United States Nuclear Regulatory Commission Official Hearing Exhibit
In the Matter of: Entergy Nuclear Operations, Inc. (Indian Point Nuclear Generating Units 2 and 3)
ASLBP #: 07-858-03-LR-BD01
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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.	December 22, 2011

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PRE-FILED WRITTEN TESTIMONY OF

RICHARD T. LAHEY, JR.

REGARDING CONSOLIDATED CONTENTION NYS-26B / RK-TC-1B

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. regarding Contention NYS-25.

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 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 nuclear reactors. Also, I am a member of a number of professional societies and have served on various 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), and the Electric Power Research Institute (EPRI). I am a member of the National Academy of Engineering (NAE), and have been elected Fellow of both the ANS and the ASME.

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, 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 awards.

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

A. Yes. They are copies of the six declarations that I previously prepared to date for the State of New York in this proceeding. They include the 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 25, the September 9, 2010 declaration submitted in support of the amended Contention

NYS26B/RK-TC-1B, and the September 30, 2011 and November 1, 2011 declarations submitted in support of 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. The Report reflects 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 concerning certain components the Indian Point reactors. The supplemental report also reflects my analysis and opinions.

Q. What is the purpose of your testimony?

A. I was 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 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.

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, and various related text books and peer-reviewed articles.

Q. I show you Exhibits NYS00146A-C, NYS00147A-D, NYS000160, NYS000161, NYS000195, and NYS000304 through NYS000369. 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, 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 Report (Exh. NYS000296) entitled "Reference Documents," which contains a list of documents. Would you describe that list?

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

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 power reactors that are located in northern Westchester County near the Village of Buchanan. The operating reactors are known as the Indian Point Unit 2 and

Indian Point Unit 3 power reactors. These Westinghouse-designed plants are 4-loop pressurized water reactors (PWRs), and they are currently rated at power levels of 3,216.4 MWt. 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. 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 its name 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, as well as 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 various materials that are used or contained in those 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" 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 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. 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, and various weldments, including welds in 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. The over-arching age-related safety issues that I am concerned with have to do with the fact that the various degradation mechanisms and the safety analyses have been done separately (i.e., in "silos"), and thus it has been implicitly assumed that there is no interaction between them. In addition, the fatigue evaluations which were presented by Entergy (i.e., those done for them by Westinghouse using the WESTEMS computer code) focused on the fatigue of various piping systems and piping fittings, but not the fatigue of reactor pressure vessel internals. Moreover, the metric used to appraise their fatigue-induced damage (CUF_{en}) only considered quasi-static low and high cycle fatigue evaluations, rather than accident-induced shock loads which may cause failures well before the standard fatigue limit is reached (i.e., $CUF_{en} = 1.0$). Also, no error analyses were presented to quantify the WESTEMS predictions for various piping systems and fittings even though some of them were extremely close to the $CUF_{en} = 1.0$ limit.

Anyway, in my opinion, probably the most serious short coming of this "siloing" approach is that synergistic

interactions between radiation-induced embrittlement, stress corrosion cracking, and fatigue-induced degradation mechanisms have not been considered; for example, the fatigue-induced failure of seriously embrittled reactor pressure vessel (RPV) internals. Also, when the plant's safety analyses were done by Entergy it was implicitly assumed that the incore geometry would remain intact during postulated accidents. Unfortunately, unlike ductile metals, seriously embrittled and fatigued RPV internals may not be able to survive the pressure and/or thermal shock loads induced by various accidents and severe 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 their plants can be safely operated beyond the 40 year design life of these plants. 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. 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 refers to the change in the mechanical properties (or 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 strength. While the initial aging effect is loss of ductility and toughness, unstable crack propagation is the eventual aging effect if a crack is present and the local applied stress intensity exceeds the reduced fracture toughness. This is particularly true for the large pressure and/or thermal shock loads associated with postulated accidents.

The USNRC's guidance document 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." That is from GALL, Revision 2 at page IX-34 (Exhibit NYS000147). I note that the phrase "low-alloy steels" includes stainless steel.

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 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 RPV internals), the temperature required for them to maintain some 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. A degradation in ductility (or embrittlement) will adversely affect a PWR's ability to withstand 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.

