


United States Nuclear Regulatory Commission Official Hearing Exhibit	
In the Matter of:	Entergy Nuclear Operations, Inc. (Indian Point Nuclear Generating Units 2 and 3)
	ASLBP #: 07-858-03-LR-BD01
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NYS000374
Submitted: June 19, 2012

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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 18, 2012

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PRE-FILED WRITTEN TESTIMONY OF

DR. RICHARD T. LAHEY, JR.

REGARDING CONTENTION NYS-38/RK-TC-5

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

Q. Please state your full name.

A. Richard T. Lahey, Jr.

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

A. I am 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 **Experience**

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

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

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

22 Q. Yes. Over the last 50 years, I have published
23 numerous books, monographs, chapters, articles, reports, and

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1 journal papers on nuclear engineering and nuclear reactor safety
2 technology. Those articles are listed in my Curricula Vitae.

3 Q. Have you received any professional awards?

4 A. Yes, I have received many honors and awards for my
5 career accomplishments, including: the E.O. Lawrence Memorial
6 Award of the Department of Energy (DOE), the Glenn Seaborg Medal
7 of the ANS and the Donald Q. Kern Award of the AIChE.

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

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

13
14 **Previous Submissions**

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

17 A. Yes. They are copies of the six declarations that I
18 previously prepared for the State of New York in this
19 proceeding. They include the initial declaration that was
20 submitted in November 2007 in support of the State's petition to
21 intervene and its initial contentions, the April 7, 2008
22 declaration in support of Contention NYS-26A, the September 15,
23 2010 declaration submitted in support of the State's

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1 supplemental bases for Contention 25, the September 9, 2010
2 declaration submitted in support of the amended Contention NYS-
3 26B/RK-TC-1B, and the September 30, 2011 and November 1, 2011
4 declarations submitted in support of the present Contention,
5 Contention NYS-38/RK-TC-5.

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

8 A. Yes. It is a copy of the Report that I prepared for
9 the State of New York in this proceeding concerning Contentions
10 NYS-25 and NYS-26B/RK-TC-1B. The Report reflects my analysis
11 and opinions.

12 Q. I show you what has been marked as Exhibit NYS000297.
13 I note that the State has provisionally designated the exhibit
14 as containing confidential information. Do you recognize that
15 document?

16 A. Yes. This is a copy of a Supplemental Report that I
17 previously prepared for the State of New York in this proceeding
18 that addresses aspects of the revised fatigue analysis that
19 Entergy and Westinghouse prepared concerning certain components
20 in the Indian Point reactors and their related systems and
21 boundaries. The Supplemental Report sets out some of my
22 concerns about the use of the WESTEMS computer code by Entergy
23 and Westinghouse to develop a cumulative fatigue analysis of

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1 certain reactor components. The Supplemental Report also
2 reflects my analysis and opinions.

3 Q. I show you a copy of what has been marked as Exhibit
4 NYS000344. Do you recognize that document?

5 A. Yes, this is a copy of the pre-filed testimony that I
6 previously submitted in December 2011 concerning Contention NYS-
7 26B/RK-TC1B.

8
9 **Overview**

10 Q. What is the purpose of your testimony?

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

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1 The purpose of my testimony today is to provide support
2 for, and my views on, certain aspects of New York's and
3 Riverkeeper's Joint Contention NYS-38/RK-TC-5 ("NYS-38/RK-TC-
4 5"), which was admitted for litigation by the Atomic Safety
5 Licensing Board. Contention NYS-38/RK-TC-5 asserts, among other
6 things, that Entergy has not demonstrated that it has a program
7 that will manage the effects of aging of critical components or
8 systems at the Indian Point nuclear power facilities and that
9 therefore the USNRC does not have a record and a rational basis
10 upon which it can determine whether to grant Entergy a renewed
11 license for the Indian Point facilities.

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

16 My testimony also critiques Entergy's proposed approach
17 that defers important aspects of a program that ostensibly seeks
18 to address the age related degradation caused by metal fatigue.

