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 9, 2015

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REVISED PRE-FILED WRITTEN TESTIMONY OF

Dr. RICHARD T. LAHEY, JR.

REGARDING JOINT 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 RICHARD T. LAHEY, JR., PhD. regarding Joint Contention NYS-38/RK-TC-5.

Q. Please state your full name.

A. Richard T. Lahey, Jr.

Q. By whom are you employed and what is your position?

A. I am retired and am currently the Edward E. Hood Professor Emeritus of Engineering at Rensselaer Polytechnic Institute (RPI), which is located in Troy, New York.

Q. Please summarize your educational and professional qualifications.

A. I have earned the following academic degrees: a B.S. in Marine Engineering from the United States Merchant Marine Academy, a M.S. in Mechanical Engineering from Rensselaer Polytechnic Institute, a M.E. in Engineering Mechanics from Columbia University, and a Ph.D. in Mechanical Engineering from Stanford University. I have held various technical and administrative positions in the nuclear industry, and I have served as both the Dean of Engineering and the Chairman of the Department of Nuclear Engineering & Science at RPI. Previously, I was responsible for nuclear reactor safety R&D (research & development) for the General Electric Company (GE), and I have extensive experience with both military (i.e., naval) and commercial pressurized water and boiling water nuclear reactors (PWR and BWR). Also, I am a member of a number of professional societies and have served on numerous expert panels. I was also an Editor of the international Journal of Nuclear Engineering & Design, which focuses on nuclear engineering and nuclear reactor safety technology. I am widely considered to be an expert in matters relating to the design, operations, safety, and aging of nuclear power plants.

Q. Which professional societies are you a member of?

A. I am a member of a number of professional societies, including: the American Nuclear Society (ANS), where I was a member of the Board of Directors and the ANS's Executive Committee, and was the founding Chair of the ANS's Thermal-Hydraulics Division; the American Society of Mechanical Engineers (ASME), where I was Chair of the Nucleonics Heat Transfer Committee, K-13; the American Institute of Chemical Engineering (AIChE), where I was the Chair of the Energy Transport Field Committee; and the American Society of Engineering Educators (ASEE), where I was Chair of the Nuclear Engineering Division.

Q. What expert panels have you served on?

A. I have served on numerous panels and committees for the: United States Nuclear Regulatory Commission (USNRC), Idaho National Engineering Laboratory (INEL), Oak Ridge National Laboratory (ORNL), National Aeronautics and Space Administration (NASA), National Research Council(NRC) and the Electric Power Research Institute (EPRI). I am a member of the National Academy of Engineering (NAE), have been elected Fellow of both the ANS and the ASME, and have been a Fulbright-Hays, Alexander

von Humboldt and Japanese Society for the Promotion of Science (JSPS) Scholar.

A. Have you published any papers in the field of nuclear engineering and nuclear reactor safety technology?

Q. Yes. Over the last 50 years, I have published numerous books, monographs, chapters, articles, reports, and journal papers on nuclear engineering and nuclear reactor safety technology. Those articles are listed in my Curricula Vitae.

Q. Have you received any professional awards?

A. Yes, I have received many honors and awards for my career accomplishments in the area of nuclear reactor thermal-hydraulics and safety technology, including: the E.O. Lawrence Memorial Award of the Department of Energy (DOE), the Glenn Seaborg Medal of the ANS and the Donald Q. Kern Award of the AIChE.

Q. I show you what has been marked as Exhibit NYS000295. Do you recognize that document?

A. Yes. It is a copy of my Curricula Vitae, which summarizes, among other things, my experience, publications, and honors & awards.

Q. I show you what has been marked as Exhibit NYS000299 to Exhibit NYS000303, and Exhibit NYS000483. Do you recognize those documents?

A. Yes. They are copies of the seven declarations that I previously prepared to date for the State of New York in this proceeding. They include my initial declaration that was submitted in November 2007 in support of the State's petition to intervene and its initial contentions, the April 7, 2008 declaration in support of Contention NYS-26A, the September 15, 2010 declaration submitted in support of the State's supplemental bases for Contention NYS-25, the September 9, 2010 declaration submitted in support of the amended Contention NYS-26B/RK-TC-1B, the September 30, 2011 and November 1, 2011 declarations submitted in support of Joint Contention NYS-38/RK-TC-5, and the February 12, 2015 declaration submitted in support of additional bases for Contention NYS-25 and Joint Contention NYS-38/RK-TC-5.

Q. I show you what has been marked as Exhibit NYS000296. 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. This Report documents my analysis and opinions.

Q. I show you what has been marked as Exhibit NYS000297. Do you recognize that document?

A. Yes. This is a copy of a Supplemental Report that I prepared for the State of New York in this proceeding that addresses aspects of the revised fatigue analysis that Entergy and Westinghouse prepared for certain components in the Indian Point reactors. The supplemental report also documents my assessment and opinions of this work.

Q. I show you what has been marked as Exhibit NYS000294, NYS000344, and NYS000440. Do you recognize those documents?

A. Yes, those documents contain my previous pre-filed testimony filed in December 2011 and June 2012 in support of Contentions NYS-25 and NYS-26B.

Q. I also show you what has been marked as Exhibit NYS000374 and NYS000453. Do you recognize those documents?

A. Yes, I do. Those documents contain my previous pre-filed testimony filed in June 2012 and November 2012 in support of aspects of Joint Contention NYS-38/RK-TC-5.

Q. What is the purpose of your current testimony?

A. I have been retained by the State of New York State to review Entergy's application to the U.S. Nuclear Regulatory Commission (USNRC) and its Staff for two renewed operating

licenses for the nuclear power plants known as Indian Point Unit 2 and Unit 3. I have reviewed the License Renewal Applications (LRAs) and subsequent filings by Entergy and the USNRC Staff. My declarations and report discuss my concerns and opinions about issuing twenty-year extended operating licenses for these facilities. My testimony seeks to identify and discuss some age-related safety concerns which have not yet been addressed by Entergy. In my opinion these concerns must be resolved to assure the health and safety of the American public, particularly those in the vicinity of the Indian Point reactors.

Also, the purpose of my testimony here is to provide support for, and my views on, aspects of New York's and Riverkeeper's Joint Contention NYS-38/RK-TC-5 ("NYS-38/RK-TC-5"), which was admitted for litigation by the Atomic Safety Licensing Board. Contention NYS-38/RK-TC-5 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 USNRC 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 to age related degradation caused by embrittlement and fatigue as being inadequate.

My testimony also critiques Entergy's proposed approach to address the age-related degradation caused by metal fatigue and for deferring or not disclosing various details of its approach.

My testimony further critiques Entergy’s proposed approach towards the age-related degradation of various components in Indian Point's steam generators during the requested twenty year period of extended operation.

In each of these areas, Entergy’s approach is inadequate or defers and avoids important aspects of an aging management program.

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

A. Yes.

Q. What is the source of those materials?

A. I have reviewed documents prepared by government agencies, Entergy, Westinghouse, the utility industry, or its associations (e.g., EPRI), and various related text books and peer-reviewed articles.

Q. I show you Exhibits NYS00146A-C, NYS00147A-D, NYS000160, NYS000161, NYS000195, NYS000304 through NYS000369, NYS000484 through NYS000525, and NYS000533, NYS000539, NYS000542, NYS000544, NYS000548, NYS000549, NYS000554, NYS000558 through NYS000561. Do you recognize these documents?

A. Yes. These are true and accurate copies of some of the documents that I referred to, used, or relied upon in preparing my report, declarations, previous testimony, 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.

Q. I direct your attention to latter part of your 2011 Report (Exh. NYS000296) entitled “Reference Documents,” which contains a list of documents. Would you describe that list?

A. Yes that section of the Report lists various salient documents that I referred to, used or relied on, in preparing my Report and the Supplemental Report.

Q. I direct your attention to the latter part of your February 12, 2015 Declaration (Exh. NYS000483) entitled “Reference Documents,” which contains a list of documents. Would you describe that list?

A. Yes, that section of the Declaration lists various additional salient documents that I referred to, used or relied on, in preparing my February 12, 2015 Declaration.

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.

The Indian Point Reactors

Q. Are you familiar with the power reactors that are the subject of this proceeding?

A. Yes.

Q. Would you briefly describe them?

A. Entergy operates two nuclear power reactors that are located in northern Westchester County near the Village of Buchanan. The operating nuclear reactors are known as the Indian Point Unit 2 and Indian Point Unit 3 reactors. These Westinghouse-designed plants are 4-loop pressurized water reactors (PWRs), and they are currently rated at power levels of 3,216.4 MW_t. Entergy also owns another reactor at the same site. That reactor is known as the Indian Point Unit 1 reactor;

however, that reactor has been shut down and no longer produces power.

Operation of a Pressurized Water Reactor

Q. Would you briefly describe the design and operation of a pressurized water reactor?

A. Pressurized water nuclear reactors have water (i.e., the primary coolant) under high pressure flowing through the core in which heat is generated by the fission process. The core is located inside a reactor pressure vessel (RPV). This heat is absorbed by the coolant and then transferred from the coolant in the primary system to lower pressure water in the secondary system via a large heat exchanger (i.e., a steam generator) which, in turn, produces steam on the secondary side. These steam generator systems, which are part of the plant’s Nuclear Steam Supply System (NSSS), are located inside a large containment structure. As 4-loop units, Indian Point Unit 2 and Indian Point Unit 3 each have four steam generators. After leaving the containment building, via main steam piping, the steam drives a turbine, which turns a generator to produce electrical power.

The reactor pressure vessel is a large steel container that holds the core (i.e., the nuclear fuel); it also serves as a key part of the primary coolant's pressure boundary.

As the name Pressurized Water nuclear Reactor (PWR) suggests, this reactor design uses a pressurizer on the primary side that performs several functions. In particular, it maintains the operating pressure on the primary side of the nuclear reactor and accommodates variations in reactor coolant volume for load changes during reactor operations, and during reactor heat-up and cool-down. The reactor coolant also moderates the neutrons produced in the core since a pressurized water nuclear reactor will not function unless the neutrons are moderated (i.e., slowed down due to collisions with the hydrogen molecules in the primary coolant).

Q. I show you what has been marked as Exhibit NYS000304. Do you recognize it?

A. Yes. It is a schematic diagram from a USNRC document that identifies the relative location of various components in a pressurized water nuclear reactor type of power plant including, from the inside to the outside, the reactor core, reactor pressure vessel, pressurizer, steam generator, containment structure, turbine, and associated piping. The diagram also

identifies the various materials that are used or contained in those components.

Q. I show you what has been marked as NYS000485 and ask you to turn to page 2, Figure 1. Do you recognize that?

A. Yes, this is a similar schematic diagram from a 2014 U.S. Department of Energy (USDOE) document; it is more recent than NYS000304 and contains additional information concerning the material that makes up the various components.

Reactor Pressure Vessel Internals

Q. I show you what has been marked as Exhibit NYS000306. Do you recognize it?

A. Yes. It is a series of schematic diagrams or figures, including Figure 3-5, from an Electric Power Research Institute (EPRI) document known as MRP-227 that identifies various components within pressurized water nuclear reactor designed by the Westinghouse Company. The title of Figure 3-5 is, “Overview of typical Westinghouse internals.”

Q. Please describe what is encompassed by the term “reactor pressure vessel (RPV) internals”?

A. The term “reactor pressure vessel internals” (i.e., RVIs) includes various structures, components, and fittings inside the reactor pressure vessel including the: core barrel

(and its welds), core baffle, intermediate shells, former plates, lower core plate and support structures, clevis bolts, fuel alignment pins, thermal shield, the lower support column and mixer, upper mixing vanes, and the upper/lower core assemblies and support column, and the control rods and their associated guide tubes, plates, and welds. Reactor pressure vessel internals (RVIs) also include the bolts that hold various components together or to other components including: the baffle-to-baffle bolts, the core barrel-to-former bolts, and baffle-to-former bolts as well as the welds or weldments that hold sections of these components together.

Q. Was the aging management of RVIs initially considered as part of the LRA for the Indian Point facilities?

A. No, it was not. Fortunately, during the course of these ASLB hearings on Indian Point the USNRC has now recognized and highlighted the importance of RVIs [see, e.g., USNRC Report, "Final Interim Guidance LR-ISG-2011-04 Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors," NRC-ISG-2011-04 (May 28, 2013) (NYS000524)].

Q. Are there any reactor components that you believe should be considered as reactor vessel internals, but that Entergy has claimed are not reactor vessel internals?

