UNITED STATES
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

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, Entergy Nuclear Indian Point 3, LLC, and Entergy Nuclear Operations, Inc.	DPR-26, DPR-64 June 7, 2012

REPORT OF
DR. DAVID J. DUQUETTE
IN SUPPORT OF
CONTENTION NYS-38/RK-TC-5

Prepared for the State of New York

Office of the Attorney General

PROFESSIONAL QUALIFICATIONS

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Dr. David J. Duquette is the John Tod Horton Professor of Engineering at Rensselaer Polytechnic Institute (RPI) and a member of the Department of Materials Science and Engineering. He is a graduate of the United States Coast Guard Academy and the Massachusetts Institute of Technology (MIT). He performed his graduate work at the Corrosion Laboratory at the Massachusetts Institute of Technology, and 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. Dr. Duquette's research is primarily in the area of corrosion science and engineering. He has supervised more than 50 graduate research dissertations in corrosion and related sciences. He is the author or co-author of more than 230 publications and 20 book chapters. He presents invited lectures internationally approximately 20 to 25 times per year. Among his awards, he has 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). He has received the Whitney Award of NACE for outstanding corrosion research, an A. V. Humboldt Senior Scientist Award from the German government, and a number of other awards from the scientific community. Dr. Duquette has just 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. In addition to his academic duties, Dr. Duquette maintains an active consulting practice, primarily in the area of corrosion and mechanical failures.

His experience with corrosion issues at nuclear plants includes consultation at Three Mile Island (TMI-1 and TMI-2), Diablo Canyon, all of the pressurized water reactors and boiling water reactors formerly operated by Commonwealth Edison (including Byron, LaSalle, Braidwood, Dresden, Quad Cities, Clinton, Zion 1&2), and Seabrook. He has 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 United States Department of Energy for 11 years for corrosion research in nuclear systems. He has supervised Ph.D. students performing research on nuclear systems for United States Navy applications at the Knolls Atomic Power Laboratory in Niskayuna, NY. He has also received invitations to visit numerous reactors because of his service on the Nuclear Waste Technical Review Board including Dresden, Savannah River, Hanford, several French plants and plants in England, Germany, Spain, and Argentina. In each of those visits high level aspects

of technical management of the facilities, including aging and maintenance of the infrastructures were discussed in detail.

A REVIEW OF ENTERGY'S PROPOSED AGING MANAGEMENT PROGRAM FOR STEAM GENERATORS AT THE INDIAN POINT POWER REACTORS

EXECUTIVE SUMMARY

Entergy has submitted a license renewal application (LRA) to the Nuclear Regulatory Commission (NRC) for permission to continue to operate two of the nuclear power generation units (IP2 and IP3) at Indian Point in Westchester County, New York (or IPEC) for an additional period of 20 years. New York State (NYS) has expressed various concerns about the safety of some of the Indian Point infrastructures because of aging of those infrastructures. Accordingly, NYS has filed contentions with the NRC regarding Entergy's proposed Aging Management Program (AMP) for the Indian Point systems, structures, and components. Among other issues, a concern exists about the approach to the aging management of the facilities' steam generators that Entergy and NRC Staff agreed to in 2011.

Based on the results of a review of documents provided by Entergy and NRC Staff to date as well as industry and engineering literature, there is a serious concern about potential cracking in the channel head assembly of the Westinghouse steam generators at Indian Point. Recent experience in similar steam generators in Europe have 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.

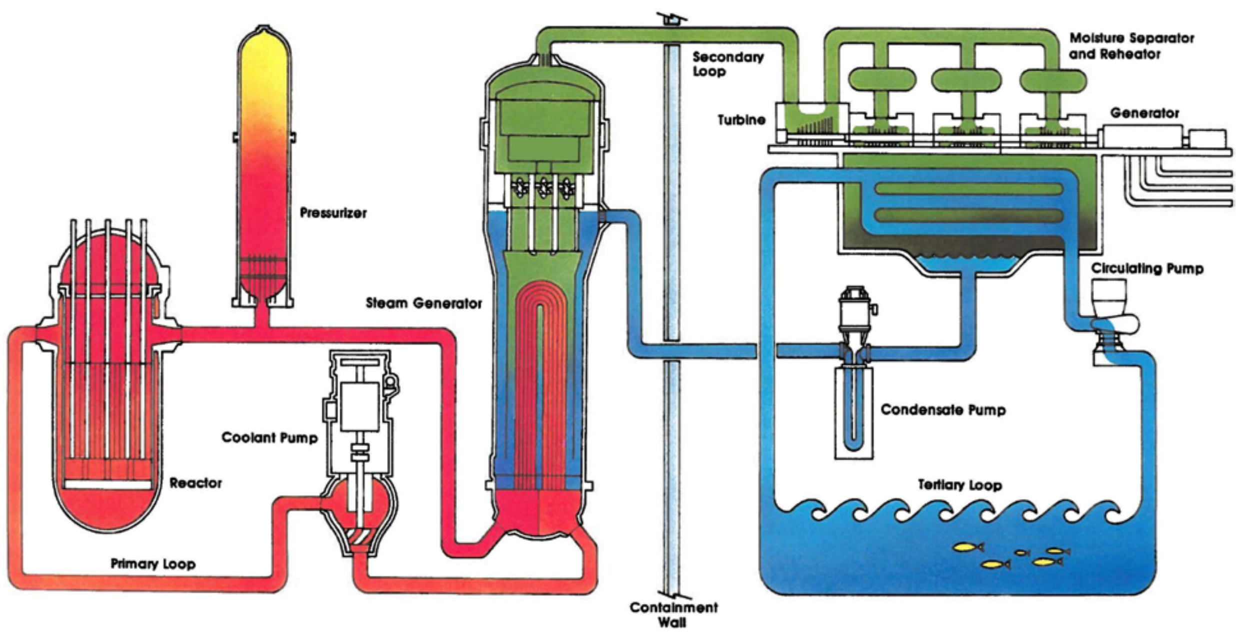
At the present time there is no qualified inspection procedure to determine the extent of cracking in the divider plates or associated channel assemblies. European inspection procedures result in high radiation doses for plant workers/inspectors.

