

**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 CONTENTION NYS-25**

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

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 Q. Please summarize your educational and professional  
2 qualifications.

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

23 Q. Which professional societies are you a member of?

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

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

12           Q.    What expert panels have you served on?

13           A.    I have served on numerous panels and committees for  
14 the: United States Nuclear Regulatory Commission (USNRC), Idaho  
15 National Engineering Laboratory (INEL), Oak Ridge National  
16 Laboratory (ORNL), National Aeronautics and Space Administration  
17 (NASA), National Research Council(NRC) and the Electric Power  
18 Research Institute (EPRI). I am a member of the National  
19 Academy of Engineering (NAE), have been elected Fellow of both  
20 the ANS and the ASME, and have been a Fulbright-Hays, Alexander  
21 von Humboldt and Japanese Society for the Promotion of Science  
22 (JSPS) Scholar.

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

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

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

7           Q.    Have you received any professional awards?

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

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

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

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

22          A.    Yes.  They are copies of the seven declarations that I  
23 previously prepared to date for the State of New York in this

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 proceeding. They include my initial declaration that was  
2 submitted in November 2007 in support of the State's petition to  
3 intervene and its initial contentions, the April 7, 2008  
4 declaration in support of Contention NYS-26A, the September 15,  
5 2010 declaration submitted in support of the State's  
6 supplemental bases for Contention 25, the September 9, 2010  
7 declaration submitted in support of the amended Contention  
8 NYS26B/RK-TC-1B, the September 30, 2011 and November 1, 2011  
9 declarations submitted in support of Joint Contention NYS-38/RK-  
10 TC-5, and the February 12, 2015 declaration submitted in support  
11 of additional bases for Contention NYS-25 and Joint Contention  
12 NYS-38/RK-TC-5.

13 Q. I show you what has been marked as Exhibit NYS000296.  
14 Do you recognize that document?

15 A. Yes. It is a copy of the Report that I prepared for  
16 the State of New York in this proceeding. This Report documents  
17 my analysis and opinions.

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

20 A. Yes. This is a copy of a Supplemental Report that I  
21 prepared for the State of New York in this proceeding that  
22 addresses aspects of the revised fatigue analysis that Entergy  
23 and Westinghouse prepared for certain components in the Indian

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 Point reactors. The supplemental report also documents my  
2 assessment and opinions of this work.

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

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

8 Q. What is the purpose of your testimony?

9 A. I have been retained by the State of New York State to  
10 review Entergy's application to the U.S. Nuclear Regulatory  
11 Commission (USNRC) and its Staff for two renewed operating  
12 licenses for the nuclear power plants known as Indian Point Unit  
13 2 and Unit 3. I have reviewed the License Renewal Applications  
14 (LRAs) and subsequent filings by Entergy and the USNRC Staff.  
15 My declarations and report discuss my concerns and opinions  
16 about issuing twenty-year extended operating licenses for these  
17 facilities. My testimony seeks to identify and discuss some  
18 age-related safety concerns which have not yet been addressed by  
19 Entergy. In my opinion these concerns must be resolved to  
20 assure the health and safety of the American public,  
21 particularly those in the vicinity of the Indian Point reactors.

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

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 A. Yes.

2 Q. What is the source of those materials?

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

7 Q. I show you Exhibits NYS00146A-C, NYS00147A-D,  
8 NYS000160, NYS000161, NYS000195, NYS000304 through NYS000369,  
9 and NYS000484 through NYS000525. Do you recognize these  
10 documents?

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

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

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

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

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

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

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

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

14 **The Indian Point Reactors**

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

17 A. Yes.

18 Q. Would you briefly describe them?

19 A. Entergy operates two nuclear power reactors that are  
20 located in northern Westchester County near the Village of  
21 Buchanan. The operating nuclear reactors are known as the  
22 Indian Point Unit 2 and Indian Point Unit 3 reactors. These  
23 Westinghouse-designed plants are 4-loop pressurized water

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*



1 reactors (PWRs), and they are currently rated at power levels of  
2 3,216.4 MW<sub>t</sub>. Entergy also owns another reactor at the same site.  
3 That reactor is known as the Indian Point Unit 1 reactor;  
4 however, that reactor has been shut down and no longer produces  
5 power.

6 **Operation of a Pressurized Water Reactor**

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

9 A. Pressurized water nuclear reactors have water (i.e.,  
10 the primary coolant) under high pressure flowing through the  
11 core in which heat is generated by the fission process. The  
12 core is located inside a reactor pressure vessel (RPV). This  
13 heat is absorbed by the coolant and then transferred from the  
14 coolant in the primary system to lower pressure water in the  
15 secondary system via a large heat exchanger (i.e., a steam  
16 generator) which, in turn, produces steam on the secondary side.  
17 These steam generator systems, which are part of the plant's  
18 Nuclear Steam Supply System (NSSS), are located inside a large  
19 containment structure. After leaving the containment building,  
20 via main steam piping, the steam drives a turbine, which turns a  
21 generator to produce electrical power.

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

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

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

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

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

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

3 **Reactor Pressure Vessel Internals**

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

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

12 Q. Please describe what is encompassed by the term  
13 "reactor pressure vessel (RPV) internals"?

14 A. The term "reactor pressure vessel internals" (i.e.,  
15 RVIs) includes various structures, components, and fittings  
16 inside the reactor pressure vessel including the: core barrel  
17 (and its welds), core baffle, intermediate shells, former  
18 plates, lower core plate and support structures, clevis bolts,  
19 fuel alignment pins, thermal shield, the lower support column  
20 and mixer, upper mixing vanes, and the upper/lower core  
21 assemblies and support column, and the control rods and their  
22 associated guide tubes, plates, and welds. Reactor pressure  
23 vessel internals (RVIs) also include the bolts that hold various

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 components together or to other components including: the  
2 baffle-to-baffle bolts, the core barrel-to-former bolts, and  
3 baffle-to-former bolts as well as the welds or weldments that  
4 hold sections of these components together.

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

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

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

16 A. Yes. Entergy has argued that the control rods are not  
17 reactor vessel internals. However, the control rods and their  
18 associated guide tubes, plates, pins and welds are located in  
19 the core region of the RPV, and the control rods are inserted  
20 into the RPV through the upper head through so-called stub  
21 tubes. The function of the control rods is to absorb excess  
22 fission neutrons (i.e., those not needed to achieve a chain  
23 reaction) so that the power level of a reactor can be

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 controlled. Accordingly, the control rods and associated  
2 components are very important RPV internals and their integrity  
3 is an extremely important safety concern. While the control  
4 rods are moving parts and can be replaced as required, many of  
5 the other associated components are not moving parts and are not  
6 normally replaced. In any event, if a shock load occurs (e.g.,  
7 during a LOCA or severe earthquake) any of these seriously  
8 embrittled structures may fail and lead to degraded core  
9 cooling. Thus, in my opinion, omitting the control rod  
10 assemblies and associated fittings from an RPV internals (RVIs)  
11 aging management program is a serious and indefensible omission.

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

14 A. Yes. They are a collection of additional schematic  
15 figures from the Electric Power Research Institute's Report MRP-  
16 227 that provide additional detail concerning various reactor  
17 pressure vessel internals and their location within the reactor  
18 pressure vessel. The reactor pressure vessel internals shown  
19 include the control rod guide tube assembly, the control rod  
20 guide cards, guide tube support pins, the control rods, baffles,  
21 formers, baffle-former assemblies, baffle-to-former bolts,  
22 corner edge bracket baffle to former bolts, core barrel to  
23 former bolts, baffle plate edge bolts, core support structures,

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 and various weldments, including welds within the reactor  
2 pressure vessel for the core barrel plates.

3 **Overview**

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

7 A. My over-arching concern relates to Entergy's "silo"  
8 type approach to evaluating the impact of various aging  
9 mechanisms such as embrittlement and fatigue, and the company's  
10 failure to consider, as part of plant safety analyses, the  
11 potential consequence of unanticipated shock loads (e.g., those  
12 due to design basis accidents) on severely fatigued and  
13 embrittled components. Entergy implicitly assumes that there is  
14 no interplay between the various material aging degradation  
15 phenomena and that degraded components will have no impact on  
16 the plants' ability to safely operate, particularly during  
17 unanticipated shock loads. For example, Entergy's fatigue  
18 evaluations, performed by Westinghouse using the WESTEMS  
19 computer code, used the metric  $CUF_{en}$  to appraise environmentally  
20 assisted fatigue in various reactor components. However, these  
21 evaluations were quasi-static low and high cycle fatigue  
22 evaluations that considered neither the effect of neutron-  
23 induced embrittlement nor the combined effects of fatigue damage

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 and other degradation mechanisms such as radiation enhanced  
2 corrosion-induced cracking. In both the fatigue analyses and  
3 the plant safety analyses, it was implicitly assumed that  
4 fatigue weakened and embrittled structures, components and  
5 fittings would respond to shock loads in the same way as if they  
6 were ductile, which is simply not true. Also, no error analyses  
7 were presented to quantify the WESTEMS predictions for the  
8 various internals, piping systems and fittings even though some  
9 of them were extremely close to the  $CUF_{en} = 1.0$  failure limit. In  
10 any event, under these circumstances, various operational and  
11 accident-induced shock loads could cause failures well before  
12 the fatigue limit is reached (i.e., when  $CUF_{en} < 1.0$ ), and  
13 therefore reliance on inspection-based fatigue monitoring does  
14 not provide adequate assurance that the degraded components will  
15 not fail.