A. Yes. Entergy has argued that the control rods are not reactor vessel internals. However, the control rods and their associated guide tubes, plates, pins and welds are located in the core region of the RPV, and the control rods are inserted into the RPV through the upper head through so-called stub tubes. The function of the control rods is to absorb excess fission neutrons (i.e., those not needed to achieve a chain reaction) so that the power level of a reactor can be controlled. Accordingly, the control rods and associated components are very important RPV internals and their integrity is an extremely important safety concern. While the control rods are moving parts and can be replaced as required, many of the other associated components are not moving parts and are not normally replaced. In any event, if a shock load occurs (e.g., during a LOCA or severe earthquake) any of these seriously embrittled structures may fail and lead to degraded core cooling. Thus, in my opinion, omitting the control rod assemblies and associated fittings from an RPV internals (RVIs) aging management program is a serious and indefensible omission.

Q. Coming back to Exhibit NYS000306, would you describe the other diagrams?

A. Yes. They are a collection of additional schematic figures from the Electric Power Research Institute's Report MRP-227 that provide additional detail concerning various reactor pressure vessel internals and their location within the reactor pressure vessel. The reactor pressure vessel internals shown include the control rod guide tube assembly, the control rod guide cards, guide tube support pins, the control rods, baffles, formers, baffle-former assemblies, baffle-to-former bolts, corner edge bracket baffle to former bolts, core barrel to former bolts, baffle plate edge bolts, core support structures, and various weldments, including welds within the reactor pressure vessel for the core barrel plates.

Overview

Q. In your expert opinion what is the most important age-related safety issue associated with the relicensing of the two Indian Point reactors?

A. My over-arching concern relates to Entergy's "silo" type approach to evaluating the impact of various aging mechanisms such as embrittlement and fatigue, and the company's failure to consider, as part of plant safety analyses, the potential consequence of unanticipated shock loads (e.g., those due to design basis accidents) on severely fatigued and

embrittled components. Entergy implicitly assumes that there is no interplay between the various material aging degradation phenomena and that degraded components will have no impact on the plants' ability to safely operate, particularly during unanticipated shock loads. For example, Entergy's fatigue evaluations, performed by Westinghouse using the WESTEMS computer code, used the metric CUF_{en} to appraise environmentally assisted fatigue in various reactor components. However, these evaluations were quasi-static low and high cycle fatigue evaluations that considered neither the effect of neutron-induced embrittlement nor the combined effects of fatigue damage and other degradation mechanisms such as radiation enhanced corrosion-induced cracking and primary water stress corrosion cracking (PWSCC). In both the fatigue analyses and the plant safety analyses, it was implicitly assumed that fatigue weakened and embrittled structures, components and fittings would respond to shock loads in the same way as if they were ductile, which is simply not true. Also, no error analyses were presented to quantify the WESTEMS predictions for the various internals, piping systems and fittings even though some of them were extremely close to the CUF_{en} = 1.0 failure limit. In any event, under these circumstances, various operational and accident-

induced shock loads could cause failures well before the fatigue limit is reached (i.e., when CUF_{en} < 1.0), and therefore reliance on inspection-based fatigue monitoring does not provide adequate assurance that the degraded components will not fail.

Once again, the most serious short-coming of this “siloing” approach is that synergistic interactions between radiation-induced embrittlement, corrosion-induced cracking, and fatigue-induced degradation mechanisms have not been considered. For example, neither Entergy’s license renewal application nor its proposed aging management plan consider the potential for, or the consequences of, fatigue-induced failure of seriously embrittled reactor pressure vessel internals (RVIs). Also, when the plant’s safety analyses were done by Entergy it was implicitly assumed that the in-core geometry would remain intact during postulated accidents. Unfortunately, unlike ductile metals, seriously embrittled and fatigued RPV internals may not be able to survive the shock loads associated with significant seismic events or the pressure and/or thermal shock loads induced by various accidents and severe operational transients. If not, they can fail and relocate, possibly causing core blockages that degrade core cooling and may lead to core melting and massive radiation releases.

Entergy has an obligation to show that its plants can be safely operated beyond their 40 year design lives. I believe that this will require much more study and analysis than has been presented to date to identify any limiting RPV internals that require repair or replacement. Nevertheless, this must be done to verify that the two Indian Point reactors can be safely operated for another 20 years beyond the design life of these plants.

Q. What do you mean by synergistic interactions between aging-related degradation mechanisms?

A. I mean that the concurrent exposure of reactor components – especially RVI components – to multiple aging mechanisms that occur in a reactor core (including fatigue, irradiation embrittlement, and corrosion) may result in cumulative material degradation that exceeds the predicted combined degradation for each aging mechanism acting alone.

Q. Are there any studies or reports that support your concern regarding synergistic aging effects?

A. Yes. However, the rather complex and interacting metal degradation mechanisms associated with fatigue, irradiation and corrosion interact is still an area of active research (e.g., how fatigue-induced cracks propagate in an

embrittled, as opposed to ductile, metal structure). In fact, the Department of Energy (DOE) and USNRC, in conjunction with various national laboratories, have recently embarked on an ambitious R&D program to understand and resolve issues related to these interacting and synergistic effects [NUREG/CR-7153, Vol. 2, "Expanded Materials Degradation Assessment (EMDA), Aging of Core Internals and Piping Systems" (October 2014), at 1-5 (Exh. NYS00484A-B)]. In addition, the federal government has also embarked on a fairly large research program, known as the Light Water Reactor Sustainability Program, which includes research into whether the different materials and LWR components can continue to perform their intended function during the extended operation of a nuclear reactor. [DOE, Light Water Sustainability Program, Material Aging and Degradation Technical Program Plan (August 2014) (Exh. NYS000485)].

Nevertheless, it is well known that, "the effects of embrittlement, especially loss of fracture toughness, make existing cracks in the affected materials and components less resistant to growth" [USNRC Letter, Grimes to Newton, at 16 (Feb. 10, 2001) (Exh. NYS000324); see Stevens, Gary L., Presentation to the ACRS on "Technical Brief on Regulatory Guidance for Evaluating the Effects of Light Water Reactor

Coolant Environments in Fatigue Analyses of Metal Components" (December 2, 2014), at 56-58 (Exh. NYS000486); Chopra, O.K., "Degradation of LWR Core Internal Materials due to Neutron irradiation," NUREG/CR-7027 (Dec. 2010) (Exh. NYS000487)], and, "irradiation embrittlement decreases the resistance to crack propagation" [Westinghouse Owners Group WCAP-14577 Rev. 1-A Report, at 3-2 (March 2001) (Exh. NYS00307A-D)]. Moreover, a recent report, prepared by Argonne National Laboratory for the USNRC, acknowledges, with respect to cast austenitic stainless steels (CASS), that "a combined effect of thermal aging and irradiation embrittlement could reduce the fracture resistance even further to a level neither of these degradation mechanisms can impart alone" [Chen, et al., "Crack Growth Rate and Fracture Toughness Tests on Irradiated Cast Stainless Steels," NUREG/CR-7184 (Revised December 2014), at xv (Exh. NYS00488A-B)]. Indeed, nuclear industry groups have now recognized the potential for synergistic aging effects in CASS RVI components [EPRI, Slides, Industry-NRC Meeting on CASS Screening Criteria for Thermal and Irradiation Embrittlement for BWR and PWR Internals" (July 15, 2014) (Exh. NYS000489)].

Q. Are synergistic aging effects limited to CASS components?

A. No. All components within the RPV are subject to multiple aging degradation mechanisms. Different materials may undergo aging in different ways, but all materials are susceptible to synergistic effects.

Q. Are these synergistic aging effects fully understood?

A. Not at all. Multiple recent reports and studies from USNRC, DOE, and associated contractors recognize the lack of understanding of the interrelationship between embrittlement, high or low cycle fatigue, and shock loads for highly fatigued and/or embrittled components made of CASS, non-cast stainless steels, or other alloys. In addition, the consequences of the interaction of embrittlement, fatigue, and the corrosion-induced degradation of various reactor pressure vessel internals (RVI), and safety-related components/systems during shock loads, remains unknown [see, e.g., NUREG/CR-6909 Rev. 1 (March 2014 (draft) (Exh. NYS000490)), at 11 ("it is not possible to quantify the impact of irradiation on the prediction of fatigue lives in PWR primary water environments compared to those in air."); NUREG/CR-7153, Vol. 2, "Expanded Materials Degradation Assessment (EMDA), Aging of Core Internals and Piping Systems" (October 2014), at 3 (Exh. NYS00484A-B)]. The Argonne National Laboratory report described above states that, "no data are

available at present with regard to the combined effect of thermal aging and irradiation embrittlement" on CASS [Chen, et al., NUREG/CR-7184, at xv (Exh. NYS00488A-B); see also Chopra, O.K., "Degradation of LWR Core Internal Materials due to Neutron irradiation," NUREG/CR-7027 (Dec. 2010) (Exh. NYS000487)]. As noted before, the same is also true for the interaction of irradiation-induced embrittlement, corrosion, and fatigue of non-cast stainless steel RVIs.

A recent paper presented at an MPA Seminar in Stuttgart, Germany confirms that, at present, the USNRC staff does not have a clear solution to the challenges posed by synergistic age-related degradation mechanisms [Stevens, Gary L., et al., "Observations and Recommendations for Further Research Regarding Environmentally Assisted Fatigue Evaluation Methods," 40th MPA-Seminar, Materials Testing Institute, University of Stuttgart, Stuttgart, Germany (October 6-7, 2014) (Exh. NYS000491)]. A recent draft report on the "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials," prepared by Argonne National Laboratory (ANL) and USNRC Staff, recognizes the "inconclusive" nature of existing data on the synergistic effects of irradiation and fatigue, and other aging mechanisms in LWR environments, and concludes that, "additional fatigue

data on reactor structural materials irradiated under LWR operating conditions are needed." [NUREG/CR-6909, Rev. 1 (March 2014 [draft]), at 11 (Exh. NYS000490)]. Furthermore, during a "Briefing on Subsequent License Renewal" to the USNRC, the USNRC's Chief of the Corrosion and Metallurgy Branch, Dr. Mirela Gravila, testified that the Piping and Core Internals Panel had recognized "significant gaps" in our technical knowledge with respect to the effects of irradiation-induced degradation of the RVI components [Trans. of Briefing on Subsequent License Renewal, at 77 (May 2014) (Exh. NYS000492)].

Q. With respect to the aging management of nuclear facilities, how has the USNRC responded to these embrittlement concerns with respect to its synergistic effects on fatigue?

A. Notwithstanding the significant concerns and considerable uncertainty regarding synergistic aging effects, the USNRC has so far declined to require that plant operators repair or replace degraded systems, structures, and fittings, opting instead to manage aging through periodic inspections, and the use of an empirical environmental factor method (F_{en}) for fatigue life when evaluating the in situ degradation of structures and components [Stevens, et al., (October 2014), at 10 (Exh. NYS000491)], a method which is not necessarily

conservative and one that certainly does not address all the synergistic effects (e.g., embrittlement) that New York State is concerned about.

Q. Would you please explain in more detail the various degradation mechanisms that you are concerned with?

A. Yes, let me begin with embrittlement.

Embrittlement

Q. Would you explain what embrittlement is?

A: Embrittlement can occur due to various mechanisms, but herein it refers primarily to the change in the mechanical properties (and structure) of materials, such as metals, that can occur over time under the bombardment of neutrons. The degree of exposure to neutrons is normally expressed in terms of a "fluence" (i.e., the neutron flux times the duration of the irradiation process). The extended exposure to neutrons causes damage to metals and makes them more brittle so that they become more susceptible to failures due to cracking or fracture. In particular, this radiation-induced damage results in a decrease in fracture toughness and ductility.

Embrittlement is an age-related degradation mechanism whereby a component experiences a decrease in ductility, a loss of facture toughness, and an increase in yield strength. While

the initial aging effect is loss of ductility and toughness, unstable crack propagation is the eventual adverse aging effect if a crack is present and the local applied stress intensity is sufficient. Moreover, when subjected to a sufficient load, a component which has been highly embrittled by neutron irradiation may experience sudden, brittle fracture well before a surface crack is detected. This is a particular problem for the large pressure and/or thermal shock loads associated with postulated accidents. For this reason, USNRC regulations set forth at 10 C.F.R. § 50.61 impose fracture toughness requirements and/or operating parameters to prevent brittle facture of reactor pressure vessels. Indeed, NUREG-1800, Rev. 2 (Table 4.1-3) (Exh. NYS000161) identifies reduced fracture toughness of reactor vessel internals as a candidate for a time limited aging analysis. Because loss of ductility due to radiation embrittlement was not considered in the design of the stainless steel reactor vessel's internal components (RVIs), it is all the more important to evaluate the degree of embrittlement of RVIs during license renewal review. [Chopra, O., Public Comment on NRC-2010-0180-0001, Availability of Draft NUREG-1800, Revision 2 and Draft NUREG-1801, Revision 2 (June 9, 2010) (Exh. NYS000493)].

