NYS000481
Submitted: June 9, 2015

UNITED STATES
NUCLEAR REGULATORY COMMISSION

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, Entergy Nuclear Indian Point 3, LLC, and Entergy Nuclear Operations, Inc.	DPR-26, DPR-64 June 9, 2015
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STATE OF NEW YORK
REVISED STATEMENT OF POSITION
CONTENTION NYS-25

Office of the Attorney General
for the State of New York
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Albany, New York 12224

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PRELIMINARY STATEMENT

In accordance with 10 C.F.R. § 2.1207(a)(1), the Atomic Safety and Licensing Board's ("Board") July 1, 2010 Scheduling Order,[1] the Board's December 9, 2014 Revised Scheduling Order,[2] and the Board's May 27, 2015 Order,[3] the State of New York (the "State") hereby submits its Revised Statement of Position on the State's admitted Contention 25 ("NYS-25"), as supplemented on September 15, 2010 and February 13, 2015, concerning the integrity of Indian Point's embrittled reactor pressure vessels and their internal components. Embrittlement of reactor pressure vessels and their internal components is one of the most important age-related phenomena that the U.S. Nuclear Regulatory Commission (NRC) must consider in its review of the license renewal application (LRA) of Entergy Nuclear Operations, Inc. ("Entergy" or "the Applicant") to extend the operating life of the two operating Indian Point reactors for an additional twenty years. Failure to consider the effects of embrittlement and the synergistic effects of embrittlement and other age-related degradation mechanisms could prevent the maintenance of a coolable core geometry which in turn, could result in a meltdown of the core, a release of radiation, and profound safety consequences for the State and its citizens.

[1] *Entergy Nuclear Operations, Inc.* (Indian Point, Units 2 and 3), Scheduling Order (July 1, 2010) (unpublished) (ML101820387).

[2] *Entergy Nuclear Operations, Inc.* (Indian Point, Units 2 and 3), Revised Scheduling Order (December 9, 2014) (unpublished) (ML14343A757).

[3] *Entergy Nuclear Operations, Inc.* (Indian Point, Units 2 and 3), Order (Granting New York's Motion for an Eight-Day Extension of the Filing Deadline) (May 27, 2015) (unpublished). The May 27, 2015 Order extended the deadline for the State and Riverkeeper to file their revised prefiled testimony, affidavits and exhibits from June 1, 2015 to June 9, 2015. *Id.*

In this proceeding, the State has satisfied the standards contained in 10 C.F.R. § 2.309 governing contention admissibility – standards that NRC and Entergy have described as "strict by design." The State, in support of its "Initial Statement of Position on Contention NYS-25," dated December 22, 2011, submitted the December 20, 2011 Report of Dr. Richard T. Lahey, Jr., Edward S. Hood Professor Emeritus of Engineering, Rensselaer Polytechnic Institute (Exh. NYS000296), the Supplemental Report by Dr. Lahey concerning Entergy's use of the WESTEMS computer code as a tool to analyze the cumulative fatigue condition of important reactor components (Exh. NYS000297), Dr. Lahey's prefiled written testimony (Exh. NYS000294), and supporting evidence. Thereafter, NRC Staff informed the Board that Staff could not present responsive testimony on the issue and the proceeding was essentially stayed for approximately three years to allow Staff to examine the issue. The State now submits Dr. Lahey's revised prefiled written testimony (Exh. NYS000482) and additional supporting evidence addressing various developments that have occurred since 2011. The State's exhibits show that Entergy's LRA should be denied because the application does not demonstrate that the reactor pressure vessel internals will remain functional during various design basis events as required by 10 C.F.R. §§ 50.54, 50.46, 54.3, 54.4, 54.21, 54.29, 54.33(a), and because the application does not provide an adequate program to monitor and manage the effects of aging degradation of the reactor pressure vessel internals as required by 10 C.F.R. §§ 54.21(a)(3), 54.21(c)(iii) and 54.29(a)(1).

BACKGROUND

A pressurized water nuclear reactor (PWR) is made up of many different systems, components and fittings. In turn, these systems components and fittings are made of many different types of materials. *See* Figure 1. PWRs 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 large steel container known as the reactor pressure vessel (RPV). The 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 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.