19 It is also my understanding that the Board has deferred
20 presentation of testimony on another aspect of this contention,
21 namely the ongoing iterative regulatory dialogue between Entergy
22 and USNRC Staff concerning important aspects of Entergy's

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1 approach to address age related degradation caused by
2 embrittlement.

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

5 A. Yes.

6 Q. What is the source of those materials?

7 A. I have reviewed documents prepared by government
8 agencies, Entergy, Westinghouse, the utility industry, or its
9 associations, and various related text books and peer-reviewed
10 articles.

11 Q. I show you Exhibits NYS00146A-C, NYS00147A-D,
12 NYS000150 through NYS000154, NYS000160, NYS000161, NYS000195,
13 NYS000304 through NYS000369 and NYS000375 through NYS000395. Do
14 you recognize these documents?

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

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1 Q. How do these documents relate to the work that you did
2 as an expert in forming opinions such as those contained in this
3 testimony?

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

7
8 **Conclusions and Opinions**

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

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

15 Q. Did you review the Supplemental Safety Evaluation
16 Report?

17 A. Yes.

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

20 A. Yes, these are 2011 communications from Entergy to the
21 USNRC Staff in response to Staff questions about the age related
22 degradation of various components at Indian Point Unit 2 and

1 Indian Point Unit 3. These are Entergy communications NL-11-
2 032, NL-11-074, NL-11-90, and NL-11-096.

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

4 A. I have reviewed the USNRC Staff's Supplemental Safety
5 Evaluation Report for Indian Point Unit 2 and Unit 3. The SSER
6 makes it clear that a number of important details and questions
7 remain unresolved concerning the aging-induced degradation of
8 various safety-related systems and components and the management
9 of that process. Unfortunately, there are virtually no details
10 given on the future analyses and/or inspections that Entergy
11 will apparently do. The absence of such details makes it
12 difficult, if not impossible, to meaningfully evaluate the
13 approach or program that Entergy proposes. In any event, the
14 dates given for Entergy and the USNRC's anticipated resolution
15 of these issues appear to be beyond the time frame for
16 submission of testimony and the evidentiary hearings in this
17 ASLB proceeding and thus will not allow for a testing of the
18 adequacy of the proposed resolution of these issues in this
19 proceeding. That timeline will also prevent the State of New
20 York from playing any meaningful role in their development or
21 resolution.

22 The details of the inspections for primary water stress
23 corrosion cracking (PWSCC) in the steam generator's divider

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1 plates will apparently not be available until after extended
2 operations are expected to begin.

3 Inspections of the steam generator's tube-to-tubesheet
4 welds in Indian Point Unit 2 for PWSCC will not be made until
5 sometime between March 2020 and March 2024, which is well after
6 the proposed extended operation period has begun. This is
7 particularly troubling since these welds form part of the
8 primary system's pressure boundary, and if they fail radiation
9 may be released to the secondary side and also to the
10 environment.

11 Inspections of the steam generator's tube-to-tubesheet
12 welds in Indian Point Unit 3 for PWSCC will not be made until
13 the first refueling outage after the reactor enters the period
14 of extended operation. Indian Point Unit 3 could enter its
15 period of extended operation in late December 2015. Based on
16 the current refueling schedules, which have Indian Point Unit 3
17 refueling in March of odd numbered years, I anticipate that the
18 first refueling outage for Indian Point Unit 3 after it enters
19 the period of extended operation in December 2015 would be in or
20 around March 2017.

21 In addition, although USNRC Staff has required Entergy and
22 Westinghouse to disclose the parameters surrounding code user
23 interventions in future applications and runs of the WESTEMS

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1 code for Indian Point, Entergy has not disclosed the parameters
2 surrounding user intervention in the previous runs of WESTEMS
3 that provided the basis for the what has been described as the
4 refined metal fatigue analysis that was previously submitted in
5 this proceeding. The absence of this information impedes and
6 prevents a meaningful analysis of the metal fatigue analysis
7 that Entergy has presented here and the aging management program
8 that Entergy has proposed.