The Consequences of Embrittlement

Q. Is embrittlement a concern for pressurized water nuclear reactors?

A. Yes. For a pressurized water nuclear reactor to operate safely, the metals involved need to be sufficiently ductile, which means that they must be able to deform without experiencing failures. When metals, such as steel, experience a significant neutron fluence, which happens to the materials in close proximity to the reactor core (e.g., the steel reactor pressure vessel's interior wall and the associated RVIs), the temperature required for them to maintain sufficient ductility is increased as the metal is continually bombarded by a neutron flux. The temperature at which there is a marked change from ductile to non-ductile behavior is often called the "nil ductility temperature" (NDT). However, even for temperatures well above the NDT, the irradiated metals continue to be damaged and further embrittled due to the neutron bombardment. Indeed, the neutron damage will not be annealed out (i.e., be neutralized) unless the damaged metals are taken to temperatures that are well above PWR operating temperatures.

Q. Could embrittlement impact a nuclear reactor's ability to respond to a transient, shock load, or an accident scenario?

A. Yes. Reduced ductility (or embrittlement) will adversely affect a PWR's ability to withstand severe seismic events and pressure and/or thermal shock loads, and thus there is a threat to the integrity of highly embrittled internal structures in the reactor pressure vessel. For example, during a recent meeting regarding Indian Point, a member of the Advisory Committee on Reactor Safeguards Plant License Renewal Subcommittee expressed concern that embrittled RVI components could fail during a seismic event. [Trans. of Advisory Committee on Reactor Safeguards, Plan License Renewal Subcommittee, at 209-210 (April 23, 2015) (Exh. NYS000526)].

Various accidents and abnormal transients can expose a reactor pressure vessel and its internal structures, components and fittings (i.e., RVIs) to significant pressure and/or thermal shock loads. If the reactor pressure vessel's internal structures (RVIs) are sufficiently degraded due to corrosion-induced cracking, fatigue and/or radiation-induced embrittlement, these shock loads can have significant consequences. Indeed, the resultant stresses from such accidents may cause the RVIs to fail structurally and relocate within the RPV. If so, the ability to effectively cool the decay heat in the core may be lost due to core blockage.

One well known safety concern associated with embrittlement is the ability of metals to withstand a thermal shock event. A thermal shock can occur in various ways, for example: (1) during loss of coolant accidents (e.g., postulated primary or secondary side LOCAs), or, (2) during a reactor SCRAM (i.e., a rapid insertion of the control rods which terminates the nuclear chain reaction). A particularly bad LOCA event is one in which there is a rapid depressurization of the secondary side (e.g., a steam line break) which causes a reactor SCRAM and thus a rapid cooling of the primary coolant via the steam generators. This type of accident can lead to severe thermal shock of the reactor pressure vessel and the associated RPV internals (RVIs).

Severe thermal shocks can also occur during a design basis accident (DBA) LOCA event (i.e., a complete breach of main coolant piping on the primary side), which rapidly depressurizes the primary side and leads to the injection of relatively cool emergency core coolant into the reactor pressure vessel (e.g., from the accumulators). As noted previously, this may lead to the sudden fracture and relocation of highly embrittled RVI structures, components and fittings, and thus impede their ability to perform their intended functions, and adversely impact their core-cooling functions. In the past, most of the

USNRC's attention has been focused on the integrity of the reactor pressure vessel. However, the RVIs are much less massive and are much closer to the core, and thus they suffer a lot more radiation damage and embrittlement. Notably, the USNRC's fluence threshold for irradiation embrittlement of the reactor pressure vessel beltline is 1×10^{17} n/cm^2 [10 C.F.R. Part 50, Appendix G; USNRC Regulatory Issue Summary 2014-11 (Exh. NYS000494)]. In contrast, Westinghouse RVIs can experience fluence in the range 1×10^{21} to 5×10^{22} n/cm^2, or higher. Thus, RVI are subject to neutron irradiation which is several orders of magnitude higher than levels known to cause reduced fracture toughness in reactor pressure vessel materials. [MRP 191 (Nov. 2006), Table 4-6 (Exh. NYS000321)].

Q. Are there other effects of embrittlement that can compromise the ability to maintain a coolable core geometry in the event of thermal or decompression shock loads following a DBA LOCA?

A. Yes. As described previously, the synergistic interactions between the metal degradation mechanisms associated with fatigue, irradiation and corrosion are not well understood. However, it is well known that irradiation embrittlement reduces fracture toughness and decreases the resistance to crack

propagation in the metal. [USNRC Letter, Grimes to Newton, at 16 (Feb. 10, 2001) (Exh. NYS000324); Westinghouse Owners Group, WCAP-14577, Rev. 1-A Report (March 2001), at 3-2 (Exh. NYS000341)].

The radiation-induced damage to some RPV internals can be extensive, since they can experience a neutron fluence of at least 10^{23} n/cm^2 at neutron energy (E) levels of E > 1 MeV (i.e., > 100 dpa) [Was (2007) (Exh. NYS000339); EPRI, Dyle (2008) (Exh. NYS000322); WOG WCAP-14577 Rev. 1-A Report (March 2001) (Exh. NYS000341)] by the end of life (EOL) for extended operations. According to one study, the crack growth rate for materials irradiated to only 3×10^{20} n/cm^2 fluence can be up to 40 times higher than that for unirradiated materials [Chopra, O.K., "Degradation of LWR Core Internal Materials due to Neutron irradiation," NUREG/CR-7027 (Dec. 2010) (Exh. NYS000487)]. It should be stressed that the fluence experienced by some RPV internals is about four orders of magnitude (i.e., ~ 10,000 times) larger than will be experienced by the inner wall of the reactor pressure vessel by the end of life (EOL) for extended operations [Rao, A.S. (USNRC), "Irradiation Assisted Degradation of LWR Core Internal Materials; Brief Review," (Apr. 14, 2015) (Exh. NYS000495)]. Thus, the RPV internals will be much more

embrittled than the RPV walls, which have historically been the focus of USNRC embrittlement concerns. A highly embrittled RPV internal component subjected to a severe earthquake or thermal/decompression shock, could thus fail and relocate within the RPV, which, in turn, could result in the loss of a coolable core geometry.

GALL, Revision 1

Q. I show you a document marked as Exhibit NYS00146A-C and entitled NUREG-1801, Revision 1, the Generic Aging Lessons Learned Report, GALL. Are you familiar with this document?

A. Yes.

Q. When did the USNRC Staff release that document?

A. In September of 2005.

Q. Does NUREG-1801, Revision 1 include an aging management program (AMP) for reactor pressure vessel internals in a pressurized water nuclear reactor?

A. No. Revision 1 of NUREG-1801 includes no aging management program description for PWR reactor pressure vessel internals (RVIs). NUREG-1801, Revision 1, Section XI.M16, entitled “PWR Vessel Internals,” instead defers to the guidance provided in Chapter IV line items as appropriate. The Chapter IV line item guidance simply recommends actions to:

> "..(1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and, (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval."

That statement appears a number of times in GALL, Revision 1, Chapter IV. For example, that statement appears on pages IV B2-4, IV B2-5, IV B2-8, IV B2-14, IV B2-16, and IV B2-17 with respect to the embrittlement of reactor pressure vessel internals.

Q. I show you what has been marked as Exhibit NYS000313, which is a July 15, 2010 submission from Entergy that forwarded a document to the Atomic Safety and Licensing Board (ASLB). Do you recognize the attachment to that submission?

A. Yes, it contains a copy of a July 14, 2010 communication, NL-10-063, from Entergy to the USNRC's document control desk that concerns embrittlement of reactor pressure vessel internals. In addition, NL-10-063 contains an "Attachment 1."

Q. Directing your attention to NL-10-063, Attachment 1, page 84 of 90, what does Entergy say there about GALL, NUREG-1801, Revision 1 and reactor pressure vessel internals?

A. Entergy states that "Revision 1 of NUREG-1801 includes no aging management program description for PWR reactor vessel internals."

Standard Review Plan, Revision 1

Q. I show you a document marked as Exhibit NYS000195 that is entitled NUREG-1800, Revision 1, USNRC Staff's Standard Review Plan (SRP). Are you familiar with this document?

A. Yes.

Q. When did the USNRC Staff release that document?

A. In September of 2005.

Q. Does the Standard Review Plan, Revision 1 recognize that the reactor pressure vessel internals could experience embrittlement?

A. Yes, the Standard Review Plan, Revision 1 at § 3.1.2.2.6 recognized that reactor pressure vessel internals could experience embrittlement.

Q. Would you elaborate?

A. In § 3.1.2.2.6 on page 3.1-5, the Standard Review Plan, Revision 1 states, "Loss of fracture toughness due to

neutron irradiation embrittlement and void swelling could occur in stainless steel and nickel alloy reactor vessel internals components exposed to reactor coolant and neutron flux."

Q. Did the Standard Review Plan, Revision 1 make provision for an aging management program (AMP) for reactor pressure vessel internals in a pressurized water reactor?

A. No, it did not. At § 3.1.3.2.6, the Standard Review Plan, Revision 1 stated that "The GALL Report recommends no further evaluation of programs to manage loss of fracture toughness due to neutron irradiation embrittlement . . ." That statement is on page 3.1-12. This is also confirmed by § 3.1.2.2.6 and Table 3.1-1 which made clear that GALL and the Standard Review Plan did not propose a specific aging management plan and repeated the language from GALL about staying up to date with industry discussions about embrittlement and submitting a plan in the future for consideration by USNRC Staff.

<u>Entergy's Opposition to NYS Contention 25</u>

Q. In November 2007 you submitted a declaration in support of the State of New York's Contention 25 concerning embrittlement. Do you know if Entergy submitted a response?

A. Yes, Entergy did.

Q. What did Entergy say in its response?

A. Entergy opposed the admission of Contention 25 and presented various arguments. One of Entergy's principal arguments was that stainless steel components are not susceptible to a decrease in fracture toughness as a result of neutron embrittlement. Entergy stated: "The core barrel, thermal shield, baffle plates and baffle former plates (including bolts) are, however, made of stainless steel and are not susceptible to a decrease in fracture toughness as a result of neutron embrittlement." [Entergy January 22, 2008 Answer at 137]. This is a surprisingly uninformed statement from the operators of a nuclear power plant. Anyway, while this may have been a popular belief many years ago, it is incorrect.

GALL, Revision 2

Q. I show you a document marked as Exhibit NYS00147A-D that is entitled Revision 2 of the Generic Aging Lessons Learned Report or GALL. Are you familiar with this document?

A. Yes, I have reviewed it.

Q. When did the USNRC Staff release that document?

A. December of 2010.

Q. What does GALL, Revision 2 say about embrittlement?

A. GALL, Revision 2 includes the following statement: "Neutron irradiation embrittlement - Irradiation by neutrons results in embrittlement of carbon and low-alloy steels. It may produce changes in mechanical properties by increasing the tensile and yield strengths with a corresponding decrease in fracture toughness and ductility. The extent of embrittlement depends on the neutron fluence, temperature, and trace material chemistry." [GALL, Revision 2 at page IX-34 (Exh. NYS000147)]. I note that the phrase "low-alloy steels" includes stainless steel.

Q. Does GALL, Revision 2 discuss the aging degradation of PWR reactor pressure vessel internals?

A. Yes. Chapter IV and Chapter XI now discuss the aging degradation of PWR reactor pressure vessel internals through various aging mechanisms including embrittlement.

Q. What does GALL, Revision 2, Chapter IV state about embrittlement of PWR reactor pressure vessel internals?

A. Chapter IV summarizes which reactor vessel internals are subject to embrittlement (and other aging mechanisms) and is organized by nuclear steam supply system vendors. There is a section ("B2") concerning components in nuclear steam supply systems designed by Westinghouse, the company that designed

those systems at Indian Point Unit 2 and Unit 3. That section recognizes that reactor pressure vessel internals in Westinghouse-designed PWRs are subject to degradation due to embrittlement. It further recognizes that for Westinghouse PWRs, reactor pressure vessel internal components made of stainless steel and nickel alloy experience a "loss of fracture toughness due to neutron irradiation embrittlement." These statements appear on GALL, Revision 2 at pages IV B2-2 to IV B2-14.

Q. Directing your attention to GALL, Revision 2, pages IV B2-12 and IV B2-13, do you see the items numbered IV.B2.RP-268 and IV.B2.RP-269?

A. Yes, those items concern reactor vessel internal components in "inaccessible locations."

Q. What is the aging effect or mechanism of concern?

A. There are a number including loss of fracture toughness due to neutron irradiation embrittlement, void swelling, and corrosion-induced cracking.

Q. And these are inaccessible RPV internals in Westinghouse PWRs?

A. Yes.

Q. Does GALL Revision 2 make any suggestions about the reactor pressure vessel components that are located in inaccessible locations?