16       Once again, the most serious short-coming of this "siloing"  
17 approach is that synergistic interactions between radiation-  
18 induced embrittlement, corrosion-induced cracking, and fatigue-  
19 induced degradation mechanisms have not been considered. For  
20 example, neither Entergy's license renewal application nor its  
21 proposed aging management plan consider the potential for, or  
22 the consequences of, fatigue-induced failure of seriously  
23 embrittled reactor pressure vessel internals (RVIs). Also, when

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 the plant's safety analyses were done by Entergy it was  
2 implicitly assumed that the in-core geometry would remain intact  
3 during postulated accidents. Unfortunately, unlike ductile  
4 metals, seriously embrittled and fatigued RPV internals may not  
5 be able to survive the shock loads associated with significant  
6 seismic events or the pressure and/or thermal shock loads  
7 induced by various accidents and severe operational transients.  
8 If not, they can fail and relocate, possibly causing core  
9 blockages that degrade core cooling and may lead to core melting  
10 and massive radiation releases.

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

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

21 A. I mean that the concurrent exposure of reactor  
22 components - especially RVI components - to multiple aging  
23 mechanisms that occur in a reactor core (including fatigue,

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*



1 irradiation embrittlement, and corrosion) may result in  
2 cumulative material degradation that exceeds the predicted  
3 combined degradation for each aging mechanism acting alone.

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

6 A. Yes. However, the rather complex and interacting  
7 metal degradation mechanisms associated with fatigue,  
8 irradiation and corrosion interact is still an area of active  
9 research (e.g., how fatigue-induced cracks propagate in an  
10 embrittled, as opposed to ductile, metal structure). In fact,  
11 the Department of Energy (DOE) and USNRC, in conjunction with  
12 various national laboratories, have recently embarked on an  
13 ambitious R&D program to understand and resolve issues related  
14 to these interacting and synergistic effects [NUREG/CR-7153,  
15 Vol. 2, "Expanded Materials Degradation Assessment (EMDA), Aging  
16 of Core Internals and Piping Systems" (October 2014), at 1-5  
17 (Exh. NYS00484A-B)]. In addition, the federal government has  
18 also embarked on a fairly large research program, known as the  
19 Light Water Reactor Sustainability Program, which includes  
20 research into whether the different materials and LWR components  
21 can continue to perform their intended function during the  
22 extended operation of a nuclear reactor. [DOE, Light Water

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 Sustainability Program, Material Aging and Degradation Technical  
2 Program Plan (August 2014) (Exh. NYS000485)].

3       Nevertheless, it is well known that, "the effects of  
4 embrittlement, especially loss of fracture toughness, make  
5 existing cracks in the affected materials and components less  
6 resistant to growth" [USNRC Letter, Grimes to Newton, at 16  
7 (Feb. 10, 2001) (Exh. NYS000324); see Stevens, Gary L.,  
8 Presentation to the ACRS on "Technical Brief on Regulatory  
9 Guidance for Evaluating the Effects of Light Water Reactor  
10 Coolant Environments in Fatigue Analyses of Metal Components"  
11 (December 2, 2014), at 56-58 (Exh. NYS000486); Chopra, O.K.,  
12 "Degradation of LWR Core Internal Materials due to Neutron  
13 irradiation," NUREG/CR-7027 (Dec. 2010) (Exh. NYS000487)], and,  
14 "irradiation embrittlement decreases the resistance to crack  
15 propagation" [Westinghouse Owners Group WCAP-14577 Rev. 1-A  
16 Report, at 3-2 (March 2001) (Exh. NYS00307A-D)]. Moreover, a  
17 recent report, prepared by Argonne National Laboratory for the  
18 USNRC, acknowledges, with respect to cast austenitic stainless  
19 steels (CASS), that "a combined effect of thermal aging and  
20 irradiation embrittlement could reduce the fracture resistance  
21 even further to a level neither of these degradation mechanisms  
22 can impart alone" [Chen, et al., "Crack Growth Rate and Fracture  
23 Toughness Tests on Irradiated Cast Stainless Steels," NUREG/CR-

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 7184 (Revised December 2014), at xv (Exh. NYS00488A-B)].

2 Indeed, nuclear industry groups have now recognized the  
3 potential for synergistic aging effects in CASS RVI components  
4 [EPRI, Slides, Industry-NRC Meeting on CASS Screening Criteria  
5 for Thermal and Irradiation Embrittlement for BWR and PWR  
6 Internals" (July 15, 2014) (Exh. NYS000489)].

7 Q. Are synergistic aging effects limited to CASS  
8 components?

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

13 Q. Are these synergistic aging effects fully understood?

14 A. Not at all. Multiple recent reports and studies from  
15 USNRC, DOE, and associated contractors recognize the lack of  
16 understanding of the interrelationship between embrittlement,  
17 high or low cycle fatigue, and shock loads for highly fatigued  
18 and/or embrittled components made of CASS, non-cast stainless  
19 steels, or other alloys. In addition, the consequences of the  
20 interaction of embrittlement, fatigue, and the corrosion-induced  
21 degradation of various reactor pressure vessel internals (RVI),  
22 and safety-related components/systems during shock loads,  
23 remains unknown [see, e.g., NUREG/CR-6909 Rev. 1 (March 2014

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 (draft) (Exh. NYS000490)), at 11 ("it is not possible to  
2 quantify the impact of irradiation on the prediction of fatigue  
3 lives in PWR primary water environments compared to those in  
4 air."); NUREG/CR-7153, Vol. 2, "Expanded Materials Degradation  
5 Assessment (EMDA), Aging of Core Internals and Piping Systems"  
6 (October 2014), at 3 (Exh. NYS00484A-B)]. The Argonne National  
7 Laboratory report described above states that, "no data are  
8 available at present with regard to the combined effect of  
9 thermal aging and irradiation embrittlement" on CASS [Chen, et  
10 al., NUREG/CR-7184, at xv (Exh. NYS00488A-B); see also Chopra,  
11 O.K., "Degradation of LWR Core Internal Materials due to Neutron  
12 irradiation," NUREG/CR-7027 (Dec. 2010) (Exh. NYS000487)]. As  
13 noted before, the same is also true for the interaction of  
14 irradiation-induced embrittlement, corrosion, and fatigue of  
15 non-cast stainless steel RVIs.

16 A recent paper presented at an MPA Seminar in Stuttgart,  
17 Germany confirms that, at present, the USNRC staff does not have  
18 a clear solution to the challenges posed by synergistic age-  
19 related degradation mechanisms [Stevens, Gary L., et al.,  
20 "Observations and Recommendations for Further Research Regarding  
21 Environmentally Assisted Fatigue Evaluation Methods," 40th MPA-  
22 Seminar, Materials Testing Institute, University of Stuttgart,  
23 Stuttgart, Germany (October 6-7, 2014) (Exh. NYS000491)]. A

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 recent draft report on the "Effect of LWR Coolant Environments  
2 on the Fatigue Life of Reactor Materials," prepared by Argonne  
3 National Laboratory (ANL) and USNRC Staff, recognizes the  
4 "inconclusive" nature of existing data on the synergistic  
5 effects of irradiation and fatigue, and other aging mechanisms  
6 in LWR environments, and concludes that, "additional fatigue  
7 data on reactor structural materials irradiated under LWR  
8 operating conditions are needed." [NUREG/CR-6909, Rev. 1 (March  
9 2014 [draft]), at 11 (Exh. NYS000490)]. Furthermore, during a  
10 "Briefing on Subsequent License Renewal" to the USNRC, the  
11 USNRC's Chief of the Corrosion and Metallurgy Branch, Dr. Mirela  
12 Gravila, testified that the Piping and Core Internals Panel had  
13 recognized "significant gaps" in our technical knowledge with  
14 respect to the effects of irradiation-induced degradation of the  
15 RVI components [Trans. of Briefing on Subsequent License  
16 Renewal, at 77 (May 2014) (Exh. NYS000492)].

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

20 A. Notwithstanding the significant concerns and  
21 considerable uncertainty regarding synergistic aging effects,  
22 the USNRC has so far declined to require that plant operators  
23 repair or replace degraded systems, structures, and fittings,

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1   opting instead to manage aging through periodic inspections, and  
2   the use of an empirical environmental factor method ( $F_{en}$ ) for  
3   fatigue life when evaluating the in situ degradation of  
4   structures and components [Stevens, et al., (October 2014), at  
5   10 (Exh. NYS000491)], a method which is not necessarily  
6   conservative and one that certainly does not address all the  
7   synergistic effects (e.g., embrittlement) that New York State is  
8   concerned about.