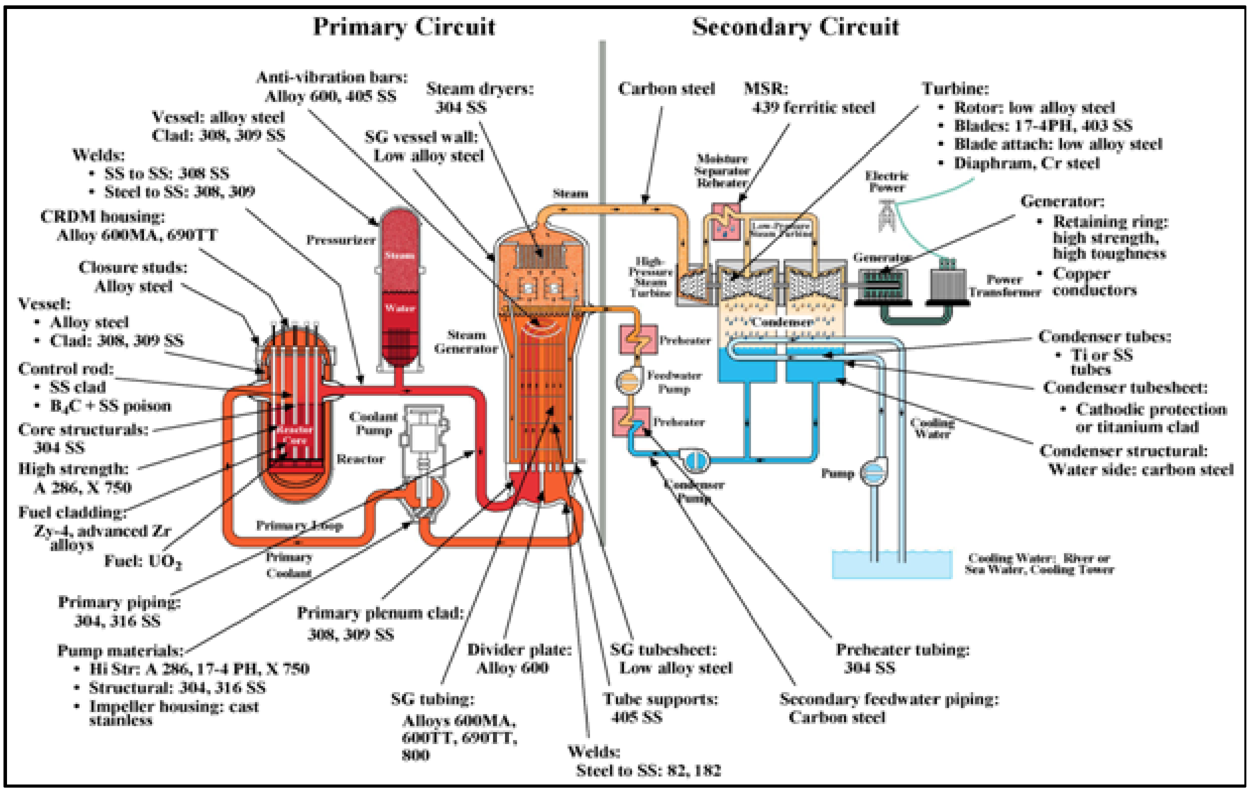

Figure 1. Overview of a PWR reactor, showing the various materials used in the reactor construction. Source: DOE, Light Water Reactor Sustainability Program: Materials Aging and Degradation Technical Program Plan, at 2, Figure 1 (August 2014) (Exh. NYS000485).

Nuclear reactor components need to function in a very harsh environment that includes extended time at high temperatures, as well as exposure to neutron irradiation, stress, vibrations, and a corrosive media. The many forms of age-related degradation are complex and vary depending on the location of the component, the material of the component, and the environment in which that component operates. *See* Figure 2.

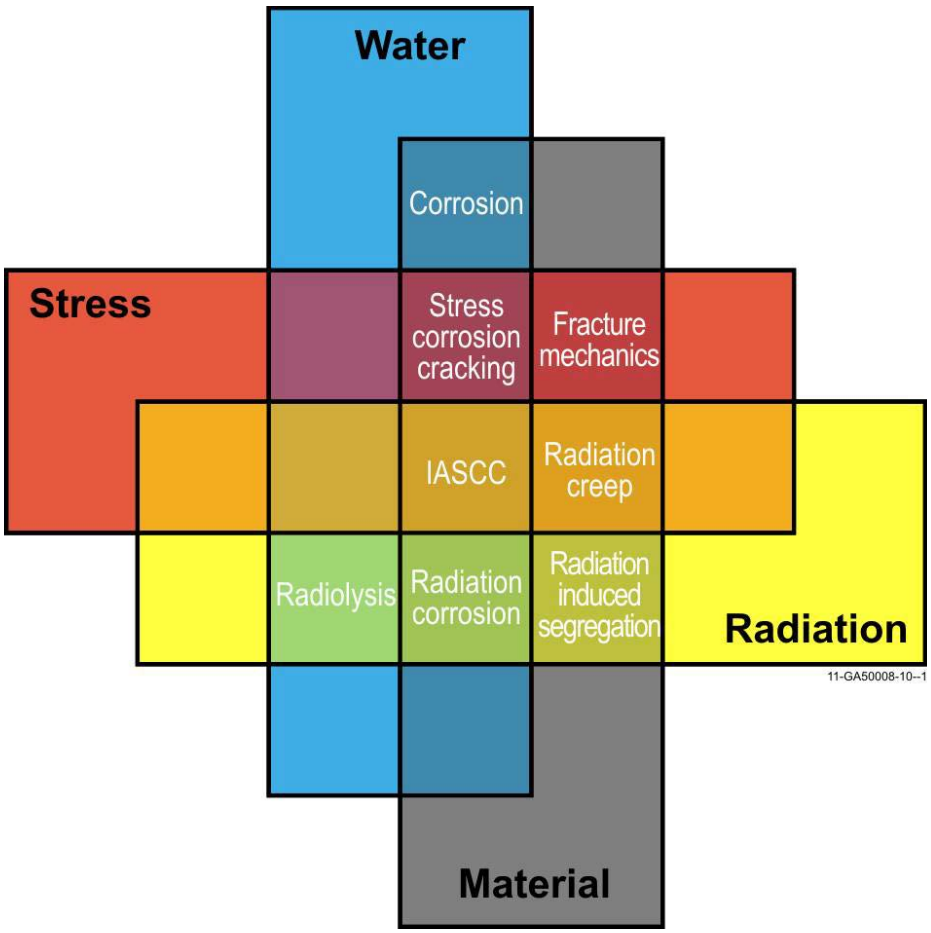

Figure 2. Demonstrative figure showing some, but not all (i.e. not explicitly including fatigue) of the interactions between degradation mechanisms in an LWR nuclear power plant. Source: DOE, Light Water Reactor Sustainability Program: Materials Aging and Degradation Technical Program Plan, at 5, Figure 2 (August 2014) (Exh. NYS000485).

In particular, a number of structures, components, and fittings are located within the RPV and are subjected to the extremely harsh conditions within the reactor core. These components, known as reactor pressure vessel internals (RPVIs or RVIs), include the core barrel (and its welds), core baffle, intermediate shells, former plates, lower core plate and support structures, clevis bolts, fuel alignment pins, thermal shield, the lower support column and mixer, upper mixing vanes, and the upper/lower core assemblies and support column, and the control rods and their associated guide tubes, plates, and welds. *See* Figure 3. RVIs also include the bolts that hold various components together or to other components including: the baffle-to-baffle bolts, the core barrel-to-former bolts, and baffle-to-former bolts as well as the welds or weldments that hold sections of these components together.