A. Yes, it recommends an “evaluation” of the internals located in inaccessible locations if other similar components “indicate aging effects that need management.”

Q. You mentioned that GALL, Revision 2, Chapter XI also discussed reactor pressure vessel internals. Where is that discussion?

A. Chapter XI contains a section numbered XI.M16A entitled “PWR Vessel Internals,” which starts at page XI M16A-1.

Q. Would you summarize that section?

A. Yes. Like Chapter IV, it recognizes that PWR reactor pressure vessel internals experience a “loss of fracture toughness due to either thermal aging or neutron irradiation embrittlement,” as well as other age-related degradation mechanisms, such as various corrosion-induced cracking mechanisms. It provides a template for license renewal applicants to include in their license renewal applications that discusses embrittlement and other aging mechanisms that degrade reactor pressure vessel internals. It recommends that applicants propose an inspection plan that is then submitted to

the USNRC Staff for review and approval. The template is derived from a document prepared as a result of an effort coordinated by the Electric Power Research Institute (EPRI) to develop guidelines concerning the inspection of reactor pressure vessel internals.

Q. Directing your attention to GALL, Revision 2, page XI M16A-3, do you see item 3, titled “Parameters Monitored/ Inspected”?

A. Yes.

Q. Would you summarize that section?

A. Yes, this section provides recommendations for an inspection plan for reactor pressure vessel internals, and specifically what I would describe as the scope or focus of the plan. This section is titled “Parameters Monitored/Inspected” and states that the recommended inspection “program does not directly monitor for loss of fracture toughness that is induced by thermal aging or neutron embrittlement.” Instead, it states that the embrittlement of reactor pressure vessel internal components is indirectly monitored through visual or volumetric inspection techniques that look for cracking (i.e., the detection of failures <u>after</u> they have occurred). It is important to note that the focus of this document is on non-

destructive testing (NDT) and non-destructive evaluation (NDE) techniques. In particular it does not consider the implications on core coolability subsequent of any shock load induced failures of highly degraded RPV internals.

MRP-227, Revision 0

Q. I show you a document marked as Exhibit NYS00307A-D. Do you recognize it?

A. Yes, I have reviewed it. It is a copy of the document prepared as a result of the nuclear industry's efforts coordinated by the Electric Power Research Institute (EPRI).

Q. What is the title of that document?

A. The document's title is, "Material Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0), 1016596, Final Report, December 2008." Unfortunately, as I discussed previously, it is focused on NDT and NDE inspection techniques rather than my aging-related safety concerns.

MRP-227-A

Q. I show you a document marked as Exhibit NRC00114A-F [MRP 227-A]. Do you recognize it?

A. Yes, I have reviewed it. It is the version of the MRP-227, Revision 0 [Exh. NYS00307A-D] that was reviewed and

approved by the USNRC Staff, and includes various edits and additional materials in response to USNRC Staff comments and questions. It was submitted to the USNRC in January 2012. Unfortunately, as I have noted previously, it is focused on NDT and NDE inspection techniques rather than my aging-related safety concerns.

Q.   Does MRP-227-A say anything about embrittlement?

A.   Yes. The industry has recognized that, “there are no recommendations for inspection to determine embrittlement level because these mechanisms cannot be directly observed” [MRP-227-A, Footnote 1 for Table 3-3 (December 2011) [Exh. NRC00114A-F]. That is, the level of degradation due to embrittlement of RPV internal components, fittings and structures, and their ability to withstand fatigue and shock loads cannot be determined using the inspection techniques proposed in MRP-227-A.

Q.   Do you have specific concerns with the approach to aging management for reactor vessel internals set forth in MRP-227-A?

A.   Yes. MRP-227-A is an inspection-based aging management plan, which I believe is inadequate. To begin with, depending on the type of component, inspection may not be possible for the entire component, or for the entire set of such

components, given the location of the components and their possible inaccessibility. For example, a visual or ultrasonic inspection of the external head of a bolt does not necessarily provide insight into the integrity of the remainder of the bolt which is not visible. Moreover, an inspection focused on one type of age-related degradation mechanism does not necessarily work for another ongoing degradation process that is affecting the same component, and the effect of shock loads on the integrity of various RVIs and primary pressure boundary systems is certainly not addressed by inspections. An inspection-based approach to aging management, such as the one developed by the nuclear industry in MRP-227 and condoned by USNRC in MRP-227-A, is useful but it fails to account for the possibility that highly embrittled and fatigued RVI components may not have signs of degradation that can be detected by an inspection, but such weakened components could nonetheless fail as a result of a severe seismic event or thermal or pressure shock load. In short, many of my concerns about the cumulative and ongoing synergistic aging effects are not adequately addressed by MRP-227-A.

Entergy’s License Renewal Application

Q. Directing your attention to Entergy’s 2007 License Renewal Application (LRA), did you find any indication in the LRA that Entergy recognized that embrittlement could affect the reactor pressure vessel?

A. Yes.

Q. Where was that?

A. The License Renewal Application at § 3.1.2.1.1 recognized that reactor pressure vessels are constructed of the following materials:

- carbon steel;
- carbon steel with stainless steel or nickel alloy;
- cladding;
- nickel alloys; and,
- stainless steel.

The same LRA section further recognized that reactor pressure vessels experience the following aging effects that require management:

- cracking;
- loss of material; and,
- reduction of fracture toughness, a term which encompasses embrittlement.

Q. Did you find any indication in the LRA that Entergy has now recognized that embrittlement could affect reactor pressure vessel internals?

A. Yes.

Q. Where was that?

A. The License Renewal Application at § 3.1.2.1.2 recognized that reactor pressure vessel internals are constructed of the following materials:

- cast austenitic stainless steel (CASS);
- nickel alloy; and,
- stainless steel.

The same LRA section further recognized that the reactor pressure vessel internals experience the following aging effects that require management:

- change in dimensions;
- cracking;
- loss of material;
- loss of preload; and,
- reduction of fracture toughness, a term which, as noted previously, encompasses embrittlement.

The 2007 LRA and the IP3 Reactor Pressure Vessel

Q. I direct your attention to License Renewal Application Appendix A, § A.3.2.1.4. Do you have that?

A. Yes.

Q. What is that section of the License Renewal Application concerned with?

A. That section concerns the IP3 reactor pressure vessel itself.

Q. And what did Entergy say there?

A. Entergy stated that a part of the IP3 pressure vessel, specifically plate B2803-3, exceeded the screening criteria for pressurized thermal shock (PTS).

Q. Did Entergy acknowledge any specific concern about the reactor pressure vessels at Indian Point?

A. Yes, Entergy acknowledged that with respect to IP3 that the reactor pressure vessel plate B2803-3 "exceeds the screening criterion by 9.9°F." [Entergy January 22, 2008 Answer at 139; citing LRA § A.3.2.1.4].

Q. What if anything did Entergy propose to do about the IP3 pressure vessel?

A. Entergy proposed to submit to USNRC Staff a safety analysis for plate B2803-3 three years before the plate reached

the reference temperature for pressurized thermal shock (RT_{PTS}) criterion.

The 2007 LRA and RPV Internals

Q. In your review of the April 2007 Indian Point License Renewal Application, did you see an aging management program (AMP) for reactor pressure vessel internals?

A. No, I did not. The 2007 License Renewal Application did not contain an aging management program that specifically focused on reactor pressure vessel internals. Rather, Appendix A stated that sometime in the future Entergy would develop an aging management program for the reactor pressure vessel internals of their plants [LRA Appendix A, § A.2.1.41 with respect to IP2, and § A.3.1.41 with respect to IP3]. This deferred approach concerning IP2 and IP3 reactor pressure vessel internals is also repeated at LRA, § 3.1.2.2.6.

Q. Do reactor pressure vessels and their associated internal structures, components and fittings experience embrittlement?

A. Yes.

Q. Are there any reactor pressure vessel internal structures that are neglected in Entergy's discussion of future programs it will develop to address such structures?

A. Yes. It should be noted that the control rods and their associated guide tubes, plates, pins, and welds are not highlighted, but they are also very important RPV internals and their integrity is an extremely important safety concern. As I have previously noted, they are located in the core region of the RPV, and are inserted into the RPV through the upper head via so-called stub tubes. Their function is to absorb excess fission neutrons (i.e., those not needed to achieve a chain reaction) so that the power level of a reactor can be controlled. The control rods themselves are currently considered by the USNRC to be moving components (which can be replaced) and are thus not required to have an aging management plan (AMP). Nevertheless, the other associated CRD structures, components and fittings need an AMP since if these highly embrittled structures, components and fittings are subjected to significant shock loads they may fail, leading to possible core cooling issues.

Q. Do you believe there are any special problems associated with providing an adequate aging management program for control rods and their associated guide tubes, plates and welds?

A. Yes. For example, because of geometric considerations, many PWRs (including IP2 and IP3) cannot meet the USNRC's required minimum coverage for the non-destructive testing (NDT) of the so-called "J-groove" welds [Entergy, Walpole, NL-09-130 (Sept. 24, 2009) (Exh. NYS000311)], and thus the integrity of these important CRD stub tube welds cannot be directly confirmed by inspection. It appears that to help address this chronic problem Entergy has ordered two new RPV heads [Telecom-USNRC/Entergy Report (March 18, 2008) (Exh. NYS000317)], but they have not yet been scheduled for installation at Indian Point [Telecom-USNRC/Entergy (March 18, 2008) (Exh. NYS000317)]. In any event, unlike the rather superficial treatment given this important safety concern by Entergy [NL-10-063 (Exh. NYS000313)], I believe that a tangible, enforceable, and viable aging management program (AMP) should be developed and implemented before re-licensing the Indian Point reactor plants for extended operations, since the integrity of these CRD welds must be assured. If not, due to the leakage of borated primary coolant through cracked welds, there can be aggressive corrosion and wasting of the unclad outer surface of the upper head of the RPVs (such as the serious event that occurred at Davis-Besse and was identified in 2002). Worse yet,

there might be an inadvertent control rod ejection (due to a massive failure of the welds in the upper RPV head), which could cause a significant reactivity excursion, leading to core melting and radiation releases.

Q. Are there places within the reactor pressure vessel that you believe warrant particular aging management attention?

A. Yes. For the relicensing of the two reactors at Indian Point, corrosion-induced cracking (e.g., SCC) and radiation-induced embrittlement of the RPVs and their associated internals is an important age-related safety concern, particularly in the so-called "belt line" region of the RPV, which is the region that is the closest to the reactor core. In addition, as noted previously, the integrity of the so-called J-welds, which are part of the control rod drive seal in the upper head of reactor pressure vessels, is important to avoid corrosion-induced failures of the upper head and the possibility of control rod ejection (and thus an uncontrolled reactivity excursion).

<u>Entergy's NL-10-063 Communication</u>

Q. I direct your attention to Exhibit NYS000313. Do you recognize it?

A. Yes, I have reviewed this document. As noted above, it contains a copy of a July 14, 2010 communication, NL-10-063, from Entergy to the USNRC document control desk that concerns embrittlement of reactor pressure vessel internals. In turn, NL-10-063 contains an "Attachment 1."

Q. Does Entergy make any statements here about embrittlement of reactor pressure vessel internals?

A. Yes. Entergy acknowledges that, "PWR internals aging degradation has been observed in European PWRs, specifically with regard to cracking of baffle-former bolting." [NL-10-063, at 89 (Exh. NYS000313)]. Entergy also states: "As with other U.S. commercial PWR plants, cracking of baffle-former bolts is recognized as a potential issue for the [Indian Point] units." [NL-10-063, at 89 (Exh. NYS000313)]. Moreover, EPRI has stated that, a "considerable amount of PWR internals aging degradation has been observed in European PWRs." [EPRI MRP-227, at A-4 (Exh. NYS00307A-D)]. Material degradation has also been observed in control rod guide tube alignment (split) pins [EPRI MRP-227, at A-4 (December 2008) (Exh. NYS00307A-D)]. It is important to note that MRP-227 has also recommended that analysis be done to show when it is acceptable to continue to operate PWRs in which there have been bolt failures (e.g., due

to embrittlement and/or fatigue). While this type of temporary, short-term "fix" might be adequate for normal operations, it may lead to structural and component failures due to the shock loads associated with various postulated accidents. If so, the failed internal structures and components may relocate, cause core blockages, or otherwise result in uncoolable core geometry, and thus lead to seriously degraded core cooling, core melting and massive radiation releases.

Q. Do you have any additional problems with the inspection program for RVIs as proposed in the MRP-227 and adopted by Entergy?