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

11       A.    Yes, let me begin with embrittlement.

12       **Embrittlement**

13       Q.    Would you explain what embrittlement is?

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

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1        Embrittlement is an age-related degradation mechanism  
2 whereby a component experiences a decrease in ductility, a loss  
3 of fracture toughness, and an increase in yield strength. While  
4 the initial aging effect is loss of ductility and toughness,  
5 unstable crack propagation is the eventual adverse aging effect  
6 if a crack is present and the local applied stress intensity is  
7 sufficient. Moreover, when subjected to a sufficient load, a  
8 component which has been highly embrittled by neutron  
9 irradiation may experience sudden, brittle fracture well before  
10 a surface crack is detected. This is a particular problem for  
11 the large pressure and/or thermal shock loads associated with  
12 postulated accidents. For this reason, USNRC regulations set  
13 forth at 10 C.F.R. § 50.61 impose fracture toughness  
14 requirements and/or operating parameters to prevent brittle  
15 fracture of reactor pressure vessels. Indeed, NUREG-1800, Rev. 2  
16 (Table 4.1-3) (Exh. NYS000161) identifies reduced fracture  
17 toughness of reactor vessel internals as a candidate for a time  
18 limited aging analysis. Because loss of ductility due to  
19 radiation embrittlement was not considered in the design of the  
20 stainless steel reactor vessel's internal components(RVIs), it  
21 is all the more important to evaluate the degree of  
22 embrittlement of RVIs during license renewal review. [Chopra,  
23 O., Public Comment on NRC-2010-0180-0001, Availability of Draft

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 NUREG-1800, Revision 2 and Draft NUREG-1801, Revision 2 (June 9,  
2 2010) (Exh. NYS000493)].

3 **The Consequences of Embrittlement**

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

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

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*



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

3 A. Yes. Reduced ductility (or embrittlement) will  
4 adversely affect a PWR's ability to withstand severe seismic  
5 events and pressure and/or thermal shock loads, and thus there  
6 is a threat to the integrity of highly embrittled internal  
7 structures in the reactor pressure vessel. For example, during  
8 a recent meeting regarding Indian Point, a member of the  
9 Advisory Committee on Reactor Safeguards Plant License Renewal  
10 Subcommittee expressed concern that embrittled RVI components  
11 could fail during a seismic event. [Trans. of Advisory Committee  
12 on Reactor Safeguards, Plan License Renewal Subcommittee, at  
13 209-210 (April 23, 2015) (Exh. NYS000526)].

14 Various accidents and abnormal transients can expose a  
15 reactor pressure vessel and its internal structures, components  
16 and fittings (i.e., RVIs) to significant pressure and/or thermal  
17 shock loads. If the reactor pressure vessel's internal  
18 structures (RVIs) are sufficiently degraded due to corrosion-  
19 induced cracking, fatigue and/or radiation-induced  
20 embrittlement, these shock loads can have significant  
21 consequences. Indeed, the resultant stresses from such  
22 accidents may cause the RVIs to fail structurally and relocate

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 within the RPV. If so, the ability to effectively cool the  
2 decay heat in the core may be lost due to core blockage.

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

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

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 impact their core-cooling functions. In the past, most of the  
2 USNRC's attention has been focused on the integrity of the  
3 reactor pressure vessel. However, the RVIs are much less  
4 massive and are much closer to the core, and thus they suffer a  
5 lot more radiation damage and embrittlement. Notably, the  
6 USNRC's fluence threshold for irradiation embrittlement of the  
7 reactor pressure vessel beltline is  $1 \times 10^{17}$  n/cm<sup>2</sup> [10 C.F.R. Part  
8 50, Appendix G; USNRC Regulatory Issue Summary 2014-11 (Exh.  
9 NYS000494)]. In contrast, Westinghouse RVIs can experience  
10 fluence in the range  $1 \times 10^{21}$  to  $5 \times 10^{22}$  n/cm<sup>2</sup>, or higher. Thus,  
11 RVI are subject to neutron irradiation which is several orders  
12 of magnitude higher than levels known to cause reduced fracture  
13 toughness in reactor pressure vessel materials. [MRP 191 (Nov.  
14 2006), Table 4-6 (Exh. NYS000321)].

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

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

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

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

5       The radiation-induced damage to some RPV internals can be  
6 extensive, since they can experience a neutron fluence of at  
7 least  $10^{23}$  n/cm<sup>2</sup> at neutron energy (E) levels of  $E > 1$  MeV (i.e.,  
8  $> 100$  dpa) [Was (2007) (Exh. NYS000339); EPRI, Dyle (2008)  
9 (Exh. NYS000322); WOG WCAP-14577 Rev. 1-A Report (March 2001)  
10 (Exh. NYS000341)] by the end of life (EOL) for extended  
11 operations. According to one study, the crack growth rate for  
12 materials irradiated to only  $3 \times 10^{20}$  n/cm<sup>2</sup> fluence can be up to 40  
13 times higher than that for unirradiated materials [Chopra, O.K.,  
14 "Degradation of LWR Core Internal Materials due to Neutron  
15 irradiation," NUREG/CR-7027 (Dec. 2010) (Exh. NYS000487)]. It  
16 should be stressed that the fluence experienced by some RPV  
17 internals is about four orders of magnitude (i.e.,  $\sim 10,000$   
18 times) larger than will be experienced by the inner wall of the  
19 reactor pressure vessel by the end of life (EOL) for extended  
20 operations [Rao, A.S. (USNRC), "Irradiation Assisted Degradation  
21 of LWR Core Internal Materials; Brief Review," (Apr. 14, 2015)  
22 (Exh. NYS000495)]. Thus, the RPV internals will be much more  
23 embrittled than the RPV walls, which have historically been the

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 focus of USNRC embrittlement concerns. A highly embrittled RPV  
2 internal component subjected to a severe earthquake or  
3 thermal/decompression shock, could thus fail and relocate within  
4 the RPV, which, in turn, could result in the loss of a coolable  
5 core geometry.

6 **GALL, Revision 1**

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

10 A. Yes.

11 Q. When did the USNRC Staff release that document?

12 A. In September of 2005.

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

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

22 "...(1) participate in the industry programs for

23 investigating and managing aging effects on reactor

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

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

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

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

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

21       Q. Directing your attention to NL-10-063, Attachment 1,  
22 page 84 of 90, what does Entergy say there about GALL, NUREG-  
23 1801, Revision 1 and reactor pressure vessel internals?

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

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

4           **Standard Review Plan, Revision 1**

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

8           A.    Yes.

9           Q.    When did the USNRC Staff release that document?

10          A.    In September of 2005.

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

14          A.    Yes, the Standard Review Plan, Revision 1 at §  
15 3.1.2.2.6 recognized that reactor pressure vessel internals  
16 could experience embrittlement.

17          Q.    Would you elaborate?

18          A.    In § 3.1.2.2.6 on page 3.1-5, the Standard Review  
19 Plan, Revision 1 states, "Loss of fracture toughness due to  
20 neutron irradiation embrittlement and void swelling could occur  
21 in stainless steel and nickel alloy reactor vessel internals  
22 components exposed to reactor coolant and neutron flux."

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

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

4 A. No, it did not. At § 3.1.3.2.6, the Standard Review  
5 Plan, Revision 1 stated that "The GALL Report recommends no  
6 further evaluation of programs to manage loss of fracture  
7 toughness due to neutron irradiation embrittlement . . ." That  
8 statement is on page 3.1-12. This is also confirmed by §  
9 3.1.2.2.6 and Table 3.1-1 which made clear that GALL and the  
10 Standard Review Plan did not propose a specific aging management  
11 plan and repeated the language from GALL about staying up to  
12 date with industry discussions about embrittlement and  
13 submitting a plan in the future for consideration by USNRC  
14 Staff.

15 **Entergy's Opposition to NYS Contention 25**

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

19 A. Yes, Entergy did.

20 Q. What did Entergy say in its response?

21 A. Entergy opposed the admission of Contention 25 and  
22 presented various arguments. One of Entergy's principal  
23 arguments was that stainless steel components are not

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*



1 susceptible to a decrease in fracture toughness as a result of  
2 neutron embrittlement. Entergy stated: "The core barrel,  
3 thermal shield, baffle plates and baffle former plates  
4 (including bolts) are, however, made of stainless steel and are  
5 not susceptible to a decrease in fracture toughness as a result  
6 of neutron embrittlement." [Entergy January 22, 2008 Answer at  
7 137]. This is a surprisingly uninformed statement from the  
8 operators of a nuclear power plant. Anyway, while this may have  
9 been a popular belief many years ago, it is incorrect.

10 **GALL, Revision 2**

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

14 A. Yes, I have reviewed it.

15 Q. When did the USNRC Staff release that document?

16 A. December of 2010.

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

18 A. GALL, Revision 2 includes the following statement:  
19 "Neutron irradiation embrittlement - Irradiation by neutrons  
20 results in embrittlement of carbon and low-alloy steels. It may  
21 produce changes in mechanical properties by increasing the  
22 tensile and yield strengths with a corresponding decrease in  
23 fracture toughness and ductility. The extent of embrittlement

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 depends on the neutron fluence, temperature, and trace material  
2 chemistry." [GALL, Revision 2 at page IX-34 (Exh. NYS000147)].  
3 I note that the phrase "low-alloy steels" includes stainless  
4 steel.

