

In the Matter of:

Entergy Nuclear Operations, Inc.  
(Indian Point Nuclear Generating Units 2 and 3)

Submitted: June 9, 2015  
Public, Redacted



<b>ASLBP #:</b> 07-858-03-LR-BD01	<b>Identified:</b> 11/5/2015
<b>Docket #:</b> 05000247   05000286	<b>Withdrawn:</b>
<b>Exhibit #:</b> NYS000562-PUB-00-BD01	<b>Stricken:</b>
<b>Admitted:</b> 11/5/2015	
<b>Rejected:</b>	
<b>Other:</b>	

UNITED STATES

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

-----x

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

-----x

REVISED PRE-FILED WRITTEN TESTIMONY OF

Dr. RICHARD T. LAHEY, JR.

REGARDING JOINT CONTENTION NYS-38/RK-TC-5

On behalf of the State of New York ("NYS" or "the State"), the Office of the Attorney General hereby submits the following testimony by RICHARD T. LAHEY, JR., PhD. regarding Joint Contention NYS-38/RK-TC-5.

Q. Please state your full name.

A. Richard T. Lahey, Jr.

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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.

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

13 Q. What expert panels have you served on?

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

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

9 Q. Have you received any professional awards?

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

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

4 A. Yes. They are copies of the seven declarations that I  
5 previously prepared to date for the State of New York in this  
6 proceeding. They include my initial declaration that was  
7 submitted in November 2007 in support of the State's petition to  
8 intervene and its initial contentions, the April 7, 2008  
9 declaration in support of Contention NYS-26A, the September 15,  
10 2010 declaration submitted in support of the State's  
11 supplemental bases for Contention NYS-25, the September 9, 2010  
12 declaration submitted in support of the amended Contention NYS-  
13 26B/RK-TC-1B, the September 30, 2011 and November 1, 2011  
14 declarations submitted in support of Joint Contention NYS-38/RK-  
15 TC-5, and the February 12, 2015 declaration submitted in support  
16 of additional bases for Contention NYS-25 and Joint Contention  
17 NYS-38/RK-TC-5.

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

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

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

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

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

19 Q. What is the purpose of your current testimony?

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

11 Also, the purpose of my testimony here is to provide  
12 support for, and my views on, aspects of New York's and  
13 Riverkeeper's Joint Contention NYS-38/RK-TC-5 ("NYS-38/RK-TC-  
14 5"), which was admitted for litigation by the Atomic Safety  
15 Licensing Board. Contention NYS-38/RK-TC-5 asserts, among other  
16 things, that Entergy has not demonstrated that it has a program  
17 that will manage the effects of aging of critical components or  
18 systems at the Indian Point nuclear power facilities and that  
19 therefore the USNRC does not have a record and a rational basis  
20 upon which it can determine whether to grant Entergy a renewed  
21 license for the Indian Point facilities.

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

1 My testimony critiques Entergy's proposed approach to age  
2 related degradation caused by embrittlement and fatigue as being  
3 inadequate.

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

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

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

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

16 A. Yes.

17 Q. What is the source of those materials?

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*



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

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

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

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

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

10                   **The Indian Point Reactors**

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

13           A.    Yes.

14           Q.    Would you briefly describe them?

15           A.    Entergy operates two nuclear power reactors that are  
16 located in northern Westchester County near the Village of  
17 Buchanan. The operating nuclear reactors are known as the  
18 Indian Point Unit 2 and Indian Point Unit 3 reactors. These  
19 Westinghouse-designed plants are 4-loop pressurized water  
20 reactors (PWRs), and they are currently rated at power levels of  
21 3,216.4 MW<sub>t</sub>. Entergy also owns another reactor at the same site.  
22 That reactor is known as the Indian Point Unit 1 reactor;

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

3 **Operation of a Pressurized Water Reactor**

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

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

9 **Reactor Pressure Vessel Internals**

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

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

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

1           A.    Yes.  They are a collection of additional schematic  
2 figures from the Electric Power Research Institute's Report MRP-  
3 227 that provide additional detail concerning various reactor  
4 pressure vessel internals and their location within the reactor  
5 pressure vessel.  The reactor pressure vessel internals shown  
6 include the control rod guide tube assembly, the control rod  
7 guide cards, guide tube support pins, the control rods, baffles,  
8 formers, baffle-former assemblies, baffle-to-former bolts,  
9 corner edge bracket baffle to former bolts, core barrel to  
10 former bolts, baffle plate edge bolts, core support structures,  
11 and various weldments, including welds within the reactor  
12 pressure vessel for the core barrel plates.