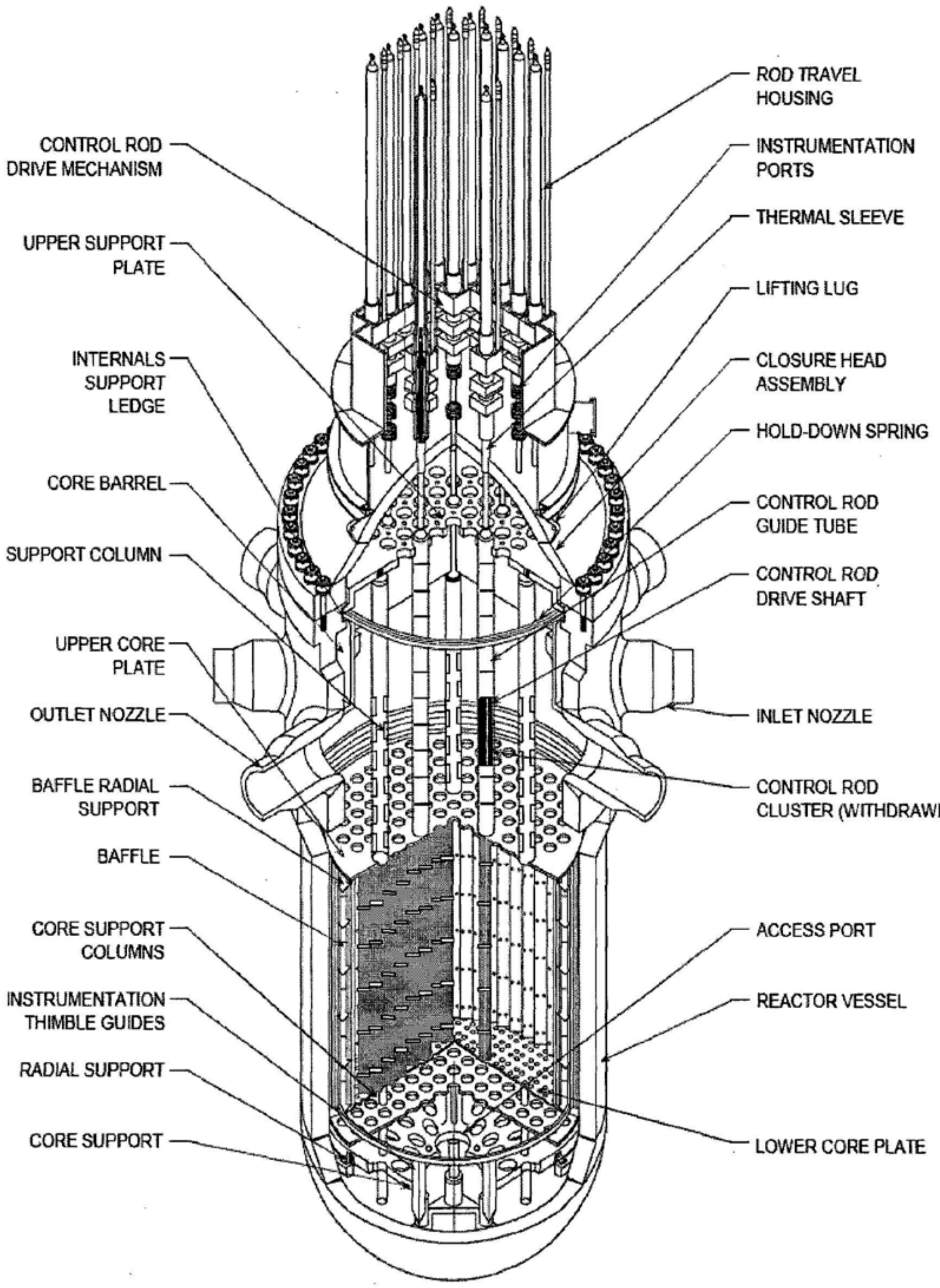

Figure 3. Overview of typical Westinghouse internals. Source: Dacimo, Fred, Entergy, letter to NRC, NL-12-037, Attachment 2, at 3 (Feb. 17, 2012) (Exh. NYS000496).

Entergy operates two Westinghouse-designed PWRs at the Indian Point site in Buchanan, New York, roughly 24 miles north of New York City. The two operating reactors are known as Indian Point Unit 2 (IP2) and Indian Point Unit 3 (IP3).[4] The Indian Point reactors are among the older operating nuclear reactors in the United States. IP2 reached the end of its initial 40-year operating license on September 28, 2013, and IP3 will reach the end of its initial operating license on December 12, 2015. In 2007, Entergy submitted an LRA seeking permission to operate both Indian Point reactors for an additional 20 years, which would make them among the first nuclear reactors to operate out to 60 years.

PROCEDURAL HISTORY

I. The State of New York's Petition to Intervene and Contention NYS-25

On November 30, 2007, the State submitted a Petition to Intervene (NYS Petition), which included proposed contentions regarding critical deficiencies in Entergy's LRA with respect to public safety, health, and the environment. State of New York Notice of Intention to Participate and Petition to Intervene (Nov. 30, 2007) (ML073400187). Among those proposed contentions was Contention NYS-25, challenging Entergy's approach to the embrittlement of the reactor pressure vessel and their internal components. NYS Petition, at 223-27. NYS-25, asserted:

> Entergy's License Renewal Application Does Not Include an Adequate Plan to Monitor and Manage the Effects of Aging Due to Embrittlement of the Reactor Pressure Vessels ("RPVs") and the Associated Internals.

NYS Petition, at 223. Contention NYS-25 alleges that the LRA does not include an adequate plan to monitor and manage the effects of aging due to embrittlement of the RPVs and the

[4] A third nuclear reactor, Indian Point Unit 1, is owned by Entergy at the site but is not operational.

associated internals as required by 10 C.F.R. § 54.21(a), and does not comply with 10 C.F.R. § 54.21(c). *Id*. The State further contends that the LRA does not establish "that Entergy performed any age-related accident analyses, or that it took embrittlement into account when it assessed the effect of transient loads or the loss of coolant accident ("LOCA")." NYS Petition, at 224.

Dr. Richard Lahey, a Professor of Engineering at Rensselaer Polytechnic Institute with extensive experience in the design, operations, safety, and aging of nuclear power plants, submitted a declaration in support of various contentions presented by the State including NYS-25. Declaration of Dr. Richard T. Lahey, Jr., at ¶¶ 6-18 (Nov. 2007) (*included in* ML073400193) (Exh. NYS000298). Dr. Lahey stated that embrittlement of the RPVs and their associated internal components is one of the most important age-related phenomena, and that failure to carefully consider the effects of embrittlement could result in a meltdown of the core. *Id*. at ¶¶ 6, 9. Furthermore, Entergy failed to document any experiments or analysis in its LRA to show that the embrittled RPV internal structures would not fail and that a coolable core geometry will be maintained subsequent to a Design Basis Accident LOCA. *Id*. at ¶¶ 14-16. According to Dr. Lahey "[t]his is a serious and unacceptable omission by Entergy because embrittled structures are known not to tolerate shock loads well." *Id*. at ¶ 16.