A. Yes. With respect to Entergy's proposal to conduct baseline examinations of the RPV internals (RVIs), it should be noted that I have previously called on Entergy to conduct such examinations and for USNRC Staff to require the conduct of such examinations before entering the period of extended operations [See November 2007 Declaration of Richard T. Lahey, Jr., at ¶¶ 24, 25 (Exh. NYS000298); see also State of New York Notice of Intention to Participate and Petition to Intervene, at 217-220, State of New York Contention-23 (Baseline Inspections)]. Fortunately, both the USNRC and Entergy now seem to have embraced the concept of baseline inspections for RPV internals,

but the proposed aging management program (AMP) as set forth in NL-10-063 lacks sufficient details to know when the baseline inspections of the RPV and its internals will begin and end, and the scope of these inspections. Thus, it is not possible to know whether the proposed baseline inspections will be comprehensive and adequate.

Q. Are there other problems that you believe need to be addressed if Entergy is to have an adequate aging management program for RPV internals?

A. Yes. My Report provides more details on my concerns with Entergy's failure to conduct an evaluation of the synergistic impacts of embrittlement, corrosion-induced cracking, and metal fatigue on the degradation of RPV internals, and its failure to consider how those interacting degradation mechanisms will impact the ability of the RPV internals to withstand the effect of thermal and decompression shock loads as a result of a DBA LOCA. I am also concerned that the design of the inspection programs -- including their frequency, the type of inspections to be conducted, the acceptance criteria and the criteria for actions to be taken in the event of a failure of a component -- does not consider these synergistic degradation mechanisms. Finally, Entergy's AMP for RPV internals does not

include specific programs with objective criteria for either preventative measures or for corrective actions to be taken when inspections show that certain components are not able to safely undergo extended plant operations.

Entergy's NL-11-107 Communication

Q. I show you what has been marked as Exhibit NYS000314. Do you recognize that document?

A. Yes, this is a copy of Entergy's September 28, 2011 communication, NL-11-107, with the USNRC's document control desk.

Q. Would you please turn to Table 5-2 at page 36 of the Attachment to NL-11-107.

A. Yes, I have that.

Q. What does the document say there?

A. In discussing the baffle-former assemblies and their related baffle-edge bolts, it recognizes that irradiated-assisted stress corrosion cracking and fatigue can cause cracking which, in turn, leads to failed or missing bolts connecting a baffle to a former.

Q. What else does communication NL-11-107 state?

A. In it, Entergy tells the USNRC that it has completed commitment number 30 wherein Entergy stated that it would submit

an inspection plan to the USNRC for reactor pressure vessel internals (RVIs) no later than two years before the plant entered the period of extended operations. However, none of my safety concerns associated with the synergistic effects of embrittlement, fatigue and corrosion on the integrity of RPV internals, and post-accident core coolability (i.e., due to shock load induced failures), were addressed. In my opinion an adequate inspection plan for RPV internals is a necessary, but not sufficient, means of assuring safe extended plant operations. Indeed, a systematic safety evaluation of the degraded RPV internals is also needed to identify the limiting structures, components and fittings that need to be repaired or replaced before the onset of extended operations.

Entergy's Amended and Revised RVI Plan, and USNRC Staff's November 2014 SSER2

Q. I direct your attention to Exhibits NYS000496 through NYS000506. Do you recognize these exhibits?

A. Yes, I have reviewed these documents. In communication NL-12-037, dated February 17, 2012 [Exh. NYS000496], the applicant submitted an amendment to its license renewal application entitled "Revised Reactor Vessel Internals Program and Inspection Plan." Thereafter, the applicant

explained and modified this proposed plan in response to various requests for information (RAIs) from the USNRC. [Exhs. NYS000497 through NYS000506]. Collectively, I will refer to this collection of communications as the applicant's "Amended and Revised RVI Plan."

Q. I direct your attention to Exhibit NYS000507. Do you recognize this exhibit?

A. Yes, I have reviewed this exhibit. It is the Second Supplemental Safety Evaluation Report, or SSER2, prepared by USNRC Staff and released in November 2014. In the SSER2, the USNRC Staff evaluated and approved the applicant's Amended and Revised RVI Plan.

Q. Does the SSER2 discuss the potential synergism between various aging mechanisms?

A. Yes, to some degree. The USNRC recognized the potential synergy between thermal and irradiation embrittlement for cast austenitic stainless steel components (CASS). [SSER2 at 3-42 (Exh. NYS000507)]. In particular, in its Safety Evaluation Report (SER) for MRP-227, the USNRC Staff acknowledged the potential for synergistic interaction between embrittlement and other aging mechanisms. For example, the USNRC noted that "the synergistic effects of SCC, fatigue, and

thermal embrittlement . . . could potentially cause greater degradation in the welds [of Combustion Engineering lower support columns] than just the consideration of IASCC (irradiation assisted stress corrosion cracking) and irradiation embrittlement alone. Degradation in these welds could then be equivalent to or greater than other components susceptible only to IASCC and irradiation embrittlement due to the synergistic effects." [SE at 15 (Exh. NYS000309)]. The USNRC staff could have – indeed, should have - made the same observation about potential synergistic aging effects for Westinghouse RVI components, fittings, and structures at IP2 and IP3.

Q. Does the applicant's Amended and Revised RVI Plan say anything about preventative actions to manage aging effects?

A. Yes. In Attachment 1 to NL-12-037, Entergy has indicated that the Amended and Revised RVI Plan "is a condition monitoring program that does not include preventative actions." [Attachment 1 to NL-12-037, at 5 (Exh. NYS000496)]. Generally, the applicant continues to approach the problem of synergistic aging effects on RVI components through "condition monitoring" (i.e., periodic inspections per MRP-227-A) rather than a comprehensive approach which includes detailed analyses and/or preventative actions (i.e., repair and replacement) ["Revised

Reactor Vessel Internals Program and Inspection Plan," Attachment 1 to NL-12-037, at 5 (Exh. NYS000496)]. This approach implies that aging effects and degradation will not be addressed until cracks or other degradation mechanisms (e.g., wear) have been directly observed ["Revised Reactor Vessel Internals Program and Inspection Plan," Attachment 1 to NL-12-037, at 5 (Exh. NYS000496)].

In short, component degradation will be addressed only after it occurs. The applicant incorrectly concludes that preventative actions, such as component replacement, are not required for most RVI components because cracking or other flaws can be detected before the failure of a component affects the safe operation of the reactor. This is apparently based on the erroneous assumption that IP2 and IP3 will continuously operate during the 20-year period of extended operation within normal "steady-state" parameters. Entergy ignores the possibility that significantly fatigued, embrittled and corrosion-weakened, or otherwise degraded, RVI components, structures, or fittings may be exposed to various shock loads which can cause them to deform or relocate and thereby impair core cooling. In fact, the applicant's reactor safety analyses implicitly assume that the reactor core will maintain a coolable geometry during emergency

core cooling system (ECCS) operation subsequent to a DBA LOCA, notwithstanding the degradation and possible deformation or relocation of various RVI components and potential flow blockages and degraded core cooling which may result.

Q. Does Entergy make any statements about the degradation of RVI components in the “Amended and Revised RVI Plan”?

A. Yes, similar to NL-10-063 (Exh. NYS000313), the applicant acknowledges, in NL-12-037, that other PWRs have experienced material degradation and failure of multiple RVI components, including cracking of baffle-former bolting, cracking in other important bolting, wear in thimble tubes, and potential wear in control rod guide tube guide plates [Attachment 1 to NL-12-037, at 8 (Exh. NYS000496)]. Also, the applicant has committed to replace one affected IP2 component – the degraded guide tube support pins (split pins) – by 2016 [“Revised Reactor Vessel Internals Program and Inspection Plan,” Attachment 1 to NL-12-037, at 8 (Exh. NYS000496); Commitment 50, Attachment 1 to NL-13-122, at 7 (Exh. NYS000502)]. Interestingly, the applicant has agreed to replace the IP2 split pins, even though they were already replaced once in 1995, and even though the applicant claims that the failure of a split pin would not compromise reactor vessel functions [Response to RAI

16, Attachment 2 to NL-12-166, at 1 (Exh. NYS000500)]. However, for many other affected RVI components, the applicant proposes a “wait-and-see” approach.

Q. Could you provide an example?

A. Yes. The applicant acknowledges that “cracking of baffle former bolts is recognized as a potential issue for the Indian Point units” [“Revised Reactor Vessel Internals Program,” Attachment 1 to NL-12-037, at 8 (Exh. NYS000496)], but the applicant does not propose to replace the degraded bolts, only to continue monitoring them [“Revised Reactor Vessel Internals Inspection Plan,” Attachment 2 to NL-12-037, at 40, tbl. 5-2 (Exh. NYS000496)]. In fact, the applicant has not yet developed inspection acceptance criteria for baffle former bolts in either IP2 or IP3 [SSER2, at 3-20 (Exh. NYS000507)]. Instead, the applicant has agreed to develop a technical justification including acceptance criteria for baffle former bolts <u>sometime</u> prior to the first round of inspections, which might not occur until 2019 for IP2 and 2021 for IP3 [SSER2, at 3-20 (Exh. NYS000507); Response to RAI 5, Attachment 1 to NL-12-089, at 11 (Exh. NYS000497)].

Another example of the applicant’s “wait-and-see” approach for the RVIs is the applicant’s proposal for managing aging

effects on the clevis insert bolts. [SSER2, at 3-23 to 3-26 (Exh. NYS000507)]. Like the split pins that the applicant is replacing in IP2 for the second time, clevis insert bolts are susceptible to primary water stress corrosion cracking (PWSCC) [MRP-227-A, Appendix A, at A-2 (Exh. NRC00114A-F)]. Failures of clevis insert bolts, apparently caused by PWSCC, were detected at a Westinghouse-designed reactor in 2010. Out of 48 clevis bolts in this reactor, 29 were partially or completely fractured but only 7 of those damaged bolts were visually detected as having failed [SSER2, at 3-25 (Exh. NYS000507)]. Despite this high rate of failure (about 60% of the total bolts were damaged) and low rate of visual detection (only about 24% of the damaged bolts were detected), the applicant proposes to manage the aging degradation of clevis insert bolts with visual (VT-3) inspections rather than pre-emptive replacement ["Revised Reactor Vessel Internals Inspection Plan," Attachment 2 to NL-12-037, tbl. 5-4, at 51 (Exh. NYS000496)].

The applicant apparently acknowledges that visual inspections will not detect the majority of clevis bolt cracks prior to failure, but justifies this approach on the grounds that "crack detection prior to bolt failure is not required due to design redundancy" [Response to RAI 17, Attachment 1 to NL-

13-122, at 8 (Exh. NYS000502)]. In fact, the applicant appears to suggest that the failure of multiple clevis insert bolts will not seriously affect the operation of the reactor. The applicant then analyzes the effect of clevis bolt failures on various other components.

The applicant's analysis of the effects of clevis bolt failures assumes that all other components will be functioning according to their design specifications, and does not consider the fact that the other components may also be undergoing degradation from various interacting aging mechanisms. Moreover, the applicant fails to consider the possibility that a shock load (e.g., due to a LOCA) may cause the sudden failure of the remaining intact clevis bolts, which, in turn, may lead to an uncoolable core geometry. In short, rather than taking proactive steps to replace the degraded clevis bolts prior to failure, the applicant proposes to wait for clevis bolt failures to occur before taking steps to address the problem, an approach which is totally unacceptable in my opinion.

The baffle former bolts and clevis insert bolts are just two examples of Entergy's overarching approach to RVI aging management, which foregoes preventative component repair or

replacement in favor of running the reactor until detectable damage or component failure occurs.

Q. Do you have any other concerns regarding specific components discussed in the Amended and Revised RVI Plan?

A. Yes. The applicant's approach for analyzing the lower support structures' functionality and fracture toughness is also flawed [Response to RAI-11-A, Attachment 1 to NL-13-052, at 1-4 (Exh. NYS000501)]. The applicant suggested that irradiation embrittlement effects would only be significant in the presence of pre-existing flaws or service induced defects, together with a stress level capable of crack propagation. In its analysis, the applicant, based on the lack of documented fractures of core support columns, "assumed that only a limited number of columns could actually contain flaws of significant size." The applicant further assumed that the columns would be subject to "nominal normal operating stresses" [SSER2, at 3-43 (Exh. NYS000507)]. When the USNRC Staff inquired about the most recent visual inspections of the core support structures, the applicant acknowledged that the CASS support column caps were inaccessible to inspection and that VT-3 visual inspection offered "no meaningful information regarding the structural integrity of the columns." [Id. at 3044.] Under these

circumstances, the applicant's conclusion that irradiation-induced cracking of core support columns is "unlikely" represents wishful thinking and is contrary to recent studies [e.g., NUREG/CR-7184, at xv (Revised December 2014) (Exh. NYS00488A-B)], which show the extreme sensitivity of crack growth rate and fracture toughness to irradiation. Moreover, it ignores the fact that these and other non-CASS RVI structures and components undergo a range of aging degradation mechanisms simultaneously under steady-state and transient conditions, and that their embrittlement or susceptibility to fracture simply cannot always be adequately detected using currently available inspection techniques.