13                           **Overview**

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*



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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

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

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

21 Q. Are synergistic aging effects limited to CASS  
22 components?

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

5           Q.    Are these synergistic aging effects fully understood?

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

1 data on reactor structural materials irradiated under LWR  
2 operating conditions are needed." [NUREG/CR-6909, Rev. 1 (March  
3 2014 [draft]), at 11 (Exh. NYS000490)]. Furthermore, during a  
4 "Briefing on Subsequent License Renewal" to the USNRC, the  
5 USNRC's Chief of the Corrosion and Metallurgy Branch, Dr. Mirela  
6 Gravila, testified that the Piping and Core Internals Panel had  
7 recognized "significant gaps" in our technical knowledge with  
8 respect to the effects of irradiation-induced degradation of the  
9 RVI components [Trans. of Briefing on Subsequent License  
10 Renewal, at 77 (May 2014) (Exh. NYS000492)].

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

14 A. Notwithstanding the significant concerns and  
15 considerable uncertainty regarding synergistic aging effects,  
16 the USNRC has so far declined to require that plant operators  
17 repair or replace degraded systems, structures, and fittings,  
18 opting instead to manage aging through periodic inspections, and  
19 the use of an empirical environmental factor method ( $F_{en}$ ) for  
20 fatigue life when evaluating the in situ degradation of  
21 structures and components [Stevens, et al., (October 2014), at  
22 10 (Exh. NYS000491)], a method which is not necessarily

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*



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

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

6 A. Yes, let me begin with embrittlement.

7 **Embrittlement**

8 Q. Would you explain what embrittlement is?

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

1 the initial aging effect is loss of ductility and toughness,  
2 unstable crack propagation is the eventual adverse aging effect  
3 if a crack is present and the local applied stress intensity is  
4 sufficient. Moreover, when subjected to a sufficient load, a  
5 component which has been highly embrittled by neutron  
6 irradiation may experience sudden, brittle fracture well before  
7 a surface crack is detected. This is a particular problem for  
8 the large pressure and/or thermal shock loads associated with  
9 postulated accidents. For this reason, USNRC regulations set  
10 forth at 10 C.F.R. § 50.61 impose fracture toughness  
11 requirements and/or operating parameters to prevent brittle  
12 fracture of reactor pressure vessels. Indeed, NUREG-1800, Rev. 2  
13 (Table 4.1-3) (Exh. NYS000161) identifies reduced fracture  
14 toughness of reactor vessel internals as a candidate for a time  
15 limited aging analysis. Because loss of ductility due to  
16 radiation embrittlement was not considered in the design of the  
17 stainless steel reactor vessel's internal components (RVIs), it  
18 is all the more important to evaluate the degree of  
19 embrittlement of RVIs during license renewal review. [Chopra,  
20 O., Public Comment on NRC-2010-0180-0001, Availability of Draft  
21 NUREG-1800, Revision 2 and Draft NUREG-1801, Revision 2 (June 9,  
22 2010) (Exh. NYS000493)].

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

1                    **The Consequences of Embrittlement**

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

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

7 **GALL, Revision 1**

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

11 A. Yes.

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

13 A. In September of 2005.

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*



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

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

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

1 Q. Directing your attention to NL-10-063, Attachment 1,  
2 page 84 of 90, what does Entergy say there about GALL, NUREG-  
3 1801, Revision 1 and reactor pressure vessel internals?

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

7 **Standard Review Plan, Revision 1**

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

11 A. Yes.

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

13 A. In September of 2005.

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

17 A. Yes, the Standard Review Plan, Revision 1 at §  
18 3.1.2.2.6 recognized that reactor pressure vessel internals  
19 could experience embrittlement.