Entergy and NRC Staff opposed the admission of the State's contention.[5] Among other things, Entergy argued that the Board should not admit NYS-25 because "[t]he core barrel,

[5] Answer of Entergy Nuclear Operations Inc. Opposing New York State Notice of Intention to Participate and Petition to Intervene (Jan. 22, 2008) (ML080300149); NRC Staff's Response to Petitions for Leave to Intervene (Jan. 22, 2008) (ML080230543).

thermal shield, baffle plates and baffle former plates (including bolts) are, however, made of stainless steel and are not susceptible to a decrease in fracture toughness as a result of neutron embrittlement." Entergy Answer, at 137. On February 22, 2008, the State submitted a reply in further support of Contention NYS-25.[6] On March 11, 2008, the Board heard oral argument concerning the admission of Contention NYS-25.[7]

On July 31, 2008, the Board issued a memorandum and order rejecting Entergy's and NRC Staff's arguments and admitting contention NYS-25. *Entergy Nuclear Operations, Inc.* (Indian Point, Units 2 and 3), LBP-08-13, 68 N.R.C. 43 (July 31, 2008) (ML082130436). The decision reviewed the State's contention and Dr. Lahey's supporting declaration. 68 N.R.C. at 129-131. The Board recognized Dr. Lahey's opinion that components in the Indian Point reactors have serious embrittlement issues that are not adequately addressed in Entergy's LRA. *Id.* at 131. The Board noted:

> Dr. Lahey states that Entergy fails to document in its LRA "any experiments or analysis to justify that the embrittled RPV internal structures will not fail and that a coolable core geometry will be maintained subsequent to a [Design Basis Accident] LOCA." According to Dr. Lahey "[t]his is a serious and unacceptable omission by Entergy because embrittled structures are known not to tolerate shock loads well."

Id. (citations omitted).

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[6] State of New York Reply in Support of Petition to Intervene (Feb. 22, 2008) (ML080600444).

[7] Transcript of Proceedings, at 401-410 (Mar. 11, 2008) (ML080740257). The parties submitted an errata sheet, which the Board subsequently accepted, for the transcript of the three-day oral argument.

II. Entergy's Revised LRA and the State's Additional Bases Regarding Reactor Vessel Internals

On July 15, 2010, Entergy submitted an amendment to the LRA to the Board. *See* Entergy Letter to ASLB enclosing Entergy Communication NL-10-063 (LRA Amendment No. 9) (ML102030120) (Exh. NYS000313). NL-10-063 forwarded to NRC Staff a document entitled "Reactor Vessel Internals Program" (or "RVI Program"). Entergy stated that "[t]he RVI Program will implement the [Electric Power Research Institute's (EPRI)] Pressurized Water Reactor Internals Inspection and Evaluation Guidelines, MRP-227." NL-10-063, Attachment 1, at 8 (Exh. NYS000313). On September 15, 2010, the State submitted a motion for leave to file additional bases to Contention NYS-25. State of New York's Motion for Leave to File Additional Bases for Previously-Admitted Contention NYS-25, etc. (Sept. 15, 2010) (ML103050402). Dr. Lahey submitted a declaration in support of the additional bases, identifying concerns with Entergy's NL-10-063 RVI Program. Declaration of Richard T. Lahey, Jr. (Sept. 15. 2010) (included in ML103050402) (Exh. NYS000301). Among other things, that declaration discussed the synergistic effects of embrittlement and fatigue. Lahey Sept. 15, 2010 Decl. at ¶¶ 13, 14, 15. In the September 15, 2010 declaration, Dr. Lahey referenced another declaration that he had then recently submitted in connection with Contention NYS-26B. *Id*. at ¶ 13 (referencing Declaration of Richard T. Lahey, Jr. [Sept. 8, 2010], included in ML102670665 [Redacted, Public Version] [Exh. NYS000300]). There, Dr. Lahey noted that in-core fatigue failures of irradiated baffle-to-former bolts had been observed in operating PWRs. Sept. 8, 2010 Lahey Decl. at ¶ 12. The State also raised concerns over the examination techniques. September

2010 Additional Bases for Previously-Admitted Contention NYS-25, ¶ 3.4 (included in ML102670665).

Entergy and NRC Staff opposed the admission of the State's proposed additional bases.[8] On October 22, 2010, the State submitted a reply in further support of the additional bases for Contention NYS-25.[9]

On April 29, 2011, the State submitted to the Board additional information that reflected concerns by NRC Staff about deficiencies in the visual and remote examination techniques that Entergy and industry had proposed to employ as part of the aging management program for the embrittlement of reactor internals. Letter from AAG J. Sipos to ASLB (Apr. 29, 2011) (ML11133A288) (Exh. NYS000370) (enclosing two memoranda documenting non-concurrences by NRC Staff members about a safety evaluation of EPRI MRP-227). In those documents, two NRC Staff members recognized that two types of examination methodologies (EVT-1 and UT) were superior to an alternative examination methodology (VT-3) recommended by EPRI in MRP-227-Rev. 0. Tregoning, R.L., "Reasons for Non-concurrence on Draft Safety Evaluation for the EPRI Topical Report (TR) Materials Reliability Program (MRP) Report 1016596 (MRP-227), Revision 0, Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines" (March 22, 2011) (Exh. NYS000508); Case, Michael J., Comments for the

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[8] NRC Staff's Answer to State of New York's Motion for Leave to File Additional Bases for Previously-Admitted Contention NYS-25 (Oct. 12, 2010) (ML102850764); Applicant's Answer to Amended Contention NYS 25 Concerning Aging Management of Reactor Pressure Vessel Internals (Oct. 12, 2010) (ML103010104).

[9] State of New York's Joint Reply to Entergy and NRC Staff's Separate Answers to the State's Additional Bases for Previously Admitted Contention NYS-25 (Oct. 22, 2010). (ML103000060).

Document Sponsor to Consider Pertaining to Non-concurrence on Draft Safety Evaluation for EPRI Topical Report (TR) Materials Reliability Program (MRP) Report 1016596 (MRP-227), Revision 0, Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines, RES/DE (March 22, 2011) (Exh. NYS000509).

On July 6, 2011, the Board issued a memorandum and order that, among other things, admitted the State's amended bases for Contention NYS-25 and rejected Entergy's and NRC Staff's arguments opposing the State's submission. *Entergy Nuclear Operations, Inc.* (Indian Point, Units 2 and 3), Memorandum and Order Ruling on Pending Motions for Leave to File New and Amended Contentions (July 6, 2011) (unpublished) (ML111870344).