Also, not all of the core support structures are accessible for inspection, so surrogate structures have been chosen by Entergy to assess age-related degradation mechanisms. For example, the girth weld of the core barrel has been proposed by the applicant as a leading indicator for irradiation-induced embrittlement (IE) and irradiation-assisted stress corrosion cracking (IASCC) of the core support column caps, even though these components are very different, and they may be exposed to different degradation mechanisms and shock loads. In fact, as pointed out recently by a member of the ACRS Plant License

Renewal Subcommittee, “[t]he relationship between a lower core barrel weld and the tops of these columns is a bit of a stretch . . . [t]hey’re totally different type of components, totally different loadings.” Moreover, to have a failure due to a seismic event “you don’t even need to have a crack if these columns are really brittle . . . .” [ACRS Plant License Renewal Subcommittee Transcript, at 209-211 (April 23, 2015) (Exh. NYS000526)].

Q. Does the applicant’s Amended and Revised RVI Plan adequately account for the potential cumulative effect of synergistic aging mechanisms on RVIs?

A. No. By merely relying on MRP 227-A for its aging management plan, the applicant has ignored the large uncertainties that exist with respect to the effects of irradiation-induced aging phenomena. [Chen, et al., at xv (Exh. NYS000488A); NUREG/CR-7153, Vol. 2: Aging of Core Internals and Piping Systems, at 181, 187, 210-211 (Exh. NYS00484A-B); Stevens, et al. (October 2014), at 9-10 (Exh. NYS000491)]. While the applicant’s Thermal Aging and Neutron Irradiation of Cast Austenitic and Stainless Steel (CASS) program generally recognizes the potential adverse synergistic effects of elevated coolant temperature and irradiation on the fracture toughness of

CASS materials, a broader recognition of this principle is needed by the applicant, since RVI components made from non-cast stainless steel will also experience the combined effects of irradiation-induced embrittlement, corrosion, and other aging mechanisms. The applicant has failed to evaluate the synergistic mechanisms that occur for many other important and vulnerable RVI components, such as the core baffles, baffle bolts, and formers. Compared to the baffles, baffle bolts, and formers, the core support columns (which are obviously very important incore structures) are located in an area of the reactor pressure vessel which is subject to less radiation fluence (and thus are less susceptible to embrittlement).

Q. Do you have any other concerns with the applicant's Amended and Revised RVI Plan?

A. Yes. The applicant proposes to rely on visual (VT-3) inspection techniques for many RVI components. However, there are significant shortcomings of this technique to detect material cracking, degradation, or wear prior to failure, as has been noted by USNRC staff [Tregoning, at 2-3 (Exh. NYS000508); Case, at 1 (Exh. NYS000509)], and illustrated by the visual detection of only 7 out of 29 fractured clevis insert bolts at a Westinghouse PWR in 2010 [SSER2, at 3-25 (Exh. NYS000507)]. In

an RAI to Entergy, the USNRC staff observed that "VT-3 visual examination may not be adequate for all components for detecting fatigue cracking prior to the occurrence of structurally significant cracking." [Attachment 1 to NL-13-052, at 5 (May 7, 2013) (Exh. NYS000501)]. Moreover, as I have noted previously, the level of embrittlement can not be detected at all using visual inspection techniques.

Fatigue

Q. Turning to fatigue, could you explain what fatigue is?

A. Yes. Fatigue is another important age-related degradation mechanism. It is one of the primary considerations when conducting a time limited aging analysis (TLAA) and an aging management program (AMP) for nuclear power plants. Fatigue of various structures, components and fittings in a nuclear reactor can result in piping and pressure boundary component and fitting ruptures, physical failures, and the relocation of loose pieces of RVI metal throughout the reactor system, which, in turn, may result in core blockages and interfere with the effective core cooling of a nuclear power plant. My main concerns about fatigue are the increased potential for a primary or secondary side LOCA, and the failure of various RPV internals (RVIs). It should be noted that the

fatigue life of a PWR component, fitting or structure is normally evaluated in terms of a cumulative usage factor (CUF) which is corrected for the degradation in fatigue life due to the reactor coolant environment (i.e., CUF_{en}). The cumulative usage factor is defined as, $CUF = N/N_{a\text{-}AIR}$, where N is the number of the various fatigue cycles that have occurred (or are expected by the end of plant life, EOL), and $N_{a\text{-}AIR}$ is the number of allowable fatigue cycles obtained from data (taken in air) at which failure (i.e., significant surface cracking) is expected. The observed degradation in fatigue that occurs due to hot reactor coolant is quantified by an environmental fatigue correction factor, $F_{en} = N_{a\text{-}AIR}/N_{a\text{-}RC}$, where $N_{a\text{-}AIR}$ is the allowable number of fatigue cycles measured in air, and $N_{a\text{-}RC}$ is the allowable fatigue cycles measured in a simulated reactor coolant (RC) enviroment; thus, $CUF_{en} = CUF \times F_{en} = N/N_{a\text{-}RC}$. The criterion for acceptance by the USNRC is that $CUF_{en} < 1.0$ by the end of life (EOL) for the component, fitting or structure in question. Anyway, the allowable cycles to failure ($N_{a\text{-}RC}$) are determined from small scale experiments using metal test samples which are exposed to simulated reactor coolant environments. Unfortunately, to date, there have NOT been any systematic fatigue experiments done in simulated reactor coolant

environments using highly embrittled metal test samples, which have less fatigue life than ductile materials. That is, the synergistic degradation effect of embrittlement has not been included in CUF_{en} (= $N/N_{a\text{-}RC}$) evaluations, thus the results are expected to be non-conservative since the denominator ($N_{a\text{-}RC}$) will be too large, and thus CUF_{en} will be too small.

Q. I show you what has been marked as Exhibit NYS000527. Do you recognize this document?

A. Yes. This is Entergy's Fatigue Monitoring Plan for IP2 and IP3.

Q. Is Entergy required to conduct fatigue evaluations of internal and external components?

A. Yes. In this proceeding, the applicant agreed, in Commitments 33, 43 and 49, to calculate the CUF_{en} for external(i.e., primary pressure boundary) and internal (RVI) components in certain locations [Dacimo, Fred, Entergy, letter to Document Control Desk, USNRC, "Reply to Request for Additional Information Regarding the License Renewal Application," NL-13-122 (September 27, 2013), at 20 (NYS000502)]. Additionally, the USNRC has recently proposed to require all applicants for license renewal to evaluate the fatigue life of limiting components beyond those originally

specified in NUREG/CR-6260, and to evaluate the effect of reactor coolant environment on the fatigue life of both external and internal (i.e., RVIs) structures and systems [79 Fed. Reg. 69,884 (November 24, 2014) (NYS000522); USNRC, Draft Regulatory Guide DG-1309 (Proposed Revision 1 of Regulatory Guide 1.207, dated March 2007), "Guidelines for Evaluating the Effects of Light-Water Reactor Coolant Environments in Fatigue Analyses of Metal Components" (November 2014) (NYS000523)].

Q. In your expert opinion, has Entergy done adequate fatigue evaluations to assure the safety of their two nuclear power plants at the Indian Point site during extended operations?

A. No. [redacted]

[redacted]

[redacted] It would appear that virtually any error would put some of the calculated values of CUF_{en} over the CUF_{en} = 1.0 fatigue failure limit.

Q. I show you a document marked as Exhibit NYS000513. Do you recognize it?

A. Yes, it is a paper presented by Westinghouse at a recent Pressure Vessels & Piping Conference of the American Society of Mechanical Engineers held in Anaheim, California in July 2014; it is entitled “License Renewal and Environmental Fatigue Screening Application” and its authors were Mark Gray and Christopher Kupper.

Q. Are you familiar with its contents?

A. Yes, I have reviewed this article and it clearly shows the iterative process used by Westinghouse in which safety margin is removed in its environmentally assisted fatigue (EAF) calculations in an effort to reduce the output or result below CUF_{en} = 1.0.

Q. Returning to our discussion about Westinghouse's fatigue evaluations of reactor coolant pressure boundary components, has Entergy addressed the issue of fatigue in the context of shock loads?

A. [redacted] Even assuming this CUF_{en} calculation is accurate, it does not account for the possibility that a highly fatigued component, which does not yet have signs of significant surface cracking, may be exposed to an unexpected seismic event or shock load that could cause it to fail. This is a good example of the type of "silo

thinking" (i.e., the fatigue and safety analyses are treated entirely separately) that NYS is concerned about.

Q. I show you your Supplemental Report, which has been marked as Exhibit NYS000297. I note that the State has provisionally designated it as containing confidential information. Would you provide a brief summary of the Report?

A. I prepared this Supplemental Report to set out some of my concerns about the use of the WESTEMS computer code to develop a cumulative fatigue analysis of certain components in the Indian Point reactors and their reactor coolant pressure boundaries.

Q. Would you briefly summarize your concerns?

A. Yes. First, I am concerned that without an error analysis it is difficult to be in a position to meaningfully analyze the results of the 2010 and subsequently refined CUF_{en} analyses presented by Entergy and Westinghouse.

Q. Why is an error analysis important?

A. It is well known that all engineering analyses are based on imperfect mathematical models of reality and various code user assumptions which inherently involve some level of error. Error analyses help readers and decision makers

understand what level of confidence to attach to the calculated results and the proposed conclusions.

Q. Is the preparation of an error analysis an accepted practice in the field of engineering?

A. Yes. Engineers frequently prepare error analyses. In my submissions in this proceeding I noted that one would normally expect to see at least a hybrid 'propagation-of-error' type of analysis [Kline & McClintock (1953) (Exh. NYS000514)] to determine the overall uncertainty in the CUF_{en} results given by Westinghouse. I also referenced a standard enineering text book, "Basic Engineering Data Collection and Analysis," pp. 310-311, by Vardeman & Jobe [2001], to demonstrate the various types of error analyses which are regularly done by engineers [Exh. NYS000347].

Q. I show you what has been marked as Exhibit NYS000515. Are you familiar with it?

A. Yes, it is a recent USNRC inspection report with notices of non-conformance for Westinghouse's Quality Assurance Program. In that report, the USNRC determined that Westinghouse failed to adequately implement its QA program in the areas of corrective actions, oversight of suppliers, and audits. Since Entergy relies on Westinghouse services to, among other things,

provide appropriate guidance on corrective action and other activities affecting safety-related functions, the USNRC's findings of non-conformance are all the more unsettling. Under these circumstances, the USNRC should insist that an error analysis be performed to ensure the validity of Westinghouse's fatigue evaluations for IP2 and IP3 components.

Q. Are you aware of an instance where an error analysis was prepared for a project at Indian Point?

A. Yes, for example, in 1980, the Consolidated Edison Company of New York prepared an error analysis in support of a proposal to add more spent fuel into the spent fuel pool at Indian Point Unit 2.

Q. I show you what has been marked as Exhibit NYS000348; do you recognize it?

A. Yes. That is a copy of the 1980 Con Edison error analysis for the re-racking of spent fuel in the Unit 2 spent fuel pool.

Q. Do you have other concerns about the refined CUF_{en} reanalysis?

A. Yes, as discussed in my Supplemental Report, I am concerned that engineering judgment or user intervention could have affected the results. I note that when USNRC Staff issued

the Supplemental Safety Evaluation Report, Staff instructed Entergy and Westinghouse, on a going forward basis, to document and disclose the use of engineering judgment and user intervention when conducting future fatigue analysis using the WESTEMS code. This is noted in Exhibit NYS000160 at page 4-2. To my knowledge, Westinghouse has provided such information for some, but not all of the fatigue evaluations performed to date. Also, USNRC Staff instructed Entergy not to use WESTEMS when conducting analyses under the ASME Standard know as NB-3600 [at 4-2, 4-3 (Exh. NYS000160)]. Furthermore, I am concerned about the analytical framework employed by the WESTEMS code. As detailed, in my Supplemental Report, I believe that the code’s thermal-hydraulic models and framework are too simplified to predict accurate results. [redacted]

Q. Do you know how Entergy is proposing to address fatigue as part of its overall aging management plan for IP2 and IP3?

A. Yes. Entergy has an existing Fatigue Monitoring Plan for addressing metal fatigue. The program is designed to monitor operational cycles and transients so that the various CUF_{en} remain below unity. The company has also proposed to include RVIs in its Fatigue Monitoring Program; however, for the reasons already stated, WESTEMS may be non-conservative, so any program that relies on values, such as the fatigue cycles to failure, derived from the WESTEMS methodology is inherently unreliable for ensuring that aging RVIs avoid failure. This is particularly true when these results include no accompanying error analysis.

