and  
20 a rational basis upon which it can determine whether to grant  
21 Entergy a renewed license for the Indian Point facilities. My  
22 testimony critiques Entergy's proposed approach towards the age  
23 related degradation of various components of Indian Point's

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1 steam generators during the requested twenty year period of  
2 extended operation.

3 Q. I show you what has been marked as Exhibit NYS000372.  
4 Do you recognize that document?

5 A. Yes. It is a copy of the report that I prepared for  
6 the State of New York in this proceeding concerning Contention  
7 NYS-38/RK-TC-5. The report reflects my review of various  
8 documents and my analysis and opinions.

9 Q. What, in general terms, does this report consist of?

10 A. This report contains a discussion of my experience, a  
11 description of Indian Point's nuclear steam supply systems  
12 including the steam generators, a review of stress corrosion  
13 cracking and its interaction with certain alloys and welds in  
14 nuclear power plants, stress corrosion cracking concerns for  
15 nuclear power plant steam generator divider plate assemblies and  
16 tube-to-tubesheet welds, Entergy's proposed approach to these  
17 concerns, and my opinions and conclusions concerning that  
18 approach.

19 Q. Have you reviewed materials in preparation for your  
20 testimony?

21 A. Yes.

22 Q. What is the source of those materials?

23 A. Many are documents prepared by government agencies,

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1 peer reviewed articles, or documents prepared by Entergy or by  
2 the nuclear power industry.

3 Q. What materials have you reviewed in preparation for  
4 your testimony?

5 A. Among the materials I have reviewed are portions of  
6 Entergy's License Renewal Application for Indian Point Unit 2  
7 and Unit 3 related to the aging management review and aging  
8 management programs for steam generators; communications between  
9 Entergy and NRC Staff concerning steam generators; the 2011  
10 Supplemental Safety Evaluation Report (SSER) for the renewal of  
11 the Indian Point operating licenses prepared by NRC Staff; a  
12 document known as the Generic Aging Lessons Learned Report  
13 (GALL), Final Report (including Revision 1 and Revision 2), a  
14 document known as the Standard Review Plan (both Revision 1 and  
15 Revision 2); numerous industry documents including Electric  
16 Power Research Institute (EPRI), Westinghouse, Nuclear Energy  
17 Institute (NEI) documents; scientific and engineering  
18 literature, and NRC documents; as well as disclosures in this  
19 proceeding related to steam generators.

20 Q. I show you NYS Exhibits NYS00146A-NYS146C [GALL Rev 1]  
21 NYS00147A-NYS00147D [GALL Rev 2], NYS000151 [NL-11-032],  
22 NYS000152 [NL-11-074], NYS000153 [NL-11-090] NYS000154 [NL-11-  
23 096], NYS000160 [SSER], NYS000161 [SRP Rev 2], NYS000195 [SRP

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1 Rev 1], NYS000199 [Feb. RAI], and NYS000375 through NYS000394.

2 Do you recognize these documents?

3 A. Yes. These are true and accurate copies of the  
4 documents that I referred to, used and/or relied upon in  
5 preparing my report and this testimony. In some cases, where  
6 the document was extremely long and only a small portion is  
7 relevant to my testimony, an excerpt of the document is  
8 provided. If it is only an excerpt, that is noted on the first  
9 page of the Exhibit or its description.

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

13 A. These documents represent the type of information that  
14 persons within my field of expertise reasonably rely upon in  
15 forming opinions of the type offered in this testimony.

16 Q. Did you review anything else in preparing your report  
17 or this testimony?

18 A. Yes, I reviewed other documents Entergy produced in  
19 this proceeding as of early June and concluded that they were  
20 not relevant in preparing my report and this testimony.

21 **Conclusions and Opinions**

22 Q. What conclusions, if any, have you reached?

23 A. In my professional judgment, and as I describe in more

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1 detail below, and in my report, based on a review of documents  
2 provided by Entergy and NRC Staff as of early June as well as  
3 industry and engineering literature, a serious concern exists  
4 about potential cracking in the divider plate assemblies and  
5 tube-to-tubesheet welds of the Westinghouse steam generators at  
6 Indian Point Unit 2 and Unit 3. Recent experience in similar  
7 steam generators in Europe has discovered primary water stress  
8 corrosion cracking (PWSCC) in Alloy 600 divider plates and in  
9 the Alloy 82/182 welds connecting the divider plates to the  
10 tubesheets. If cracks in the divider plates or in the divider  
11 plate welds propagate into the Alloy 600 cladding of the  
12 tubesheets it is likely that they will propagate into the tube-  
13 to-tubesheet welds and accordingly compromise the pressure  
14 boundary, resulting in contamination of the secondary water with  
15 primary water.