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

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

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

12 A. Chapter IV summarizes which reactor vessel internals  
13 are subject to embrittlement (and other aging mechanisms) and is  
14 organized by nuclear steam supply system vendors. There is a  
15 section ("B2") concerning components in nuclear steam supply  
16 systems designed by Westinghouse, the company that designed  
17 those systems at Indian Point Unit 2 and Unit 3. That section  
18 recognizes that reactor pressure vessel internals in  
19 Westinghouse-designed PWRs are subject to degradation due to  
20 embrittlement. It further recognizes that for Westinghouse  
21 PWRs, reactor pressure vessel internal components made of  
22 stainless steel and nickel alloy experience a "loss of fracture  
23 toughness due to neutron irradiation embrittlement." These

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 statements appear on GALL, Revision 2 at pages IV B2-2 to IV B2-  
2 14.

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

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

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

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

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

14 A. Yes.

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

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

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

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

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

3           Q.    Would you summarize that section?

4           A.    Yes.  Like Chapter IV, it recognizes that PWR reactor  
5   pressure vessel internals experience a "loss of fracture  
6   toughness due to either thermal aging or neutron irradiation  
7   embrittlement," as well as other age-related degradation  
8   mechanisms, such as various corrosion-induced cracking  
9   mechanisms.  It provides a template for license renewal  
10   applicants to include in their license renewal applications that  
11   discusses embrittlement and other aging mechanisms that degrade  
12   reactor pressure vessel internals.  It recommends that  
13   applicants propose an inspection plan that is then submitted to  
14   the USNRC Staff for review and approval.  The template is  
15   derived from a document prepared as a result of an effort  
16   coordinated by the Electric Power Research Institute (EPRI) to  
17   develop guidelines concerning the inspection of reactor pressure  
18   vessel internals.

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

22          A.    Yes.

23          Q.    Would you summarize that section?

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1       A.    Yes, this section provides recommendations for an  
2 inspection plan for reactor pressure vessel internals, and  
3 specifically what I would describe as the scope or focus of the  
4 plan. This section is titled "Parameters Monitored/Inspected"  
5 and states that the recommended inspection "program does not  
6 directly monitor for loss of fracture toughness that is induced  
7 by thermal aging or neutron embrittlement." Instead, it states  
8 that the embrittlement of reactor pressure vessel internal  
9 components is indirectly monitored through visual or volumetric  
10 inspection techniques that look for cracking (i.e., the  
11 detection of failures after they have occurred). It is  
12 important to note that the focus of this document is on non-  
13 destructive testing (NDT) and non-destructive evaluation (NDE)  
14 techniques. In particular it does not consider the implications  
15 on core coolability subsequent of any shock load induced  
16 failures of highly degraded RPV internals.

17       **MRP-227, Revision 0**

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

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

23       Q.    What is the title of that document?

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

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

7           **MRP-227-A**

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

10          A.    Yes, I have reviewed it. It is the version of the  
11 MRP-227, Revision 0 [Exh. NYS00307A-D] that was reviewed and  
12 approved by the USNRC Staff, and includes various edits and  
13 additional materials in response to USNRC Staff comments and  
14 questions. It was submitted to the USNRC in January 2012.  
15 Unfortunately, as I have noted previously, it is focused on NDT  
16 and NDE inspection techniques rather than my aging-related  
17 safety concerns.

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

19          A.    Yes. The industry has recognized that, "there are no  
20 recommendations for inspection to determine embrittlement level  
21 because these mechanisms cannot be directly observed" [MRP-227-  
22 A, Footnote 1 for Table 3-3 (December 2011) [Exh. NRC00114A-F].  
23 That is, the level of degradation due to embrittlement of RPV

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 internal components, fittings and structures, and their ability  
2 to withstand fatigue and shock loads cannot be determined using  
3 the inspection techniques proposed in MRP-227-A.

4 Q. Do you have specific concerns with the approach to  
5 aging management for reactor vessel internals set forth in MRP-  
6 227-A?

7 A. Yes. MRP-227-A is an inspection-based aging  
8 management plan, which I believe is inadequate. To begin with,  
9 depending on the type of component, inspection may not be  
10 possible for the entire component, or for the entire set of such  
11 components, given the location of the components and their  
12 possible inaccessibility. For example, a visual or ultrasonic  
13 inspection of the external head of a bolt does not necessarily  
14 provide insight into the integrity of the remainder of the bolt  
15 which is not visible. Moreover, an inspection focused on one  
16 type of age-related degradation mechanism does not necessarily  
17 work for another ongoing degradation process that is affecting  
18 the same component, and the effect of shock loads on the  
19 integrity of various RVIs and primary pressure boundary systems  
20 is certainly not addressed by inspections. An inspection-based  
21 approach to aging management, such as the one developed by the  
22 nuclear industry in MRP-227 and condoned by USNRC in MRP-227-A,  
23 is useful but it fails to account for the possibility that

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 highly embrittled and fatigued RVI components may not have signs  
2 of degradation that can be detected by an inspection, but such  
3 weakened components could nonetheless fail as a result of a  
4 severe seismic event or thermal or pressure shock load. In  
5 short, many of my concerns about the cumulative and ongoing  
6 synergistic aging effects are not adequately addressed by MRP-  
7 227-A.

8 **Entergy's License Renewal Application**

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

13 A. Yes.

14 Q. Where was that?

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

- 18 • carbon steel;
- 19 • carbon steel with stainless steel or nickel alloy;
- 20 • cladding;
- 21 • nickel alloys; and,
- 22 • stainless steel.



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

- 4 • cracking;
- 5 • loss of material; and,
- 6 • reduction of fracture toughness, a term which  
7 encompasses embrittlement.

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

11 A. Yes.

12 Q. Where was that?

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

- 16 • cast austenitic stainless steel (CASS);
- 17 • nickel alloy; and,
- 18 • stainless steel.

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

- 22 • change in dimensions;
- 23 • cracking;

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

- 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].

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

3 A. Entergy proposed to submit to USNRC Staff a safety  
4 analysis for plate B2803-3 three years before the plate reached  
5 the reference temperature for pressurized thermal shock (RT<sub>PTS</sub>)  
6 criterion.

7 **The 2007 LRA and RPV Internals**

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

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

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

23 A. Yes.

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

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

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

21       Q.    Do you believe there are any special problems  
22 associated with providing an adequate aging management program

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 for control rods and their associated guide tubes, plates and  
2 welds?

3       A.    Yes.   For example, because of geometric  
4 considerations, many PWRs (including IP2 and IP3) cannot meet  
5 the USNRC's required minimum coverage for the non-destructive  
6 testing (NDT) of the so-called "J-groove" welds [Entergy,  
7 Walpole, NL-09-130 (Sept. 24, 2009) (Exh. NYS000311)], and thus  
8 the integrity of these important CRD stub tube welds cannot be  
9 directly confirmed by inspection. It appears that to help  
10 address this chronic problem Entergy has ordered two new RPV  
11 heads [Telecom-USNRC/Entergy Report (March 18, 2008) (Exh.  
12 NYS000317)], but they have not yet been scheduled for  
13 installation at Indian Point [Telecom-USNRC/Entergy (March 18,  
14 2008) (Exh. NYS000317)]. In any event, unlike the rather  
15 superficial treatment given this important safety concern by  
16 Entergy [NL-10-063 (Exh. NYS000313)], I believe that a tangible,  
17 enforceable, and viable aging management program (AMP) should be  
18 developed and implemented before re-licensing the Indian Point  
19 reactor plants for extended operations, since the integrity of  
20 these CRD welds must be assured. If not, due to the leakage of  
21 borated primary coolant through cracked welds, there can be  
22 aggressive corrosion and wasting of the unclad outer surface of  
23 the upper head of the RPVs (such as the serious event that

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 occurred at Davis-Besse and was identified in 2002). Worse yet,  
2 there might be an inadvertent control rod ejection (due to a  
3 massive failure of the welds in the upper RPV head), which could  
4 cause a significant reactivity excursion, leading to core  
5 melting and radiation releases.

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

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

20 **Entergy's NL-10-063 Communication**

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

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

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

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

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 short-term "fix" might be adequate for normal operations, it may  
2 lead to structural and component failures due to the shock loads  
3 associated with various postulated accidents. If so, the failed  
4 internal structures and components may relocate, cause core  
5 blockages, or otherwise result in uncoolable core geometry, and  
6 thus lead to seriously degraded core cooling, core melting and  
7 massive radiation releases.