EPRI has recently begun a program (2012) to determine the susceptibility of divider plates and related structures and assemblies to PWSCC and the results of that research are not scheduled to be available until 2016, well into the period of extended operation of IP2 and IP3 at Indian Point. Entergy's proposed plan for steam generator divider plate assemblies, tubesheets, and welds contains several unknowns. At present, neither Indian Point nor NRC 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.

OVERVIEW OF THE INDIAN POINT NUCLEAR STEAM SUPPLY SYSTEM INCLUDING THE STEAM GENERATORS

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.[1] 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.[2] A schematic drawing of a Westinghouse Pressurized Water Reactor Nuclear Steam Supply System is shown below:



Nuclear Steam Supply System
MB 3618A

[1] Entergy Indian Point License Renewal Application (LRA), at pages 1-6, 2.3-2, 2.3-6 (April 2007).

[2] LRA, at p. 2.3-2 & 2.3-6.

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. [3] The divider plate is joined to the channel head and the tubesheet through a stub runner.

Multiple indications of cracking in the welds that connect the stub runner to the divider plate and the stub runner to the tubesheet have been identified in French and Swedish steam generators of similar construction to those at Indian Point. Cracking has also been observed in the divider plate in the European steam generators.

According to public documents on file with NRC, Indian Point Unit 2 and Unit 3 were constructed with Westinghouse Model 44 steam generators. In 1989, Indian Point Unit 3 installed Westinghouse Model 44F steam generators. In January 2001, following a steam generator tube rupture, Indian Point Unit 2 installed Westinghouse Model 44F steam generators.[4] The steam generators at the Indian Point reactors are constructed primarily of carbon (low alloy) steel. The heat transfer tubes are Inconel: Alloy 600 for IP2, and Alloy 690 for IP3. The tubes were thermally treated after tube-forming operations. The April 2007 LRA stated that the interior surfaces of the channel heads and nozzles are clad with austenitic stainless steel, and the tube sheet surfaces in contact with reactor coolant are clad with Inconel.[5] In 2011, Entergy informed NRC that at both Indian Point Unit 2 and Indian Point Unit 3 the steam generator divider plates are Inconel 600 (Alloy 600) and that it assumed that the weld material for the divider plate assemblies was Alloy 82/182 weld material.[6]