20 Q. Would you elaborate?

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

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

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

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

22 A. Yes, Entergy did.

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

1 Q. What did Entergy say in its response?

2 A. Entergy opposed the admission of Contention 25 and  
3 presented various arguments. One of Entergy's principal  
4 arguments was that stainless steel components are not  
5 susceptible to a decrease in fracture toughness as a result of  
6 neutron embrittlement. Entergy stated: "The core barrel,  
7 thermal shield, baffle plates and baffle former plates  
8 (including bolts) are, however, made of stainless steel and are  
9 not susceptible to a decrease in fracture toughness as a result  
10 of neutron embrittlement." [Entergy January 22, 2008 Answer at  
11 137]. This is a surprisingly uninformed statement from the  
12 operators of a nuclear power plant. Anyway, while this may have  
13 been a popular belief many years ago, it is incorrect.

14 **GALL, Revision 2**

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

18 A. Yes, I have reviewed it.

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

20 A. December of 2010.

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

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

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

1 those systems at Indian Point Unit 2 and Unit 3. That section  
2 recognizes that reactor pressure vessel internals in  
3 Westinghouse-designed PWRs are subject to degradation due to  
4 embrittlement. It further recognizes that for Westinghouse  
5 PWRs, reactor pressure vessel internal components made of  
6 stainless steel and nickel alloy experience a "loss of fracture  
7 toughness due to neutron irradiation embrittlement." These  
8 statements appear on GALL, Revision 2 at pages IV B2-2 to IV B2-  
9 14.

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

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

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

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

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

21 A. Yes.

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

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

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

12 Q. Would you summarize that section?

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

9 A. Yes.

10 Q. Would you summarize that section?

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*



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

5 **MRP-227, Revision 0**

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

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

11 Q. What is the title of that document?

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

18 **MRP-227-A**

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

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

16 Q. Do you have specific concerns with the approach to  
17 aging management for reactor vessel internals set forth in MRP-  
18 227-A?

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

1 components, given the location of the components and their  
2 possible inaccessibility. For example, a visual or ultrasonic  
3 inspection of the external head of a bolt does not necessarily  
4 provide insight into the integrity of the remainder of the bolt  
5 which is not visible. Moreover, an inspection focused on one  
6 type of age-related degradation mechanism does not necessarily  
7 work for another ongoing degradation process that is affecting  
8 the same component, and the effect of shock loads on the  
9 integrity of various RVIs and primary pressure boundary systems  
10 is certainly not addressed by inspections. An inspection-based  
11 approach to aging management, such as the one developed by the  
12 nuclear industry in MRP-227 and condoned by USNRC in MRP-227-A,  
13 is useful but it fails to account for the possibility that  
14 highly embrittled and fatigued RVI components may not have signs  
15 of degradation that can be detected by an inspection, but such  
16 weakened components could nonetheless fail as a result of a  
17 severe seismic event or thermal or pressure shock load. In  
18 short, many of my concerns about the cumulative and ongoing  
19 synergistic aging effects are not adequately addressed by MRP-  
20 227-A.

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

1                    **Entergy's License Renewal Application**

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

6            A.     Yes.

7            Q.     Where was that?

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

- 11                    • carbon steel;
- 12                    • carbon steel with stainless steel or nickel alloy;
- 13                    • cladding;
- 14                    • nickel alloys; and,
- 15                    • stainless steel.

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

- 19                    • cracking;
- 20                    • loss of material; and,
- 21                    • reduction of fracture toughness, a term which  
22                    encompasses embrittlement.

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

4 A. Yes.

5 Q. Where was that?

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

- 9 • cast austenitic stainless steel (CASS);
- 10 • nickel alloy; and,
- 11 • stainless steel.

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

- 15 • change in dimensions;
- 16 • cracking;
- 17 • loss of material;
- 18 • loss of preload; and,
- 19 • reduction of fracture toughness, a term which, as  
20 noted previously, encompasses embrittlement.

21  
22  
*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

1                    The 2007 LRA and the IP3 Reactor Pressure Vessel

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

4            A.    Yes.

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

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

9            Q.    And what did Entergy say there?

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

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

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

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

1 the reference temperature for pressurized thermal shock (RT<sub>PTS</sub>)  
2 criterion.

