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

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

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

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

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

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

A. Yes.

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.

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

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 steam generators at nuclear power plants?

A. Yes.

Q. Please describe that experience.

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

Overview

Q. What is the purpose of your testimony?

A. The purpose of my testimony is to provide support for, and my views on, an aspect of New York’s Contention 38 (“NYS-38”), which was admitted for litigation by the Atomic Safety Licensing Board (“ASLB”). Contention NYS-38 asserts, among other things, that Entergy has not demonstrated that it has a program that will manage the effects of aging of critical components or systems at the Indian Point nuclear power facilities and that therefore the NRC does not have a record and a rational basis upon which it can determine whether to grant Entergy a renewed license for the Indian Point facilities. My testimony critiques Entergy’s proposed approach towards the age related degradation of various components of Indian Point's steam generators during the requested twenty year period of extended operation.

Q. I show you what has been marked as Exhibit NYS000372. 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.

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

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

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

A. Yes.

Q. What is the source of those materials?

A. Many are documents prepared by government agencies, peer reviewed articles, or documents prepared by Entergy or by the nuclear power industry.

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

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

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

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

Q. Did you review anything else in preparing your report 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.

Conclusions and Opinions

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

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

At the present time there is no qualified inspection procedure to determine the extent of cracking in the divider plates or associated channel head assemblies or the propagation of cracking from the tubesheet cladding to the tube-to-tubesheet weld. European inspection procedures result in high radiation doses for plant workers/inspectors. According to an August 4, 2011 NEI Steam Generator Task Force presentation to NRC, the stresses that may initiate PWSCC or to lead to the propagation of either PWSCC or fatigue cracks from the divider plate assemblies into the tube-to-tubesheet welds have not even been 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.

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

Stress Corrosion Cracking

Q. What is stress corrosion cracking?

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

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

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

A. Yes. Cracking of Alloy 600 steam generator tubes was originally observed in the vicinity of the tubesheets and tube support plates in steam generators because of the expansive characteristics of the corrosion products of the carbon steel tubesheets and support plates in the crevices between the support plates and the tubesheets and the rolled-in tubes. The expansion of the corrosion products imparted large stresses on the mill annealed Alloy 600 tubes resulting in plastic deformation of the tubes (denting). Cracking in the deformed tubes in the tubesheet region was brought under some measure of control by judicious water treatment campaigns. However, 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 components in nuclear reactors, and to the present day numerous attempts at quantifying the specific mechanisms of the susceptibility of Alloy 600 to primary water stress corrosion cracking were attempted but only with limited success. It is clear that metallurgical, environmental, and loading variables all contribute to the susceptibility of Alloy 600 to primary water stress corrosion cracking.

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

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

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

Indian Point Power Generation Systems & Steam Generators

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

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

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?

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

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.

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

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.

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

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

Q. Did there come a time when Indian Point facilities 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.

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

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 current Indian Point Unit 2 steam generators?

A.   Yes, Entergy has stated that the current Indian Point Unit 2 steam generators use Alloy 600 for the tubes and 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.   Has Entergy disclosed the material used in the current 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 are affected by primary water stress corrosion cracking?

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

<u>Reactor Coolant Pressure Boundary</u>

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

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

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

> as pressure vessels, piping, pumps, and valves, which are:
>
> (1) Part of the reactor coolant system,
>
> or
>
> (2) Connected to the reactor coolant system, up to and including any and all of the following:
>
> (i) The outermost containment isolation valve in system piping which penetrates primary reactor containment,
>
> (ii) The second of two valves normally closed during normal reactor operation in system piping which does not penetrate primary reactor containment,
>
> (iii) The reactor coolant system safety and relief valves."

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

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

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

Current Concerns About Primary Water Stress Corrosion Cracking

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

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

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

A. Yes.

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

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

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

A. Yes, it is a copy of the NRC Staff's Supplemental Safety Evaluation Report that they issued at the end of August 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 Westinghouse steam generator divider plates assemblies and the tube-to-tubesheet welds.

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

A. Yes.

"The staff noted that, although these SG divider plate assembly cracks might not have a significant safety impact in and of themselves, these cracks could affect adjacent items that are part of the reactor coolant pressure boundary, such as the tubesheet and the channel head, if they propagate to the boundary with these items. The staff further noted that for the tubesheet, PWSCC cracks in the divider plate assemblies fabricated from Alloy 600 and its associated weld metals could propagate to the tubesheet cladding, with possible consequences to the integrity of the tube-to-tubesheet welds. Furthermore, for the channel head, the PWSCC cracks in the divider plate assemblies could propagate to the SG triple point (i.e. the point where the divider plate and tube sheet meet with the shell) and potentially affect the pressure boundary of the SG channel head."

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

A. Yes.

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

EPRI and Westinghouse have cited reports of cracking in the divider plate assemblies in French steam generators (Saint Laurent, Gravelines, Chinon) and in a Swedish steam generator (Ringhals) that have similar design and construction details to U.S. reactors. The cracking has been observed in the divider plate itself, in the full penetration welds connecting the stub runner to the tubesheet and connecting the stub runner to the divider plate. In the French steam generators, the cracks are reported to have occurred in the heat affected zone of the stub runner to divider plate weld and have been observed to run nearly the length of the divider plate (~6 feet). Perhaps of more concern, as the cracks approach the triple point of the tubesheet-channel head complex, the cracks tend to curve upwards. It has been suggested that this PWSCC could compromise the pressure boundary of the steam generator by propagating through the channel head via corrosion fatigue after the PWSCC crack has initiated. Cracks that form in the divider plate, the stub runner, and/or the associated welds may propagate into the tubesheet, allowing mixing of the primary water with the secondary water and accordingly compromising the integrity of the reactor coolant pressure boundary. Given the crack path, another possibility is propagation of PWSCC into the tubesheet cladding that would then propagate into the tube to tubesheet weld and subsequently into the Alloy 600 tubes. This phenomenon is of particular concern for the IP2 replacement steam generators that were constructed in the 1980s with Alloy 600 tubes. Moreover, the steam generators at both IP2 and IP3 have Alloy 600 divider plates and Alloy 82/182 welds.

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:

"The industry plans are to study the potential for divider plate crack growth and develop a resolution to the concern through the EPRI Steam Generator Management Program Engineering and Regulatory Technical Advisory Group. This industry-lead effort is expected to begin in 2011 and be completed within two years."

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

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

A. No, it did not.

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

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

Option 1 (Analysis)

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

Option 2 (Inspection)

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

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

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

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

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.

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

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

A. In this section EPRI acknowledges that primary water stress corrosion cracking (PWSCC) that initiates in Alloy 600 and associated weld materials in the steam generator could propagate to pressure boundaries such as the tube-to-tubesheet weld or the carbon steel materials in the bowl. The document further acknowledges that the industry lacks understanding of the impact of cracks that may compromise safe operations as the steam generators age, and proposes a research program to address this issue in AMR’s. EPRI also admits that there are “no qualified techniques to inspect the steam generator channel head”, and that the inspection methods currently used in Europe to inspect the steam generator divider plates result in 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?

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

Q. Has Entergy agreed that it needs an Aging Management

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?

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

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.

Q. Are you aware that NRC Staff, in its August 2011 Supplemental Safety Evaluation Report, found that Entergy's proposed approach "acceptable" and concluded that Entergy has demonstrated that the effects of primary water stress corrosion cracking in the divider plate assemblies in the steam generators will be adequately managed so that their intended functions would be maintained during the requested period of extended 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 testimony regarding contention NYS-38/RK-TC-5?

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

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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 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)

D.J. Duquette

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