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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.	March 4, 2016

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SUPPLEMENTAL WRITTEN TESTIMONY OF

DR. RICHARD T. LAHEY, JR.

REGARDING CONTENTIONS

NYS-25, NYS-26B/RK-TC-1B, AND NYS-38/RK-TC-5

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List of Proposed New Exhibits

Proposed Exhibit No.	Document
NYS000583	"Indian Point Units 2 and 3 Inspection Response Plan for Aging Management of MRP-227-A Primary and Expansion Components" (WCAP-17941-P, Rev. 1) (IPECPROP00085900), January 12, 2016 Short title: "Inspection Response Plan"
NYS000584	"Background and Technical Basis Supporting Engineering Flaw Acceptance Criteria for Indian Point Unit-2 Reactor Vessel Internals MRP-227-A Primary and Expansion Components" (WCAP-17949-P, Rev. 0) (IPECPROP00085359), January 12, 2016 Short title: "Flaw Acceptance Criteria for IP-2"
NYS000585	"Background and Technical Basis Supporting Engineering Flaw Acceptance Criteria for Indian Point Unit-3 Reactor Vessel Internals MRP-227-A Primary and Expansion Components" (WCAP-17951-P, Rev. 0) (IPECPROP000856629), January 12, 2016 Short title: "Flaw Acceptance Criteria for IP-3"
NYS000586	"Determination of Acceptable Baffle-Former Bolting for Indian Point Units 2 and 3" (WCAP-18048-P, Rev. 0) (IPECPROP00086434), January 12, 2016 Short title: "Bolting Analysis"

Proposed Exhibit No.	**Document**
NYS000587	E. Eason and R. Pathania, "Disposition Curves for Irradiation-Assisted Stress Corrosion Cracking of Austenitic Stainless Steels in Light Water Reactor Environments," ASME PVP2015-4532, Proceedings of the ASME 2015 Pressure Vessels and Piping Conference, July 19-23, 2015 Short title: "Eason and Pathania Paper"
NYS000588	USNRC Indian Point Nuclear Generating Unit-3 – License Renewal Inspection Report 05000286/2015011, Nov. 19, 2015 (ML15323A026) Short title: "IP-3 Inspection Report"

List of Acronyms

Acronym	**Meaning**
AMP	aging management plan
ASLB, the Board	Atomic Safety Licensing Board
BWR	boiling water nuclear reactor
CGR	crack growth rate
DBA	design basis accident
EPRI	Electric Power Research Institute
FMEA	failure modes and effects analysis
IASCC	irradiation-induced stress corrosion cracking
IP-2	Indian Point nuclear facility Unit-2
IP-3	Indian Point nuclear facility Unit-3
LBB	leak-before-break
LOCA	loss-of-coolant accident
LRA	license renewal application
NYS, the State	the State of New York
PEO	period of extended operation
PWR	pressurized water nuclear reactor
RTA	real-time analysis
RVI	reactor vessel internal
SSE	safe-shutdown earthquake
UFSAR	Updated Final Safety Analysis Report

I. Introduction

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., Ph.D., regarding Contentions NYS-25, NYS-26B/RK-TC-1B, and NYS-38/RK-TC-5.

Q. Please state your full name.

A. Richard T. Lahey, Jr.

Q. Dr. Lahey, you have previously provided your educational and professional qualifications and submitted testimony in this proceeding, correct?

A. Yes. I have previously submitted testimony and reports in this proceeding. My education and professional qualifications and experience are described in my previously submitted Curricula Vitae and were summarized in my previous testimony.

Q. I show you proposed exhibits NYS000583 through NYS000588. Do you recognize those documents?

A. Yes. These documents consist of four technical reports authored by Westinghouse on behalf of Entergy, one technical paper authored by E. Eason and R. Pathania, and one USNRC inspection report for Indian Point Unit-3.

Q. Have you reviewed proposed exhibits NYS000583 through NYS000588?

A. Yes.

Q. Have these documents caused you to change the testimony and opinions that you have previously submitted in this proceeding in connection with Contentions NYS-25, NYS-26B/RK-TC-1B and NYS-38/RK-TC-5?

A. No. In fact, I believe that these documents support the concerns, opinions, and testimony that I have presented previously in this proceeding. It is my opinion that the documents are relevant to the technical and legal matters at issue in this proceeding, and that they provide substantial support for the opinions that I have presented to the Atomic Safety and Licensing Board (the "Board") in my testimony and written reports in this proceeding. In my opinion, these documents provide further bases for the Board to conclude that Entergy's license renewal application ("LRA") for the Indian Point facilities fails to adequately manage the effects of aging degradation such as embrittlement, fatigue, and irradiation assisted stress corrosion cracking ("IASCC") on reactor vessel internal ("RVI") systems, structures and components. They also describe the facilities' approach to fatigue analysis and confirm that certain aging management program plan ("AMP") details will be deferred until well into the period of extended operation ("PEO").