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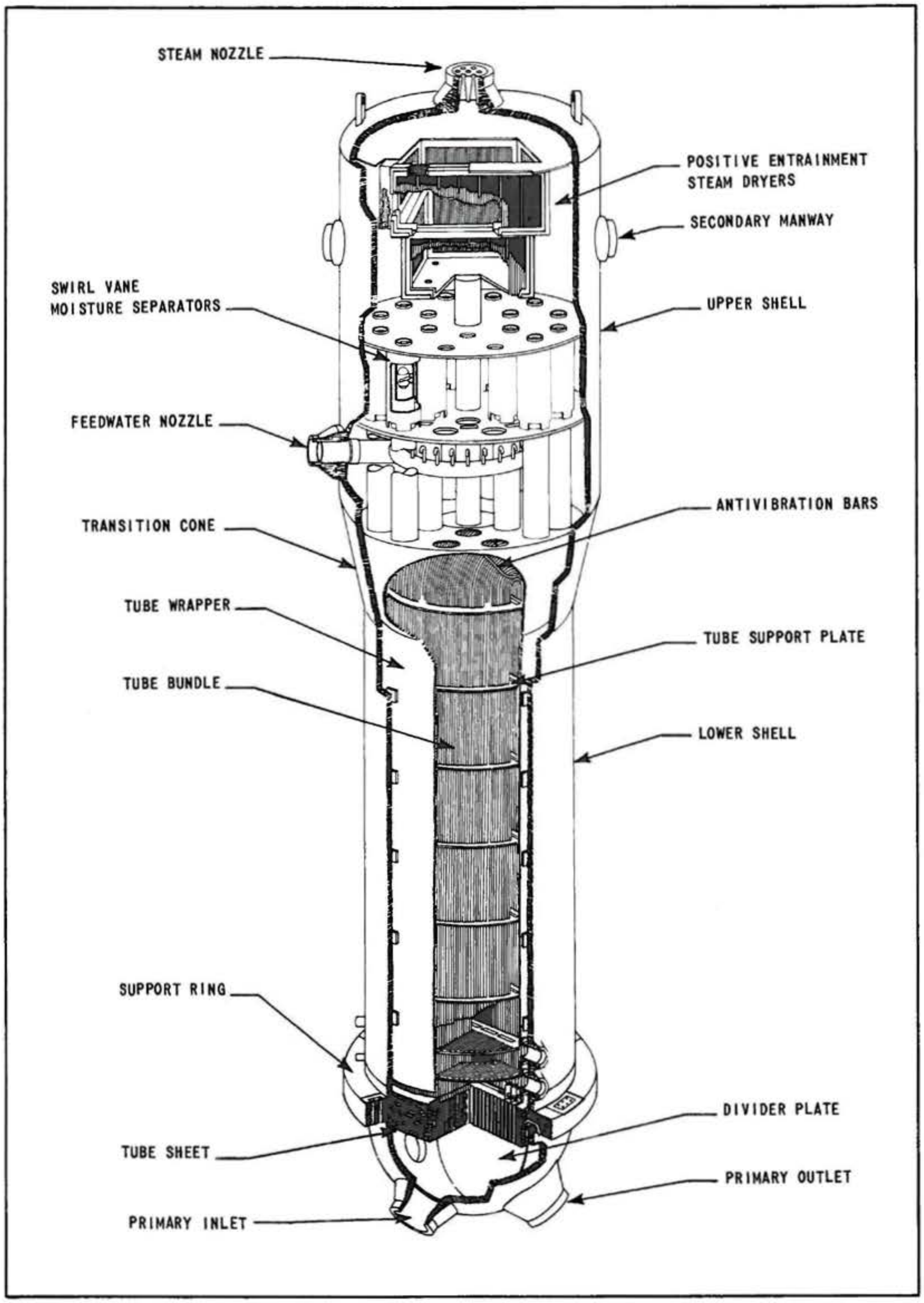
[3] LRA, at p. 2.3-4 & 2.3-8.

[4] LRA, at p. 2.3-21.

[5] LRA, at p. 2.3-21.

[6] Entergy NL-11-032 communication with NRC Staff, Response to Request for Additional Information (March 28, 2011) ML110960360, Attachment 1 at p. 20 of 27.

A diagram of a Westinghouse steam generator is shown below:



Westinghouse STEAM GENERATOR

TECHNICAL BACKGROUND ON STRESS CORROSION CRACKING

Background

Stress corrosion cracking (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.

In the nuclear energy production area, cracking of stressed nickel based alloys, in particular Alloy 600, in primary water was identified as early as the 1950's.[7] It was originally called pure water stress corrosion cracking and was later relabeled as primary water stress corrosion cracking (PWSCC). Cracking of Alloy 600 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, including at least one documented rupture at Indian Point 2 on February 15, 2000.[8] This cracking has been ascribed to ovalization during the fabrication process that resulted in residual tensile stresses.[9] 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 PWSCC were attempted but only with limited

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[7] M. M. Coriou, et al, "Corrosion Fissurante sous Contrainte de L'Inconel dans L'Eau a Haute Temperature", 3e Colloque de Metallurgie Corrosion (Seche et Aqueuse, 1959).

[8] Steam Generator Tube Operational Experience (from NRC Website (Nov. 2, 2000)), ML013100106.

[9] History of Westinghouse Model 44 Steam Generators (from NRC website (Nov. 2, 2000)), ML013100106.

success. It is clear that metallurgical, environmental, and loading variables all contribute to the susceptibility of Alloy 600 to PWSCC. As early as 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.[10] Some success has been achieved with specific thermal treatments of the alloy, and the introduction of improved water chemistries. In many cases, particularly for steam generator tubes, Alloy 600 was replaced with a more PWSCC resistant alloy designated Alloy 690.[11] However, there are numerous components in an operating nuclear reactor that still contain Alloy 600. For example, a Companion Guide to the ASME Boiler Code has identified PWSCC concerns in steam generator tubes, heater thermal sleeves and penetrations in the pressurizer, penetrations for the control rod drive mechanisms in reactor pressure vessel heads and in other components of the reactors all of which are fabricated from Alloy 600.[12] 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 PWSCC. Cracking in welds has been observed in the butt welds of control rod drive mechanisms in 1991 (Bugey-3), in 1999, (Ringhals Unit 3), and in 2000 (V.C. Summer and Ringhals Unit 3), indicating that, while the welds may have somewhat more resistance to cracking, they are certainly not immune.[13]

PWSCC in Divider Plate Assemblies

In a recent publication H. Cothron of EPRI cited French reports of cracking in the divider plate assemblies in French steam generators (Saint Laurent, Gravelines, Chinon) and a Swedish steam generator (Ringhals) that have similar design and

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[10] NRC Generic Letter No. 85-02, "Staff Recommended Actions Stemming From NRC Integrated Program for the Resolution of Unresolved Safety Issues Regarding Steam Generator Tube Integrity" (April 17, 1985), ML031150391.