9 Furthermore, the State has previously raised concerns about
10 the locations within the reactor coolant pressure boundary that
11 are examined with respect to fatigue. In 2011 Entergy indicated
12 that it will undertake further steps to identify additional
13 limiting locations that are subject to fatigue in the reactor
14 coolant pressure boundary. Based on a May 15, 2012 Entergy
15 letter to the Board, it appears that Entergy and Westinghouse
16 will not complete the initial steps of this analysis until mid
17 September 2012. It is my understanding that Entergy has not yet
18 disclosed the results of any such review. The absence of this
19 information also impedes and prevents an meaningful analysis of
20 the metal fatigue analysis that Entergy has presented here and
21 the aging management program that Entergy has proposed.

22
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1 **The Indian Point Reactors**

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

4 A. Yes.

5 Q. Would you briefly describe them?

6 A. Entergy operates two power reactors that are located
7 in northern Westchester County near the Village of Buchanan.
8 The operating reactors are known as the Indian Point Unit 2 and
9 Indian Point Unit 3 power reactors. These Westinghouse-designed
10 plants are 4-loop pressurized water reactors (PWRs), and they
11 are currently rated at power levels of 3,216.4 MWt. Entergy
12 also owns another reactor at the same site. That reactor is
13 known as the Indian Point Unit 1 reactor; however, that reactor
14 has been shut down and no longer produces power.

15
16 **Operation of a Pressurized Water Reactor**

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

19 A. Pressurized water nuclear reactors have water (i.e.,
20 the primary coolant) under high pressure flowing through the
21 core in which heat is generated by the fission process. The
22 core is located inside a reactor pressure vessel (RPV). This
23 heat is absorbed by the coolant and then transferred from the

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1 coolant in the primary system to lower pressure water in the
2 secondary system via a large heat exchanger (i.e., a steam
3 generator) which, in turn, produces steam on the secondary side.
4 These steam generator systems, which are part of the plant's
5 Nuclear Steam Supply System (NSSS), are located inside a large
6 containment structure. After leaving the containment building,
7 via main steam piping, the steam drives a turbine, which turns a
8 generator to produce electrical power.

9 As I mentioned, Indian Point Unit 2 and Indian Point Unit 3
10 each have a four loop Nuclear Steam Supply System or NSSS. Each
11 loop contains, among other components, a pressurizer, a steam
12 generator, and a reactor coolant pump. Thus, the two operating
13 Indian Point reactors collectively have eight steam generators.

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

17 As its name (PWR) suggests, this reactor design uses a
18 pressurizer on the primary side that performs several functions.
19 In particular, it maintains the operating pressure on the
20 primary side of the nuclear reactor and accommodates variations
21 in reactor coolant volume for load changes during reactor
22 operations, as well as reactor heat-up and cool-down. The
23 reactor coolant also moderates the neutrons produced in the core

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1 since a pressurized water nuclear reactor will not function
2 unless the neutrons are moderated (i.e., slowed down due to
3 collisions with the hydrogen molecules in the primary coolant).

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

6 A. Yes. It is a schematic diagram from a USNRC document
7 that identifies the relative location of various components in a
8 pressurized water nuclear reactor type of power plant including,
9 from the inside to the outside, the: reactor core, reactor
10 pressure vessel, pressurizer, steam generator, containment
11 structure, turbine, and associated piping. The diagram also
12 identifies various materials that are used or contained in those
13 components.

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

16 A. Yes. It is a schematic diagram of a Westinghouse
17 Nuclear Steam Supply System. Among other things, this diagram
18 depicts the reactor coolant pressure boundary. The components
19 on the primary side appear in red or orange, and the components
20 on the secondary side are in blue or green.