Q. In your opinion, what is the significance of the proposed new exhibits?

A. The documents call into question whether Entergy's currently proposed amended and revised AMP for RVIs and the associated Inspection Plan for the Indian Point nuclear facilities, e.g., exhibits NYS000496-NYS000507 ("Amended and Revised RVI AMP"), provide reasonable assurance against the failure of various RVI systems, structures, and components. The documents also suggest that Westinghouse's approach to developing flaw acceptance criteria and the associated inspection intervals under Entergy's Amended and Revised RVI AMP may be inadequate and non-conservative. For example, [redacted]. In my testimony below, I will begin by describing how three of the proposed new exhibits provide specific details on the implementation of Entergy's

Amended and Revised RVI AMP and Inspection Plan. Next, I will discuss my concern regarding Westinghouse's plant-specific bolting pattern analysis for the baffle-former bolts at Indian Point Unit-2 ("IP-2") and Unit-3 ("IP-3"). Thereafter, I will describe my concern that Entergy's Amended and Revised RVI AMP contains several non-conservative assumptions that may cause IASCC cracks to grow faster at IP-2 and IP-3 than would be accounted for by Entergy under its Amended and Revised RVI AMP. Finally, I discuss USNRC's recent inspection report for IP-3.

II. <u>The Inspection Response Plan and Flaw Acceptance Criteria</u>

Q. I show you three of the proposed new exhibits, each of which relates to Entergy's Amended and Revised RVI AMP. The documents are:

- "Indian Point Units 2 and 3 Inspection Response Plan for Aging Management of MRP-227-A Primary and Expansion Components," a document that has been marked as proposed exhibit NYS000583, and identified by Westinghouse as WCAP-17941-P, Rev. 1 ("Inspection Response Plan");
- "Background and Technical Basis Supporting Engineering Flaw Acceptance Criteria for Indian Point Unit-2 Reactor Vessel Internals MRP-227-A Primary and Expansion Components," a document that has been marked as proposed exhibit NYS000584, and

identified by Westinghouse as WCAP-17949-P, Rev. 0 ("Flaw Acceptance Criteria for IP-2"); and,

- "Background and Technical Basis Supporting Engineering Flaw Acceptance Criteria for Indian Point Unit-3 Reactor Vessel Internals MRP-227-A Primary and Expansion Components," a document that has been marked as proposed exhibit NYS000585, and identified by Westinghouse as WCAP-17951-P, Rev. 0 ("Flaw Acceptance Criteria for IP-3").

Please describe how the Inspection Response Plan (NYS000583) and Flaw Acceptance Criteria for IP-2 (NYS000584) and IP-3 (NYS000585) each relate to the implementation of Entergy's Amended and Revised RVI AMP?

A. According to the Inspection Response Plan and the Flaw Acceptance Criteria for IP-2 and IP-3, these documents, [redacted].

1. The Inspection Response Plan

Q. I show you the Inspection Response Plan (NYS000583). Would you please explain your understanding of how this document relates to Entergy's Amended and Revised RVI AMP?

A. Yes. [redacted]

[redacted]

[redacted]

2. The Flaw Acceptance Criteria for IP-2 and IP-3

Q. I now show you two documents, the Flaw Acceptance Criteria for IP-2 (NYS000584), and the Flaw Acceptance Criteria for IP-3 (NYS000585). Would you explain your understanding of how these documents relate to Entergy's Amended and Revised RVI AMP?

A. [redacted]

[redacted] As a guide to how Entergy will ultimately address any cracks or flaws in RVIs identified as a result of inspections, these documents serve as a key piece of Entergy's Amended and Revised RVI AMP.

Q. What, if any, concerns do you have about Entergy's Amended and Revised RVI AMP, as described in the Inspection Response Plan and the two Flaw Acceptance Criteria documents for IP-2 and IP-3?

A. [redacted]

[redacted]; and Entergy License Renewal Application – Revised Reactor Vessel Internals Program and Inspection Plan, NL-12-037, Attach. 2, at Table 5-2 (Primary components) and Table 5-3 (Expansion components).

In my view, waiting 10 years between inspections is not a reasonable or prudent approach for plants operating beyond their originally intended design life. As I have discussed in my prior testimony, many systems, structures, and components at Indian Point will be significantly degraded as they approach the end of their PEO. In particular, some RVIs will have accumulated sufficient neutron fluence such that the effects of various age-related degradation mechanisms such as irradiation-induced embrittlement and IASCC will be a concern. Because degradation is expected to accelerate with aging, I believe more frequent inspections are warranted, particularly as these RVI systems, structures, and components reach the later stages of the proposed PEO when exposure to radiation is expected to meet or exceed thresholds for the development of significant embrittlement and IASCC.