[11] Steam Generator Tube Operational Experience (from NRC Website (Nov. 2, 2000)), ML013100106.

[12] Gorman, et al., "Companion Guide to ASME Boiler & Pressure Vessel Code, Chapter 44, PWR Reactor Vessel Alloy 600 Issue."

[13] G. Roussel, "Management of the Nickel-Base Alloy Cracking in Butt Welds at Belgian Nuclear Power Plants" in Ageing Issues in Nuclear Power Plants, NuPeer, 2005.

construction details to U.S. reactors.[14] 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. 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 steam generators which were replaced in 2001 with steam generators constructed with Alloy 600 tubes. Moreover, both IP2 and IP3 have Alloy 600 divider plates and Alloy 82/182 welds. The susceptibility to PWSCC of divider plate assemblies was recognized by Westinghouse in 2003 in the design of the AP1000 reactor. In a presentation to NRC Staff dated June 11, 2003, Westinghouse indicated that the application of Alloy 690TT were extended to steam generator divider plates because of the incidence of PWSCC in Alloy 600 and its associated Alloy 82 and 182 welds.[15]

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[14] EPRI Final Report 1014982 (redacted) Divider Plate Cracking in Nuclear Reactors, Results of Phase 1: Analysis of Primary Water Stress Corrosion Cracking and Mechanical Fatigue in the Alloy 600 Stub Runner to Divider Plate Weld Material, Non-Proprietary Version, June 2007, ADAMS ML072970190, at 5-1 (References) *citing:* [1.] MRP-EDF-SGPP05, Saint-Laurent B NPP – Unit 1 – SG #52 – Loop#2 – Examination of specimen harvested from a hammered partition plate; [2.] MRP-EDF-SGPP01, Saint-Laurent B NPP – Unit 1 – SG#52 – Loop#2 – Examination of two specimens taken out from the SG channel head divider plate; [3.] MRP-EDF-SGPP03, Gravelines NPP – Unit 1 –Destructive examination of the triple point of the SG2 channel head; [4.] MRP-EDF-SGPP02, Dampierre 1/2 NPP – Steam generator #27 retired from loop 3 when the SG's were replaced in 1990 – Chemical, metallurgical and mechanical characterization of the weld joining the partition stub, the divider plate and the channel head bowl; [5.] MRP-EDF-SGPP04, Chinon B NPP – Unit 4 – Charaterization of indications discovered on the SG #2 stud/partition plate weld surface.

[15] Westinghouse presentation to NRC Staff, "AP1000 Draft Safety Evaluation Report LBB Open Items," (July 2003).

Entergy's LRA Related to Divider Plates

In its license renewal application (LRA) Entergy indicates that at IP2 and IP3 the steam generators are constructed of carbon steel with Alloy 600 tubes at IP2 and Alloy 690 tubes at IP3.[16] The tubesheet surfaces are clad with Inconel and the tube-to-tubesheet joints are welded. The original LRA did not specify if the Inconel is Alloy 600, 690, or some other Inconel alloy. Entergy's original LRA also does not indicate the alloy used for welding the tubesheet joints. Upon issuance of an RAI by NRC Staff, Entergy clarified that the alloys use in the divider plates and channel head assemblies are Alloy 600 and the related weld Alloys 82 and 182.[17] In the license renewal application,[18] Entergy has identified cracking of the nickel alloy clad steam generator divider plate exposed to reactor coolant as subject to its Aging Management Review (AMR) and presumably its Aging Management Program (AMP) under NUREG-1800.[19] In the LRA, Entergy claims that water chemistry control will manage cracking of the nickel alloy steam generator divider plates exposed to the reactor coolant (primary water). The divider separates the inlet coolant from the effluent. Through-wall cracking of the divider will not *per se* compromise the pressure boundary of the steam generator. However, 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. Note that in Table 2.3.1-4 of the LRA Entergy does indicate that the channel head, the divider plate, and the tubesheet each constitutes a pressure boundary for IP2 and IP3.[20] Through wall cracking of the divider will also compromise the intended heat transfer function of the generator and tubes.

Entergy's AMP Related to Divider Plates

NRC Staff's Generic Aging Lessons Learned Report (GALL) NUREG-1801 specifies that certain aspects of an aging management program (AMP) must be addressed in

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[16] LRA, section 2.3.1.4.