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

11 A. Yes. With respect to Entergy's proposal to conduct  
12 baseline examinations of the RPV internals (RVIs), it should be  
13 noted that I have previously called on Entergy to conduct such  
14 examinations and for USNRC Staff to require the conduct of such  
15 examinations before entering the period of extended operations  
16 [See November 2007 Declaration of Richard T. Lahey, Jr., at ¶¶  
17 24, 25 (Exh. NYS000298); see also State of New York Notice of  
18 Intention to Participate and Petition to Intervene, at 217-220,  
19 State of New York Contention-23 (Baseline Inspections)].

20 Fortunately, both the USNRC and Entergy now seem to have  
21 embraced the concept of baseline inspections for RPV internals,  
22 but the proposed aging management program (AMP) as set forth in  
23 NL-10-063 lacks sufficient details to know when the baseline

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*



1 inspections of the RPV and its internals will begin and end, and  
2 the scope of these inspections. Thus, it is not possible to  
3 know whether the proposed baseline inspections will be  
4 comprehensive and adequate.

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

8 A. Yes. My Report provides more details on my concerns  
9 with Entergy's failure to conduct an evaluation of the  
10 synergistic impacts of embrittlement, corrosion-induced  
11 cracking, and metal fatigue on the degradation of RPV internals,  
12 and its failure to consider how those interacting degradation  
13 mechanisms will impact the ability of the RPV internals to  
14 withstand the effect of thermal and decompression shock loads as  
15 a result of a DBA LOCA. I am also concerned that the design of  
16 the inspection programs -- including their frequency, the type  
17 of inspections to be conducted, the acceptance criteria and the  
18 criteria for actions to be taken in the event of a failure of a  
19 component -- does not consider these synergistic degradation  
20 mechanisms. Finally, Entergy's AMP for RPV internals does not  
21 include specific programs with objective criteria for either  
22 preventative measures or for corrective actions to be taken when

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 inspections show that certain components are not able to safely  
2 undergo extended plant operations.

3 **Entergy's NL-11-107 Communication**

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

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

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

11 A. Yes, I have that.

12 Q. What does the document say there?

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

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

19 A. In it, Entergy tells the USNRC that it has completed  
20 commitment number 30 wherein Entergy stated that it would submit  
21 an inspection plan to the USNRC for reactor pressure vessel  
22 internals (RVIs) no later than two years before the plant  
23 entered the period of extended operations. However, none of my

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 safety concerns associated with the synergistic effects of  
2 embrittlement, fatigue and corrosion on the integrity of RPV  
3 internals, and post-accident core coolability (i.e., due to  
4 shock load induced failures), were addressed. In my opinion an  
5 adequate inspection plan for RPV internals is a necessary, but  
6 not sufficient, means of assuring safe extended plant  
7 operations. Indeed, a systematic safety evaluation of the  
8 degraded RPV internals is also needed to identify the limiting  
9 structures, components and fittings that need to be repaired or  
10 replaced before the onset of extended operations.

11 **Entergy's Amended and Revised RVI Plan, and**  
12 **USNRC Staff's November 2014 SSER2**  
13

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

16 A. Yes, I have reviewed these documents. In  
17 communication NL-12-037, dated February 17, 2012 [Exh.  
18 NYS000496], the applicant submitted an amendment to its license  
19 renewal application entitled "Revised Reactor Vessel Internals  
20 Program and Inspection Plan." Thereafter, the applicant  
21 explained and modified this proposed plan in response to various  
22 requests for information (RAIs) from the USNRC. [Exhs.  
23 NYS000497 through NYS000506]. Collectively, I will refer to

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 this collection of communications as the applicant's "Amended  
2 and Revised RVI Plan."

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

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

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

12 A. Yes, to some degree. The USNRC recognized the  
13 potential synergy between thermal and irradiation embrittlement  
14 for cast austenitic stainless steel components (CASS). [SSER2  
15 at 3-42 (Exh. NYS000507)]. In particular, in its Safety  
16 Evaluation Report (SER) for MRP-227, the USNRC Staff  
17 acknowledged the potential for synergistic interaction between  
18 embrittlement and other aging mechanisms. For example, the  
19 USNRC noted that "the synergistic effects of SCC, fatigue, and  
20 thermal embrittlement . . . could potentially cause greater  
21 degradation in the welds [of Combustion Engineering lower  
22 support columns] than just the consideration of IASCC  
23 (irradiation assisted stress corrosion cracking) and irradiation

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 embrittlement alone. Degradation in these welds could then be  
2 equivalent to or greater than other components susceptible only  
3 to IASCC and irradiation embrittlement due to the synergistic  
4 effects." [SE at 15 (Exh. NYS000309)]. The USNRC staff could  
5 have - indeed, should have - made the same observation about  
6 potential synergistic aging effects for Westinghouse RVI  
7 components, fittings, and structures at IP2 and IP3.

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

10 A. Yes. In Attachment 1 to NL-12-037, Entergy has  
11 indicated that the Amended and Revised RVI Plan "is a condition  
12 monitoring program that does not include preventative actions."  
13 [Attachment 1 to NL-12-037, at 5 (Exh. NYS000496)]. Generally,  
14 the applicant continues to approach the problem of synergistic  
15 aging effects on RVI components through "condition monitoring"  
16 (i.e., periodic inspections per MRP-227-A) rather than a  
17 comprehensive approach which includes detailed analyses and/or  
18 preventative actions (i.e., repair and replacement) ["Revised  
19 Reactor Vessel Internals Program and Inspection Plan,"  
20 Attachment 1 to NL-12-037, at 5 (Exh. NYS000496)]. This  
21 approach implies that aging effects and degradation will not be  
22 addressed until cracks or other degradation mechanisms (e.g.,  
23 wear) have been directly observed ["Revised Reactor Vessel

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 Internals Program and Inspection Plan," Attachment 1 to NL-12-  
2 037, at 5 (Exh. NYS000496)].

3 In short, component degradation will be addressed only  
4 after it occurs. The applicant incorrectly concludes that  
5 preventative actions, such as component replacement, are not  
6 required for most RVI components because cracking or other flaws  
7 can be detected before the failure of a component affects the  
8 safe operation of the reactor. This is apparently based on the  
9 erroneous assumption that IP2 and IP3 will continuously operate  
10 during the 20-year period of extended operation within normal  
11 "steady-state" parameters. Entergy ignores the possibility that  
12 significantly fatigued, embrittled and corrosion-weakened, or  
13 otherwise degraded, RVI components, structures, or fittings may  
14 be exposed to various shock loads which can cause them to deform  
15 or relocate and thereby impair core cooling. In fact, the  
16 applicant's reactor safety analyses implicitly assume that the  
17 reactor core will maintain a coolable geometry during emergency  
18 core cooling system (ECCS) operation subsequent to a DBA LOCA,  
19 notwithstanding the degradation and possible deformation or  
20 relocation of various RVI components and potential flow  
21 blockages and degraded core cooling which may result.

22 Q. Does Entergy make any statements about the degradation  
23 of RVI components in the "Amended and Revised RVI Plan"?

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1           A.    Yes, similar to NL-10-063 (Exh. NYS000313), the  
2 applicant acknowledges, in NL-12-037, that other PWRs have  
3 experienced material degradation and failure of multiple RVI  
4 components, including cracking of baffle-former bolting,  
5 cracking in other important bolting, wear in thimble tubes, and  
6 potential wear in control rod guide tube guide plates  
7 [Attachment 1 to NL-12-037, at 8 (Exh. NYS000496)]. Also, the  
8 applicant has committed to replace one affected IP2 component -  
9 the degraded guide tube support pins (split pins) - by 2016  
10 ["Revised Reactor Vessel Internals Program and Inspection Plan,"  
11 Attachment 1 to NL-12-037, at 8 (Exh. NYS000496); Commitment 50,  
12 Attachment 1 to NL-13-122, at 7 (Exh. NYS000502)].  
13 Interestingly, the applicant has agreed to replace the IP2 split  
14 pins, even though they were already replaced once in 1995, and  
15 even though the applicant claims that the failure of a split pin  
16 would not compromise reactor vessel functions [Response to RAI  
17 16, Attachment 2 to NL-12-166, at 1 (Exh. NYS000500)]. However,  
18 for many other affected RVI components, the applicant proposes a  
19 "wait-and-see" approach.

20           Q.    Could you provide an example?

21           A.    Yes. The applicant acknowledges that "cracking of  
22 baffle former bolts is recognized as a potential issue for the  
23 Indian Point units" ["Revised Reactor Vessel Internals Program,"

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 Attachment 1 to NL-12-037, at 8 (Exh. NYS000496)], but the  
2 applicant does not propose to replace the degraded bolts, only  
3 to continue monitoring them ["Revised Reactor Vessel Internals  
4 Inspection Plan," Attachment 2 to NL-12-037, at 40, tbl. 5-2  
5 (Exh. NYS000496)]. In fact, the applicant has not yet developed  
6 inspection acceptance criteria for baffle former bolts in either  
7 IP2 or IP3 [SSER, at 3-20 (Exh. NYS000507)]. Instead, the  
8 applicant has agreed to develop a technical justification  
9 including acceptance criteria for baffle former bolts sometime  
10 prior to the first round of inspections, which might not occur  
11 until 2019 for IP2 and 2021 for IP3 [SSER2, at 3-20 (Exh.  
12 NYS000507); Response to RAI 5, Attachment 1 to NL-12-089, at 11  
13 (Exh. NYS000497)].