21 Q. What is the reactor coolant pressure boundary?

22 A. The USNRC provides a definition of the reactor coolant
23 pressure boundary in its regulations 10 C.F.R. § 50.2. In

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1 essence, the reactor coolant pressure boundary refers to a
2 physical barrier or boundary between the reactor coolant system
3 in the "primary loop" of nuclear steam supply system and the
4 "secondary loop" of the nuclear steam supply system. As I
5 noted, one can see this boundary line in Westinghouse diagram
6 (Exhibit NYS000375) that represents the primary loop in red or
7 yellow and the secondary loop in green or blue. It is critical
8 not to breach the reactor coolant pressure boundary and allow
9 the reactor coolant to escape.

10
11 **Steam Generators, Their Components, and**
12 **Primary Water Stress Corrosion Cracking**

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

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

17 Q. Would you please describe the role of the steam
18 generators in the Indian Point nuclear steam supply systems?

19 A. Each reactor coolant loop contains a vertical shell
20 and U-tube steam generator. Reactor coolant enters the inlet
21 side of the channel head at the bottom of the steam generator
22 through the inlet nozzle, is forced upward through the
23 tubesheet, flows through the U-tubes, returns through the

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1 tubesheet to an outlet plenum and leaves the generator through a
2 bottom nozzle. The inlet and outlet plena in the steam
3 generator are separated by a partition or divider plate. The
4 divider plate is joined to the plenum head and the tubesheet
5 through a stub runner.

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

8 A. Yes, in the License Renewal Application Tables 2.3.1-
9 4-IP2/IP3 of the LRA Entergy states that the channel head, the
10 divider plate, tubes, and the tubesheet each constitutes a
11 pressure boundary for Indian Point Unit 2 and Indian Point Unit
12 3. Entergy acknowledged that the tubes also perform a heat
13 transfer function. Those tables are located in the License
14 Renewal Application at pages 2.3-36, 2.3-39, respectively.

15 Q. Are you familiar with the term "primary water stress
16 corrosion cracking"?

17 A. Yes. Primary water stress corrosion cracking is a
18 recognized aging phenomenon for many alloy/environmental
19 combinations. It presents a challenge since it occurs in
20 otherwise ductile alloys, but only in very specific
21 environments. Occurrence of the phenomenon requires the
22 simultaneous presence of stress, whether residual or applied,
23 and a specific alloy /environment combination.

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1 Q. Has primary water stress corrosion cracking occurred
2 in the nuclear energy production area?

3 A. Yes. In operating nuclear reactors, cracking of
4 stressed nickel based alloys, such as Alloy 600, has occurred.
5 It should also be noted that Alloy 600 components are generally
6 welded with Alloys 82 or 182, which are derivatives of Alloy 600
7 that have also been found to be susceptible to primary water
8 stress corrosion cracking.

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

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

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

21 A. Yes, this is a copy of NL-11-032, which was Entergy's
22 March 28, 2011 initial response to the USNRC Staff's request for
23 additional information. In this document starting on page 20 of

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1 Attachment 1, Entergy discusses, among other things, the
2 material composition of the steam generator divider plates and
3 associated welds.

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

6 A. Yes, in the License Renewal Application at page 2.3-
7 21, Entergy stated that the current Indian Point Unit 2 steam
8 generators use Alloy 600 for the heat transfer tubes and that
9 the current Indian Point Unit 3 steam generators use Alloy 690
10 for their tubes.

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

13 A. In addition to the heat transfer tubes, primary water
14 stress corrosion cracking could also affect other components or
15 assemblies that use Alloy 600 or welds that use Alloy 82/182
16 weld material that, as I noted, are derivatives of Alloy 600.
17 In the August 30, 2011 Supplemental Safety Evaluation Report at
18 page 3-21, the USNRC Staff has also expressed concern about the
19 propagation of primary water stress corrosion cracking in
20 tubesheets that have Alloy 600 cladding or related weld even
21 when the heat transfer tubes are made from Alloy 690TT material.
22 According to Staff, "a crack initiated in this region, close to
23 the tube, may propagate into or through the weld, causing a

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1 failure of the weld and of the reactor coolant pressure
2 boundary." These areas of concern would include the channel
3 head-to-tubesheet-to-tube complex, including the divider plate
4 assembly and the tube-to-tubesheet welds.