3 **The 2007 LRA and RPV Internals**

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

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

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

19 A. Yes.

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*



1           A.    Yes.  For example, because of geometric  
2 considerations, many PWRs (including IP2 and IP3) cannot meet  
3 the USNRC's required minimum coverage for the non-destructive  
4 testing (NDT) of the so-called "J-groove" welds [Entergy,  
5 Walpole, NL-09-130 (Sept. 24, 2009) (Exh. NYS000311)], and thus  
6 the integrity of these important CRD stub tube welds cannot be  
7 directly confirmed by inspection.  It appears that to help  
8 address this chronic problem Entergy has ordered two new RPV  
9 heads [Telecom-USNRC/Entergy Report (March 18, 2008) (Exh.  
10 NYS000317)], but they have not yet been scheduled for  
11 installation at Indian Point [Telecom-USNRC/Entergy (March 18,  
12 2008) (Exh. NYS000317)].  In any event, unlike the rather  
13 superficial treatment given this important safety concern by  
14 Entergy [NL-10-063 (Exh. NYS000313)], I believe that a tangible,  
15 enforceable, and viable aging management program (AMP) should be  
16 developed and implemented before re-licensing the Indian Point  
17 reactor plants for extended operations, since the integrity of  
18 these CRD welds must be assured.  If not, due to the leakage of  
19 borated primary coolant through cracked welds, there can be  
20 aggressive corrosion and wasting of the unclad outer surface of  
21 the upper head of the RPVs (such as the serious event that  
22 occurred at Davis-Besse and was identified in 2002).  Worse yet,

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

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

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

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

21 Fortunately, both the USNRC and Entergy now seem to have  
22 embraced the concept of baseline inspections for RPV internals,

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

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

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

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

13 A. Yes, I have that.

14 Q. What does the document say there?

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

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

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

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*



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

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

1 Reactor Vessel Internals Program and Inspection Plan,"  
2 Attachment 1 to NL-12-037, at 5 (Exh. NYS000496)]. This  
3 approach implies that aging effects and degradation will not be  
4 addressed until cracks or other degradation mechanisms (e.g.,  
5 wear) have been directly observed ["Revised Reactor Vessel  
6 Internals Program and Inspection Plan," Attachment 1 to NL-12-  
7 037, at 5 (Exh. NYS000496)].

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

7 A. Yes, similar to NL-10-063 (Exh. NYS000313), the  
8 applicant acknowledges, in NL-12-037, that other PWRs have  
9 experienced material degradation and failure of multiple RVI  
10 components, including cracking of baffle-former bolting,  
11 cracking in other important bolting, wear in thimble tubes, and  
12 potential wear in control rod guide tube guide plates  
13 [Attachment 1 to NL-12-037, at 8 (Exh. NYS000496)]. Also, the  
14 applicant has committed to replace one affected IP2 component -  
15 the degraded guide tube support pins (split pins) - by 2016  
16 ["Revised Reactor Vessel Internals Program and Inspection Plan,"  
17 Attachment 1 to NL-12-037, at 8 (Exh. NYS000496); Commitment 50,  
18 Attachment 1 to NL-13-122, at 7 (Exh. NYS000502)].

19 Interestingly, the applicant has agreed to replace the IP2 split  
20 pins, even though they were already replaced once in 1995, and  
21 even though the applicant claims that the failure of a split pin  
22 would not compromise reactor vessel functions [Response to RAI

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

4 Q. Could you provide an example?

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

21 Another example of the applicant's "wait-and-see" approach  
22 for the RVIs is the applicant's proposal for managing aging

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

1 effects on the clevis insert bolts. [SSER2, at 3-23 to 3-26  
2 (Exh. NYS000507)]. Like the split pins that the applicant is  
3 replacing in IP2 for the second time, clevis insert bolts are  
4 susceptible to primary water stress corrosion cracking (PWSCC)  
5 [MRP-227-A, Appendix A, at A-2 (Exh. NRC00114A-F)]. Failures of  
6 clevis insert bolts, apparently caused by PWSCC, were detected  
7 at a Westinghouse-designed reactor in 2010. Out of 48 clevis  
8 bolts in this reactor, 29 were partially or completely fractured  
9 but only 7 of those damaged bolts were visually detected as  
10 having failed [SSER2, at 3-25 (Exh. NYS000507)]. Despite this  
11 high rate of failure (about 60% of the total bolts were damaged)  
12 and low rate of visual detection (only about 24% of the damaged  
13 bolts were detected), the applicant proposes to manage the aging  
14 degradation of clevis insert bolts with visual (VT-3)  
15 inspections rather than pre-emptive replacement ["Revised  
16 Reactor Vessel Internals Inspection Plan," Attachment 2 to NL-  
17 12-037, tbl. 5-4, at 51 (Exh. NYS000496)].