14 Another example of the applicant's "wait-and-see" approach  
15 for the RVIs is the applicant's proposal for managing aging  
16 effects on the clevis insert bolts. [SSER2, at 3-23 to 3-26  
17 (Exh. NYS000507)]. Like the split pins that the applicant is  
18 replacing in IP2 for the second time, clevis insert bolts are  
19 susceptible to primary water stress corrosion cracking (PWSCC)  
20 [MRP-227-A, Appendix A, at A-2 (Exh. NRC00114A-F)]. Failures of  
21 clevis insert bolts, apparently caused by PWSCC, were detected  
22 at a Westinghouse-designed reactor in 2010. Out of 48 clevis  
23 bolts in this reactor, 29 were partially or completely fractured

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*



1 but only 7 of those damaged bolts were visually detected as  
2 having failed [SSER2, at 3-25 (Exh. NYS000507)]. Despite this  
3 high rate of failure (about 60% of the total bolts were damaged)  
4 and low rate of visual detection (only about 24% of the damaged  
5 bolts were detected), the applicant proposes to manage the aging  
6 degradation of clevis insert bolts with visual (VT-3)  
7 inspections rather than pre-emptive replacement ["Revised  
8 Reactor Vessel Internals Inspection Plan," Attachment 2 to NL-  
9 12-037, tbl. 5-4, at 51 (Exh. NYS000496)].

10 The applicant apparently acknowledges that visual  
11 inspections will not detect the majority of clevis bolt cracks  
12 prior to failure, but justifies this approach on the grounds  
13 that "crack detection prior to bolt failure is not required due  
14 to design redundancy" [Response to RAI 17, Attachment 1 to NL-  
15 13-122, at 8 (Exh. NYS000502)]. In fact, the applicant appears  
16 to suggest that the failure of multiple clevis insert bolts will  
17 not seriously affect the operation of the reactor. The  
18 applicant then analyzes the effect of clevis bolt failures on  
19 various other components.

20 The applicant's analysis of the effects of clevis bolt  
21 failures assumes that all other components will be functioning  
22 according to their design specifications, and does not consider  
23 the fact that the other components may also be undergoing

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 degradation from various interacting aging mechanisms.  
2 Moreover, the applicant fails to consider the possibility that a  
3 shock load (e.g., due to a LOCA) may cause the sudden failure of  
4 the remaining intact clevis bolts, which, in turn, may lead to  
5 an uncoolable core geometry. In short, rather than taking  
6 proactive steps to replace the degraded clevis bolts prior to  
7 failure, the applicant proposes to wait for clevis bolt failures  
8 to occur before taking steps to address the problem, an approach  
9 which is totally unacceptable in my opinion.

10 The baffle former bolts and clevis insert bolts are just  
11 two examples of Entergy's overarching approach to RVI aging  
12 management, which foregoes preventative component repair or  
13 replacement in favor of running the reactor until detectable  
14 damage or component failure occurs.

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

17 A. Yes. The applicant's approach for analyzing the lower  
18 support structures' functionality and fracture toughness is also  
19 flawed [Response to RAI-11-A, Attachment 1 to NL-13-052, at 1-4  
20 (Exh. NYS000501)]. The applicant suggested that irradiation  
21 embrittlement effects would only be significant in the presence  
22 of pre-existing flaws or service induced defects, together with  
23 a stress level capable of crack propagation. In its analysis,

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 the applicant, based on the lack of documented fractures of core  
2 support columns, "assumed that only a limited number of columns  
3 could actually contain flaws of significant size." The  
4 applicant further assumed that the columns would be subject to  
5 "nominal normal operating stresses" [SSER2, at 3-43 (Exh.  
6 NYS000507)]. When the USNRC Staff inquired about the most  
7 recent visual inspections of the core support structures, the  
8 applicant acknowledged that the CASS support column caps were  
9 inaccessible to inspection and that VT-3 visual inspection  
10 offered "no meaningful information regarding the structural  
11 integrity of the columns." [Id. at 3044.] Under these  
12 circumstances, the applicant's conclusion that irradiation-  
13 induced cracking of core support columns is "unlikely"  
14 represents wishful thinking and is contrary to recent studies  
15 [e.g., NUREG/CR-7184, at xv (Revised December 2014) (Exh.  
16 NYS00488A-B)], which show the extreme sensitivity of crack  
17 growth rate and fracture toughness to irradiation. Moreover, it  
18 ignores the fact that these and other non-CASS RVI structures  
19 and components undergo a range of aging degradation mechanisms  
20 simultaneously under steady-state and transient conditions, and  
21 that their embrittlement or susceptibility to fracture simply  
22 cannot always be adequately detected using currently available  
23 inspection techniques.

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1       Also, not all of the core support structures are accessible  
2 for inspection, so surrogate structures have been chosen by  
3 Entergy to assess age-related degradation mechanisms. For  
4 example, the girth weld of the core barrel has been proposed by  
5 the applicant as a leading indicator for irradiation-induced  
6 embrittlement (IE) and irradiation-assisted stress corrosion  
7 cracking (IASCC) of the core support column caps, even though  
8 these components are very different, and they may be exposed to  
9 different degradation mechanisms and shock loads. In fact, as  
10 pointed out recently by a member of the ACRS Plant License  
11 Renewal Subcommittee, "[t]he relationship between a lower core  
12 barrel weld and the tops of these columns is a bit of a stretch  
13 . . . [t]hey're totally different type of components, totally  
14 different loadings." Moreover, to have a failure due to a  
15 seismic event "you don't even need to have a crack if these  
16 columns are really brittle . . . ." [ACRS Plant License Renewal  
17 Subcommittee Transcript, at 209-211 (April 23, 2015) (Exh.  
18 NYS000526)].

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

22       A. No. By merely relying on MRP 227-A for its aging  
23 management plan, the applicant has ignored the large

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 uncertainties that exist with respect to the effects of  
2 irradiation-induced aging phenomena. [Chen, et al., at xv (Exh.  
3 NYS000488A); NUREG/CR-7153, Vol. 2: Aging of Core Internals and  
4 Piping Systems, at 181, 187, 210-211 (Exh. NYS00484A-B);  
5 Stevens, et al. (October 2014), at 9-10 (Exh. NYS000491)].  
6 While the applicant's Thermal Aging and Neutron Irradiation of  
7 Cast Austenitic and Stainless Steel (CASS) program generally  
8 recognizes the potential adverse synergistic effects of elevated  
9 coolant temperature and irradiation on the fracture toughness of  
10 CASS materials, a broader recognition of this principle is  
11 needed by the applicant, since RVI components made from non-cast  
12 stainless steel will also experience the combined effects of  
13 irradiation-induced embrittlement, corrosion, and other aging  
14 mechanisms. The applicant has failed to evaluate the  
15 synergistic mechanisms that occur for many other important and  
16 vulnerable RVI components, such as the core baffles, baffle  
17 bolts, and formers. Compared to the baffles, baffle bolts, and  
18 formers, the core support columns (which are obviously very  
19 important incore structures) are located in an area of the  
20 reactor pressure vessel which is subject to less radiation  
21 fluence (and thus are less susceptible to embrittlement).

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

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Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1           A.    Yes.  The applicant proposes to rely on visual (VT-3)  
2 inspection techniques for many RVI components.  However, there  
3 are significant shortcomings of this technique to detect  
4 material cracking, degradation, or wear prior to failure, as has  
5 been noted by USNRC staff [Tregoning, at 2-3 (Exh. NYS000508);  
6 Case, at 1 (Exh. NYS000509)], and illustrated by the visual  
7 detection of only 7 out of 29 fractured clevis insert bolts at a  
8 Westinghouse PWR in 2010 [SSER2, at 3-25 (Exh. NYS000507)].  In  
9 an RAI to Entergy, the USNRC staff observed that "VT-3 visual  
10 examination may not be adequate for all components for detecting  
11 fatigue cracking prior to the occurrence of structurally  
12 significant cracking." [Attachment 1 to NL-13-052, at 5 (May 7,  
13 2013) (Exh. NYS000501)].  Moreover, as I have noted previously,  
14 the level of embrittlement can not be detected at all using  
15 visual inspection techniques.

