


In the Matter of: Entergy Nuclear Operations, Inc.
(Indian Point Nuclear Generating Units 2 and 3)

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UNITED STATES

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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In re: Docket Nos. 50-247-LR; 50-286-LR

License Renewal Application Submitted by ASLBP No. 07-858-03-LR-BD01

Entergy Nuclear Indian Point 2, LLC, DPR-26, DPR-64

Entergy Nuclear Indian Point 3, LLC, and

Entergy Nuclear Operations, Inc. June 9, 2015

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REVISED PRE-FILED WRITTEN TESTIMONY OF

Dr. RICHARD T. LAHEY, JR.

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

On behalf of the State of New York ("NYS" or "the State"), the Office of the Attorney General hereby submits the following testimony by RICHARD T. LAHEY, JR., PhD. regarding Consolidated Contention NYS-26B/RK-TC-1B.

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.

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

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

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

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

12 Q. What expert panels have you served on?

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

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

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

7 Q. Have you received any professional awards?

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

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

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

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

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

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

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

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

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

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

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

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

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

8 Q. What is the purpose of your testimony?

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

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

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1 A. Yes.

2 Q. What is the source of those materials?

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

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

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

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

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

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

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

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

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

14 **The Indian Point Reactors**

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

17 A. Yes.

18 Q. Would you briefly describe them?

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

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

6 Operation of a Pressurized Water Reactor

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

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

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

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

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

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

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

3 **Reactor Pressure Vessel Internals**

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

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

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

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

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

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

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

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

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

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

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

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

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

3 **Overview**

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

13 Q. Are these synergistic aging effects fully understood?

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

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

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

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

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

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

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

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

11 A. Yes, let me begin with embrittlement.

12 **Embrittlement**

13 Q. Would you explain what embrittlement is?

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

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

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

3 **The Consequences of Embrittlement**

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

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

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

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

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

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

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

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

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1 impact their core-cooling functions. In the past, most of the
2 USNRC's attention has been focused on the integrity of the
3 reactor pressure vessel. However, the RVIs are much less
4 massive and are much closer to the core, and thus they suffer a
5 lot more radiation damage and embrittlement. Notably, the
6 USNRC's fluence threshold for irradiation embrittlement of the
7 reactor pressure vessel beltline is 1×10^{17} n/cm² [10 C.F.R. Part
8 50, Appendix G; USNRC Regulatory Issue Summary 2014-11 (Exh.
9 NYS000494)]. In contrast, Westinghouse RVIs can experience
10 fluence in the range 1×10^{21} to 5×10^{22} n/cm², or higher. Thus,
11 RVI are subject to neutron irradiation which is several orders
12 of magnitude higher than levels known to cause reduced fracture
13 toughness in reactor pressure vessel materials. [MRP 191 (Nov.
14 2006), Table 4-6 (Exh. NYS000321)].

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

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

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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/cm² 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 3×10^{20} n/cm² fluence can be up to 40
13 times higher than that for unirradiated materials [Chopra, O.K.,
14 "Degradation of LWR Core Internal Materials due to Neutron
15 irradiation," NUREG/CR-7027 (Dec. 2010) (Exh. NYS000487)]. It
16 should be stressed that the fluence experienced by some RPV
17 internals is about four orders of magnitude (i.e., $\sim 10,000$
18 times) larger than will be experienced by the inner wall of the
19 reactor pressure vessel by the end of life (EOL) for extended
20 operations [Rao, A.S. (USNRC), "Irradiation Assisted Degradation
21 of LWR Core Internal Materials; Brief Review," (Apr. 14, 2015)
22 (Exh. NYS000495)]. Thus, the RPV internals will be much more
23 embrittled than the RPV walls, which have historically been the

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

6 **GALL, Revision 1**

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

10 A. Yes.

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

12 A. In September of 2005.

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

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

22 "...(1) participate in the industry programs for
23 investigating and managing aging effects on reactor