16       At the present time there is no qualified inspection  
17 procedure to determine the extent of cracking in the divider  
18 plates or associated channel head assemblies or the propagation  
19 of cracking from the tubesheet cladding to the tube-to-tubesheet  
20 weld. European inspection procedures result in high radiation  
21 doses for plant workers/inspectors. According to an August 4,  
22 2011 NEI Steam Generator Task Force presentation to NRC, the  
23 stresses that may initiate PWSCC or to lead to the propagation

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1 of either PWSCC or fatigue cracks from the divider plate  
2 assemblies into the tube-to-tubesheet welds have not even been  
3 determined.

4 EPRI has recently begun a program to determine the  
5 susceptibility of divider plates and related structures and  
6 assemblies to PWSCC, but the results of that research are not  
7 scheduled to be available until 2016, well into the periods of  
8 extended operation for Indian Point Unit 2 and Unit 3.

9 Entergy's proposed plan for steam generator divider plate  
10 assemblies, tubesheets, and welds contains several unknowns. At  
11 present, neither Indian Point nor NRC, EPRI, or the industry  
12 have demonstrated that the age related degradation of divider  
13 plate assemblies, tubesheets, and welds can be adequately  
14 managed.

15 Until the magnitude of the problem is assessed and a  
16 qualified inspection program is developed, the Entergy Aging  
17 Management Program at Indian Point cannot be considered adequate  
18 to assure the safety of the site to workers at the facility and  
19 to the general public.

## 20 **Stress Corrosion Cracking**

21 Q. What is stress corrosion cracking?

22 A. Stress corrosion cracking (or SCC) is a well-  
23 documented phenomenon for many alloy/environmental combinations.

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1 It is a particularly insidious phenomenon since it occurs in  
2 otherwise ductile alloys, but only in very specific  
3 environments. Occurrence of the phenomenon requires the  
4 simultaneous presence of stress, whether residual or applied,  
5 and a specific alloy /environment combination. The phenomenon  
6 is generally unpredictable for new combinations of alloys and  
7 environments and is often only identified through experience.  
8 It was originally called pure water stress corrosion cracking  
9 and was later relabeled as primary water stress corrosion  
10 cracking (or PWSCC).

11 Q. Can you briefly describe the experience of stress  
12 corrosion cracking in the nuclear energy production area?

13 A. Yes. Cracking of Alloy 600 steam generator tubes was  
14 originally observed in the vicinity of the tubesheets and tube  
15 support plates in steam generators because of the expansive  
16 characteristics of the corrosion products of the carbon steel  
17 tubesheets and support plates in the crevices between the  
18 support plates and the tubesheets and the rolled-in tubes. The  
19 expansion of the corrosion products imparted large stresses on  
20 the mill annealed Alloy 600 tubes resulting in plastic  
21 deformation of the tubes (denting). Cracking in the deformed  
22 tubes in the tubesheet region was brought under some measure of  
23 control by judicious water treatment campaigns. However,

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1 cracking in the U-bends of Alloy 600 tubes has also been  
2 observed at nuclear power plants, including a documented rupture  
3 of a steam generator tube at Indian Point 2 on February 15,  
4 2000.

5       Since the first observations of cracked Alloy 600  
6 components in nuclear reactors, and to the present day numerous  
7 attempts at quantifying the specific mechanisms of the  
8 susceptibility of Alloy 600 to primary water stress corrosion  
9 cracking were attempted but only with limited success. It is  
10 clear that metallurgical, environmental, and loading variables  
11 all contribute to the susceptibility of Alloy 600 to primary  
12 water stress corrosion cracking.

13       In 1985, the NRC issued a generic letter to PWR licensees  
14 and potential licensees recommending actions for the resolution  
15 of unresolved safety issues regarding steam generator tube  
16 integrity. Some success has been achieved with specific thermal  
17 treatments of the alloy, and the introduction of improved water  
18 chemistries. In many cases where steam generator tubes were  
19 made of Alloy 600, reactor owners replaced those steam  
20 generators with steam generators with tubes that were made from  
21 a more PWSCC-resistant alloy designated Alloy 690.

