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.

Experience

Q. Please summarize your educational and professional qualifications.

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

Q. Which professional societies are you a member of?

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

Q. What expert panels have you served on?

A. I have served on numerous panels and committees for the United States Nuclear Regulatory Commission (USNRC), Idaho National Engineering Laboratory (INEL), Oak Ridge National Laboratory (ORNL), and the Electric Power Research Institute (EPRI). I am a member of the National Academy of Engineering (NAE), and have been elected Fellow of both the ANS and the ASME.

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

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

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

Q. Have you received any professional awards?

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

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

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

Previous Submissions

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

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

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

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

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

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

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

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

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

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

Overview

Q. What is the purpose of your testimony?

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

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

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

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

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

approach to address age related degradation caused by embrittlement.

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

A. Yes.

Q. What is the source of those materials?

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

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

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

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

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

Conclusions and Opinions

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

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

Q. Did you review the Supplemental Safety Evaluation Report?

A. Yes.

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

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

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

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

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

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

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

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

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

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

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

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

The Indian Point Reactors

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

A. Yes.

Q. Would you briefly describe them?

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

Operation of a Pressurized Water Reactor

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

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

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

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

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

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

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

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

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

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

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

Q. What is the reactor coolant pressure boundary?

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

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

Steam Generators, Their Components, and
<u>Primary Water Stress Corrosion Cracking</u>

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

A. Yes, in my opinion it would.

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

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

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

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

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

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

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

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

WESTEMS and "User Intervention"

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

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

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

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

A. Yes, I have those documents.

Q. Would you identify them for the record?

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

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

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

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

A. They do.

Q. Would you describe the term user intervention?

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

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

A. Yes.

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

A. Yes.

Q. Would you describe those conclusions?

A. Certainly.

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

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

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

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

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

A. Yes, I have those documents.

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

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

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

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

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

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

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

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

Identification of Limiting Locations for Fatigue Analysis

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

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

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

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

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

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

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

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

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

A. Yes, I have those pages.

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

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

Q. Do you know how Entergy responded?

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

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

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

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

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

Conclusion

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

A. Yes.

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

A. No. In my opinion it does not.

Q. What is the basis for your statement?

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

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

A. No, they have not.

Q. Does this conclude your testimony?

A. Yes, it does.

UNITED STATES

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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In re:	Docket Nos. 50-247-LR; 50-286-LR
License Renewal Application Submitted by	ASLBP No. 07-858-03-LR-BD01
Entergy Nuclear Indian Point 2, LLC,	DPR-26, DPR-64
Entergy Nuclear Indian Point 3, LLC, and	
Entergy Nuclear Operations, Inc.	June 18, 2012

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DECLARATION OF RICHARD T. LAHEY, JR.

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

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

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Pre-filed Written
Testimony of Richard T. Lahey, Jr.
Contention NYS-38/RK-TC-5