16           **Fatigue**

17           Q.    Turning to fatigue.  Could you explain what fatigue  
18 is?

19           A.    Yes.  Fatigue is another important age-related  
20 degradation mechanism.  It is one of the primary considerations  
21 when conducting a time limited aging analysis (TLAA) and an  
22 aging management program (AMP) for nuclear power plants.  
23 Fatigue of various structures, components and fittings in a

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Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 nuclear reactor can result in piping and pressure boundary  
2 component and fitting ruptures, physical failures, and the  
3 relocation of loose pieces of RVI metal throughout the reactor  
4 system, which, in turn, may result in core blockages and  
5 interfere with the effective core cooling of a nuclear power  
6 plant. My main concerns about fatigue are the increased  
7 potential for a primary or secondary side LOCA, and the failure  
8 of various RPV internals (RVIs). It should be noted that the  
9 fatigue life of a PWR component, fitting or structure is  
10 normally evaluated in terms of a cumulative usage factor (CUF)  
11 which is corrected for the degradation in fatigue life due to  
12 the reactor coolant environment (i.e.,  $CUF_{en}$ ). The cumulative  
13 usage factor is defined as,  $CUF = N/N_{a-AIR}$ , where N is the number  
14 of the various fatigue cycles that have occurred (or are  
15 expected by the end of plant life, EOL), and  $N_{a-AIR}$  is the number  
16 of allowable fatigue cycles obtained from data (taken in air) at  
17 which failure (i.e., significant surface cracking) is expected.  
18 The observed degradation in fatigue that occurs due to hot  
19 reactor coolant is quantified by an environmental fatigue  
20 correction factor,  $F_{en} = N_{a-AIR}/N_{a-RC}$ , where  $N_{a-AIR}$  is the allowable  
21 number of fatigue cycles measured in air, and  $N_{a-RC}$  is the  
22 allowable fatigue cycles measured in a simulated reactor coolant  
23 (RC) environment; thus,  $CUF_{en} = CUF \times F_{en} = N/N_{a-RC}$ . The criterion

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 for acceptance by the USNRC is that  $CUF_{en} < 1.0$  by the end of  
2 life (EOL) for the component, fitting or structure in question.  
3 Anyway, the allowable cycles to failure ( $N_{a-RC}$ ) are determined  
4 from small scale experiments using metal test samples which are  
5 exposed to simulated reactor coolant environments.  
6 Unfortunately, to date, there have not been any systematic  
7 fatigue experiments done in simulated reactor coolant  
8 environments using highly embrittled metal test samples, which  
9 have less fatigue life than ductile materials. That is, the  
10 synergistic degradation effect of embrittlement has not been  
11 included in  $CUF_{en}$  ( $= N/N_{a-RC}$ ) evaluations, thus the results are  
12 expected to be non-conservative since the denominator ( $N_{a-RC}$ ) will  
13 be too large, and thus  $CUF_{en}$  will be too small.

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

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

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

20 A. Yes. In this proceeding, the applicant agreed, in  
21 Commitments 33, 43 and 49, to calculate the  $CUF_{en}$  for  
22 external(i.e., primary pressure boundary) and internal (RVI)  
23 components in certain locations [Dacimo, Fred, Entergy, letter

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*



1 to Document Control Desk, USNRC, "Reply to Request for  
2 Additional Information Regarding the License Renewal  
3 Application," NL-13-122 (September 27, 2013), at 20  
4 (NYS000502)]. Additionally, the USNRC has recently proposed to  
5 require all applicants for license renewal to evaluate the  
6 fatigue life of limiting components beyond those originally  
7 specified in NUREG/CR-6260, and to evaluate the effect of  
8 reactor coolant environment on the fatigue life of both  
9 external and internal (i.e., RVIs) structures and systems [79  
10 Fed. Reg. 69,884 (November 24, 2014) (NYS000522); USNRC, Draft  
11 Regulatory Guide DG-1309 (Proposed Revision 1 of Regulatory  
12 Guide 1.207, dated March 2007), "Guidelines for Evaluating the  
13 Effects of Light-Water Reactor Coolant Environments in Fatigue  
14 Analyses of Metal Components" (November 2014) (NYS000523)].

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

19 A. No. [REDACTED]  
20 [REDACTED]  
21 [REDACTED]  
22 [REDACTED]  
23 [REDACTED]

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

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[REDACTED]

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 [REDACTED]

2 [REDACTED]

3 [REDACTED]

4 [REDACTED]

5 [REDACTED]

6 [REDACTED]

7 [REDACTED] Unfortunately, Westinghouse did

8 not perform error analyses to quantify the modeling

9 uncertainties and the effect of code user interactions in its

10 fatigue calculations using WESTEMS. It would appear that

11 virtually any error would put some of the calculated values of

12  $CUF_{en}$  over the  $CUF_{en} = 1.0$  fatigue failure limit.

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

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

21 Q. Are you familiar with its contents?

22 A. Yes, I have reviewed this article and it clearly shows  
23 the iterative process used by Westinghouse in which safety

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 margin is removed in its environmentally assisted fatigue (EAF)  
2 calculations in an effort to reduce the output or result below  
3  $CUF_{en} = 1.0$ .

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

8 A. No, [REDACTED]

9 [REDACTED]  
10 [REDACTED]  
11 [REDACTED]  
12 [REDACTED]  
13 [REDACTED]  
14 [REDACTED]  
15 [REDACTED]  
16 [REDACTED]  
17 [REDACTED]  
18 [REDACTED]  
19 [REDACTED]  
20 [REDACTED] Even

21 assuming this  $CUF_{en}$  calculation is accurate, it does not account  
22 for the possibility that a highly fatigued component, which does  
23 not yet have signs of significant surface cracking, may be

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 exposed to an unexpected seismic event or shock load that could  
2 cause it to fail. This is a good example of the type of "silo  
3 thinking" (i.e., the fatigue and safety analyses are treated  
4 entirely separately) that NYS is concerned about.

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

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

14 Q. Would you briefly summarize your concerns?

15 A. Yes. First, I am concerned that without an error  
16 analysis it is difficult to be in a position to meaningfully  
17 analyze the results of the 2010 and subsequently refined CUF<sub>en</sub>  
18 analyses presented by Entergy and Westinghouse.

19 Q. Why is an error analysis important?

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

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Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

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

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

5 A. Yes. Engineers frequently prepare error analyses. In  
6 my submissions in this proceeding I noted that one would  
7 normally expect to see at least a hybrid 'propagation-of-error'  
8 type of analysis [Kline & McClintock (1953) (Exh. NYS000514)] to  
9 determine the overall uncertainty in the CUF<sub>en</sub> results given by  
10 Westinghouse. I also referenced a standard engineering text  
11 book, "Basic Engineering Data Collection and Analysis," pp. 310-  
12 311, by Vardeman & Jobe [2001], to demonstrate the various types  
13 of error analyses which are regularly done by engineers [Exh.  
14 NYS000347].

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

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

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 activities affecting safety-related functions, the USNRC's  
2 findings of non-conformance are all the more unsettling. Under  
3 these circumstances, the USNRC should insist that an error  
4 analysis be performed to ensure the validity of Westinghouse's  
5 fatigue evaluations for IP2 and IP3 components.

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

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

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

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

17 Q. Do you have other concerns about the refined CUF<sub>en</sub>  
18 reanalysis?

19 A. Yes, as discussed in my Supplemental Report, I am  
20 concerned that engineering judgment or user intervention could  
21 have affected the results. I note that when USNRC Staff issued  
22 the Supplemental Safety Evaluation Report, Staff instructed  
23 Entergy and Westinghouse, on a going forward basis, to document

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 and disclose the use of engineering judgment and user  
2 intervention when conducting future fatigue analysis using the  
3 WESTEMS code. This is noted in Exhibit NYS000160 at page 4-2.  
4 To my knowledge, Westinghouse has provided such information for  
5 some, but not all of the fatigue evaluations performed to date.  
6 Also, USNRC Staff instructed Entergy not to use WESTEMS when  
7 conducting analyses under the ASME Standard known as NB-3600 [at  
8 4-2, 4-3 (Exh. NYS000160)]. Furthermore, I am concerned about  
9 the analytical framework employed by the WESTEMS code. As  
10 detailed, in my Supplemental Report, I believe that the code's  
11 thermal-hydraulic models and framework are too simplified to  
12 predict accurate results. [REDACTED]

13 [REDACTED]  
14 [REDACTED]  
15 [REDACTED]  
16 [REDACTED]  
17 [REDACTED]  
18 [REDACTED]

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

22 A. Yes. Entergy has an existing Fatigue Monitoring Plan  
23 for addressing metal fatigue. The program is designed to

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*



1 monitor operational cycles and transients so that the various  
2  $CUF_{en}$  remain below unity. The company has also proposed to  
3 include RVIs in its Fatigue Monitoring Program; however, for the  
4 reasons already stated, WESTEMS may be non-conservative, so any  
5 program that relies on values, such as the fatigue cycles to  
6 failure, derived from the WESTEMS methodology is inherently  
7 unreliable for ensuring that aging RVIs avoid failure. This is  
8 particularly true when these results include no accompanying  
9 error analysis.

#### 10 **Conclusion**

11 Q. Could you summarize your general concerns with the  
12 applicant's license renewal application?

13 A. Yes, I am very concerned that Entergy has continually  
14 eroded the safety margins and conservatisms built into the  
15 current licensing basis for the Indian Point reactors. For  
16 example, Entergy has relied on  $CUF_{en}$  calculations that remove  
17 various conservatisms but are still very close to unity (the  
18 fatigue failure limit). Entergy also relies on the detection of  
19 degradation, wear or cracking prior to component failure, rather  
20 than repairing or replacing the aging parts - particularly the  
21 RVIs - preemptively. Moreover, Entergy implicitly assumes that  
22 the plant will operate in a steady-state, and has not taken into

1 account unanticipated severe seismic events or thermal/pressure  
2 shock loads which can cause failures to occur.

