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	PRE-FILED WRITTEN TESTIMONY OF
12	DR. DAVID J. DUQUETTE
13	REGARDING CONTENTION NYS-38/RK-TC-5
14	On behalf of the State of New York ("NYS" or "the State"),
15	the Office of the Attorney General hereby submits the following
16	testimony by Dr. David J. Duquette regarding Contention NYS-
17	38/RK-TC-5.
18	Q. Please state your name and address.
19	A. David J. Duquette, Materials Engineering Consulting
20	Services, 4 North Lane, Loudonville, New York 12211.
21	Experience
22	Q. What is your educational background?
23	A. My educational and professional experience are
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detailed in the attached curriculum vitae (CV)(Exhibit 1 2 NYS000166) and report (Exhibit NYS000372); also attached is a list of my publications, awards, and other professional 3 4 activities. I am a graduate of the United States Coast Guard 5 Academy and the Massachusetts Institute of Technology (MIT). I performed my graduate work at the Corrosion Laboratory at the б 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 My research is primarily in the area of corrosion Α. 14 science and engineering. I have supervised more than 50 graduate research dissertations in corrosion and related 15 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 President Bush in 2002. The Nuclear Waste Technical Review 21 22 Board was created by Congressional legislation to provide 23 scientific oversight and advice on spent nuclear fuel and high Pre-filed Written Testimony of David J. Duquette

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

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

8 I have been elected a Fellow of three learned Α. 9 societies, ASMI (formerly the American Society of Metals), NACE 10 (formerly known as the National Association of Corrosion Engineers) and ECS (the Electrochemical Society). 11 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.

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Q. Please describe that experience.

 A. I have served on Electric Power Research Institute
 (EPRI) panels for corrosion control in nuclear power systems,
 and was funded by EPRI for 5 years and by the Department of
 Energy for 11 years for corrosion research in nuclear systems. *Pre-filed Written Testimony of David J. Duquette*

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I have supervised Ph.D. students performing research on nuclear 1 2 systems for U.S. Navy applications at the Knolls Atomic Power Laboratory located in upstate New York. As part of my work, I 3 have also had personal tours of numerous reactors and related 4 5 waste facilities because of my service on the Nuclear Waste Technical Review Board. These reactors included Dresden, б 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 infrastructures were discussed in detail. 11

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

Q. Do you have experience with respect to steamgenerators at nuclear power plants?

21 A. Yes.

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22 Q. Please describe that experience.

A. I have examined the issue of corrosion and materials Pre-filed Written Testimony of David J. Duquette Contention NYS-38/RK-TC-5

1 degradation of steam generators at various U.S. nuclear plants 2 including Three Mile Island Unit 1 (stress corrosion cracking of steam generator), Seabrook (corrosion of steam generator), and 3 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 water reactor designs and utilize steam generators manufactured 8 9 by Westinghouse, General Electric, and Babcock & Wilcox.

10 Overview

11

Q. What is the purpose of your testimony?

12 Α. 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 Licensing Board ("ASLB"). Contention NYS-38 asserts, among 15 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 facilities and that therefore the NRC does not have a record and 19 20 a rational basis upon which it can determine whether to grant Entergy a renewed license for the Indian Point facilities. 21 My 22 testimony critiques Entergy's proposed approach towards the age 23 related degradation of various components of Indian Point's Pre-filed Written Testimony of David J. Duquette

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

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

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

9 Ο. What, in general terms, does this report consist of? 10 Α. 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.

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Q. What is the source of those materials?

A. Many are documents prepared by government agencies, Pre-filed Written Testimony of David J. Duquette Contention NYS-38/RK-TC-5

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

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

5 Α. Among the materials I have reviewed are portions of Entergy's License Renewal Application for Indian Point Unit 2 б 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 Revision 2); numerous industry documents including Electric 15 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.

Q. I show you NYS Exhibits NYS00146A-NYS146C [GALL Rev 1] NYS00147A-NYS00147D [GALL Rev 2], NYS000151 [NL-11-032], NYS000152 [NL-11-074], NYS000153 [NL-11-090] NYS000154 [NL-11-096], NYS000160 [SSER], NYS000161 [SRP Rev 2], NYS000195 [SRP Pre-filed Written Testimony of David J. Duquette

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Rev 1], NYS000199 [Feb. RAI], and NYS000375 through NYS000394.
 Do you recognize these documents?

These are true and accurate copies of the 3 Α. Yes. 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.