Conclusions Regarding Fatigue and Embrittlement Issues

Q. Could you summarize your general concerns with the various fatigue and embrittlement issues in the applicant’s license renewal application?

A. Yes, I am very concerned that Entergy has continually eroded the safety margins and conservatisms built into the current licensing basis for the Indian Point reactors. For example, Entergy has relied on CUF_{en} calculations that remove

various conservatisms but are still very close to unity (the fatigue failure limit). Entergy also relies on the detection of degradation, wear or cracking prior to component failure, rather than repairing or replacing the aging parts – particularly the RVIs – preemptively. Moreover, Entergy implicitly assumes that the plant will operate in a steady-state, and has not taken into account unanticipated severe seismic events or thermal/pressure shock loads which can cause failures to occur.

As reactors and their constituent components age, it becomes very important to preserve – rather than erode – operational safety margins. Uncertainties exist in all systems, and calculation or modeling mistakes are always possible. For example, the USNRC recently became aware that certain methodologies prescribed in its NUREG-0800 Branch Technical Position (BTP) 5-3 for estimating the initial fracture toughness of reactor vessel materials may be non-conservative. [See, e.g., Troyer, et al., "An Assessment of Branch Technical Position 5-3 to Determine Unirradiated RTNDT for SA-508 Cl.2 Forgings," Paper No. PCP2014-28897, Proceedings of the ASME 2014 Pressure Vessels and Piping Conference, Anaheim, California (July 20-24, 2014) (Exh. NYS00516); Letter from Pedro Salas, Regulatory Affairs Director, AREVA, to USNRC regarding Potential

Non-conservatism in NRC Branch Technical Position 5-3 (January 30, 2014)(Exh. NYS000517); USNRC, Slides, “Assessment of BTP 5-3 Protocols to Estimate RTNDT(u) and USE (June 4, 2014) (Exh. NYS000518); NUREG-0800, Rev. 2 (Exh. NYS000521)]. [redacted] Anyway, since unexpected errors of this type do occur, maintaining safety margins helps to guard against potentially adverse impacts due to precisely this type of unexpected finding of non-conservatism in safety evaluations. Lastly, I would like to note that at a recent American Society of Mechanical Engineers (ASME) Pressure Vessels & Piping Conference, USNRC staff also highlighted newly-identified non-conservatisms in sections of the ASME Code regarding fracture toughness applicable to nuclear reactor operations. [Kirk, M. et al., “Assessment of Fracture Toughness Models for Ferritc Steels Used in Section XI of the ASME Code Relative to Current Data-Based Model,” PVP 2014-28540 (Exh. NYS000520)]. This is yet another reason to preserve, rather than erode safety margins in the aging management of light water nuclear reactors (e.g., PWRs).

Q. You have reviewed NUREG-1801, GALL Report Revision 1 (Exh. NYS00146A-C); NUREG-1800, Standard Review Plan Revision 1 (Exh. NYS000195); NUREG-1801, GALL Report Revision 2 (Exh. NYS00147A-D); NUREG-1800, Standard Review Plan Revision 2 (Exh. NYS000161); EPRI's MRP-227 Revision 0 (Exh. NYS00307A-D); EPRI's MRP-227-A (Exh. NYS000507); Entergy's July 2010 NL-10-063 communication, Entergy's February 2012 NL-12-037 communication (Exh. NYS000313) and subsequent communications constituting its Amended and Revised RVI Plan (Exhs. NYS000496-506); USNRC Staff's June 22, 2011 Safety Evaluation of MRP-227 Revision 0 (Exh. NYS000309); NUREG-1930, USNRC Staff's August 30, 2011 Supplemental Safety Evaluation (Exh. NYS000160); and NUREG-1930, USNRC Staff's November 2014 Second Supplement Safety Evaluation Report for the Indian Point License Renewal Application (Exh. NYS000507); and Entergy's NL-11-107 communication (Exh. NYS000314), correct?

A. Yes.

Q. Do you have any opinion about those documents with respect to the degradation of important primary system components such as reactor pressure vessel internals (RVIs)?

A. Yes.

Q. Please summarize your testimony.

A. As I stated in my initial November 2007 declaration in support of the State of New York's Contentions 25 and 26, my April 2008 declaration in support of Contention NYS-26A, my September 2010 declarations in support of the State's supplemental filings on Contentions NYS-25 and NYS-26B/RK-TC-1B, my 2011 declarations in support of Contention NYS-38/RK-TC-5, in my previously filed testimony on Contentions NYS-25, 26 and 38, and my February 2015 declaration in support of the State's further supplemental filings on Contentions NYS-25 and Contention NYS-38/RK-TC-5, in my professional judgment Entergy has failed to demonstrate that it has adequately accounted for the aging phenomena of embrittlement and fatigue for structures, components and fittings inside the reactor pressure vessels (i.e., RVIs) at Indian Point Unit 2 and Indian Point Unit 3. My professional judgment has not fundamentally changed based upon Entergy's July 14, 2010 submission of License Renewal Application, Amendment No. 9 [NL-10-063 (Exh. NYS000313)], Entergy's September 28, 2011 submission of NL-11-107 [Exh. NYS000314], or Entergy's Amended and Revised RVI Plan, consisting of the February 17, 2012 submission of NL-12-037 [Exh. NYS000496] as amended by its subsequent communications [Exhs. NYS000497-506] and approved by the USNRC Staff in the

SSER2 [Exh. NYS000507]. I do not believe that Entergy's July 15, 2010 communication to the Board [NL-10-063 (Exh. NYS000313)] concerning a new AMP for RPV internals, or its September 28, 2011 communication [NL-11-107 (Exh. NYS000314)], are adequate to address the safety concerns and technical issues that I have raised herein. They do not address my age-related safety concerns, nor do they recognize the importance of the various synergistic degradation mechanisms that I am concerned with. The Amended and Revised RVI Plan, which the USNRC Staff evaluated and approved in the November 2014 SSER2, also does not resolve my concerns over the simultaneous and synergistic age-related degradation mechanisms that may affect various RVI components and structures.

While some age-related safety issues might eventually be resolved analytically or experimentally, in many cases it appears that the easiest and most cost-effective way to resolve them is to simply repair or replace the most seriously degraded structures, components and fittings, and this approach is what NYS has been proposing for some time (particularly for the degraded RVIs).

I want to stress that during the course of my involvement in these relicensing proceedings I have discovered what I

believe to be some important new age-related safety concerns which, to the best of my knowledge, have not been previously considered in relicensing proceedings. These concerns include: the synergistic effect on the degradation and integrity of RPV internals (RVIs) of radiation-induced embrittlement, corrosion and fatigue, and the potential for the unanticipated failure of RPV internals (RVIs) due to a severe seismic event or accident-induced thermal and/or pressure shock loads, and the implications of the failure of RPV internal structures, components and fittings (i.e., RVIs) on post-accident core coolability. While in the past many of these issues and concerns have been noted separately, the implications of their synergistic interaction has apparently been overlooked and not evaluated (i.e., they have been evaluated in “silos”). Since I first raised these technical issues in 2007, the USNRC, DOE and various nuclear industry groups have slowly begun to recognize their significance. In fact, the evaluation and study of these important issues is underway, but major uncertainties still exist. As a consequence, I believe that these important age-related safety concerns must be resolved in order to have assurance that the Indian Point reactors can operated safely beyond their design life of 40 years. Indeed, I believe that

the most vulnerable RPV internals (RVIs) need to be carefully identified and repaired or replaced prior to extended operations since it is beyond the current state-of-the-art to perform realistic and accurate calculations on the relocation of failed RPV internals (RVIs) and the resultant potential for core blockages and degraded core cooling.

Steam Generator Issues

Q. Let us now turn your attention to the specific steam generator issues which have been a focus of Contention NYS-38/RK-TC-5. Earlier in your testimony you mentioned that Indian Point Unit 2 and Indian Point Unit 3 include steam generators. How many steam generators are there at Indian Point?

A. Indian Point Unit 2 and Unit 3 each have Westinghouse-designed 4-loop nuclear steam systems, and each of those units has four U-tube steam generators. Therefore, those two units collectively have eight Westinghouse steam generators.

Steam Generators, Their Components, and Function

Q. I show you what has been marked as Exhibit NYS000376. Do you recognize it?

A. Yes, it is a diagram of a Westinghouse steam generator.

Q. Would you please describe the role of the steam

generators in the Indian Point nuclear steam supply systems?

A. At Indian Point, each reactor coolant loop contains a vertical shell and U-tube phase-change heat exchanger (i.e., steam generator). Reactor coolant enters an inlet plenum in the bottom of the steam generator through an inlet nozzle. The primary coolant then flows upward through the tubesheet and the various U-tubes, returning through the tubesheet to the outlet plenum, from which it leaves the steam generator through a bottom nozzle. The U-tubes are welded to the tubesheet, and the inlet and outlet plena in the steam generator are separated by a partition called a divider plate. The divider plate is joined to the lower head of the steam generator and the tubesheet via a stub runner and divider plate assembly.

Q. How many tubes are in each of the Westinghouse steam generators at Indian Point?

A. There are numerous U-tubes in each steam generator. According to Entergy reports, there are 3,214 tubes in each of the eight steam generators. [Entergy Indian Point 2 Steam Generator Program (July 2007) at 4; Entergy Indian Point 3 Steam Generator Program (July 2007) at 4 (Exhs. NYS000554, NYS000533).]

Q. How many tube-to-tubesheet welds are there in each

steam generator?

A. Because each U-tube passes through the tubesheet twice (once on the inlet side and once on the outlet side), there are twice as many tube-to-tubesheet weld locations as there are U-tubes.

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

A. Yes, in the License Renewal Application at Tables 2.3.1-4-IP2 and 2.3.1-4-IP3 Entergy states that the lower head of a steam generator, the divider plate, U-tubes, the tubesheet, and the tube-tubesheet welds each constitute a pressure boundary for Indian Point Unit 2 and Indian Point Unit 3. Significantly, all these boundaries, except the divider plate, represent important primary pressure boundaries. Entergy also acknowledged that the U-tubes also perform an important heat transfer function. Those tables are located in the License Renewal Application at pages 2.3-36, 2.3-39, respectively (Exh. NYS000558).

Q. I show you what has been marked as Exhibit NYS000375. Do you recognize it?

A. Yes. It is a schematic diagram of a Westinghouse Nuclear Steam Supply System (NSSS). Among other things, this

diagram depicts the reactor coolant pressure boundary. The components on the primary side appear in red (primary side liquid) or yellow (primary side steam), and the components on the secondary side are in blue (secondary side liquid) or green (secondary side steam).

Q. What is the reactor coolant pressure boundary?

A. The USNRC provides a definition of the reactor coolant pressure boundary in its regulations at 10 C.F.R. § 50.2. In essence, the reactor coolant pressure boundary refers to a physical barrier or boundary between the reactor coolant system on the "primary side" of the reactor and the environment, or the "secondary side" of the nuclear steam supply system, which eventually communicates with the environment. As I have noted previously, one can see this boundary line in the Westinghouse diagram (Exhibit NYS000375) that represents the primary loop in red or yellow and also the secondary loop in green or blue. It is critical not to breach the reactor's primary coolant pressure boundary and allow the radioactive reactor coolant to escape to the ambient.

Q. I show you two documents that have been marked as Exhibits NYS000456 and NYS000560? Do you recognize them?

A. Yes, these are summary charts identifying the

operating history, material properties, and other characteristics of the eight steam generators in use at Indian Point Unit 2 and Unit 3. The charts summarize statements about the steam generators based on submissions in this proceeding or other public documents.

Primary Water Stress Corrosion Cracking

Q. Are you familiar with the term "primary water stress corrosion cracking (PWSCC)"?

A. Yes. Primary water stress corrosion cracking is a well-known LWR aging phenomenon for many metal-alloy/ environmental combinations. It presents a challenge since it degrades and embrittles otherwise ductile alloys, but only in very specific environments. Occurrence of this phenomenon requires the simultaneous presence of stress, whether residual or applied, and a specific metal-alloy/environment combination.

Q. Has primary water stress corrosion cracking occurred in pressurized water reactors (PWRs)?

A. Yes. In operating PWRs the PWSCC of stressed nickel-based alloys, such as Alloy 600, has occurred. It should also be noted that Alloy 600 components are generally welded using Alloys 82 or 182 rods (which are derivatives of Alloy 600) that have also been found to be susceptible to PWSCC.

Q. Has Entergy disclosed the material present in the divider plates in the steam generators at Indian Point?

A. Yes, in 2011 in response to a request for information by USNRC Staff, Entergy stated that the current Indian Point Unit 2 steam generators use Alloy 600 for the divider plates and that it assumed that the weld material for the divider plate assemblies was Alloy 82/182 weld material. Entergy also stated that the Indian Point Unit 3 steam generators use Alloy 600 for the divider plates and that it assumed that the weld material for the divider plate assemblies was Alloy 82/182 weld material.