Various accidents can expose a reactor pressure vessel and its internal structures, components and fittings to significant pressure and/or thermal shock loads. If the reactor pressure vessel’s internal structures are sufficiently degraded due to stress corrosion cracking (SCC), fatigue and/or radiation-induced embrittlement, such accidents can have significant consequences. Indeed, the resultant stresses from such accidents may cause the reactor pressure vessel internals to

fail structurally. 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 a loss of coolant accidents (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 which causes a reactor SCRAM and thus a rapid cooling of the primary coolant via the steam generators. This accident leads to a severe pressurized thermal shock of the reactor pressure vessel and the associated RPV internals.

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 failure of a highly embrittled reactor pressure vessel internal structures, components and fittings and thus the inability to subsequently cool the core. It should be noted that in the past most of the attention has been focused on the

integrity of the reactor pressure vessel. However, the RPV internals are much less massive and are much closer to the core, and thus suffer a lot more radiation damage.

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

A. Yes. How the rather complex, interacting metal degradation mechanisms associated with fatigue, irradiation and corrosion interact is still an area of research (e.g., how fatigue-induced surface cracks propagate in an embrittled, as opposed to ductile, metal structure). 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, pg. 16 (Feb. 10, 2001) (Exh. NYS000324)]. Indeed, it is also well known that, "… irradiation embrittlement decreases the resistance to crack propagation," [Westinghouse Owners Group WCAP-14577 Rev. 1-A Report, pg. 3-2 (March 2001) (Exh. NYS00307A-D)]. Anyway, the radiation-induced damage to RPV internals can be extensive, since they can experience a neutron fluence of at least 10^{23} n/cm2 at neutron energy (E) levels of E > 1 MeV (i.e., > 100 dpa) [Was, 2007

(NYS000339); EPRI (Dyle), 2008 (NYS000322); WOG WCAP-14577 Rev. 1-A Report, March 2001 (NYS000341)] by the end of life (EOL) for extended operations. It should be noted 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. Thus, the RPV internals will be much more embrittled.

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 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. 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 USNRC 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, the 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 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.

Entergy's Opposition to NYS Contention 25

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, it 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. While this was a popular belief some years ago, unfortunately 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 2010.

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

A. Yes. Chapter IV and Chapter XI 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 stress corrosion 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 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 stress corrosion 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 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. It is important to note that the focus of this document is on non-destructive testing (NDT) and non-destructive evaluation (NDE) rather than on the implications on core coolability subsequent to any 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 techniques rather than my aging-related safety concerns.

Entergy's License Renewal Application

Q. Directing you 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 , and,
- nickel alloy
- stainless steel

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

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

Q. Did you find any indication in the LRA that Entergy 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 (or CASS)
- nickel alloy
- 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
- 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 IP-3 reactor pressure vessel itself.

Q. And what did Entergy say there?

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

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

A. Yes, Entergy acknowledged that with respect to IP-3 that the reactor pressure vessel plate B2803-3 "exceeds the screening criterion by 9.9°F." That statement is in 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 IP-3 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 in the future Entergy would develop an aging management program for the reactor pressure vessel internals of their plants. That is stated at LRA Appendix A, § A.2.1.41 with respect to IP-2 and § A.3.1.41 with respect to IP3. This deferred approach concerning IP-2 and IP-3 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. 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 need to achieve a chain reaction) so that the power level of a reactor can be controlled.

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. Because of geometric considerations, many PWRs (including IP-2 and IP-3) 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)], 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 [Telecon-USNRC/Entergy Report, March 18, 2008 (Exh. NYS000317)], but they have not yet been scheduled for installation at Indian Point [Telecon-USNRC/Entergy, March

18, 2008]. In any event, unlike the rather superficial treatment given this important safety concern by Entergy [NL-10-063 (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 the 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, stress corrosion cracking (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 uncontrolled reactivity excursion).

Entergy's NL-10-063 Communication

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 pg. 89. Moreover, EPRI has stated that, “considerable amount of PWR internals aging degradation has been observed in European PWRs.” EPRI MRP-227, at A-4 (NYS00307A-D). 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 pg. 89. Moreover, material degradation has also been observed in control rod guide tube alignment (split) pins. EPRI MRP-227, at A-4.