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

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

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

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

21 Conclusions and Opinions

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

A. In my professional judgment, and as I describe in more Pre-filed Written Testimony of David J. Duquette Contention NYS-38/RK-TC-5

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 industry and engineering literature, a serious concern exists 3 4 about potential cracking in the divider plate assemblies and 5 tube-to-tubesheet welds of the Westinghouse steam generators at Indian Point Unit 2 and Unit 3. Recent experience in similar б 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 Pre-filed Written Testimony of David J. Duquette

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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.

EPRI has recently begun a program to determine the susceptibility of divider plates and related structures and assemblies to PWSCC, but the results of that research are not scheduled to be available until 2016, well into the periods of 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.

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

20 Stress Corrosion Cracking

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Q. What is stress corrosion cracking?
A. Stress corrosion cracking (or SCC) is a welldocumented phenomenon for many alloy/environmental combinations. Pre-filed Written
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1 It is a particularly insidious phenomenon since it occurs in 2 otherwise ductile alloys, but only in very specific environments. Occurrence of the phenomenon requires the 3 4 simultaneous presence of stress, whether residual or applied, 5 and a specific alloy /environment combination. The phenomenon is generally unpredictable for new combinations of alloys and б 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).

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

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13 Α. Cracking of Alloy 600 steam generator tubes was Yes. 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, Pre-filed Written Testimony of David J. Duquette

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

Since the first observations of cracked Alloy 600 5 components in nuclear reactors, and to the present day numerous б 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 Pre-filed Written Testimony of David J. Duquette Contention NYS-38/RK-TC-5

1 600. For example, PWSCC concerns exist for steam generator 2 tubes, steam generator divider plates, heater thermal sleeves and penetrations in the pressurizer, penetrations for the 3 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

Q. Can you briefly describe the design of the IndianPoint power generation system?

13 According to Entergy's License Renewal Application Α. 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 Pre-filed Written Testimony of David J. Duquette

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is produced to drive the turbine electric power generators.

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Q. I show you Exhibit NYS000375. Do you recognize it?
A. Yes, it is a schematic drawing of a Westinghouse
Pressurized Water Reactor Nuclear Steam Supply System that
identifies the various components and the reactor coolant
pressure boundary.

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

9 Α. 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 through the inlet nozzle, is forced upward through the 12 13 tubesheet, flows through the U-tubes, returns through the 14 tubesheet to an outlet channel and leaves the generator through a bottom nozzle. The inlet and outlet channels in the steam 15 16 generator are separated by a partition or divider plate. The divider plate is joined to the channel head and the tubesheet 17 through a stub runner. 18

Q. I show you Exhibit NYS000376. Do you recognize it?
A. Yes, it is a diagram of a Westinghouse steam generator
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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A. These are two NRC documents, one entitled History of "Westinghouse Model 44 Steam Generators," and the second entitled "Steam Generator Tube Operational Experience." They describe the use of the Westinghouse Model 44 Steam Generator, that model's use of Alloy 600 material, incidents of tube ruptures of steam generators using Alloy 600, and the 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?

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

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

A. Yes. According to Entergy and NRC documents, in 1989,
thirteen years after it began operations, Indian Point Unit 3
replaced its Westinghouse Model 44 steam generators with
Westinghouse Model 44F steam generators that use Alloy 690 for
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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Westinghouse in the 1980s, but that it did not install them at that time. In February 2000, a tube ruptured on steam generator Number 24, and the plant shut down and remained offline for 11 months. During that outage, Indian Point Unit 2 replaced its original Westinghouse steam generators with the ones it received from Westinghouse in 1980s.

Q. Has Entergy disclosed the material used in the current8 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?

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

Q. What types of steam generators parts or locations areaffected by primary water stress corrosion cracking?

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A. In addition to the heat transfer tubes, which we have Pre-filed Written Testimony of David J. Duquette Contention NYS-38/RK-TC-5

1 already discussed, primary water stress corrosion cracking could 2 also affect other components or assemblies that use Alloy 600 or welds that use Alloy 82/182 weld material that, as I noted, are 3 derivatives of Alloy 600. In the August 2011 Supplemental 4 5 Safety Evaluation Report at page 3-21, the NRC Staff has also expressed concern about the propagation of primary water stress б 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 Are you familiar with the term reactor coolant Ο. 17 pressure boundary? Yes, the NRC has a definition of this term in its 18 Α. regulations at 10 C.F.R. § 50.2. That regulation provides: 19 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.
	Pre-filed Written