Q. Do you have any other concerns about Entergy's Amended and Revised RVI AMP, as described in the Inspection Response Plan and the Flaw Acceptance Criteria for IP-2 and IP-3?

A. Yes. As I have discussed in my prior testimony, any inspection-based aging management program has some probability that an existing flaw will go undetected. For example, [redacted]. Undetected cracks are a concern because inspections will not occur for another 10 years. Not only can existing cracks -- both detected and undetected -- grow, but new cracks can develop and grow in the period between inspections. Embrittlement results in decreased fracture toughness, with a corresponding decrease in the critical flaw size that can lead to accelerated component failure. Thus my concern is that with embrittled components, cracks may grow to critical size, and the components may fail before corrective action is taken to ensure component integrity or assembly functionality. Although Westinghouse's flaw evaluation procedures under Entergy's Amended and Revised RVI AMP [redacted]

[redacted], Entergy cannot possibly evaluate and monitor any cracks that it has failed to identify.

III. <u>Westinghouse's Baffle-Former Bolt Analysis</u>

Q. Dr. Lahey, as described above, one of your concerns relates to Entergy's and Westinghouse's approach to evaluating baffle-former bolt failures at the Indian Point facilities. How does Entergy plan to address baffle-former bolt failures in the two Indian Point reactors?

A. Proposed exhibit NYS000586, a Westinghouse report entitled "Determination of Acceptable Baffle-Former Bolting for Indian Point Units 2 and 3" (WCAP-18048-P, Rev. 0) ("Bolting Analysis"), [redacted]. This document is the site-specific bolting analysis that Entergy witnesses referenced during the November 2015 evidentiary hearing. *See* Transcript of ASLB Hearing, at 5239, 5314 (Nov. 17, 2016). At the time of this ASLB hearing, my understanding was that this site-specific bolting analysis was not available.

Q. In your opinion, are bolting failures a concern at Indian Point?

A. Yes. I have previously testified that I am concerned about the possible failure of baffle-former bolts during an extended 20-year period of operation, particularly because failures of baffle-former bolts have been observed in other similar nuclear power plants, and due to my prevailing concern that Westinghouse's analyses do not consider the combined or synergistic effects of multiple aging degradation mechanisms acting together, and the effect of very energetic shock loads. Indeed, the failure of baffle-former bolts is a significant issue for aging pressurized water nuclear reactor ("PWR") plants worldwide, and certainly for the two Indian Point plants. Initially identified in European PWRs in 1989, numerous flaws in baffle-former bolts have, in recent years, been observed in a number of domestic PWRs as well. For example, failed baffle-former bolts have been found in domestic PWRs at Ginna, D.C. Cook, Surry, Prairie Island, Point Beach, and Robinson. Among other things, bolt failures can contribute to fuel damage and loose parts. Moreover, bolt failures can greatly degrade core cooling capabilities and compromise safe plant operations.

Q. How does the Bolting Analysis (NYS000586) address these concerns at Indian Point?

A. The Bolting Analysis [redacted]

Q. What are the LOCA conditions that Westinghouse applies in the Bolting Analysis?

A. [redacted]

[redacted]

Q. So, in your opinion, is a bolting analysis that is based on [redacted] an adequate approach?

A. No. In my opinion, analysis of the single and two-phase the shock loads typically associated with the original design basis accident ("DBA") LOCA (i.e., as specified in 10 C.F.R. 50 Appendix K, an instantaneous double-ended guillotine break of the most limiting primary coolant line, in particular, the cold leg), is much more appropriate to help ensure that the RVI components will perform their intended functions during the 20 years of extended operation. It is therefore my opinion that Entergy's and Westinghouse's proposed methodology [redacted] is non-conservative and ill-advised from a safety perspective for a 20-year period of extended operation beyond the original 40-year license term.