5 Q. In your opinion, would primary water stress corrosion
6 cracking of the divider plates, weld, or channel head assemblies
7 impact the intended function of the steam generators?

8 A. Yes, in my opinion it would.

9 First, primary water stress corrosion cracking of a divider
10 plate or its weld could compromise the ability of the divider
11 plate to direct fluid through the tubesheet into the heat
12 transfer tubes and hence impede one of the intended functions of
13 the tubes and the steam generator to transfer heat and thus to
14 provide a heat sink for the heat generated in the core. I would
15 consider the loss of that intended function to be a significant
16 safety concern since shock-load-induced failures of the divider
17 plate have apparently not been analyzed (e.g., the
18 thermal/pressure shock loads experience during various
19 postulated LOCA events), but such events may lead to gross
20 failures of cracked divider plates.

21 Second, the USNRC Staff recognized that a primary water
22 stress corrosion crack in the divider plate could propagate into
23 a tubesheet and a tube-to-tubesheet weld. Such a crack in the

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1 lower steam generator assembly area could compromise another
2 intended function of the steam generator, namely the maintenance
3 of the reactor coolant pressure boundary between the primary
4 loop and the secondary loop in the nuclear steam supply system.

5 Q. Do you have any opinion about the sufficiency of the
6 approach that Entergy has proposed here regarding primary water
7 stress corrosion cracking of steam generator components at the
8 Indian Point facilities?

9 A. As I stated last Fall, Entergy's proposal leaves many
10 important details and questions unresolved. As the Supplemental
11 Safety Evaluation Report confirms (at p. 3-19), Entergy proposes
12 that it will perform "an inspection of steam generators for both
13 units to assess the condition of the divider plate assembly."
14 However, this proposal does not describe the inspection
15 methodology nor the number of steam generators to be inspected.
16 Indeed, it is quite vague and does not provide details.
17 Similarly, it does not describe the acceptance criteria for such
18 inspection or the corrective action criteria for divider plates
19 that fail the inspection.

20 Turning to the issue of cracks spreading from tubesheet
21 cladding to tube-to-tubesheet welds, Entergy again proposes an
22 approach that is short on details. Specifically, Entergy
23 proposes to "develop a plan" that will use one of two options:

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1 (1) "perform an analytical evaluation" to establish that
2 tubesheet cladding and welds are not susceptible to primary
3 water stress corrosion cracking, or redefine the reactor coolant
4 pressure boundary; or (2) perform a one time inspection of a
5 representative number of tube-to-tubesheet welds in each steam
6 generator to determine if primary water stress corrosion
7 cracking is present. This plan that Entergy has proposed to
8 develop leaves many questions unanswered, including: the basis
9 for the analytical analysis, how Entergy can change the
10 definition of the reactor coolant pressure boundary after these
11 facilities have relied on that definition for many years, and,
12 the methodology of the alternative one-time inspection.

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

16
17 **WESTEMS and "User Intervention"**

18 Q. Turning to the issue of fatigue, could you explain
19 what fatigue is?

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

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1 Fatigue of various structures, components and fittings in a
2 nuclear reactor can result in pipe ruptures, physical failures,
3 and the relocation of loose pieces of metal throughout the
4 reactor system, which, in turn, may result in core blockages and
5 interfere with the safe operation of a nuclear power plant. The
6 main concerns about fatigue are the increased potential for a
7 primary or secondary side LOCA, and the failure of various RPV
8 internals.

9 Q. I would like to direct your attention to NYS000160,
10 NYS000151, NYS000152, NYS00153, and NYS00154. Do you have those
11 documents?