22       However, there are many other components in an operating  
23 nuclear plant and steam supply system that still contain Alloy

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1 600. For example, PWSCC concerns exist for steam generator  
2 tubes, steam generator divider plates, heater thermal sleeves  
3 and penetrations in the pressurizer, penetrations for the  
4 control rod drive mechanisms in reactor pressure vessel heads,  
5 and other components of the reactors that are fabricated from  
6 Alloy 600. It should also be noted that Alloy 600 components  
7 are generally welded with Alloys 82 or 182, derivatives of Alloy  
8 600 that have also been found to be susceptible to primary water  
9 stress corrosion cracking (PWSCC).

10 **Indian Point Power Generation Systems & Steam Generators**

11 Q. Can you briefly describe the design of the Indian  
12 Point power generation system?

13 A. According to Entergy's License Renewal Application  
14 Indian Point Unit 2 and Unit 3 each employ a pressurized water  
15 reactor (PWR) design and a four loop nuclear steam supply system  
16 (NSSS) furnished by Westinghouse Electric Corporation. The  
17 reactor coolant system consists of four similar transfer loops  
18 connected in parallel to the reactor vessel. Each loop contains  
19 a reactor coolant pump and a steam generator. The system also  
20 includes a pressurizer, a pressurized relief tank, connecting  
21 piping, and instrumentation necessary for operational control.  
22 The reactor coolant system transfers the heat generated in the  
23 core of the reactor vessel to the steam generators, where steam

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1 is produced to drive the turbine electric power generators.

2 Q. I show you Exhibit NYS000375. Do you recognize it?

3 A. Yes, it is a schematic drawing of a Westinghouse  
4 Pressurized Water Reactor Nuclear Steam Supply System that  
5 identifies the various components and the reactor coolant  
6 pressure boundary.

7 Q. Would you please describe the role of the steam  
8 generators in the Indian Point nuclear steam supply systems?

9 A. Each reactor coolant loop contains a vertical shell  
10 and U-tube steam generator. Reactor coolant enters the inlet  
11 side of the channel head at the bottom of the steam generator  
12 through the inlet nozzle, is forced upward through the  
13 tubesheet, flows through the U-tubes, returns through the  
14 tubesheet to an outlet channel and leaves the generator through  
15 a bottom nozzle. The inlet and outlet channels in the steam  
16 generator are separated by a partition or divider plate. The  
17 divider plate is joined to the channel head and the tubesheet  
18 through a stub runner.

19 Q. I show you Exhibit NYS000376. Do you recognize it?

20 A. Yes, it is a diagram of a Westinghouse steam generator  
21 that identifies the various components within a generator.

22 Q. I show you Exhibits NYS000377 and NYS000378. Would  
23 you describe them?

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1           A.    These are two NRC documents, one entitled History of  
2 "Westinghouse Model 44 Steam Generators," and the second  
3 entitled "Steam Generator Tube Operational Experience." They  
4 describe the use of the Westinghouse Model 44 Steam Generator,  
5 that model's use of Alloy 600 material, incidents of tube  
6 ruptures of steam generators using Alloy 600, and the  
7 replacement of Model 44 steam generators.

8           Q.    Are you aware of the type of steam generator that was  
9 initially used at Indian Point Unit 2 and Unit 3 when they began  
10 operation?

11          A.    Yes. According to Entergy and NRC documents, Indian  
12 Point Unit 2 and Unit 3 were constructed with Westinghouse Model  
13 44 steam generators.

14          Q.    Did there come a time when Indian Point facilities  
15 changed the steam generators?

16          A.    Yes. According to Entergy and NRC documents, in 1989,  
17 thirteen years after it began operations, Indian Point Unit 3  
18 replaced its Westinghouse Model 44 steam generators with  
19 Westinghouse Model 44F steam generators that use Alloy 690 for  
20 its tubes.

21          Indian Point Unit 2 used Westinghouse Model 44 steam  
22 generators from 1973 to 2000. I understand that Indian Point  
23 Unit 2 received four additional Model 44 steam generators from

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1 Westinghouse in the 1980s, but that it did not install them at  
2 that time. In February 2000, a tube ruptured on steam generator  
3 Number 24, and the plant shut down and remained offline for 11  
4 months. During that outage, Indian Point Unit 2 replaced its  
5 original Westinghouse steam generators with the ones it received  
6 from Westinghouse in 1980s.