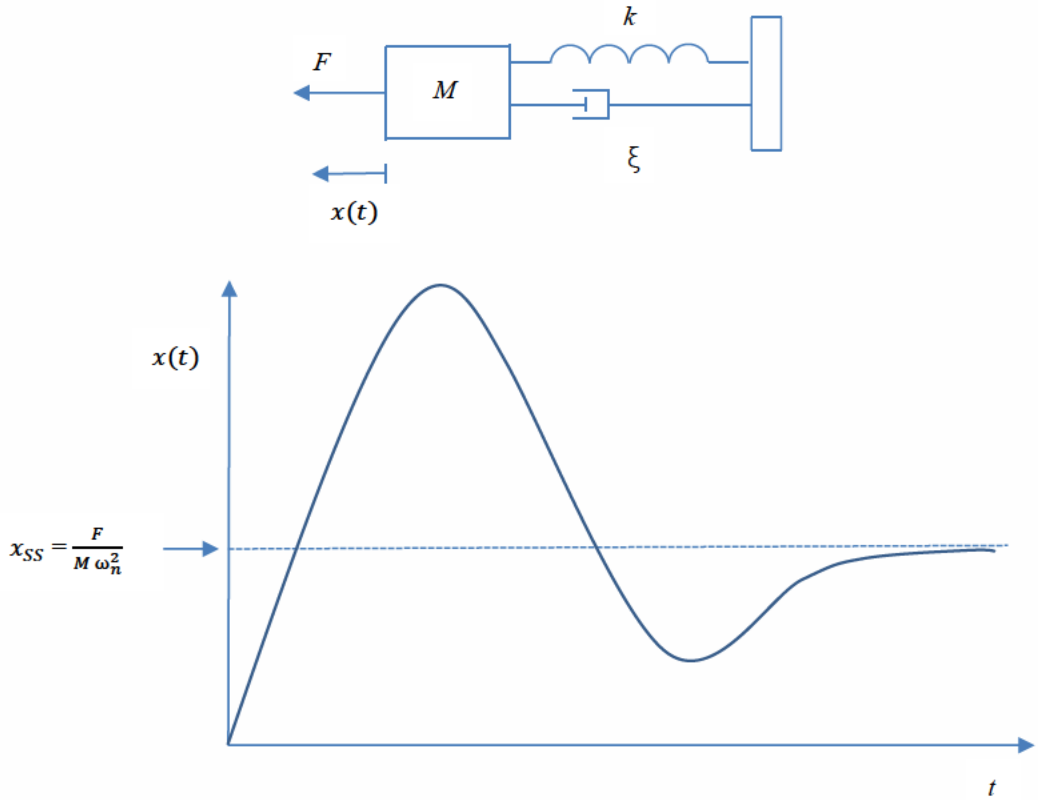
Q. How do the shock loads differ between, for example, [redacted]?

A. Large energetic, instantaneous DBA LOCA cold leg breaks generate substantially higher in-vessel shock loads than [redacted]. To understand the implications of this, I note that in a typical structural dynamics analysis, a structure may be modeled as a spring-mass-dashpot system:

$$\ddot{x} + 2\zeta\omega_n\dot{x} + \omega_n^2 x = \frac{F}{M}$$

where $\ddot{x}$ and $\dot{x}$ are, respectively, the second and first ordinary derivatives with respect to time (t) of the displacement (x) of a structure. The term $\omega_n = (\frac{k}{M})^{1/2}$ is the natural frequency of the structure (which has an equivalent spring constant of k, a damping rate of ζ, and a mass of M), and the transient force (F) acting on a structure is given by $F = \Delta pA$, where Δp is the transient pressure differential acting on a structure of area A. During the subcooled decompression stage of a DBA LOCA, the initial Δp across the baffle plate/bolting complex (and, for that matter, other in-vessel structures) can be given by a Heaviside step function (U). In particular, if ΔP is the magnitude of the largest transient Δp, then, $F(t) = \Delta PAU(t) = \Delta PA$ if $t \geq 0.0$, and zero for $t < 0.0$.

For this type of impulsive forcing function, an underdamped system, such as the baffle/bolting complex in a PWR, responds as shown schematically in the hand-drawn diagram I prepared during the November 2015 evidentiary hearing (exhibit NYS000582). For convenience, I have reproduced this schematic below:



For step change in F: $F(t) = \Delta PAU(t)$

Note that there will be significant inertial overshoot of the displacement, $x(t)$ before it returns to an equilibrium (steady-state) displacement, x_{ss}. This figure shows a hypothetical situation in which the ΔP is maintained constant, while for a PWR the impulsive forcing function will be more like that associated with a Dirac delta function, $F = \Delta PA\delta(t)$. However, the initial inertial overshoot will be similar to the step change solution but the solution will finally return to its original state, x_{ss} = 0.0. Naturally, the smaller the size of the pipe break, the less the inertial overshoot. Additionally, if the Δp(t) occurs over a finite period of time (Δt) [redacted], the induced overshoot for these ramps (as opposed to steps) in Δp(t) will also be reduced. In any event, the magnitude of the transient overshoot in the displacement, $x(t)$ of the structure determines the maximum strain (and stress) that the structure is exposed to. Because the baffle-former bolts during the period of extended operations will be highly fatigued and embrittled, their stress-strain curve will initially be quite steep (i.e., the linear elastic part will have a large slope due to irradiation-induced hardening), and the ultimate stress (which for highly embrittled stainless steel bolts will be almost the same as the yield stress) will be larger than for a corresponding ductile structure. However, physical failure of the structure is expected once the strain associated with the ultimate stress has been exceeded. Thus, for the large displacements (and thus, for the large strains) associated with energetic shock loads such as those for a DBA LOCA, failure of many of the baffle-former bolts in IP-2 and IP-3 may occur. If so, there may no longer be an intact core geometry and thus adequate core cooling will no longer be assured. In addition, the two-phase LOCA loads, which occur after the subcooled decompression loads, may lead to crushing of the fuel rod grids and the inability to insert the control rods (which are required to safely shut down the reactor). *See* "Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions under Faulted Load Conditions," WCAP-15030-NP-A (Mar. 2, 1999), § K at 2-6 (ENT000655).