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 an otherwise 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 RPVIs 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, 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; see also State of New York Notice of Intention to Participate and Petition to Intervene, at pgs. 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, stress corrosion 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 cumulative 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 acceptance 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 acceptable.

Fatigue

Q. Let us now consider 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 for nuclear power plants. Fatigue of various structures, components and fittings in a nuclear reactor can result in pipe ruptures, physical failures, and the relocation of loose pieces of metal throughout the reactor system, which, in turn, may result in core blockages and interfere with the safe operation of a nuclear power plant. The main concerns about fatigue are the increased potential for a

primary or secondary side LOCA, and the failure of various RPV internals.

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. As I have testified to earlier in my Overview remarks, they have focused on the fatigue of some selected piping systems and components and, even then, have calculated some near limiting values (e.g., CUF_{en} = 0.9961 for the RHR line of IP-3). No error analyses were presented to quantify the modeling uncertainties and the effect of code user interactions in WESTEMS, however, it appears that virtually any error would put the calculated value of CUF_{en} over the CUF_{en}=1.0 fatigue limit. In any event, Entergy has not done a systematic fatigue evaluation of limiting RPV internals, considering the synergistic effect of radiation-induced embrittlement and stress corrosion cracking on fatigue-induced failures, and the possible failures due to accident-induced pressure and/or thermal shock loads. These latter safety evaluations are particularly important in the evaluation of core coolability subsequent to postulated accidents (i.e., the potential for core blockages due to the relocation of failed RPV internals).

Q. I show you your Supplemental Report, which has been marked as Exhibit NYS000297. I note that the State has provisionally designated 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 refined CUF_{en} analysis 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 a detailed 'propagation-of-error' type of analysis to determine the overall uncertainty in the CUF_{en} results given by Westinghouse. I referenced a text, "Basic Engineering Data Collection and Analysis," pp. 310-311, by Vardeman & Jobe, to support this position, which is Exhibit NYS000347.

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. That is set our in Exhibit NYS000160 at page 4-2. 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). 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.

Entergy's NL-11-107 Communication

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

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 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 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, 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. In addition, a systematic evaluation of the degraded RPV internals is needed to identify the limiting structures, components and fittings that need to be repaired or replaced before the onset of extended operations.

Conclusion

Q. You have reviewed the GALL Report Revision 1, the Standard Review Plan Revision 1, the GALL Report Revision 2, the Standard Review Plan Revision 2, EPRI's MRP-227 Revision 0, Entergy's July 2010 NL-10-063 communication, NRC Staff's June 22, 2011 Safety Evaluation of MRP-227 Revision 0, NRC Staff's August 30, 2011 Supplemental Safety Evaluation for the Indian Point License Renewal Application, and Entergy's NL-11-107 communication, correct?

A. Yes.

Q. Do you have any opinion about those documents with respect to the degradation of reactor pressure vessel internals?

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 Contention 25 and in my September 2010 declarations in support of the State's supplemental filings, in my professional judgment Entergy has failed to demonstrate that it had adequately accounted for the aging phenomena of embrittlement for structures, components and fittings inside the reactor pressure vessels 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] and Entergy's September 28, 2011 submission of NL-11-107. I do not believe the Entergy's July 15, 2010 communication to the Board [NL-10-063] concerning a new AMP for RPV internals is adequate to address the safety concerns and technical issues that I have raised herein. NL-11-107 likewise does not address my age-related safety concerns, nor does it recognize the importance of the various synergistic degradation mechanisms that I am concerned with.

Q. Does this conclude your testimony?

A. Not quite. 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. In particular, the synergistic effect on the degradation and integrity of RPV internals of radiation-induced embrittlement, corrosion and fatigue, and the potential for the failure of RPV internals due to accident-induced pressure and/or shock loads, and the implications of the failure of RPV internal structures, components and fittings on post-accident core coolability. While in the past many of these issues and concerns have been noted separately, the implications of their interaction has

apparently been overlooked and not evaluated (i.e., they have been evaluated in “silos”). 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 need to be 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 and the resultant potential for core blockages and degraded core cooling.

Q. Does this complete your testimony.

A. Yes, it does.

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.	December 22, 2011

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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)

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