[17] Entergy, NL-11-032, Communication with NRC Staff, Response to Request for Additional Information (March 28, 2011), ML110960360.

[18] LRA, Section 2.3.1.4.

[19] LRA, Table 3.1.1 (p. 3.1-38).

[20] LRA, p. 2.3-36, 2.3-39.

an LRA as discussed in Section 3.1.2.2 of the NRC Staff's Standard Review Plan (SRP) NUREG-1800.[21] Section 3.1.2.2.13 of NUREG 1800 specifically addresses cracking due to primary water stress corrosion cracking. Entergy states that its commitment to compliance with section 3.1.2.2.13 of NUREG 1800 is included in Sections A.2.1.40 and A.3.1.40 of the LRA's UFSAR Supplement Appendix A.

NRC Commentary

Divider Plates

In August 2011, NRC Staff issued a supplement to the Safety Evaluation Report (SER) for IP2 and IP3.[22] In the supplement, NRC Staff addressed Entergy's responses to the Staff's Request for Additional Information (RAI). With reference to the divider plate assemblies the Staff, in its earlier RAI, noted that foreign operating experience with similar designs to Indian Point's steam generators had identified cracking due to PWSCC in steam generator divider plate assemblies fabricated from Alloy 600. It was noted that the cracking was observed even with proper water chemistry. The NRC Staff determined that the Water Chemistry–Primary and Secondary Program might not be effective in managing cracking due to PWSCC in steam generator divider plate assemblies.[23]

Entergy's March 2011 response to NRC Staff questions included the following:

1. The divider plates at both IP2 and IP 3 are Alloy 600 and associated Alloy 82/182 weld alloys.
2. The industry plans to study the potential for divider plate crack growth and to develop a resolution to the concern through the EPRI Steam Generator Management Program Engineering and Regulatory Advisory Group.
3. Until a resolution is achieved by EPRI, Indian Point stated that it would inspect all of its steam generators. [24]

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[21] LRA, p. 3.1-6.

[22] NRC Staff Supplemental Safety Evaluation Report, NUREG-1930 (August 30, 2011) ML11243A109.

[23] NRC Staff Supplemental Safety Evaluation Report, NUREG-1930, at p. 3-18 - 3-19.

[24] Entergy NL-11-032 communication with NRC Staff, Response to Request for Additional Information (March 28, 2011) ML110960360, Attachment 1 at p. 20 of 27.

> In July 2011 Entergy committed to the following at IPEC:
>
> "IPEC will perform an inspection of steam generators for both units to assess the condition of the divider plate assembly. The examination technique used will be capable of detecting PWSCC in the steam generator divider plate assemblies. The IP2 steam generator divider plate inspections will be completed within the first ten years of extended operation (PEO), *i.e.*, prior to September 28, 2023. The IP3 steam generator divider plate inspections will be completed within the first refueling outage following the beginning of the PEO."[25]

The NRC Staff found this response to be acceptable and decided that the concerns expressed in the RAI are resolved. The Staff further concluded that the effects of aging on the steam generator divider plates will be adequately managed. However, there was no indication of how the inspection of the divider plate assemblies will be accomplished.

Tube-to-Tubesheet Welds

In its August 9, 2011 supplemental response[26] to NRC Staff's Request for Additional Information and subsequent conferences between Staff and Entergy, Entergy proposed that it "will develop a plan" using one of two options:

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

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[25] Entergy NL-11-074 communication to NRC Staff, Response to Request for Additional Information (July14, 2011) ML11201A160.

[26] Entergy NL-11-096 communication to NRC Staff, Clarification for Request for Additional Information (RAI), Aging Management Programs, Indian Point Nuclear Generating Unit Nos. 2 & 3, Docket Nos. 50-247 and 50-286, License Nos. DPR-26 and DPR-64 (Aug. 9, 2011) ML11229A803.

> 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.

The NRC Staff found this "commitment" acceptable and concluded that this section of the RAI is resolved.

New York State Concerns

Neither Entergy's AMP, its response to the NRC Staff RAI's, nor the response of the Staff to Entergy's so-called commitments are compelling. Entergy claims to be committed to an AMP that will ensure the integrity of the steam generator pressure boundary and provide for the safety of its workers and of the surrounding citizenry.

NRC, Westinghouse, NEI, and Entergy all admit to being aware of the presence of cracked divider plates in Electricité de France's (EdF) steam generators.