Because Entergy's evaluations (performed by Westinghouse) assumes [redacted]. As such, these transient loads are non-conservative and serve to undermine the validity of any future real-time analyses of acceptable bolting patterns performed for IP-2 and IP-3. Significantly, [redacted], such as the DBA LOCA which was required by 10 C.F.R. 50 Appendix K in the original design of the Indian Point reactors. *See* Updated Final Safety Analysis Report ("UFSAR") for IP-2, § 14.3.4.3.1 at 113 (NRC000206); § 4.1.2.4.2 (ENT000634). [redacted]

Q. In your opinion, should Entergy's and Westinghouse's consideration of dynamic shock loads in the Bolting Analysis include DBA LOCA events such as an instantaneous double-ended guillotine break in the main coolant piping?

A. Yes. I am aware that USNRC regulations require an applicant to consider the impact of various size pipe ruptures, including an instantaneous double-ended guillotine rupture of the cold leg in the primary system of a PWR when assessing plant safety under LOCA conditions. *See, e.g.*, 10 C.F.R. 50, Appendices A, K. However, I also understand that in 1998, the USNRC Staff authorized the analysis of acceptable baffle-former-barrel bolting distributions based on smaller auxiliary line breaks and slower LBB break opening times. *See* "Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions under Faulted Load Conditions," WCAP-15030-NP-A (Mar. 2, 1999) (ENT000655), [redacted]

[redacted] Nevertheless, as I have indicated in my previously filed testimony, [redacted] were never intended to be applied to evaluations designed to demonstrate core cooling capability and internal component integrity, and certainly not for aged and degraded components that have been in use for over 40 years, the original design life of these plants. Instead, this methodology was developed for use in "pipe whip" evaluations within the containment. Indeed, it is totally inconsistent to use [redacted] for the analysis of potential RVI failures because, for example, significant baffle-former bolt failures may lead to an uncoolable core geometry, while the DBA LOCA analysis required by 10 C.F.R. 50, Appendix K implicitly assumes an intact core geometry. Thus, the PWR plant safety analysis requires the assumption of an instantaneous double-ended guillotine break of the cold leg for both the transient loads on RVIs and the transient core thermal-hydraulics. Anything else would be inadequate, and, in my opinion, Entergy and Westinghouse [redacted].

Q. In your opinion, does Westinghouse's Bolting Analysis (NYS000586) demonstrate that the possible failure of baffle-former bolts at IP-2 and IP-3 will be adequately addressed?

A. Absolutely not. Interestingly, the report reveals that [redacted]

[redacted]

Q. Throughout your testimony in this proceeding, you have expressed your concern that Entergy is not considering adequate shock loads in the various analyses it has conducted under its LRA. [redacted], are there other shock loads you believe Entergy should consider in its analysis of acceptable baffle-former bolting at the Indian Point facilities?

A. Yes. The safe operation of a nuclear plant requires analyses of the plant's responses to various postulated equipment failures or malfunctions. It is important to select a sufficiently broad spectrum of accident and transient events to evaluate. To be conservative, it is important to identify the accidents or events that give rise to the most limiting or challenging conditions. As a nuclear safety expert, my opinion is that it is vital to consider the most limiting event -- which for the Indian Point plants is a double-ended guillotine break of the cold leg between the reactor coolant pump and the reactor vessel, *see* IP-2 UFSAR, § 14.3.2.1 at 94 (NRC000206), to ensure that all the RVI systems, structures and components which can impact plant safety are capable of withstanding those conditions. Indeed this approach is fully consistent with USNRC directives: "A plant may always choose to use as a bounding bolting configuration calculated for a pipe break size greater than the largest size included under LBB." *See* "Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions under Faulted Load Conditions," WCAP-15030-NP-A (Mar. 2, 1999), § 2.1.1.1 at 2-7, 2-8 (ENT000655).

Q. What type of LOCAs did Entergy consider in its development of Revised and Amended RVI AMP for Indian Point?

A. As I have noted previously, [redacted].

Q. Entergy appears to have a fail-safe system to ensure core cooling. Does the existence of this system alleviate your concern?