III. MPR-227-A and NRC Staff's Supplemental Safety Evaluation Report

On July 27, 2011, NRC Staff notified the Board that the Staff had completed a Safety Evaluation Report on EPRI's MRP-227 document. Letter from NRC Counsel S. Turk to ASLB, (July 27, 2011) (ML11208C309) (*enclosing* NRC RIS 2011-07, License Renewal Submittal Information for Pressurized Water Reactor Internals Aging Management [ML111990086] [Exh. NYS000310]). RIS 2011-07 stated in part, "LRAs for pressurized water reactor (PWR) plants have identified that an aging management program (AMP) is needed to manage the effects of aging for reactor vessel internal (RVI) components that are within the scope of license renewal, in accordance with 10 CFR Part 54, 'Requirements for Renewal of Operating Licenses for Nuclear Power Plants.'"[10] The version of MRP-227 approved by NRC Staff and amended or

[10] In June 2011, Staff prepared a Safety Evaluation of EPRI's MRP-227-Rev 0. (Exh. NYS000309). EPRI had previously withdrawn a confidentiality designation for MRP-227. EPRI MRP 2010-016 letter (Mar. 2, 2010) (Exh. NYS000308).

revised to respond to NRC Staff's concerns was designated MRP-227-A. EPRI, Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A) (ML120170453) (Exhibit NRC000114-A-F) (submitted January 2012).

On August 31, 2011, NRC Staff informed the Board and the parties that Staff had issued a supplement to the Safety Evaluation Report concerning the application to renew the operating licenses for IP2 and IP3. Letter from NRC Staff Counsel S. Turk to ASLB (ML11243A109) (enclosing NUREG-1930, Supplement 1 [SSER] [Exh. NYS000160]).[11]

As noted above and in accordance with the Board's scheduling orders, on December 22, 2011, the State submitted pre-filed testimony, exhibits, and a statement of position in support of Contention NYS-25.

IV. Entergy's Revised and Amended RVI Plan and NRC Staff's SSER2

On January 27, 2012, NRC Staff informed the Board, the State of New York, and other participants in the proceeding that Entergy planned to submit additional information regarding its RVI Program, that the Staff expected that it would need to ask questions about the proposal, and that Staff's review of this matter might be the subject of a second Supplement to the Safety Evaluation Report for the Indian Point LRA. NRC Staff letter to ASLB from Sherwin Turk, Staff Counsel (Jan. 27, 2012) (ML12027A115). On February 8, 2012, NRC Staff informed the Board and parties that it would not be able to state a position on Contention NYS-25 until it received and reviewed the additional information that Entergy planned to submit concerning Entergy's proposed RVI Program and Inspection Plan. NRC Staff's Statement in Response to

[11] The State and Riverkeeper filed a Joint Contention NYS-38/RK-TC-5 following the issuance of the SSER.

the ALSB's Order of February 3, 2012 (Feb. 8, 2012) (ML12039A298). On February 16, 2012, based on NRC Staff's reporting of "unresolved safety issues" relating to contention NYS-25 and the "dynamic nature of the NRC Staff's uncompleted safety reviews," the Board placed contention NYS-25 onto the second hearing track that already included NYS-38/RK-TC-5 and RK-EC-8, which had been placed in abeyance. *Entergy Nuclear Operations, Inc.* (Indian Point Nuclear Generating Units 2 and 3), Order Granting NRC Staff's Unopposed Time Extension Motion and Directing Filing of Status Updates (February 16, 2012), at 2 (unpublished) (ML12047A308).

On February 17, 2012, Entergy once again amended its LRA to include a "Revised RVI Program and Inspection Plan" based on EPRI's newly revised MRP-227-A. Dacimo, Fred, Entergy, letter to Document Control Desk, NRC, NL-12-037 (ML12060A312) (Exh. NYS000496). Over the next 32 months, Entergy and NRC Staff engaged in a series of communications regarding Entergy's revised RVI Program and Inspection Plan, which resulted in various modifications, amendments, clarifications and additional commitments. Exhs. NYS000497 to NYS000506. In November 2014, NRC Staff released Supplement 2 to its Safety Evaluation Report (SSER2), which discussed Staff's updated review and acceptance of Entergy's amended and modified aging management plan for IP2 and IP3. NRC, NUREG-1930, "Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3, Supplement 2" (November 2014) (Exhibit NYS000507). Taken together, Entergy's February 2012 "Revised RVI Program and Inspection Plan," as modified and amended by the various subsequent communications with NRC Staff and approved by NRC Staff in SSER2, will be referred to as the "Amended and Revised RVI Program."

V. <u>The State's February 2015 Supplement to Contention NYS-25</u>

On February 13, 2015, the State moved to supplement Contention NYS-25 in response to Entergy's Amended and Revised RVI Program and NRC Staff's SSER2. State of New York's Motion to Supplement Previously-Admitted Contention NYS-25 (February 13, 2015) (ML15044A493); New York State February 2015 Supplement to Previously-Admitted Contention NYS-25 (February 13, 2015) (ML15044A491). The State contended that Entergy's plan for managing aging of RVI set forth in the Amended and Revised RVI Program was inadequate. The Motion to Supplement was supported by a Declaration from Dr. Lahey describing his continuing concerns with Entergy's Revised and Amended RVI Program, and providing references to a variety of recently published documents supporting his position. Declaration of Richard T. Lahey (February 13, 2015) (ML15044A499 [redacted version]) (Exh. NYS000483).

Entergy opposed the State's Motion to Supplement. Entergy's Consolidated Answer Opposing Intervenors' Motions to Amend Contentions NYS-25 and NYS-38/RK-TC-5 (March 10, 2015) (ML15069A677). NRC Staff did not oppose the Motion to Supplement with respect to Contention NYS-25, except in two limited respects. NRC Staff's Answer to (1) State of New York's Motion to Supplement Contention NYS-25, and (2) State of New York and Riverkeeper Inc.'s Joint Motion to Supplement Contention NYS-38/RK-TC-5 (March 10, 2015) (ML15069A590). The State submitted a reply in further support of the admission of the February 2015 Supplement to Contention NYS-25. State of New York Reply in Support of Admission of the February 2015 Supplement to Previously-Admitted Contention NYS-25 (March 17, 2015) (ML15076A574). The Board admitted the State's February 2015 supplement to Contention NYS-25 in its entirety, rejecting Entergy's and NRC Staff's arguments to the contrary. *Entergy Nuclear Operations, Inc.* (Indian Point Nuclear Generating Units 2 and 3), Memorandum and Order (Granting Motions for Leave to File Amendments to Contentions NYS-25 and NYS-38/RK-TC-5) (March 31, 2015) (unpublished) (ML15090A771 [redacted version]).