Q. I show you Exhibit NYS000151; would you describe the document?

A. Yes, this is a copy of NL-11-032, which was Entergy's March 28, 2011 initial response to the USNRC Staff's request for additional information. In this document starting on page 20 of Attachment 1, Entergy discusses, among other things, the material composition of the steam generator divider plates and associated welds.

Q. Has Entergy disclosed the composition of the heat transfer tubes in Indian Point's steam generators?

A. Yes, in the License Renewal Application at page 2.3-21, Entergy stated that the current Indian Point Unit 2 steam

generators use Alloy 600 for the heat transfer tubes and that the current Indian Point Unit 3 steam generators use Alloy 690 for the tubes.

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, 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 have noted previously, are derivatives of Alloy 600. In the August 30, 2011 Supplemental Safety Evaluation Report at page 3-21, the USNRC Staff has also expressed concern about the propagation of primary water stress corrosion cracking in tubesheets that have Alloy 600 cladding, or the related welds, even when the U-tubes are made from Alloy 690 material. According to the 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." The specific areas of concern include the channel head-to-tubesheet-to-tube complex, including the divider plate assembly and the tube-to-tubesheet welds.

Q. In your opinion, would primary water stress corrosion cracking of the divider plates, weld, or channel head assemblies

impact the intended function of the steam generators?

A. Yes, in my opinion it could.

First, the shock-load-induced failure of a divider plate and/or its welds, which have been subjected to thermal fatigue and primary water stress corrosion cracking (PWSCC), could compromise the ability of the divider plate to direct fluid through the U-tubes and hence impede one of the intended functions of the tubes and the steam generator, namely, to provide a heat sink for the heat generated in the core. I would consider the loss of that intended function to be a significant safety concern since shock-load-induced failures of the divider plates have apparently not been analyzed (e.g., the thermal/pressure shock loads experience during various postulated LOCA events), however such events may lead to gross failures of cracked divider plates. [redacted]

[redacted] While I agree with this narrow conclusion, it is not at all what I am concerned about. Instead, as I have indicated previously, what I am worried about is that when a seriously age-weakened (e.g., due to thermal fatigue and PWSCC-induced embrittlement) divider plate is subjected to a severe thermal or pressure shock load that it may fail catastrophically, opening a large flow area between the hot and cold plena of the steam generator. If so, the affected steam generator could be lost as a heat sink, and this, in turn, may seriously compromise subsequent core cooling.

Also, the USNRC Staff has proposed that a primary water stress corrosion crack in the divider plate might propagate into a tubesheet and a tube-to-tubesheet weld. If so, such a crack in the lower steam generator assembly area could compromise another important function of the steam generator, namely the maintenance of the reactor coolant pressure boundary between the primary loop and the secondary loop in the nuclear steam supply system.

Entergy's Approach to Steam Generator Issues

Q. Do you have any opinion about the sufficiency of the approach that Entergy has proposed regarding primary water

stress corrosion cracking of steam generator components at the Indian Point facilities?

A. Yes. As I stated in the Fall of 2011, and in my previously filed testimony on Contention NYS-38/RK-TC-5, Entergy's proposal leaves many important details and questions unresolved. As the Supplemental Safety Evaluation Report confirms (at p. 3-19) [Exh. NYS000160], Entergy proposes that it will perform "an inspection of steam generators for both units to assess the condition of the divider plate assembly." However, this proposal does not describe the inspection methodology nor the number of steam generators to be inspected. Indeed, it is quite vague and does not provide details. Similarly, it does not describe the acceptance criteria for such inspection or the corrective action criteria for divider plates that fail the inspection.

Turning to the issue of cracks spreading from tubesheet cladding to tube-to-tubesheet welds, Entergy again proposes an approach that is short on details. Specifically, Entergy proposes to "develop a plan" that will use one of two options: (1) "perform an analytical evaluation" to establish that tubesheet cladding and welds are not susceptible to primary water stress corrosion cracking, or redefine the reactor coolant

pressure boundary to exclude the tube-to-tubesheet welds; or, (2) perform a one-time inspection of a representative number of tube-to-tubesheet welds in each steam generator to determine if primary water stress corrosion cracking is present. This plan that Entergy has proposed to develop leaves many questions unanswered, including: (1) the basis for the proposed analysis; (2) how Entergy can simply change the definition of the reactor coolant pressure boundary for the tube-to-tubesheet welds after pressurized water nuclear reactors (PWRs) have relied on that definition for many years; and, (3) the methodology of the alternative one-time inspection.

In each regard, Entergy has not presented an aging management program (AMP), but rather has presented a vague, conceptual approach.

Q.   Dr. Lahey I show you what has been marked as Exhibit NYS000160. Do you recognize it?

A.   Yes, this is a copy of the USNRC Staff's August 2011 Supplemental Safety Evaluation Report (SSER) for the requested renewal of the operating licenses for the Indian Point reactors [NUREG-1930, Supp. 1].

Q.   Did you review the Supplemental Safety Evaluation Report?

A. Yes.

Q. I also show you Exhibits NYS000151, NYS000152, NYS00153, and NYS00154. Do you recognize them?

A. Yes, these are 2011 communications from Entergy to the USNRC Staff in response to Staff questions about the age-related degradation of various components at Indian Point Unit 2 and Indian Point Unit 3. These are Entergy communications NL-11-032, NL-11-074, NL-11-90, and NL-11-096.

Q. Did you reach any conclusions based on that review?

A. I have reviewed the USNRC Staff's 2011 Supplemental Safety Evaluation Report for Indian Point Unit 2 and Unit 3. The SSER makes it clear that a number of important details and questions remain unresolved concerning the aging-induced degradation of various safety-related systems and components and the management of that process. Unfortunately, there are virtually no details given on the future analyses and/or inspections that Entergy will apparently do. The absence of such details makes it difficult, if not impossible, to meaningfully evaluate the approach or program that Entergy proposes. In any event, the dates given for Entergy and the USNRC's anticipated resolution of some these issues appear to be beyond the time frame for submission of testimony and the

evidentiary hearings in this ASLB proceeding and thus will not allow for a testing of the adequacy of the proposed resolution of these issues in this proceeding. That timeline will also prevent the State of New York from playing a meaningful role in their development or resolution.

For example, the details of the inspections for primary water stress corrosion cracking (PWSCC) in the steam generator's divider plates and associated assemblies will apparently not be available until after extended operations begin. Moreover, Commitment 41 provides no meaningful details as to the methodology and criteria for such inspections.

In a similar way, under Commitment 42 Option 2, inspections of the four steam generators' tube-to-tubesheet welds in Indian Point Unit 3 for PWSCC will not be made until the first refueling outage after the reactor enters the period of extended operation. Indian Point Unit 3 could enter its period of extended operation in late December 2015. Based on the current refueling schedules, which have Indian Point Unit 3 refueling in March of odd numbered years, I anticipate that the first refueling outage for Indian Point Unit 3 after it enters the period of extended operation in December 2015 would be in or around March 2017.

With respect to tube-to-tubesheet welds in Indian Point Unit 2’s four steam generators, under Commitment 42 Option 2 inspections for PWSCC were to take place sometime between March 2020 and March 2024, which is well after the proposed extended operation period has begun. As I previously noted, this is particularly troubling since these welds do in fact form part of the primary system’s pressure boundary, and if they fail radiation will be released to the secondary side and subsequently to the environment. However, although the tube-to-tubesheet welds clearly form a part of the reactor coolant pressure boundary, Entergy and the USNRC Staff have pursued a different approach under Option 1 and have recently agreed to “redefine” that boundary to exclude the numerous tube-to-tubesheet weld locations in each of the four steam generators at Indian Point Unit 2. [Entergy Letter NL-14-001 (Jan. 16, 2014)(Exh. NYS000539), NRC IP2 Operating License Amendment No. 277 (Sept. 5, 2014) (Exh. NYS000542).] As noted in my previously filed rebuttal testimony [Exh. NYS000453 at 11-12], changing this definition does not resolve NYS’ concerns or eliminate the physical pathway through which radiation can be released to the environment.

Q. [redacted]

[redacted]

A. [redacted]

[redacted] [redacted]

[redacted] [redacted]

Conclusion Regarding Steam Generator Issues

Q. Do you have any response to Entergy's and the USNRC Staff's submissions concerning their approach to the steam generators at the two Indian Point reactors?

A. Yes. After reading their 2012 testimony, it remains my opinion that Entergy's approach to aging management still contains many of the unknowns and questions with respect to the safety concerns that I have raised in my previous NYS-38 filings.

For example, despite their testimony, neither Entergy nor the USNRC Staff provide sufficient details about what inspection technique they will use to inspect the divider plates and assemblies and the tube-to-tube sheet welds in the Westinghouse steam generators at Indian Point.

Moreover, while the implications on the primary pressure boundary integrity of the tube-to-tubesheet welds, and the integrity of the steam generator's divider plates due to thermal fatigue and PWSCC-induced crack propagation, are obviously important, another important age-related safety issue that has apparently not yet been addressed is the potential for the thermal-fatigue-weakened and PWSCC-embrittled steam generator divider plates to experience gross failure due to various

accident-induced thermal/pressure shock loads (e.g., a steam line break and reactor SCRAM). Although Entergy's previous testimony referred to two EPRI reports, EPRI Report 1020988, "Steam Generator Management Program: Phase II Divider Plate Cracking Engineering Study", November 2010 (Exh. ENT000523), and, EPRI Report No. 1025133, "Steam Generator Management Program: Assessment of Channel Head Susceptibility to Primary Water Stress Corrosion Cracking", June 2012 (Exh. ENT000524), those confidential reports do not resolve my concerns. [redacted] This is an important safety concern, since if there is gross failure of the divider plate then the primary coolant may bypass the steam generator's U-tubes thus eliminating the ultimate heat sink. If this occurs, the natural circulation cooling relied upon, for example, during station blackout (SBO) or during various anticipated transient without SCRAM (ATWS) events, will not occur and thus the core may overheat and melt, leading to the release of significant radioactive materials. While it is

true that a feed-and–bleed mode of core cooling can be used in this event, this emergency procedure may not be adequate to prevent core damage during rapid transients. Clearly, this important safety issue needs to be resolved as part of the ASLB relicensing hearings on the two Indian Point reactors. For the reasons expressed in my earlier rebuttal testimony [NYS000453 at 9-11, 12-14], it is my opinion that Entergy [ENT000521] and the USNRC Staff [NRC000161] have not addressed this concern.

It appears to me that the reason that the industry is not concerned with this important failure-mode is likely because no one has considered the degradation due the synergistic effects of the PWSCC-induced embrittlement and thermal fatigue that the divider plates have been subjected to, and no one has carefully evaluated the severe shock loads that the degraded divider plates may experience due to various postulated accidents (e.g., a steam line break and an associated SCRAM). It should be noted that to do these analyses properly requires special computer codes that can accurately track the thermal and pressure transients on the divider plate. In any event, the quasi-static analysis that EPRI has done for LOCA loads does not address my concerns at all. Moreover, the thermal-hydraulics code that EPRI normally uses, RETRAN, is totally inadequate for this

purpose, as it numerically smears steep spatial transient variations in both temperature and pressure. Also, since WESTEMS was used and is relied on for the fatigue analyses (i.e., CUF_{en} results) we have the same issues [redacted] the lack of an error analysis, etc., as has been discussed in Contention NYS-26B. In any event, a much more detailed analysis of the SG divider plates is clearly needed.

Closing Observation

Q. Do you have any further comments?

A. Yes. I note that an IP3 steam generator feedwater line failed in May 2015 and that the plant had to shut down as a result [Event Report 51046 (Exh. NYS000548)]. Two days later, a transformer failure caused the plant to again shut down [Event Report 51060 (Exh. NYS000561)], and thus the plant experienced two unexpected failures in one week. This information highlights the fact that age-related degradation concerns are not hypothetical, they actually occur and can have real-world consequences. Thus, an adequate aging management program (AMP) is essential to assure safe and reliable operations during plant life extension. Unfortunately, I do not believe that such a program has been developed to date for IP2 and IP3.

Q. Does this complete your testimony?

A. Yes, it does. I do, however, reserve the right to supplement my testimony if new information is disclosed or introduced.

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 9, 2015

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DECLARATION OF RICHARD T. LAHEY, JR.

I, Richard T. Lahey, Jr., 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)

R. T. Lahey

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Dr. Richard T. Lahey, Jr.
The Edward E. Hood Professor Emeritus of Engineering
Rensselaer Polytechnic Institute, Troy, NY 12180
(518) 495-3884, laheyr@rpi.edu