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

1 13-122, at 8 (Exh. NYS000502)]. In fact, the applicant appears  
2 to suggest that the failure of multiple clevis insert bolts will  
3 not seriously affect the operation of the reactor. The  
4 applicant then analyzes the effect of clevis bolt failures on  
5 various other components.

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*



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

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

12 A. No. By merely relying on MRP 227-A for its aging  
13 management plan, the applicant has ignored the large  
14 uncertainties that exist with respect to the effects of  
15 irradiation-induced aging phenomena. [Chen, et al., at xv (Exh.  
16 NYS000488A); NUREG/CR-7153, Vol. 2: Aging of Core Internals and  
17 Piping Systems, at 181, 187, 210-211 (Exh. NYS00484A-B);  
18 Stevens, et al. (October 2014), at 9-10 (Exh. NYS000491)].

19 While the applicant's Thermal Aging and Neutron Irradiation of  
20 Cast Austenitic and Stainless Steel (CASS) program generally  
21 recognizes the potential adverse synergistic effects of elevated  
22 coolant temperature and irradiation on the fracture toughness of

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

8 **Fatigue**

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

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

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

13 A. No. [REDACTED]

[REDACTED]

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

1  
2  
3  
4  
5  
6  
7  
8  
9  
10  
11  
12  
13  
14  
15  
16  
17  
18  
19  
20  
21  
22

[REDACTED]

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

1 [REDACTED]  
2 [REDACTED]  
3 [REDACTED]  
4 [REDACTED]  
5 [REDACTED] It

6 would appear that virtually any error would put some of the  
7 calculated values of  $CUF_{en}$  over the  $CUF_{en} = 1.0$  fatigue failure  
8 limit.

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

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

17 Q. Are you familiar with its contents?

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*



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

5 A. [REDACTED]

6 [REDACTED]

7 [REDACTED]

8 [REDACTED]

9 [REDACTED]

10 [REDACTED]

11 [REDACTED]

12 [REDACTED]

13 [REDACTED]

14 [REDACTED]

15 [REDACTED]

16 [REDACTED]

17 [REDACTED]

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

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

12 Q. Would you briefly summarize your concerns?

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

17 Q. Why is an error analysis important?

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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  
4 accepted practice in the field of engineering?

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

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

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

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

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

15 [REDACTED]  
16 [REDACTED]  
17 [REDACTED]  
18 [REDACTED]  
19 [REDACTED]  
20 [REDACTED]

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

15 **Conclusions Regarding Fatigue and Embrittlement Issues**

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

1 Non-conservatism in NRC Branch Technical Position 5-3 (January  
2 30, 2014)(Exh. NYS000517); USNRC, Slides, "Assessment of BTP 5-3  
3 Protocols to Estimate RTNDT(u) and USE (June 4, 2014) (Exh.  
4 NYS000518); NUREG-0800, Rev. 2 (Exh. NYS000521)]. [REDACTED]

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*



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

17 A. Yes.

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

21 A. Yes.

22 Q. Please summarize your testimony.

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

1 SSER2 [Exh. NYS000507]. I do not believe that Entergy's July  
2 15, 2010 communication to the Board [NL-10-063 (Exh. NYS000313)]  
3 concerning a new AMP for RPV internals, or its September 28,  
4 2011 communication [NL-11-107 (Exh. NYS000314)], are adequate to  
5 address the safety concerns and technical issues that I have  
6 raised herein. They do not address my age-related safety  
7 concerns, nor do they recognize the importance of the various  
8 synergistic degradation mechanisms that I am concerned with.  
9 The Amended and Revised RVI Plan, which the USNRC Staff  
10 evaluated and approved in the November 2014 SSER2, also does not  
11 resolve my concerns over the simultaneous and synergistic age-  
12 related degradation mechanisms that may affect various RVI  
13 components and structures.