SUMMARY OF ARGUMENT

Contention NYS-25 is based on several applicable regulations. First, 10 C.F.R. § 54.4(a)(1) requires that the applicant have an aging management plan ("AMP") that will ensure the following functions:

> (i) The integrity of the reactor coolant pressure boundary;
>
> (ii) The capability to shut down the reactor and maintain it in a safe shutdown condition; [and]
>
> (iii) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those referred to in § 50.34(a)(1), §50.67(b)(2), or § 100.11 of this chapter, as applicable;

Second, 10 C.F.R. § 54.21(a)(3) requires that the applicant "demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis ("CLB") for the period of extended operation." Third, 10 C.F.R. § 54.21(c) (iii) specifically requires that "the applicant shall demonstrate that . . . (iii) [t]he effects of aging on the intended function(s) will be adequately managed for the period of extended operation." Entergy has failed to make the required demonstrations and, thus, the Board cannot find, pursuant to 10 C.F.R. § 54.29(a)(1), that:

> there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the CLB . . . [including] managing the effects of aging during the

> period of extended operation on the functionality of structures and components that have been identified to require review under §54.21(a)(1).

The focus of Contention NYS-25 is Entergy's deficient AMP for RPVIs. The principal support for Contention NYS-25 is the report, declarations, and prefiled testimony of Dr. Richard T. Lahey, Jr., the Edward E. Hood Professor Emeritus of Engineering at Rensselaer Polytechnic Institute (RPI) in Troy, New York, and numerous documents, all of which demonstrate that: (1) Entergy's AMP for RPVIs is not based on an analysis that addresses the critical issue of the synergistic degradation of RPVIs caused by the combination of embrittlement, metal fatigue, irradiation-assisted stress corrosion cracking ("IASCC"), and primary water stress corrosion cracking ("PWSCC"); (2) Entergy's analysis fails to adequately consider the full range of transient shock loads (thermal and decompression) to which RPVIs will be subjected in the event of various postulated accidents, such as a design basis accident ("DBA"), and thus fails to develop a plan which considers those shock loads, and their resultant impact on core coolability, in setting either inspection, acceptance or corrective action criteria; (3) the AMP does not include a commitment to take preventative actions or to implement corrective actions, nor does it provide specific, enforceable acceptance criteria for some components; and (4) the AMP relies on fatigue predictions which are non-conservative and may not accurately predict fatigue-induced component failures. Report of Dr. Richard T. Lahey, Jr. (Exh. NYS000296); February 2015 Lahey Declaration (Exh. NYS000483); Revised Lahey PFT (Exh. NYS000482). The State also relies on Dr. Lahey's previous declarations that have been filed in this proceeding, as well as NRC, Entergy, and industry documents (Exhs. NYS000300, 301).

Dr. Lahey demonstrates the importance of RPVIs and explains how the failure of RPVIs during a DBA may result in the loss of a coolable geometry and thus core meltdown and a catastrophic release of radiation to the environment. Revised Lahey PFT, at 16, 25-27. Many of the RPVIs hold critical reactor components in place and their failure, during an accident, can cause those components, such as baffles, former plates, core support plates, thermal shields, control rods, guide tubes, plates and welds to either deform and/or create core blockages within the RPV pressure boundary. *See* Lahey Report, at ¶¶ 12-13, 16-28, 38-40 (NYS000296).

Entergy's response to Contention NYS-25 has evolved in an iterative process from the initial LRA in 2007. The initial LRA contained no AMP for RPVIs. After the State submitted its Petition including Contention NYS-25, Entergy submitted a proposal that mostly included adoption of EPRI's MRP-227 inspection plan (Exh. NYS00307A-D), with a representation that, in the future, Entergy would endeavor to develop some modifications to MRP-227 required to comply with concerns raised by the NRC. *See* LRA at 3.1.2.1.2 (which does not list an AMP for RPVIs); NL-10-063, Attachment 1 at 7 (which includes an AMP for RPVIs) (Exh. NYS000313); NL-11-107 Attachment 1 (which purports to include the inspection plan for RPVIs) (Exh. NYS000314); and NL-11-101, Attachment 1 at 4 ("This inspection plan will include the inspections specified in MRP-227, as modified by the conditions and limitations and applicant/licensee action items in the NRC SER on MRP-227, Revision 0. . . . Following issuance of MRP-227-A, Entergy will review the inspection plan to determine the need for revision, and will modify the inspection plan to include the necessary revisions, if any"). Thereafter, following the State's 2010 supplement to Contention NYS-25, Entergy submitted the Revised and Amended RVI Plan, which principally relies on the inspection program set forth in MRP-227-A (Exh. NRC000114-A-F; *see* Exhs. NYS000496 to NYS000506).

Entergy's evolving position fails to address the problem of embrittlement of RPVIs and lacks analyses of some of the most important aspects of RPVI degradation, including the absence of an adequate assessment of the synergistic effects of embrittlement, metal fatigue and stress corrosion cracking and the absence of specific criteria for acceptance, prevention and corrective actions. Entergy's inspection-based approach to RVI management fails to address the possibility that highly embrittled and fatigued RVI components, which do not show detectable signs of aging such as cracks, will fail when subjected to a sudden shock. Additionally, Entergy's assessment of environmental assisted fatigue values for various components is deeply flawed, as it fails to account for embrittlement, relies on a systematic removal of conservatisms and fails to include any assessment of its potential error range. Even using this flawed methodology, the fatigue values for some components approaches unity. Entergy's systematic erosion of safety margins fails to ensure that unanticipated non-conservatisms inherent in all predictive models will not be detected through catastrophic component failure. As the Commission has held, when it comes to reactor safety requirements, "[w]e do not simply take the applicant at its word." *Entergy Nuclear Vt. Yankee, LLC* (Vt. Yankee Nuclear Power Station), CLI-10-17, 72 N.R.C. 1, slip op. at 45 (July 8, 2010).

RELEVANT FACTS

The RPVI AMP that is currently before the Board for review consists of the Revised and Amended RVI Plan, developed between 2012 and 2014 and approved by NRC Staff in the SSER2. Thus, the adequacy of the AMP for RPVIs must stand or fall on the adequacy of these documents. With regard to the issues that are relevant here the following facts are established by Dr. Lahey's Report (Exh. NYS000296), his Revised Prefiled Direct Testimony (Exh.

NYS000482), his February 2015 and other declarations, and numerous exhibits:

1. Neutron fluence is a degradation phenomenom that causes embrittlement of metal components, that will increasingly affect important components at Indian Point if the two facilities operate beyond their original 40-year expected life. Lahey Report at ¶17 (Exh. NYS000296). Embrittlement is just one of a variety of degradation mechanisms that operate within the harsh environment of the RPV interior. February 2015 Lahey Declaration at ¶10 (Exh. NYS000483); *see* Figure 3, *supra*;

2. RPVIs are more susceptible to embrittlement than the RPV, for which Entergy has already had to seek an exemption from ASME End of Life Charpy impact Upper Shelf Energy limits. Lahey Report at ¶¶ 19-20 (Exh. NYS000296). While NRC regulations establish that an RPV may experience reduced fracture toughness due to irradiation embrittlement at a fluence level of 1 X 10^{17} n/cm^2 over the life of the component, 10 C.F.R. Part 50, Appendices G & H; RIS 2014-11 (Exh. NYS000494), the RVI components can experience fluence at a level several orders of magnitude (i.e., 10^4 [ten-thousand] to 10^6 [a million] times) greater, in the range of 1 X 10^{21} to 1 x 10^{23} n/cm^2 by the end of their period of extended operation. Lahey PFT at 27 (Exh. NYS000482); LWRS (August 2014), at 18 (Exh. NYS000485); *see* Figure 4.