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

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

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

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

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

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

4 Standard Review Plan, Revision 1

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

8 A. Yes.

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

10 A. In September of 2005.

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

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

17 Q. Would you elaborate?

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

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

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

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

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

19 A. Yes, Entergy did.

20 Q. What did Entergy say in its response?

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

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

10 **GALL, Revision 2**

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

14 A. Yes, I have reviewed it.

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

16 A. December of 2010.

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

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

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

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

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

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

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

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

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

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

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

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

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

14 A. Yes.

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

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

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

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

3 Q. Would you summarize that section?

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

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

22 A. Yes.

23 Q. Would you summarize that section?

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

17 **MRP-227, Revision 0**

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

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

23 Q. What is the title of that document?

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

7 **MRP-227-A**

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

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

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

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

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

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

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

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

8 **Entergy's License Renewal Application**

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

13 A. Yes.

14 Q. Where was that?

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

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

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

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

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

11 A. Yes.

12 Q. Where was that?

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

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

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

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

- 1 • loss of material;
- 2 • loss of preload; and,
- 3 • reduction of fracture toughness, a term which, as
- 4 noted previously, encompasses embrittlement.

5 **The 2007 LRA and the IP3 Reactor Pressure Vessel**

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

8 A. Yes.

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

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

13 Q. And what did Entergy say there?

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

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

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

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

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

7 **The 2007 LRA and RPV Internals**

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

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

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

23 A. Yes.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

11 A. Yes, I have that.

12 Q. What does the document say there?

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

20 Q. Could you provide an example?

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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1 uncertainties that exist with respect to the effects of
2 irradiation-induced aging phenomena. [Chen, et al., at xv (Exh.
3 NYS000488A); NUREG/CR-7153, Vol. 2: Aging of Core Internals and
4 Piping Systems, at 181, 187, 210-211 (Exh. NYS00484A-B);
5 Stevens, et al. (October 2014), at 9-10 (Exh. NYS000491)].

6 While the applicant's Thermal Aging and Neutron Irradiation of
7 Cast Austenitic and Stainless Steel (CASS) program generally
8 recognizes the potential adverse synergistic effects of elevated
9 coolant temperature and irradiation on the fracture toughness of
10 CASS materials, a broader recognition of this principle is
11 needed by the applicant, since RVI components made from non-cast
12 stainless steel will also experience the combined effects of
13 irradiation-induced embrittlement, corrosion, and other aging
14 mechanisms. The applicant has failed to evaluate the
15 synergistic mechanisms that occur for many other important and
16 vulnerable RVI components, such as the core baffles, baffle
17 bolts, and formers. Compared to the baffles, baffle bolts, and
18 formers, the core support columns (which are obviously very
19 important incore structures) are located in an area of the
20 reactor pressure vessel which is subject to less radiation
21 fluence (and thus are less susceptible to embrittlement).

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

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

16 **Fatigue**

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

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

23 Fatigue of various structures, components and fittings in a

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

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

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

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

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

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

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

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

19 A. No. [REDACTED]

20 [REDACTED]
21 [REDACTED]
22 [REDACTED]
23 [REDACTED]

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[REDACTED]

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[REDACTED]

[REDACTED] Unfortunately, Westinghouse did not perform error analyses to quantify the modeling uncertainties and the effect of code user interactions in its fatigue calculations using WESTEMS. It would appear that virtually any error would put some of the calculated values of CUF_{en} over the $CUF_{en} = 1.0$ fatigue failure limit.

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

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

Q. Are you familiar with its contents?

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

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

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

8 A. No, [REDACTED]
9 [REDACTED]
10 [REDACTED]
11 [REDACTED]
12 [REDACTED]
13 [REDACTED]
14 [REDACTED]
15 [REDACTED]
16 [REDACTED]
17 [REDACTED]
18 [REDACTED]
19 [REDACTED]

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

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

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

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

14 Q. Would you briefly summarize your concerns?

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

19 Q. Why is an error analysis important?

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

10 **Conclusion**

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

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

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

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

21 [REDACTED]

22 [REDACTED]

23 [REDACTED]

Anyway,

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

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

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

8 A. Yes.

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

12 A. Yes.

13 Q. Please summarize your testimony.

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

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

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

8 Q. Does this conclude your testimony?

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

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

16 Q. Does this complete your testimony.

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

20

1 UNITED STATES

2 NUCLEAR REGULATORY COMMISSION

3 BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

10 -----X
11 **DECLARATION OF RICHARD T. LAHEY, JR.**

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

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



17
18
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