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

7 **Steam Generator Issues**

8 Q. Let us now turn your attention to the specific steam  
9 generator issues which have been a focus of Contention NYS-  
10 38/RK-TC-5. Earlier in your testimony you mentioned that Indian  
11 Point Unit 2 and Indian Point Unit 3 include steam generators.  
12 How many steam generators are there at Indian Point?

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

17 **Steam Generators, Their Components, and Function**

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

20 A. Yes, it is a diagram of a Westinghouse steam  
21 generator.  
22

23 Q. Would you please describe the role of the steam

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

1 generators in the Indian Point nuclear steam supply systems?

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

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

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

1 steam generator?

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

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

8 A. Yes, in the License Renewal Application at Tables  
9 2.3.1-4-IP2 and 2.3.1-4-IP3 Entergy states that the lower head  
10 of a steam generator, the divider plate, U-tubes, the tubesheet,  
11 and the tube-tubesheet welds each constitute a pressure boundary  
12 for Indian Point Unit 2 and Indian Point Unit 3. Significantly,  
13 all these boundaries, except the divider plate, represent  
14 important primary pressure boundaries. Entergy also  
15 acknowledged that the U-tubes also perform an important heat  
16 transfer function. Those tables are located in the License  
17 Renewal Application at pages 2.3-36, 2.3-39, respectively (Exh.  
18 NYS000558).

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

6 Q. What is the reactor coolant pressure boundary?

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

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

22 A. Yes, these are summary charts identifying the

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*



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

6 **Primary Water Stress Corrosion Cracking**

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

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

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

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

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

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

21 A. Yes, in the License Renewal Application at page 2.3-  
22 21, Entergy stated that the current Indian Point Unit 2 steam

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

4 Q. What types of steam generators parts or locations are  
5 affected by primary water stress corrosion cracking?

6 A. In addition to the heat transfer tubes, primary water  
7 stress corrosion cracking could also affect other components or  
8 assemblies that use Alloy 600 or welds that use Alloy 82/182  
9 weld material that, as I have noted previously, are derivatives  
10 of Alloy 600. In the August 30, 2011 Supplemental Safety  
11 Evaluation Report at page 3-21, the USNRC Staff has also  
12 expressed concern about the propagation of primary water stress  
13 corrosion cracking in tubesheets that have Alloy 600 cladding,  
14 or the related welds, even when the U-tubes are made from Alloy  
15 690 material. According to the Staff, "a crack initiated in  
16 this region, close to the tube, may propagate into or through  
17 the weld, causing a failure of the weld and of the reactor  
18 coolant pressure boundary." The specific areas of concern  
19 include the channel head-to-tubesheet-to-tube complex, including  
20 the divider plate assembly and the tube-to-tubesheet welds.

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

1 impact the intended function of the steam generators?

2 A. Yes, in my opinion it could.

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

16 [REDACTED]  
17 [REDACTED]  
18 [REDACTED]  
19 [REDACTED]  
20 [REDACTED]  
21 [REDACTED]  
22 [REDACTED]

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

20 **Entergy's Approach to Steam Generator Issues**

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

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

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

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

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

21 Q. Did you review the Supplemental Safety Evaluation  
22 Report?

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

1 A. Yes.

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

4 A. Yes, these are 2011 communications from Entergy to the  
5 USNRC Staff in response to Staff questions about the age-related  
6 degradation of various components at Indian Point Unit 2 and  
7 Indian Point Unit 3. These are Entergy communications NL-11-  
8 032, NL-11-074, NL-11-90, and NL-11-096.