A. No. Although Entergy has emergency core cooling systems in IP-2 and IP-3, they are not fail-safe. These engineered safety coolant injection systems are intended to maintain core cooling in the event of various pipe ruptures, including a DBA LOCA, but my concern is that due to the associated shock loads generated during a DBA LOCA some of the highly embrittled and fatigued reactor vessel internals (e.g., the baffle-former bolting) may fail such that there will no longer be a coolable core geometry. Moreover, as I have previously testified, important primary coolant pressure boundary components (e.g., the accumulator line and nozzles) that are degraded due to fatigue and stress corrosion cracking and/or embrittled due to thermal embrittlement of the welds, may fail under large LOCA-induced shock loads. [redacted], it is unclear whether a coolable core can be maintained in the event of a large, energetic line break in IP-2 and IP-3. In my opinion, [redacted] full dynamic effects of a blowdown following a DBA LOCA in order to confirm that reactor vessel internals and reactor coolant piping can withstand the most limiting LOCA load in combination with a SSE, which may initiate the LOCA event. That is, if a DBA LOCA event ever occurs, it is most likely during the PEO when an aged and degraded PWR experiences a significant seismic event. Thus the safety evaluation of this postulated event is essential.

Q. In your opinion, does the Bolting Analysis (NYS000586) adequately address your previous concerns regarding the impact of embrittlement, or the loss of ductility, on the ability of baffle-former bolts to withstand loads?

A. No. [redacted] rather than outline corrective actions such as repair or replacement to address cracked bolts. [redacted]

[redacted] My concern is that [redacted], it would likely be insufficient to enable the bolts to sustain an energetic impulsive shock load, such as that associated with a DBA LOCA.

Q: Does Westinghouse address undetected cracks in bolts?

A: [redacted].

Q. How do Entergy and Westinghouse propose to address these undetected flaws and the risk that failed bolts pose to the integrity of the structure and core coolability?

A. [redacted]

[redacted]

[redacted] In my opinion, it is necessary to maintain a substantial margin throughout the PEO because the number of baffle-former bolts exceeding the stress and IASCC thresholds is expected to increase -- not decrease -- over time.

Q. Did the Bolting Analysis identify conditions where additional evaluation is required when a flaw is detected?

A. [redacted]

IV. <u>Entergy's Flaw Acceptance Criteria & the Eason and Pathania Paper</u>

Q. Dr. Lahey, as discussed above, another concern you raise relates to potential non-conservatisms in the flaw acceptance criteria for certain RVI structures and components, and particularly to the crack growth rate model used by Westinghouse in developing these acceptance criteria. How does

Westinghouse address crack growth in its flaw acceptance criteria?

A. As indicated in the Flaw Acceptance Criteria for IP-2 (NYS000584) and the Flaw Acceptance Criteria for IP-3 (NYS000585), for certain components, [redacted]

- [redacted]
- [redacted]
- [redacted]

• [redacted]

Q. I present to you a document marked as exhibit NYS000587, a paper by E. Eason and R. Pathania, entitled "Disposition Curves for Irradiation-Assisted Stress Corrosion Cracking of Austenitic Stainless Steels in Light Water Reactor Environments," PVP2015-4532, from the Proceedings of the ASME 2015 Pressure Vessels and Piping Conference ("Eason and Pathania Paper"). Have you reviewed this document?

A. Yes.

Q. Are you familiar with the authors of this paper?

A. I have never met them but I understand that Mr. Pathania, one of the co-authors, is a technical executive with the Electric Power Research Institute ("EPRI"). Additionally, he is EPRI's "Roadmap Owner" for its BWR and PWR irradiated materials testing and the degradation models for the reactor internals project. *See* http://mydocs.epri.com/docs/Portfolio/P2016/Roadmaps/NUC_MAT_01-BWR-PWR-Irradiated-Materials-Testing.pdf.

Q. In your opinion, what in the Eason and Pathania Paper is relevant to issues in this proceeding?

A. The Eason and Pathania Paper presents new IASCC crack growth rate disposition curves for both boiling water reactors ("BWRs") and PWRs. The paper summarizes the results of a multi-year international effort sponsored by EPRI to collect, rank, and model IASCC crack growth rate data. *See* Eason and Pathania Paper at 1. The paper summarizes over 800 IASCC crack growth rate data points collected from six laboratories worldwide, and reviewed and ranked by an international panel of known IASCC experts. *Id.* at 2.

According to the Eason and Pathania Paper, the new crack growth rate disposition curves presented in the paper reflect "an improvement" over the earlier BWRVIP-99-A and MRP-227-A disposition curves. *Id.* at 9. The earlier BWRVIP-99-A and MRP-227-A crack growth rate disposition curves were developed circa 2001 from then-available data, and were primarily used for estimating crack growth rates in BWR environments. *Id.* at 2. However, according to the Eason and Pathania Paper, substantially more data is now available, and is reflected in their paper. *Id.* at 2.

Q. What information regarding IASCC crack growth rate disposition curves does the Eason and Pathania Paper disclose?

A. Figure-3 of the Eason and Pathania Paper presents a PWR primary water crack growth rate disposition curve based on the newly available data. *Id.* at 4. The new curve is plotted at a temperature of 325°C and an irradiated yield stress of 700 MPa. *Id.* at 4. For comparison, Figure-3 also presents a plot of the older MRP-227-A disposition curve. *Id.* at 4.