3 As reactors and their constituent components age, it  
4 becomes very important to preserve - rather than erode -  
5 operational safety margins. Uncertainties exist in all systems,  
6 and calculation or modeling mistakes are always possible. For  
7 example, the USNRC recently became aware that certain  
8 methodologies prescribed in its NUREG-0800 Branch Technical  
9 Position (BTP) 5-3 for estimating the initial fracture toughness  
10 of reactor vessel materials may be non-conservative. [See,  
11 e.g., Troyer, et al., "An Assessment of Branch Technical  
12 Position 5-3 to Determine Unirradiated RTNDT for SA-508 Cl.2  
13 Forgings," Paper No. PCP2014-28897, Proceedings of the ASME 2014  
14 Pressure Vessels and Piping Conference, Anaheim, California  
15 (July 20-24, 2014) (Exh. NYS00516); Letter from Pedro Salas,  
16 Regulatory Affairs Director, AREVA, to USNRC regarding Potential  
17 Non-conservatism in NRC Branch Technical Position 5-3 (January  
18 30, 2014)(Exh. NYS000517); USNRC, Slides, "Assessment of BTP 5-3  
19 Protocols to Estimate RTNDT(u) and USE (June 4, 2014) (Exh.  
20 NYS000518); NUREG-0800, Rev. 2 (Exh. NYS000521)]. [REDACTED]

21 [REDACTED]

22 [REDACTED]

23 [REDACTED]

Anyway,

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 since unexpected errors of this type do occur, maintaining  
2 safety margins helps to guard against potentially adverse  
3 impacts due to precisely this type of unexpected finding of non-  
4 conservatism in safety evaluations. Lastly, I would like to  
5 note that at a recent American Society of Mechanical Engineers  
6 (ASME) Pressure Vessels & Piping Conference, USNRC staff also  
7 highlighted newly-identified non-conservatisms in sections of  
8 the ASME Code regarding fracture toughness applicable to nuclear  
9 reactor operations. [Kirk, M. et al., "Assessment of Fracture  
10 Toughness Models for Ferritic Steels Used in Section XI of the  
11 ASME Code Relative to Current Data-Based Model," PVP 2014-28540  
12 (Exh. NYS000520)]. This is yet another reason to preserve,  
13 rather than erode safety margins in the aging management of  
14 light water nuclear reactors (e.g., PWRs).

15 Q. You have reviewed NUREG-1801, GALL Report Revision 1  
16 (Exh. NYS00146A-C); NUREG-1800, Standard Review Plan Revision 1  
17 (Exh. NYS000195); NUREG-1801, GALL Report Revision 2 (Exh.  
18 NYS00147A-D); NUREG-1800, Standard Review Plan Revision 2 (Exh.  
19 NYS000161); EPRI's MRP-227 Revision 0 (Exh. NYS00307A-D); EPRI's  
20 MRP-227-A (Exh. NYS000507); Entergy's July 2010 NL-10-063  
21 communication, Entergy's February 2012 NL-12-037 communication  
22 (Exh. NYS000313) and subsequent communications constituting its  
23 Amended and Revised RVI Plan (Exhs. NYS000496-506); USNRC

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 Staff's June 22, 2011 Safety Evaluation of MRP-227 Revision 0  
2 (Exh. NYS000309); NUREG-1930, USNRC Staff's August 30, 2011  
3 Supplemental Safety Evaluation (Exh. NYS000160); and NUREG-1930,  
4 USNRC Staff's November 2014 Second Supplement Safety Evaluation  
5 Report for the Indian Point License Renewal Application (Exh.  
6 NYS000507); and Entergy's NL-11-107 communication (Exh.  
7 NYS000314), correct?

8 A. Yes.

9 Q. Do you have any opinion about those documents with  
10 respect to the degradation of reactor pressure vessel internals  
11 (RVIs)?

12 A. Yes.

13 Q. Please summarize your testimony.

14 A. As I stated in my initial November 2007 declaration in  
15 support of the State of New York's Contentions 25 and 26, my  
16 April 2008 declaration in support of Contention NYS-26A, my  
17 September 2010 declarations in support of the State's  
18 supplemental filings on Contentions NYS-25 and NYS-26B/RK-TC-1B,  
19 and my previously filed testimony on Contentions NYS-25 and NYS-  
20 26B/RK-TC-1B, and my February 2015 declaration in support of the  
21 State's further supplemental filings on Contention NYS-25, in my  
22 professional judgment Entergy has failed to demonstrate that it  
23 has adequately accounted for the aging phenomena of

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 embrittlement and fatigue for structures, components and  
2 fittings inside the reactor pressure vessels (i.e., RVIs) at  
3 Indian Point Unit 2 and Indian Point Unit 3. My professional  
4 judgment has not fundamentally changed based upon Entergy's July  
5 14, 2010 submission of License Renewal Application, Amendment  
6 No. 9 [NL-10-063 (Exh. NYS000313)], Entergy's September 28, 2011  
7 submission of NL-11-107 [Exh. NYS000314], or Entergy's Amended  
8 and Revised RVI Plan, consisting of the February 17, 2012  
9 submission of NL-12-037 [Exh. NYS000496] as amended by its  
10 subsequent communications [Exhs. NYS000497-506] and approved by  
11 the USNRC Staff in the SSER2 [Exh. NYS000507]. I do not believe  
12 that Entergy's July 15, 2010 communication to the Board [NL-10-  
13 063 (Exh. NYS000313)] concerning a new AMP for RPV internals, or  
14 its September 28, 2011 communication [NL-11-107 (Exh.  
15 NYS000314)], are adequate to address the safety concerns and  
16 technical issues that I have raised herein. They do not address  
17 my age-related safety concerns, nor do they recognize the  
18 importance of the various synergistic degradation mechanisms  
19 that I am concerned with. The Amended and Revised RVI Plan,  
20 which the USNRC Staff evaluated and approved in the November  
21 2014 SSER2, also does not resolve my concerns over the  
22 simultaneous and synergistic age-related degradation mechanisms  
23 that may affect various RVI components and structures.

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

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

8       Q.   Does this conclude your testimony?

9       A.   Not quite. I want to stress that during the course of  
10 my involvement in these relicensing proceedings I have  
11 discovered what I believe to be some important new age-related  
12 safety concerns which, to the best of my knowledge, have not  
13 been previously considered in relicensing proceedings. These  
14 concerns include: the synergistic effect on the degradation and  
15 integrity of RPV internals (RVIs) of radiation-induced  
16 embrittlement, corrosion and fatigue, and the potential for the  
17 unanticipated failure of RPV internals (RVIs) due to a severe  
18 seismic event or accident-induced thermal and/or pressure shock  
19 loads, and the implications of the failure of RPV internal  
20 structures, components and fittings (i.e., RVIs) on post-  
21 accident core coolability. While in the past many of these  
22 issues and concerns have been noted separately, the implications  
23 of their synergistic interaction has apparently been overlooked

*Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25*

1 and not evaluated (i.e., they have been evaluated in "silos").  
2 Since I first raised these technical issues in 2007, the USNRC,  
3 DOE and various nuclear industry groups have slowly begun to  
4 recognize their significance. In fact, the evaluation and study  
5 of these important issues is underway, but major uncertainties  
6 still exist. As a consequence, I believe that these important  
7 age-related safety concerns must be resolved in order to have  
8 assurance that the Indian Point reactors can operated safely  
9 beyond their design life of 40 years. Indeed, I believe that  
10 the most vulnerable RPV internals (RVIs) need to be carefully  
11 identified and repaired or replaced prior to extended operations  
12 since it is beyond the current state-of-the-art to perform  
13 realistic and accurate calculations on the relocation of failed  
14 RPV internals (RVIs) and the resultant potential for core  
15 blockages and degraded core cooling.

16 Q. Does this complete your testimony.

17 A. Yes, it does. I do, however, reserve the right to  
18 supplement my testimony if new information is disclosed or  
19 introduced.  
20

1 UNITED STATES

2 NUCLEAR REGULATORY COMMISSION

3 BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

4 -----X

5 In re: Docket Nos. 50-247-LR; 50-286-LR  
6 License Renewal Application Submitted by ASLBP No. 07-858-03-LR-BD01  
7 Entergy Nuclear Indian Point 2, LLC, DPR-26, DPR-64  
8 Entergy Nuclear Indian Point 3, LLC, and  
9 Entergy Nuclear Operations, Inc. June 9, 2015

10 -----X

11 DECLARATION OF RICHARD T. LAHEY, JR.

12 I, Richard T. Lahey, Jr., do hereby declare under penalty  
13 of perjury that my statements in the foregoing testimony and my  
14 statement of professional qualifications are true and correct to  
15 the best of my knowledge and belief.

16 Executed in Accord with 10 C.F.R. § 2.304(d)

17 

18 \_\_\_\_\_  
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Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Contention NYS-25