7 Q. Has Entergy disclosed the material used in the current  
8 Indian Point Unit 2 steam generators?

9 A. Yes, Entergy has stated that the current Indian Point  
10 Unit 2 steam generators use Alloy 600 for the tubes and for the  
11 divider plates. It also stated that it assumed that the weld  
12 material for the divider plate assemblies was Alloy 82/182 weld  
13 material.

14 Q. Has Entergy disclosed the material used in the current  
15 Indian Point Unit 3 steam generators?

16 A. Yes, Entergy has stated that the Indian Point Unit 3  
17 steam generators use Alloy 690 for the tubes and Alloy 600 for  
18 the divider plates. It also stated that it assumed that the  
19 weld material for the divider plate assemblies was Alloy 82/182  
20 weld material.

21 Q. What types of steam generators parts or locations are  
22 affected by primary water stress corrosion cracking?

23 A. In addition to the heat transfer tubes, which we have

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1 already discussed, primary water stress corrosion cracking could  
2 also affect other components or assemblies that use Alloy 600 or  
3 welds that use Alloy 82/182 weld material that, as I noted, are  
4 derivatives of Alloy 600. In the August 2011 Supplemental  
5 Safety Evaluation Report at page 3-21, the NRC Staff has also  
6 expressed concern about the propagation of primary water stress  
7 corrosion cracking in tubesheets that have Alloy 600 cladding or  
8 related weld even when the heat transfer tubes are made from  
9 Alloy 690TT material. According to Staff, "a crack initiated in  
10 this region, close to the tube, may propagate into or through  
11 the weld, causing a failure of the weld and of the reactor  
12 coolant pressure boundary." These areas of concern would  
13 include the channel head to tubesheet to tube complex, including  
14 the divider plate assembly and the tube-to-tubesheet welds.

15 **Reactor Coolant Pressure Boundary**

16 Q. Are you familiar with the term reactor coolant  
17 pressure boundary?

18 A. Yes, the NRC has a definition of this term in its  
19 regulations at 10 C.F.R. § 50.2. That regulation provides:

20 "Reactor coolant pressure boundary means all those  
21 pressure-containing components of boiling and  
22 pressurized water-cooled nuclear power reactors, such

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1 as pressure vessels, piping, pumps, and valves, which  
2 are:

3 (1) Part of the reactor coolant system,

4 or

5 (2) Connected to the reactor coolant system, up to and  
6 including any and all of the following:

7 (i) The outermost containment isolation valve in  
8 system piping which penetrates primary reactor  
9 containment,

10 (ii) The second of two valves normally closed during  
11 normal reactor operation in system piping which does  
12 not penetrate primary reactor containment,

13 (iii) The reactor coolant system safety and relief  
14 valves."

15 Stated differently, the reactor coolant pressure boundary refers  
16 to a physical barrier or boundary between the reactor coolant  
17 system on the "primary loop" of nuclear steam supply system and  
18 the "secondary loop" of the nuclear steam supply system. You  
19 can see this boundary line in the Westinghouse NSSS diagram  
20 (Exhibit NYS000375) that represents the primary loop in red or  
21 yellow and the secondary loop in green or blue. It is critical  
22 not to breach the reactor coolant pressure boundary and allow  
23 reactor coolant to escape.

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1 Q. Did Entergy's License Renewal Application discuss the  
2 function of the steam generators' components?

3 A. Yes, in the License Renewal Application Tables 2.3.1-  
4 4-IP2/IP3 of the LRA Entergy states that the channel head, the  
5 divider plate, tubes, and the tubesheet each constitutes a  
6 pressure boundary for Indian Point Unit 2 and Indian Point Unit  
7 3. They indicated that the tubes also perform a heat transfer  
8 function. Those tables are located in the License Renewal  
9 Application at pages 2.3-36, 2.3-39, respectively.