12 A. Yes, I have those documents.

13 Q. Would you identify them for the record?

14 A. Yes. These documents include the August 2011 USNRC
15 Staff Supplemental Safety Evaluation Report (NUREG-1930, Supp.
16 1) and Entergy's 2011 responses to the Staff's request for
17 additional information, including Entergy communications NL-11-
18 032, NL-11-074, NL-11-90, and NL-11-096.

19 Q. Do these documents discuss the use of the WESTEMS
20 computer code?

21 A. Yes, they discuss Entergy's and Westinghouse's use of
22 WESTEMS at Indian Point as part of the license renewal process.

23 Q. Do they also discuss the term "user intervention"?

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1 A. They do.

2 Q. Would you describe the term user intervention?

3 A. Yes. The term "user intervention" refers to, among
4 other things, the use of assumptions and engineering judgment in
5 the process of calculating the CUF_{en} values using codes such as
6 WESTEMS.

7 Q. Did you review NYS000160, NYS000151, NYS000152,
8 NYS00153, and NYS00154 with respect to the use of WESTEMS and
9 user intervention in the Indian Point license renewal process

10 A. Yes.

11 Q. And did you draw any conclusions from that review?

12 A. Yes.

13 Q. Would you describe those conclusions?

14 A. Certainly.

15 Entergy relies on WESTEMS, a proprietary computer program
16 developed by Westinghouse, as an essential part of Entergy's
17 CUF_{en} analysis for the Indian Point facilities. Entergy agreed
18 with the USNRC that the piping system stress model (NB-3600) in
19 WESTEMS will not be used until the USNRC staff resolves some
20 issues concerning its validity. Rather, a finite element method
21 (FEM) "design by analysis" approach (NB-3200) will be used
22 instead. Unfortunately, this finite element method-based
23 computational approach requires numerous assumptions concerning

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1 the stress-inducing thermal transients and the loads/moments
2 from the piping system; such assumptions must be developed and
3 applied by the WESTEMS code user to the component being
4 analyzed. These assumptions could materially affect the results
5 raising questions concerning their reliability and validity.
6 Thus, it is necessary to have disclosed in advance the
7 assumptions to be used in the analysis and the basis for using
8 those assumptions in order to ascertain whether the approach
9 being proposed will meet the required safety standards for an
10 adequate AMP. Given the role that such user-developed
11 assumptions play in the process, and the fact that Entergy has
12 apparently not done an "error analysis" of the WESTEMS results,
13 there remain important questions concerning the reliability and
14 validity of these results.

15 In its August 2011 Supplemental Safety Evaluation (at 4-2),
16 USNRC Staff has required Entergy to create records that document
17 and justify any assumptions and engineering judgments developed
18 and used in the CUF_{en} calculations. Such assumptions and
19 engineering judgments affect the WESTEMS results. A systematic
20 and methodical explanation of these assumptions and engineering
21 judgments is essential in evaluating the adequacy of the CUF_{en}
22 calculations done for IP-2 & IP-3 using WESTEMS.

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1 Entergy has agreed that any user intervention in future
2 WESTEMS evaluations will be explained and justified.
3 Unfortunately, nothing was said about the previous WESTEMS
4 evaluations that were done for IP-2 & IP-3 and the affect that
5 user interventions had on those CUF_{en} results (for which no
6 "error analysis" has been given). Moreover, as the Supplemental
7 Safety Evaluation Report makes clear (at 4-2), the documentation
8 of any new user interventions will not be disclosed or
9 implemented until close to the end of the current licensing
10 terms (i.e., September, 2013 and December, 2015). In addition,
11 Entergy has not disclosed the specific criteria it will use in
12 deciding whether to make a user intervention and what standards
13 will control the extent of these interventions. Thus, the State
14 of New York is being effectively excluded from reviewing this
15 important process.

16 Q. Dr. Lahey, I direct your attention to Exhibits
17 NYS000296 and NYS000297, which are your December 2011 reports.

18 A. Yes, I have those documents.

19 Q. Turning to Exhibit NYS000296 would you describe your
20 observations and conclusions about WESTEMS and user
21 interventions and assumptions?