In 2007, EPRI published a report entitled "Divider Plate Cracking in Steam Generators: Results of Phase I: Analysis of Primary Water Stress Corrosion Cracking and Mechanical Fatigue in the Alloy Stub Runner to Divider Plate Weld Material" (EPRI 1014982).[27] The report was followed by a second report in 2008 by

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[27] H. Cothron, EPRI Final Report 1014982 (redacted) Divider Plate Cracking in Steam Generators, Results of Phase 1: Analysis of Primary Water Stress Corrosion Cracking and Mechanical Fatigue in the Alloy 600 Stub Runner to Divider Plate Weld Material, Non-Proprietary Version (June 2007), ML072970190; H. Cothron, EPRI Final Report 1014982 (unredacted) Divider Plate Cracking in Steam Generators, Results of Phase 1: Analysis of Primary Water Stress Corrosion Cracking and Mechanical Fatigue in the Alloy 600 Stub Runner to Divider Plate Weld Material (June 2007) (available on EPRI website).

H. Cothron of EPRI entitled "Divider Plate Cracking in Steam Generators- Results of Phase II: Cracked Divider Plate on LOCA and Non-LOCA analyses (EPRI report 1016552 – redacted).[28] EPRI issued another report in 2009.[29]

Westinghouse made a presentation to the NEI Steam Generator Task Force, detailing the state of knowledge of divider assembly PWSCC in the French and Swedish reactors.[30] In June 2009, Westinghouse issued a report entitled "H*: Alternate Repair Criteria for Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model 44F) (Westinghouse-WCAP-17091-NP). That report addressed the possibilities of the consequences of cracks in the divider plate assemblies in the same model steam generators in use at Indian Point.

The NRC made a presentation to a bilateral exchange with Japan in September 2010 identifying the cracking issue in the French units.[31] That presentation included the important observation that no inspections had been performed in the U.S. and that the question of whether the cracks in the divider plate could grow into the channel head or tubesheet cladding was still open. The NRC opined that cracking of the steam generator shell or the tube-to-tubesheet weld could have safety consequences.

In Nuclear Energy Institute (NEI) presentations on February 18, 2011 and on August 4, 2011, H. Cothron of EPRI presented updates on the state of knowledge of EdF's steam generators.[32] It is important to note that the cracks that were observed in the EdF steam generators were only observed by post mortem

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[28] H. Cothron, EPRI Report 1016552 (redacted) Divider Plate Cracking in Steam Generators, Results of Phase II: Evaluation of a Cracked Divider Plate on LOCA and Non-LOCA Analyses (November 2008), ML083650073.

[29] H. Cothron, EPRI Report 1019040 (redacted) Steam Generator Management Program Steam Generator Divider Plate Cracking Engineering Study, Non-Proprietary Version (December 31, 2009), ML100491594.

[30] C. Cassino, Westinghouse Presentation, NEI Steam Generator Task Force Divider Plate Cracking Issue Update (July 25, 2007), ML0726060144.

[31] M. Evans and R. Taylor, Bilateral Exchange, USNRC Presentation (Sept. 15, 2010)

[32] NEI Steam Generator Task Force – NRC/Industry Update, (Feb. 18, 2011, August 4, 2011).

inspections of retired steam generators and that a crack was observed that was 6 feet long. Thus the cracks appear to only have been observed after the reactors were decommissioned and were not detected during plant operation.

Entergy claims to be committed to inspection of the divider plates. However, there is no evidence that Indian Point (or other U.S. utility) has successfully performed inspections of divider plates in a steam generator at a plant that is fully commissioned and operating. Westinghouse, in its presentation in 2007, stated that there are no qualified U.S. divider plate inspection or repair criteria and no qualified U.S. divider plate inspection tools.[33] To my knowledge that is also true today. Cothron in her presentation to the Steam Generator Task Force in August 2011 admitted that there are still no qualified techniques for inspection in the US and that inspections inside the steam generator bowl will result in high doses to workers.

It is also important to note that the PWSCC cracks in the divider plate assemblies in the French steam generators appear to be deviating upwards, into the vicinity of the tubesheet. A through-crack in a divider plate would result in a kind of short circuit, mixing hot and cold coolant water in an area below the tubesheet, effectively below the steam generator itself. However, if the crack deviates upwards into the tubesheet cladding material, there will be no barrier to crack propagation into the tube-to-tubesheet welds thus compromising the pressure boundary and mixing primary water and secondary water. At IP2 where the tubes are Alloy 600, propagation of cracks through the Alloy 600 cladding of the tubesheets may compromise the tubes.

At this time, Entergy admits that it does not know the susceptibility of the tube-to-tubesheet welds to PWSCC and is proposing to either perform an analysis of the susceptibility of the welds to PWSCC or to perform an undisclosed number of inspections within an extended period of time. Further, the specific nature of these inspections has not been revealed by Entergy.

PWSCC initiation in the tube-to-tubesheet welds may lead to a rapid compromise of the pressure boundary with subsequent mixing of primary water with secondary water. It is well known that if a stress corrosion crack is propagating it will generally require less driving force (energy) than might be required to nucleate a crack. It is for this very reason that extensive studies of crack propagation rates

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[33] C. Cassino, Westinghouse Presentation, NEI Steam Generator Task Force Divider Plate Cracking Issue Update (July 25, 2007) at 20.

generally accompany testing of new materials in environments where SCC is suspected. Alloys that exhibit resistance to crack initiation often show marked susceptibility to rapid crack propagation rates. For the divider plate PWSCC, the crack propagation rate is important but it is the deviation in crack growth direction that is particularly worrisome.