10 **Current Concerns About Primary Water Stress Corrosion Cracking**

11 Q. I show you Exhibit NYS000199; do you recognize it?

12 A. Yes, this is a set of questions prepared by NRC Staff  
13 and sent to Entergy in February 2011.

14 Q. Directing your attention to page 9, would you please  
15 read aloud the last full paragraph?

16 A. Yes.

17 "In some foreign steam generators with a similar  
18 design to that of Indian Point Units 2 and 3 steam  
19 generators, extensive cracking due to PWSCC has been  
20 identified in SG divider plate assemblies made with  
21 Alloy 600, even with proper primary water chemistry.  
22 Specifically, cracks have been detected in the stub  
23 runner, very close to the tubesheet/stub runner weld

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1 with depths of almost a third of the divider plate  
2 thickness. Therefore, the staff noted that the Water  
3 Chemistry Control - Primary and Secondary Program may  
4 not be effective in managing the aging effect of  
5 cracking due to PWSCC in SG divider plate assemblies."

6 Q. I show you Exhibit NYS000160. Do you recognize  
7 it?

8 A. Yes, it is a copy of the NRC Staff's Supplemental  
9 Safety Evaluation Report that they issued at the end of August  
10 2011. Among other things, at pages 3-18 to 3-19 and 3-20 to 3-  
11 23, it discusses Entergy's revised proposal concerning the  
12 Westinghouse steam generator divider plates assemblies and the  
13 tube-to-tubesheet welds.

14 Q. Directing your attention to the third paragraph on  
15 page 3-18, would you please read that aloud?

16 A. Yes.

17 "The staff noted that, although these SG divider plate  
18 assembly cracks might not have a significant safety  
19 impact in and of themselves, these cracks could affect  
20 adjacent items that are part of the reactor coolant  
21 pressure boundary, such as the tubesheet and the  
22 channel head, if they propagate to the boundary with  
23 these items. The staff further noted that for the

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1 tubesheet, PWSCC cracks in the divider plate  
2 assemblies fabricated from Alloy 600 and its  
3 associated weld metals could propagate to the  
4 tubesheet cladding, with possible consequences to the  
5 integrity of the tube-to-tubesheet welds.  
6 Furthermore, for the channel head, the PWSCC cracks in  
7 the divider plate assemblies could propagate to the SG  
8 triple point (i.e. the point where the divider plate  
9 and tube sheet meet with the shell) and potentially  
10 affect the pressure boundary of the SG channel head."

11 Q. Directing your attention to the second, third, and  
12 fourth sentences in the first full paragraph on page 3-21, would  
13 you please read that aloud?

14 A. Yes.

15 "The staff's concern is that, if the tubesheet  
16 cladding is Alloy 600 or the associated Alloy 600 weld  
17 materials, the region of the autogenous tube-to-  
18 tubesheet weld may have insufficient chromium content  
19 to prevent initiation of PWSCC, even when the SG tubes  
20 are made from Alloy 690TT. Consequently, a crack  
21 initiated in this region, close to a tube, may  
22 propagate into or through the weld, causing a failure  
23 of the weld and of the reactor coolant pressure

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1 boundary (RCPB). This could occur in once-through  
2 SGs, as well as in recirculating SGs such as those  
3 used at both of the applicant's units."

4 Q. Are these NRC statements consistent with your  
5 understanding of the recent experience with stress corrosion  
6 cracking?

7 A. Yes. As discussed in more detail in my accompanying  
8 report, recent EPRI reports and other documents have begun to  
9 report incidents of primary water stress corrosion cracking in  
10 steam generators with a similar design to the Indian Point Unit  
11 2 and Unit 3 steam generators.

12 EPRI and Westinghouse have cited reports of cracking in the  
13 divider plate assemblies in French steam generators (Saint  
14 Laurent, Gravelines, Chinon) and in a Swedish steam generator  
15 (Ringhals) that have similar design and construction details to  
16 U.S. reactors. The cracking has been observed in the divider  
17 plate itself, in the full penetration welds connecting the stub  
18 runner to the tubesheet and connecting the stub runner to the  
19 divider plate. In the French steam generators, the cracks are  
20 reported to have occurred in the heat affected zone of the stub  
21 runner to divider plate weld and have been observed to run  
22 nearly the length of the divider plate (~6 feet). Perhaps of  
23 more concern, as the cracks approach the triple point of the