22 A. As I discussed in this report, for example at
23 paragraphs 35 and 36, it is apparent that the WESTEMS code is

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1 based on rather simple models (particularly the thermal-
2 hydraulic models) and that the thermal stress results for CUF_{en}
3 are strongly influenced by the code user's assumptions,
4 manipulations and interventions. In fact, there is a lot of
5 "engineering judgment" implicit in the CUF_{en} results, and, since
6 an "error analysis" has not been done to bound the uncertainty,
7 and many results are disturbingly close to the $CUF_{en} = 1.0$ limit,
8 I do not believe that one can trust these results to assure the
9 safety of the IP-2 and IP-3 during extended plant operations.
10 Indeed, these results are quite uncertain and this uncertainty
11 should be quantified by doing parametric runs and/or a detailed
12 "error analysis". Moreover, because the effect of various shock
13 loads on the failure of these fatigue-weakened components,
14 structures, and fittings has not been considered, it is unclear
15 that the health and safety of the American public is being
16 adequately protected. As previously noted, these results are
17 quite uncertain and this uncertainty needs to be quantified by
18 doing a detailed "error analysis".

19 Q. Now, I would like to turn to Exhibit NYS000297, which
20 is your Supplemental Report. I note that this exhibit has been
21 provisionally designated as potentially containing proprietary
22 information. Would you briefly describe your conclusions
23 contained in this report?

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1 A. Yes. The Supplemental Report sets out some of my
2 concerns about the use of the WESTEMS computer code by Entergy
3 and Westinghouse to develop a cumulative fatigue analysis of
4 certain components in the reactors and their reactor coolant
5 pressure boundaries. Among other things, the report describes
6 my concerns about the use of engineering judgment, assumptions,
7 and user interventions as part of the WESTEMS analysis.

8 Q. Do you have any additional conclusions or observations
9 concerning the use of WESTEMS and user intervention in the
10 Indian Point license renewal application process?

11 A. There is a difference between stating that one will
12 develop a program that will comply with the parameters in GALL
13 and actually disclosing the details, judgments, assumptions, and
14 user interventions that underlay the program and the analyses,
15 including computer codes such as WESTEMS, that are critical to
16 the program. Only through the latter can one test an
17 applicant's claim that its proposed program is consistent with
18 GALL. In fact, it is not possible to demonstrate that an aging
19 management program is effective or consistent with GALL unless
20 the details, judgments, assumptions, and user interventions have
21 been disclosed. This is particularly important since neither
22 Westinghouse nor Entergy appear willing to perform an "error

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1 analysis" of WESTEMS results to justify any claims of
2 compliance.

3
4 **Identification of Limiting Locations for Fatigue Analysis**

5 Q. Earlier in your overview of your testimony you stated
6 that you had concerns about the locations selected for metal
7 fatigue analysis. Could you expand on that?

8 A. Yes. For CUF_{en} fatigue calculations it is important to
9 fully understand the assumptions that will be made and the
10 criteria that will be used in determining which locations will
11 produce the most limiting conditions.

12 It is my understanding that as a result of additional
13 review by USNRC Staff of the Indian Point license renewal
14 application, the USNRC Staff has raised concerns that Entergy
15 may not have chosen the sites of the most limiting fatigue
16 conditions and Entergy has agreed to reanalyze the locations it
17 has previously identified and to determine if more limiting
18 conditions exist at other sites. If so, detailed further
19 analysis will be required.

20 Unfortunately, the exact time for reporting the results of
21 this future review/analysis was not specified, but it will
22 apparently be shortly before extended operations are expected to
23 begin. Postponing the disclosure of the details of that

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1 review/analysis until Indian Point Unit 2 is on the cusp of
2 extended operation will prevent those matters from being tested
3 and resolved in these ASLB hearings and greatly handicaps, if
4 not precludes, the State of New York from any meaningful role in
5 their development and resolution. Moreover, the assumptions to
6 be used and the criteria to be applied for these future reviews,
7 and whether they were properly designed to identify limiting
8 locations and the conditions of such locations, are left for
9 consideration at a later day -- apparently by the USNRC and
10 Entergy, but not the State and other interested parties.