In the NEI presentations EPRI has apparently sought to alleviate fears of divider plate cracking in U.S. reactors by pointing out the differences between the French steam generators and domestic steam generators. For example, EPRI has indicated that the hydrostatic pressure tests in France were performed at 1.33 times design pressure, whereas U.S. steam generators are tested at a nominal ratio of 1.25 times design pressure. Also EPRI has indicated that the French steam generators have divider plate thicknesses of 1.33 inches while the Model 44F steam generators at Indian Point have divider plate thicknesses of less than 1.5 inches.[34] It is my opinion that these differences are not significant. The hydrostatic test pressure in the French steam generators was only about 6% larger than in the U.S. steam generators and the plate thickness is within less than 13% if it is fully 1.5 inches. It is my opinion that neither of these differences can account for susceptibility of the French divider plates to PWSCC, or of any indication of immunity in U.S. steam generators of the same basic design and construction.

The issues of PWSCC and the consequences of PWSCC in the divider plates of steam generators are still unresolved in 2012. As recently as February 16, 2012 EPRI made a topical presentation to a joint Steam Generator Task Force Nuclear Regulatory Commission Biannual Meeting.[35] Participants in the meeting included EPRI, Westinghouse, Entergy, NEI, Areva, and NRC. In that presentation it was noted that the GALL report, Revision 2 added the divider plate and tube-to-tubesheet weld as items to be evaluated in AMPs.[36] It was also noted that applications reviewed using GALL Revision 2 require a commitment to inspect

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[34] C. Cassino, Westinghouse Presentation, NEI Steam Generator Task Force Divider Plate Cracking Issue Update (July 25, 2007) at 4.

[35] EPRI presentation, Steam Generator Task Force / Nuclear Regulatory Commission Biannual Meeting (February 16, 2012), ML12047A296.

[36] It is my understanding that GALL, Revision 1 was issued by NRC Staff in September 2005 and that GALL, Revision 2 was issued by NRC Staff in December 2010. *See* NUREG-1801, GALL, Revision 2, at 3 (discussing sequence of issuances) ML103490041.

these areas once they are in the period of extended operation and the steam generators have been in service for more than 20 years. Notably plants whose applications were reviewed prior to issuance of GALL Revision 2 have not made those commitments. Revision 2 of the GALL report had apparently not been issued when Entergy submitted its LRA and its commitments are treated in the LRA under Revision 1. However, it is good engineering practice to address newly identified engineering problems, particularly those that involve safety, as they arise.

The EPRI Nuclear Sector Roadmaps–Materials Aging and Degradations, issued in January 2012,[37] states that the objectives of the Roadmap are to:

- Maximize the operating life of BWR and PWR passive components;
- Predict component degradation mechanisms and their rate of occurrence to inform decisions on mitigation, repair or replacement options;
- Account for the impact on plant operations associated with implementing materials aging management activities;
- Develop data and physically based predictive models for remaining useful life assessments;
- Identify and disposition (sic) degradation mechanism knowledge gaps through fundamental R&D; and
- Conduct research, evaluate and optimize joining, fabrication and repair processes; (sic).

The Roadmap includes a section entitled "Aging management of Alloy 600 and Alloy 82/182 in steam generator channel head assembly."[38] In this section EPRI acknowledges that 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 AMRs. EPRI also admits that there are

[37] EPRI, Nuclear Sector Roadmaps (January 2012) at p. 7 (Materials Aging And Degradations, Action Plan Roadmap Summary).

[38] EPRI, Nuclear Sector Roadmaps (January 2012) at p. 36-38 (In Use: Aging Management of Alloy 600 and Alloy 82/182 in the Steam Generator Channel Head Assembly).

"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 worker doses. The Roadmap proposes to develop:

(a) a review and compilation of existing information;

(b) analytical modeling to determine the maximum stress distributions in steam generators;

(c) more effective inspection techniques;

(d) updating of the Steam Generator Integrity Assessment Guidelines;

(e) address tubesheet cladding crack propagation; and

(f) perform mockup testing to determine possible repair techniques.

Aspects of this proposal are included in EPRI's 2012 Research Portfolio under the auspices of the Steam Generator Management Program.[39] The Roadmap includes a timeline that indicates that issuance of the aging management program of Alloy 600 and Alloy 82/182 in the steam generator channel head assemblies would not be available until 2016.

Safety Issues

In addition to the potential release of radioactive materials from a steam generator that has failed its reactor coolant pressure boundary, the inspection, and certainly repair, of the divider plates, channel head assemblies and the triple points where the dividers join the tubesheets will likely result in significant worker doses.