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*



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

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

1 With respect to tube-to-tubesheet welds in Indian Point  
2 Unit 2's four steam generators, under Commitment 42 Option 2  
3 inspections for PWSCC were to take place sometime between March  
4 2020 and March 2024, which is well after the proposed extended  
5 operation period has begun. As I previously noted, this is  
6 particularly troubling since these welds do in fact form part of  
7 the primary system's pressure boundary, and if they fail  
8 radiation will be released to the secondary side and  
9 subsequently to the environment. However, although the tube-to-  
10 tubesheet welds clearly form a part of the reactor coolant  
11 pressure boundary, Entergy and the USNRC Staff have pursued a  
12 different approach under Option 1 and have recently agreed to  
13 "redefine" that boundary to exclude the numerous tube-to-  
14 tubesheet weld locations in each of the four steam generators at  
15 Indian Point Unit 2. [Entergy Letter NL-14-001 (Jan. 16,  
16 2014)(Exh. NYS000539), NRC IP2 Operating License Amendment No.  
17 277 (Sept. 5, 2014) (Exh. NYS000542).] As noted in my  
18 previously filed rebuttal testimony [Exh. NYS000453 at 11-12],  
19 changing this definition does not resolve NYS' concerns or  
20 eliminate the physical pathway through which radiation can be  
21 released to the environment.

22 Q. 

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

1  
2  
3  
4  
5  
6  
7  
8  
9  
10  
11  
12  
13  
14  
15  
16  
17  
18  
19  
20  
21  
22

[REDACTED]

A. [REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

1            Conclusion Regarding Steam Generator Issues

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

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

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

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

1 accident-induced thermal/pressure shock loads (e.g., a steam  
2 line break and reactor SCRAM). Although Entergy's previous  
3 testimony referred to two EPRI reports, EPRI Report 1020988,  
4 "Steam Generator Management Program: Phase II Divider Plate  
5 Cracking Engineering Study", November 2010 (Exh. ENT000523),  
6 and, EPRI Report No. 1025133, "Steam Generator Management  
7 Program: Assessment of Channel Head Susceptibility to Primary  
8 Water Stress Corrosion Cracking", June 2012 (Exh. ENT000524),  
9 those confidential reports do not resolve my concerns. [REDACTED]

10 [REDACTED]

11 [REDACTED]

12 [REDACTED]

13 [REDACTED]

14 [REDACTED]

15 This is an important safety concern, since if there is gross  
16 failure of the divider plate then the primary coolant may bypass  
17 the steam generator's U-tubes thus eliminating the ultimate heat  
18 sink. If this occurs, the natural circulation cooling relied  
19 upon, for example, during station blackout (SBO) or during  
20 various anticipated transient without SCRAM (ATWS) events, will  
21 not occur and thus the core may overheat and melt, leading to  
22 the release of significant radioactive materials. While it is

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

1 true that a feed-and-bleed mode of core cooling can be used in  
2 this event, this emergency procedure may not be adequate to  
3 prevent core damage during rapid transients. Clearly, this  
4 important safety issue needs to be resolved as part of the ASLB  
5 relicensing hearings on the two Indian Point reactors. For the  
6 reasons expressed in my earlier rebuttal testimony [NYS000453 at  
7 9-11, 12-14], it is my opinion that Entergy [ENT000521] and the  
8 USNRC Staff [NRC000161] have not addressed this concern.

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

1 purpose, as it numerically smears steep spatial transient  
2 variations in both temperature and pressure. Also, since  
3 WESTEMS was used and is relied on for the fatigue analyses  
4 (i.e., CUF<sub>en</sub> results) we have the same issues [REDACTED]  
5 [REDACTED] the lack of an error  
6 analysis, etc., as has been discussed in Contention NYS-26B. In  
7 any event, a much more detailed analysis of the SG divider  
8 plates is clearly needed.

9 Closing Observation

10 Q. Do you have any further comments?

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

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*

1 Q. Does this complete your testimony?

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

5

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*



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  
13 penalty of perjury that my statements in the foregoing testimony  
14 and my statement of professional qualifications are true and  
15 correct to the best of my knowledge and belief.

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

17 

18 \_\_\_\_\_  
19 Dr. Richard T. Lahey, Jr.  
20 The Edward E. Hood Professor Emeritus of Engineering  
21 Rensselaer Polytechnic Institute, Troy, NY 12180  
22 (518) 495-3884, lahey@rpi.edu

*June 2015 Revised Pre-filed Written  
Testimony of Richard T. Lahey, Jr.  
Joint Contention NYS-38/RK-TC-5*