11 Moreover, it also appears that, as before, this review will
12 focus on structures, components and fittings outside the RPV and
13 will thus not include a comprehensive consideration of the
14 fatigue of important RPV internal structures, components and
15 fittings.

16 Q. I show you Exhibit NYS000355. Do you recognize that
17 document?

18 A. Yes. This is a report entitled NUREG/CR-6260
19 "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected
20 Nuclear Power Plant Components." This document identifies
21 generic locations to examine for metal fatigue; however,
22 additional location may be required to be examined.

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1 Q. I show you Exhibit NYS000160, which is the August 2011
2 Supplemental Safety Evaluation Report and direct your attention
3 to page 4-1 and 4-2.

4 A. Yes, I have those pages.

5 Q. Would you describe your understanding of what the
6 USNRC Staff has required?

7 A. As a result of additional review of the Indian Point
8 license renewal application, USNRC Staff raised a question about
9 whether the locations included in the metal fatigue analysis for
10 the Indian Point facilities were the most limiting locations for
11 the facilities.

12 Q. Do you know how Entergy responded?

13 A. Based on the Supplemental Safety Evaluation Report at
14 page 4-1, it is my understanding that Entergy proposed to review
15 the fatigue evaluations for ASME Code Class 1 components to
16 determine whether the NUREG/CR-6260 locations are the limiting
17 locations for Indian Point Unit 2 and Indian Point Unit 3.
18 Further, according to the Supplement safety Evaluation Report,
19 Entergy agreed that if more limiting locations are identified,
20 Entergy will evaluate the most limiting location for the effects
21 of reactor coolant environment on the fatigue usage.

22 However, based on a May 15, 2012 Entergy letter to the
23 Atomic Safety and Licensing Board (ASLB), it appears that

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1 Entergy and Westinghouse will not complete the initial steps of
2 this analysis until mid-September 2012 (Exhibit NYS000395). It
3 is my understanding that Entergy has not yet disclosed the
4 results of any such review.

5 Q. Do you have any conclusions about this process and
6 time lime?

7 A. The schedule for this process and the absence of this
8 information to date impedes a meaningful evaluation of the metal
9 fatigue analysis that Entergy has presented here and the aging
10 management program that Entergy has proposed. Moreover, the
11 apparent unwillingness of Entergy to provide an "error analysis"
12 or user intervention information for their CUF_{en} results makes it
13 impossible to assess the validity of their conclusions with
14 respect to limiting locations.

15
16 **Conclusion**

17 Q. Dr. Lahey, have you reviewed the USNRC's Staff Safety
18 Evaluation Report ("SER") and Supplemental Safety Evaluation
19 Report ("SSER") as it relates to the issue metal fatigue and
20 Entergy's proposal for how to address that issue?

21 A. Yes.

1 Q. Does it provide an adequate basis for acceptance of
2 Entergy's claim that it has demonstrated that it has a
3 sufficient AMP to address the metal fatigue issue?

4 A. No. In my opinion it does not.

5 Q. What is the basis for your statement?

6 A. The SER and SSER basically accept, as valid, what
7 Entergy has proposed to address the metal fatigue issue.
8 Therefore, all the criticisms I have identified in this
9 testimony and in my prior testimony related to metal fatigue
10 apply with equal force to the SER and SSER. In addition,
11 because the USNRC Staff has apparently ignored the importance of
12 shock-load-induced failures and the synergistic effects of
13 embrittlement and metal fatigue on the integrity of reactor
14 pressure vessel internals, the credibility and technical
15 validity of the USNRC Staff's position is severely compromised.
16 I have discussed this matter in detail in my previous testimony
17 with respect to these two related degradation phenomena.

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

22 A. No, they have not.

23 Q. Does this conclude your testimony?

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A. Yes, it does.

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 18, 2012

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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23

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