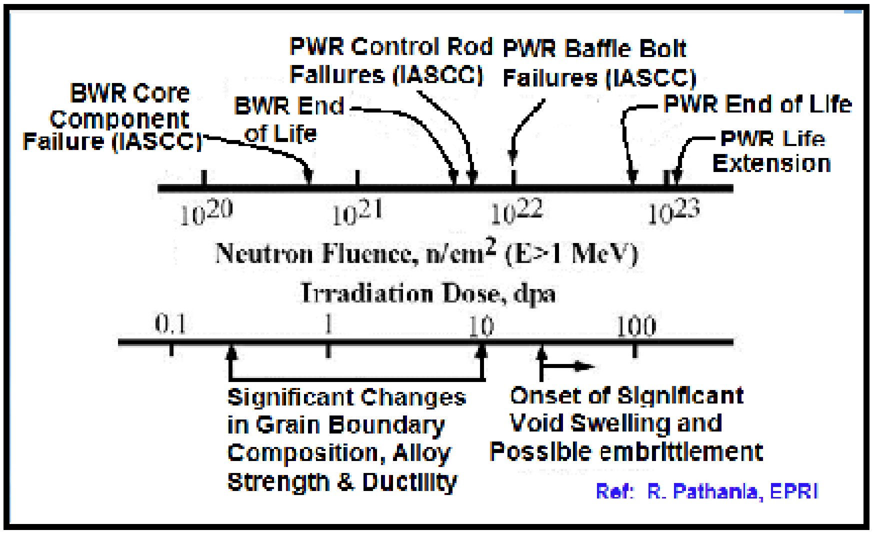

Figure 4. Logarithmic chart depicting expected fluence experienced by typical RPV internals, as well as expected onset of various material degradation effects. Source: NRC, Slides, "Irradiation Assisted Degradation of LWR Core Internal Materials: Brief Overview, Presented at Office of Nuclear Regulatory Research Seminar (April 14, 2015), at 3 (Exh. NYS000495).

3. As the RPVIs age, embrittlement causes the nil ductility temperature (NDT) to rise, thus expanding the temperature range at which RPVIs lose ductility. Revised Lahey PFT, at 24 (Exh. NYS000482); Lahey Report at ¶ 18 (Exh. NYS000296);

4. Embrittled components are more prone to crack propagation. Lahey Report at ¶ 19 (Exh. NYS000296); Revised Lahey PFT, at 18-19, 22-23; NRC Letter, Grimes to Newton, at 16 (Feb. 10, 2001) (Exh. NYS000324); Westinghouse Owners Group WCAP-14577 Rev. 1-A Report, at 3-2 (March 2001) (Exh. NYS00307A-D); Stevens, Gary L., Presentation to the ACRS on "Technical Brief on Regulatory Guidance for Evaluating the Effects of Light Water Reactor Coolant Environments in Fatigue Analyses of Metal Components" (December 2, 2014), at 56-58 (Exh. NYS000486); Chopra, O.K., "Degradation of LWR Core Internal Materials due to Neutron irradiation,"

NUREG/CR-7027 (Dec. 2010) (Exh. NYS000487);

5. The control rod drives and their associated guide tubes, plates, pins, and welds have been identified as particularly susceptible to stress corrosion cracking, which cracking is made more severe by embrittlement. Lahey Report at ¶¶ 25-27 (Exh. NYS000296);

6. There are factors other than embrittlement degrading the structural integrity of RPVIs, including metal fatigue and stress corrosion cracking. Lahey Report at ¶¶ 17 and 19 (Exh. NYS000296); *see* Figure 3, *supra*.

7. The exact manner and extent to which various age-related degradation mechanisms – including embrittlement and fatigue – interact synergistically is largely unknown, and is currently the subject of numerous large-scale research and development efforts being conducted by the NRC, the Department of Energy (DOE), and various national labs. Revised Lahey PFT, at 17-22 (Exh. NYS000482); February 2015 Lahey Declaration, at ¶¶10-15 (Exh. NYS000483); *see generally* Exhs. NYS000484A-B, NYS000485.

8. In the event of a DBA LOCA, seismic event, or other shock load, RPVIs can be subjected to substantial external forces. Lahey Report at ¶ 23 (Exh. NYS000296); Lahey Revised PFT, at 25-26 (Exh. NYS000482).

9. Maintaining the integrity of RPVIs following postulated accidents, such as a DBA LOCA or seismic event, is essential to maintaining a coolable core geometry. Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3, NUREG-1930 ("SER"), Vol. 1, at 2-41 (November 2009)

(NYS000326A-F) ("if certain reactor vessel internals failed, they could potentially inhibit core coolability during an accident"); Lahey Report at ¶23 (Exh. NYS000296).

10. Despite a series of iterative AMPs for the RPVIs, Entergy has failed to submit analysis in support of its AMPs that addresses the synergistic effects of embrittlement, metal fatigue, stress corrosion, and unique RPVI material on the degradation of RPVIs during the period of extended operation. Lahey Report at ¶ 16 (Exh. NYS000296);

11. In NL-10-063 Entergy stated that its RPVI AMP consisted of its agreement to follow MRP-227 and, when available and if applicable, the modifications to MRP-227 identified in MRP-227-A (Exh. NYS000313);

12. MRP-227 purports to be an internals inspection and evaluation guide for PWR RPVIs that requires individual plant development to create a true plant-specific AMP (MRP-227 at v [Exh. NYS000307A-D]). MRP-227-A is the version of MRP-227 approved by NRC Staff and amended or modified to respond to NRC Staff comments. Exh. NRC000114-A-F. MRP-227-A continues to rely on the inspection of RPVIs to detect signs of aging;