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1 tubesheet-channel head complex, the cracks tend to curve  
2 upwards. It has been suggested that this PWSCC could compromise  
3 the pressure boundary of the steam generator by propagating  
4 through the channel head via corrosion fatigue after the PWSCC  
5 crack has initiated. Cracks that form in the divider plate, the  
6 stub runner, and/or the associated welds may propagate into the  
7 tubesheet, allowing mixing of the primary water with the  
8 secondary water and accordingly compromising the integrity of  
9 the reactor coolant pressure boundary. Given the crack path,  
10 another possibility is propagation of PWSCC into the tubesheet  
11 cladding that would then propagate into the tube to tubesheet  
12 weld and subsequently into the Alloy 600 tubes. This phenomenon  
13 is of particular concern for the IP2 replacement steam  
14 generators that were constructed in the 1980s with Alloy 600  
15 tubes. Moreover, the steam generators at both IP2 and IP3 have  
16 Alloy 600 divider plates and Alloy 82/182 welds.

17 **Entergy's Proposed Approach**

18 Q. Following NRC Staff's question in 2011 what, if  
19 anything, did Entergy propose to do concerning the reports about  
20 primary water stress corrosion cracking?

21 A. Among other things, in its March 28, 2011 submission  
22 (Exhibit NYS000151), Entergy told NRC that:

23 "The industry plans are to study the potential for

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1 divider plate crack growth and develop a resolution to  
2 the concern through the EPRI Steam Generator  
3 Management Program Engineering and Regulatory  
4 Technical Advisory Group. This industry-lead effort  
5 is expected to begin in 2011 and be completed within  
6 two years."

7 Acknowledging that the EPRI investigation of the issue is under  
8 development and not yet completed, Entergy also stated that it  
9 would "inspect all Indian Point steam generators to assess the  
10 condition of the divider plate assembly. The examination  
11 technique used will be capable of detecting PWSCC in the steam  
12 generator divider plate assembly welds."

13 Q. Did Entergy provide any further information about the  
14 inspections it proposed to perform on the steam generator  
15 divider plate assemblies?

16 A. No, it did not.

17 Q. Did Entergy make any proposals with respect to Staff  
18 concern about the propagation of primary water stress corrosion  
19 cracking in a steam generator that contains Alloy 690TT heat  
20 transfer tubes?

21 A. Yes. In 2011, Entergy proposed two options as  
22 follows:

23 Option 1 (Analysis)

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1 "IPEC will perform an analytical evaluation of the  
2 steam generator tube-to-tubesheet welds in order to  
3 establish a technical basis for either determining  
4 that the tubesheet cladding and welds are not  
5 susceptible to PWSCC, or redefining the pressure  
6 boundary in which the tube-to-tubesheet weld is no  
7 longer included and, therefore, is not required for  
8 reactor coolant pressure boundary function. The  
9 redefinition of reactor coolant pressure boundary must  
10 be approved by the NRC as part of a license amendment  
11 request."

12 Option 2 (Inspection)

13 "IPEC will perform a one-time inspection of a  
14 representative number of tube-to-tubesheet welds in  
15 each steam generator to determine if PWSCC cracking is  
16 present. If weld cracking is identified:

- 17 a. The condition will be resolved through repair or  
18 engineering evaluation to justify continued service,  
19 as appropriate, and  
20 b. An ongoing monitoring program will be established  
21 to perform routine tube-to-tubesheet weld inspections  
22 for the remaining life of the generators.

23 Q. Have you seen any information about the status of the

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1 EPRI investigation into the issue of primary water stress  
2 corrosion cracking in divider plate assemblies?

3 A. Yes, I have read recent EPRI documents that report  
4 that the investigation and development of a proposed management  
5 program may have not be completed until at least 2016 at which  
6 point both Indian Point Unit 2 and Indian Point Unit 3 will be  
7 beyond their initial 40 year operating license term.

8 Q. I show you Exhibits NYS000393 and NYS000394? Do you  
9 recognize them?

10 A. Yes, these documents are two EPRI documents. The  
11 first is entitled "Nuclear Sector Roadmaps" (January 2012) and  
12 contains sections entitled "In Use: Aging Management of Alloy  
13 600 and Alloy 82/182 in the Steam Generator Channel Head  
14 Assembly" and "Materials Aging And Degradations, Action Plan  
15 Roadmap Summary." The second is entitled "EPRI, 2012 Research  
16 Portfolio, Steam Generator Management." These documents discuss  
17 the EPRI investigation and its timeline.

18 Q. Directing your attention to Exhibit NYS000393 would  
19 you summarize the section entitled "In Use: Aging Management of  
20 Alloy 600 and Alloy 82/182 in the Steam Generator Channel Head  
21 Assembly"?