The Eason and Pathania Paper states that the new PWR primary water crack growth rate disposition curve "is about a factor of 5.6 higher than the dashed MRP-227-A curve." *Id.* at 4. The Eason and Pathania Paper goes on to state that a higher irradiated yield stress of 970 MPa (rather than 700 MPa) would shift the new PWR primary water crack growth rate disposition curve in Figure-3 further upward by a factor of 2.3. *Id.* at 5.

Taken together, the Eason and Pathania Paper suggests that the PWR primary water crack growth rate disposition curve could increase the older MRP-227-A IASCC crack growth rate disposition curve by a factor of at least 10, depending on the operating temperatures and irradiated yield stress. *Id.* at 4-5, 9. Stated another way, the Eason and Pathania Paper indicates that IASCC cracks can grow up to an order of magnitude faster in the PWR primary water operating environment compared to earlier (mostly BWR) data that was used in the MRP-227-A disposition curves.

Q. How does this finding relate to Entergy's Amended and Revised RVI AMP?

A. The Eason and Pathania Paper suggests that the crack growth rates [redacted] under Entergy's Amended and Revised RVI AMP are non-conservative. As I explained above, the Eason and Pathania Paper indicates, based on over 800 IASCC crack growth rate data points, that IASCC crack growth rates in PWR primary water environments are between five and 10 times greater than those set forth in MRP-227-A. [redacted].

Q. To the extent that [redacted] IASCC crack growth rates that are non-conservative, what are the implications for the adequacy of Entergy's Amended and Revised RVI AMP?

A. In my opinion, the new IASCC crack growth rate disposition curves developed by EPRI, and discussed in the Eason and Pathania Paper, call into serious question the conservatism of Entergy's Amended and Revised RVI AMP. [redacted], the plan is non-conservative and fails to provide reasonable assurance that RVI component cracking due to aging degradation mechanisms such as IASCC will be adequately managed.

[redacted]

[redacted]

[redacted] I am concerned, however, that possible non-conservatism in these flaw acceptance criteria means that both identified and undetected cracks may grow faster than Entergy assumes but will not be identified during [redacted] follow-up inspections.

There are at least three specific circumstances under Entergy's Amended and Revised RVI AMP that, in my opinion, may result in the failure of RVIs due to the faster PWR IASCC crack growth rates identified in the Eason and Pathania Paper:

- First, a flaw or surface crack in an RVI system, structure, or component that is not detected during baseline visual inspections. This is a real possibility because any visual inspection technique has an inherent limit of detection, as I discussed above and in my prior testimony. Under Entergy's Amended and Revised RVI AMP, the next inspection will not occur for 10 years. Should the flaw or crack grow to critical size before this next inspection, the component may fail. In particular, given the results presented in the recent Eason and

Pathania Paper, it is possible, and even likely, that the flaw or surface crack will grow at a faster rate [redacted]

- Second, a new flaw or surface crack in an RVI system, structure, or component that develops after the baseline visual inspection. Again, because the next inspection will not occur for 10 years, I am concerned that the component may fail if the flaw or crack grows to critical size before the next inspection. In particular, it is possible, even likely, that such a crack will grow faster [redacted] in light of the accelerated PWR IASCC crack growth rate disposition curve presented in the Eason and Pathania Paper.

- Third, a flaw or surface crack that is detected by visual inspection and is indicated for Entergy's Corrective Action Program, [redacted]

[redacted]. I am concerned that such a crack may meet Entergy's and Westinghouse's calculated [redacted] allowable crack size set forth under the Flaw Acceptance Criteria for IP-2 and IP-3, but then grow faster than anticipated. In particular, towards the end of the PEO, when the fluence exposure of RVI systems, structures, and components will be in the range at which IASCC is a serious concern, it is possible that Entergy will not be monitoring or inspecting a developing crack at all. Additional inspections at the end of the PEO are important because the estimated fluence threshold for IASCC is 3 dpa, or 2×10^{21} n/cm^2 (E > 1 MeV). *See* MRP-191 at 3-3, Table 3-2 (NYS000321). For many RVI components that threshold may be reached in the latter part of the PEO, when no inspections are expected to take place. *Id.* at 4-22 to 4-29, Table 4-6. For example, with respect to girth welds, predicted neutron fluence is expected to be 4 dpa towards the end of the PEO. *See* USNRC Supplemental Safety Evaluation Report (SSER2) (NYS000507), at 3-46.

In addition, Entergy's use of [redacted] for developing its flaw acceptance criteria underscores my prevailing concern, as discussed above and in my previous testimony, that Entergy's Amended and Revised RVI AMP does not adequately protect against the failure of fatigued and embrittled components that experience significant shock loads (i.e., those due to a DBA LOCA). [redacted]. In my opinion, Entergy should, at a minimum, implement an AMP that takes into account the latest information on crack growth rates, apply appropriate DBA LOCA loads, and incorporate planned inspections in the years approaching the end of the PEO, when IASCC may become a very serious concern. Additionally, Entergy should implement shorter inspection intervals combined with regular, follow-up examinations to minimize the number of cracks that escape detection and to properly monitor crack growth.