In a recent publication, EPRI has suggested a program to:

1. Determine the integrity of the steam generator when cracks propagate to the channel head and to develop and demonstrate an inspection technique to determine if cracks exist in the channel head;
2. Use finite element modeling to determine the maximum stress distributions in a steam generator channel head assembly so that the possibility of fatigue crack growth can be determined;
3. Develop a more efficient and less dose intensity method for inspection inside the steam generator bowl;

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[39] EPRI, 2012 Research Portfolio, Steam Generator Management.

4. Update the EPRI Steam Generator Integrity Assessment Guidelines to include integrity and assessment guidance for the steam generator bowl and divider plate-channel head assembly;
5. Study crack propagation in the tubesheet cladding;
6. Determine the susceptibility of the tube to tubesheet weld to primary water stress corrosion cracking.[40]

All of these studies are addressed at both regulatory and safety considerations associated with potential cracking of the Alloy 600 material in the channel head assembly.

Entergy appears to be well acquainted with this document, issued in 2011, but has only paid lip service to its ramifications. Indian Point has not proposed a specific inspection procedure except to say that it will be guided by industry standards. Industry standards have not yet been established.

It is important to note that, from a safety point of view, Entergy has not proposed nor is it committed to the following aspects of the EPRI recommended programmatic aspects of an aging management plan to address crack initiation and propagation in the channel head assemblies including divider plates.

1. Develop a program to perform an inspection of steam generators for both IP2 and IP3 to assess the condition of the divider plate assembly.
2. Develop or implement an examination technique that is capable of detecting PWSCC in the steam generator divider plate assemblies for either IP2 or IP3.
3. Develop an analytical evaluation of the steam generator tube-to-tubesheet welds to determine a technical basis for determining if they are susceptible to PWSCC.
4. Develop a specific time sensitive program of a significant number of tube-to-tubesheet welds to determine if they are susceptible to PWSCC.
5. Develop the repair or engineering evaluations that will justify continued operation if cracks are discovered.
6. Develop an ongoing monitoring program to perform routine inspections of tube-to-tubesheet weld inspections.

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[40] EPRI, Nuclear Sector Roadmaps (January 2012) at p. 36-38 (In Use: Aging Management of Alloy 600 and Alloy 82/182 in the Steam Generator Channel Head Assembly).

7. Use the data obtained to determine the remaining life of a steam generator.

Conclusions and Opinions

It is abundantly clear that NRC, EPRI, and the nuclear power industry, including Entergy, all accept the premise that there is a high probability that PWSCC will occur in the divider plates in Westinghouse steam generators including those at Indian Point , and is likely to progress into the channel head assembly, effectively compromising the pressure boundary and allowing primary water to mix with secondary water. Further, it is well established that the industry does not know the magnitude of the problem and has no effective means of dealing with it. The most aggressive timeline for addressing the problem, proposed by EPRI, will implement an inspection protocol no earlier than 2016.

There is no question but that the occurrence of PWSCC of the divider plates and the possibility of crack propagation from the divider plate into the channel head assembly constitutes a safety issue. EPRI has admitted that the current method of inspection used in Europe leads to a high radiation dose to workers performing the inspection. If the PWSCC is not detected in a reasonable time period the possibility of crack propagation that may compromise the pressure boundary becomes a hazardous situation to the employees of a nuclear power plant and consequently to the public and to the environment.

The relicensing of nuclear power plants, some which are more than 40 years old, is essentially entering uncharted territory. As the plants age, new problems will arise and management of some of those problems will undoubtedly prove to be difficult. The PWSCC of the divider plates and the associated divider plate assemblies in PWR steam generators has only recently been identified and is only one of many potential problems that may arise.

Entergy submitted its LRA at a time when PWSCC of divider plates was only being identified in European steam generators. At that time the GALL report that was in effect predated the current Revision 2 that mandates inspection of the divider plates. However, it is good engineering practice to adopt solutions to new problems independent of the regulatory horizon, especially if they are safety related. The discovery that PWSCC of divider plates and of divider plate welds may compromise the pressure boundary in a steam generator should alert Entergy and the NRC that a potential problem exists that must be addressed. Accordingly, it is important that prior to issuing a new license to IPEC, the NRC should, as a minimum, require Entergy to adopt the requirements of the latest GALL report, including Revision 2

that mandates inspection of the bowl of Westinghouse steam generators including the divider plate associated weldments.

It is my opinion that the relicensing of IP2 and IP3 should not occur until Entergy, and the industry, show that they can effectively understand, identify, inspect and perform an assessment of the magnitude of the problem and establish mitigation and control processes to manage the incidence of PWSCC in the divider plates in the Westinghouse steam generators. The EPRI research program will hopefully answer many of the questions surrounding this potential problem but it will not be ready until 2016, well into the period of extended operation of IP2 and IP3 at Indian Point. At the present time there is inadequate knowledge to indicate that the problem can be managed at all. Until the successful development of such a program, Entergy's Aging Management Program (AMP) for the Westinghouse steam generators at Indian Point is critically flawed.

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