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Q. Did Entergy's License Renewal Application discuss the
 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 Therefore, the staff noted that the Water thickness. Chemistry Control - Primary and Secondary Program may 3 4 not be effective in managing the aging effect of 5 cracking due to PWSCC in SG divider plate assemblies." I show you Exhibit NYS000160. Do you recognize б Ο. 7 it? 8 Yes, it is a copy of the NRC Staff's Supplemental Α. 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 3-23, it discusses Entergy's revised proposal concerning the 11 12 Westinghouse steam generator divider plates assemblies and the 13 tube-to-tubesheet welds. 14 Directing your attention to the third paragraph on Ο. 15 page 3-18, would you please read that aloud? 16 Α. Yes. 17 "The staff noted that, although these SG divider plate 18 assembly cracks might not have a significant safety impact in and of themselves, these cracks could affect 19 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 The staff further noted that for the 23 these items. Pre-filed Written Testimony of David J. Duquette Contention NYS-38/RK-TC-5

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

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

A. Yes. As discussed in more detail in my accompanying
report, recent EPRI reports and other documents have begun to
report incidents of primary water stress corrosion cracking in
steam generators with a similar design to the Indian Point Unit
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 (Ringhals) that have similar design and construction details to 15 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 nearly the length of the divider plate (~6 feet). Perhaps of 22 23 more concern, as the cracks approach the triple point of the Pre-filed Written Testimony of David J. Duquette

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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 the pressure boundary of the steam generator by propagating 3 through the channel head via corrosion fatigue after the PWSCC 4 5 crack has initiated. Cracks that form in the divider plate, the stub runner, and/or the associated welds may propagate into the б 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 cladding that would then propagate into the tube to tubesheet 11 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.

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Entergy's Proposed Approach

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

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

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"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 Management Program Engineering and Regulatory 3 Technical Advisory Group. This industry-lead effort 4 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 technique used will be capable of detecting PWSCC in the steam 11 12 generator divider plate assembly welds." 13 Did Entergy provide any further information about the Ο. 14 inspections it proposed to perform on the steam generator divider plate assemblies? 15 16 No, it did not. Α. 17 Did Entergy make any proposals with respect to Staff 0. 18 concern about the propagation of primary water stress corrosion 19 cracking in a steam generator that contains Alloy 690TT heat transfer tubes? 20 21 Α. In 2011, Entergy proposed two options as Yes. follows: 22 23 Option 1 (Analysis) Pre-filed Written Testimony of David J. Duquette

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1 "IPEC will perform an analytical evaluation of the 2 steam generator tube-to-tubesheet welds in order to establish a technical basis for either determining 3 that the tubesheet cladding and welds are not 4 susceptible to PWSCC, or redefining the pressure 5 boundary in which the tube-to-tubesheet weld is no 6 7 longer included and, therefore, is not required for reactor coolant pressure boundary function. 8 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 The condition will be resolved through repair or a. 18 engineering evaluation to justify continued service, 19 as appropriate, and 20 An ongoing monitoring program will be established b. 21 to perform routine tube-to-tubesheet weld inspections for the remaining life of the generators. 22 23 Have you seen any information about the status of the 0. Pre-filed Written Testimony of David J. Duquette Contention NYS-38/RK-TC-5

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

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

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

10 Α. 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 Roadmap Summary." The second is entitled "EPRI, 2012 Research 15 16 Portfolio, Steam Generator Management." These documents discuss 17 the EPRI investigation and its timeline.

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

1 and associated weld materials in the steam generator could 2 propagate to pressure boundaries such as the tube-to-tubesheet weld or the carbon steel materials in the bowl. The document 3 4 further acknowledges that the industry lacks understanding of 5 the impact of cracks that may compromise safe operations as the steam generators age, and proposes a research program to address б 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.

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

15 Yes. It is important to develop an Aging Management Α. 16 Program with substantive, meaningful, and enforceable standards. 17 The fact that Indian Point in the past has experienced some primary water stress corrosion issues with Alloy 600 material in 18 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 Pre-filed Written Testimony of David J. Duquette Contention NYS-38/RK-TC-5

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

5 Α. Ostensibly it has, but from a practical and engineering perspective it has not, nor has it addressed safety б 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 divider plate assemblies, tubesheets, and welds contains several 11 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.

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

> Pre-filed Written Testimony of David J. Duquette Contention NYS-38/RK-TC-5

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

6 Are you aware that NRC Staff, in its August 2011 Ο. 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 will be adequately managed so that their intended functions 11 would be maintained during the requested period of extended 12 13 operation?

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

Q. Have you now completed your initial testimonyregarding contention NYS-38/RK-TC-5?

19 A. Yes. However, I reserve the right to express further20 opinions if new evidence is introduced or disclosed.

21

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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)
	David J. Duquette, Ph.D. Materials Engineering Consulting Services 4 North Lane Loudonville, New York 12211 Tel: 518 276 6490 Fax: 518 462 1206 Email: duqued@rpi.edu
	June 14, 2012
	Pre-filed Written Testimony of David J. Duquette Contention NYS-38/RK-TC-5
	30