22 A. In this section EPRI acknowledges that primary water  
23 stress corrosion cracking (PWSCC) that initiates in Alloy 600

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1 and associated weld materials in the steam generator could  
2 propagate to pressure boundaries such as the tube-to-tubesheet  
3 weld or the carbon steel materials in the bowl. The document  
4 further acknowledges that the industry lacks understanding of  
5 the impact of cracks that may compromise safe operations as the  
6 steam generators age, and proposes a research program to address  
7 this issue in AMR's. EPRI also admits that there are "no  
8 qualified techniques to inspect the steam generator channel  
9 head", and that the inspection methods currently used in Europe  
10 to inspect the steam generator divider plates result in  
11 significant doses to workers.

12 Q. Do you believe an aging management program is  
13 necessary to manage the primary water stress corrosion cracking  
14 aging degradation of steam generators at Indian Point?

15 A. Yes. It is important to develop an Aging Management  
16 Program with substantive, meaningful, and enforceable standards.  
17 The fact that Indian Point in the past has experienced some  
18 primary water stress corrosion issues with Alloy 600 material in  
19 its steam generators, detailed in my report, indicates to me  
20 that there are already corrosion risks at the facility and that  
21 appropriate measures must be taken to prevent steam generator  
22 components from failing in the future.

23 Q. Has Entergy agreed that it needs an Aging Management

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1 Program at Indian Point to address primary water stress  
2 corrosion cracking of divider plate assemblies and its  
3 propagation to pressure boundaries such as the tube-to-tubesheet  
4 weld or the carbon steel materials in the bowl?

5 A. Ostensibly it has, but from a practical and  
6 engineering perspective it has not, nor has it addressed safety  
7 considerations. Entergy has not proposed a specific inspection  
8 procedure for Indian Point except to say that it will be guided  
9 by industry standards. Industry standards have not yet been  
10 established. Entergy's proposed plan for steam generator  
11 divider plate assemblies, tubesheets, and welds contains several  
12 unknowns. It provides no detail about the inspection methods or  
13 technique (visual, volumetric, or surface inspection),  
14 acceptance criteria, monitoring and trending protocols, or  
15 corrective action responses that it might employ. The absence  
16 of such details prevents meaningful evaluation of the proposed  
17 approach. At present, Indian Point (and NRC) have not  
18 demonstrated that the age related degradation of divider plate  
19 assemblies, tubesheets, and welds resulting from primary water  
20 stress corrosion cracking can be adequately managed.

21 Q. What, generally, is your conclusion about the adequacy  
22 of Entergy's proposal for steam generators primary water stress  
23 corrosion cracking at Indian Point?

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1           A.     While Entergy advanced a proposal it is not an Aging  
2 Management Program. There is nothing in the proposal at all to  
3 determine what Entergy is committing to do. It is wholly  
4 deficient. It seems more like a "wait and see" placeholder  
5 proposal while EPRI works on the question.

6           Q.     Are you aware that NRC Staff, in its August 2011  
7 Supplemental Safety Evaluation Report, found that Entergy's  
8 proposed approach "acceptable" and concluded that Entergy has  
9 demonstrated that the effects of primary water stress corrosion  
10 cracking in the divider plate assemblies in the steam generators  
11 will be adequately managed so that their intended functions  
12 would be maintained during the requested period of extended  
13 operation?

14          A.     Yes, I am aware that it what NRC Staff stated, and for  
15 the reasons stated in my testimony and my report, I do not and  
16 cannot agree with that conclusion.

17          Q.     Have you now completed your initial testimony  
18 regarding contention NYS-38/RK-TC-5?

19          A.     Yes. However, I reserve the right to express further  
20 opinions if new evidence is introduced or disclosed.

1 UNITED STATES

2 NUCLEAR REGULATORY COMMISSION

3 BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

4 -----X

5 In re: Docket Nos. 50-247-LR; 50-286-LR  
6 License Renewal Application Submitted by ASLBP No. 07-858-03-LR-BD01  
7 Entergy Nuclear Indian Point 2, LLC, DPR-26, DPR-64  
8 Entergy Nuclear Indian Point 3, LLC, and  
9 Entergy Nuclear Operations, Inc. June 14, 2012

10 -----X

11 DECLARATION OF DAVID J. DUQUETTE

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

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



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