Q. Do you have any other opinions with regard to the new PWR primary water crack growth rate disposition curve presented in the Eason and Pathania Paper?

A. Yes. As Entergy concedes, in the event that its LRA for the Indian Point reactors is granted, it will not be compelled to undertake future actions to incorporate the new crack growth rate curves for PWRs into its Amended and Revised RVI AMP. *See* Testimony of Entergy Witnesses Nelson F. Azevedo, Robert J. Dolansky, Alan B. Cox, Jack R. Strosnider, Timothy J. Griesbach, Randy G. Lott, and Mark A. Gray Regarding Contention NYS-25 at A136 (ENT000616). Rather, such actions by Entergy would be entirely voluntary. *Id.*

Q. Does Entergy's Amended and Revised RVI AMP and Inspection Plan incorporate margin into its flaw acceptance methodology?

A. [redacted]

[redacted].

I am also concerned that this possible non-conservatism in margin in Westinghouse's flaw acceptance criteria creates a source of uncertainty in addition to the non-conservatisms that I have discussed previously. According to Entergy, the ASME code's adjustment factors of 2 on stress and 20 on cycles in the fatigue design curves provide "substantial margin." *See* *Entergy Revised Testimony on NYS-26B/RK-TC-1B (ENT000679), 43 (A70). However, as explained in NUREG-6909, Revision 1, the factors of 2 on stress and 20 on cycles used in the ASME Code Section III air fatigue design curves were intended to cover the effects of variables that influence fatigue lives (i.e., material variability, different heats, surface finish, size, mean stress, and loading sequence) and are not per se safety margins. *See* "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials – Draft Report," NUREG-6909, Rev. 1 (NYS000490A), at XXV, 5. Because these factors were not investigated in the laboratory tests that provided the data for the curves, it is therefore not clear how much conservatism is actually embedded in the ASME code. *See* NYS000490B at 147-148.

Studies on vessel steels, piping and other components have shown that ASME Code Section III fatigue design procedures do not always contain large conservatisms and that cracks can initiate before they are predicted to occur. *Id.* In my opinion, Entergy's reliance on the "inherent" margins in the ASME code for fatigue evaluations is misplaced.

V. The IP-3 Inspection Report

Q. I now show you a document marked as exhibit NYS000588, which has the title, "USNRC Indian Point Nuclear Generating Unit-3 – License Renewal Inspection Report 05000286/2015011," and is dated November 19, 2015 ("IP-3 Inspection Report"). Can you describe this document?

A. The IP-3 Inspection Report appears to be a document authored by USNRC Staff that summarizes the Staff's view concerning the status of Entergy's commitments for IP-3. More specifically, the IP-3 Inspection Report sets forth Staff's conclusion that Entergy has fulfilled its remaining commitments with respect to metal fatigue (Commitment 49), but that its commitment to develop a plant-specific safety analysis for reactor vessel plate B2803-03 three years prior to reaching the pressurized thermal shock reference temperature (Commitment 32) still remains outstanding.

With respect to Commitment 49, which requires Entergy to recalculate cumulative usage factors for certain reactor vessel internals taking into account environmental effects, the IP-3 Inspection Report notes that initial application of the maximum stainless steel F_{en} of 15.348 to the CUF values for those components resulted in CUF_{en} values in excess of the limit of 1.0 for five components and locations (i.e., upper support plate assembly, upper support plate flange, lower core plate, lower core support plate, and lower support columns) for IP-2 and three components and locations (i.e., upper support assembly, instrumentation columns, and lower support columns) for IP-3. The report states that refined fatigue calculations were performed for these components and locations to qualify them for service. Significantly, the report summarizes the manner in which Entergy has systematically removed conservatisms in order to reduce the CUF_{en} values to below 1.0. The document also presents the USNRC Inspector's view that such refinement through reduction of conservatisms "obviates the need for further CUF re-analysis, and/or repair or replacement." *See* IP-3 Inspection Report (NYS000583), at 7. Clearly this iterative process is troubling since the so-called conservatisms which were eliminated may well be needed design margins.

VI. Conclusion

Q. Would you summarize your testimony?

A. In summary, it is my opinion that these new documents provide additional bases for the Board to conclude that Entergy's Amended and Revised RVI AMP for IP-2 and IP-3 fails to adequately manage the effects of aging, and does not provide reasonable assurance that the safety functions of these plants will be maintained during the period of extended operation.

Q. It that the end of your supplementary testimony today?

A. Yes it is. However, I reserve the right to supplement my testimony if new information is disclosed or introduced.

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.	March 4, 2016

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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)

R. T. Lahey

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