13. Entergy, in NL-11-107 provided a plant-specific AMP for RPVIs (Exh. NYS000314);

14. The 2011 plant-specific AMP for Indian Point's RPVIs contained numerous instances in which no actual acceptance criteria were provided or the criteria were wholly subjective. *See, e.g.*, NL-11-107, Attachment 1, Physical Measurements Examination Acceptance Criteria , at 25 (Exh. NYS000314) ("Specific acceptance

criteria will be developed as required, and thus are not provided generically in this plan"); Control Rod Guide Tube Assembly, *id*. at 47 ("The specific relevant condition is wear that could lead to loss of control rod alignment and impede control assembly insertion"); Baffle Former Assembly (Baffle Former Bolts), *id*. at 49 ("The examination acceptance criteria for the UT of the baffle-former bolts shall be established as part of the examination technical justification"), Thermal Shield Assembly, *id*. at 51 (The specific relevant conditions for thermal shield flexures are excessive wear, fracture, or complete separation"). MRP-227 is similarly vague when it comes to enforceable and objective standards for corrective actions and acceptance criteria. MRP-227 is replete with phrases such as "should," "could," "may," and "assumed." These terms render MRP-227, which is itself only a guideline, unenforceable. *See, e.g.*, MRP-227 at 6-9; 6-1 (Exh. NYS000307A-D) ("various options … are available for the disposition of conditions detected during examinations … that are unable to satisfy the examination acceptance criteria"). *See also* Lahey Report at ¶ 16 (Exh. NYS000296).

15. Entergy's 2011 AMP did not specify what actions would take place when inspections revealed that acceptance criteria had not been achieved. For example, under Examination Acceptance Criteria for Visual (VT-3) Examination, Entergy merely stated, "The disposition can include a supplementary examination to further characterize the relevant condition, an engineering evaluation to show that the component is capable of continued operation with a known relevant condition, or repair/replacement to remediate the relevant condition." NL-11-107 at 23 (Exh. NYS000314).

16. Entergy's 2011 AMP did not include any preventative action measures.

NL-10-063, Attachment 1 at 86 (Exh. NYS000313) ("The Reactor Vessel Internals Program is a condition monitoring program that does not include preventive actions").

17. Although Entergy agreed to do some baseline inspections for the baffle former assembly, the timing of those inspections is potentially well beyond the date on which extended operation would commence, NL-11-107, Attachment 1 at 36-38 (Exh. NYS000314), indicating that baseline inspections will occur in a range of 20-40 EFPY.

18. Without timely and properly conducted baseline inspections of RPVIs, it is not possible to accurately measure the progression of degradation of components over the period of extended operation and to take timely action to prevent failure of the component. Lahey Report at ¶¶ 28-29 (Exh. NYS000296).

19. Entergy's 2011 AMP for RPVIs was inadequate with respect to the embrittlement of the control rod drives and their associated guide tubes, plates, pins, and welds. Lahey Report at ¶¶ 25-27, n. 5 (Exh. NYS000296); MRP-227 at v (Exh. NYS000307A-D);

20. Entergy's 2011 AMP for RPVIs was not based upon an analysis that considers the combined degradation effects of embrittlement, metal fatigue and stress corrosion cracking in determining the frequency of inspections, the criteria for acceptance of components after inspections, the criteria for when to expand inspections, the criteria for preventative actions and the criteria for when and how to repair or replace RPVI components. Lahey Report at ¶ 16 (Exh. NYS000296);

21. In February 2012, Entergy submitted a "Revised Reactor Vessel Internals Program and Inspection Plan" to NRC, which revised its AMP for RPVI components.

NL-12-037 (Exh. NYS000496). In the Revised Plan, Entergy acknowledges various "material degradation concerns" for reactors operating beyond 40 years, and concedes that "cracking of baffle former bolts is recognized as a potential issue for the IPEC units." Revised Reactor Vessel Internals Program, Attachment 1 to NL-12-037, at 8 (Exh. NYS000496). The Revised Inspection Plan, which purports to comply with EPRI's MRP-227-A guidelines, proposes to manage these aging effects through periodic inspections, and manifestly "does not include preventive actions." Attachment 1 to NL-12-037, at 5 (Exh. NYS000496).

22. The scope and details of the Revised Plan were developed and clarified through numerous bi-lateral meetings between Entergy and Staff and a series of NRC Staff Requests for Additional Information (RAIs) and Entergy responses, including, but not limited to, responses dated June 14, 2012 (NL-12-089) (Exh. NYS000497), September 28, 2012 (NL-12-134) (Exh. NYS000498), October 17, 2012 (NL-12-140) (Exh. NYS000499), November 20, 2012 (NL-12-166) (Exh. NYS000500), May 7, 2013 (NL-13-052) (Exh. NYS000501), September 27, 2013 (NL-13-122) (Exh. NYS000502), January 28, 2014 (NL-14-013) (Exh. NYS000503), June 9, 2014 (NL-14-067) (Exh. NYS000504), August 5, 2014 (NL14-093) (Exh. NYS000505), and September 8, 2014 (NL-14-117) (Exh. NYS000506). NRC Staff approved the "Revised RVI Program and Inspection Plan," as modified or clarified through Entergy's responses to RAIs, in November 2014 through Supplement 2 to the Safety Evaluation Report (SSER2) (Exh. NYS000507). Entergy's RVI AMP, as proposed in February 2012, modified or amended by the various subsequent communications, and approved by NRC Staff in the SSER2,

will be referred to as the "Revised and Amended RVI Plan."

23. The Revised and Amended RVI Plan does not include acceptance criteria for use when evaluating inspection results for all components. For example, Entergy has merely committed to develop acceptance criteria for baffle former bolts sometime prior to 2019 for IP2 and 2021 for IP3. Response to RAI 5, Attachment 1 to NL-12-089, at 11 (Exh. NYS000497). NRC Staff has approved this approach, SSER2, at 3-20 (Exh. NYS000507), even though cracking of baffle former bolts has been observed at European PWRs and Entergy acknowledges that it could be a problem at Indian Point. Attachment 1 to NL-12-037, at 8 (Exh. NYS000496).

24. For most other components, the Revised and Amended RVI Plan reflects a "wait-and-see" approach, in which Entergy proposes to wait for cracks or other visible wear to develop in RPVI components before deciding whether preventative steps are necessary. Attachment 2 to NL-12-037 (Exh. NYS000496); Revised Lahey PFT, at 54-55 (Exh. NYS000482). Indeed, for clevis insert bolts, Entergy accepts that crack detection before bolt failure is probably not possible, but proposes to wait for bolt failure to occur during the period of extended operation, under the assumption that bolt failures will not affect the safe operation of the IP facilities. SSER2, at 3-24 to 3-25 (Exh. NYS000507); Response to RAI 17, Attachment 1 to NL-13-122, at 8 (Exh. NYS000502). Entergy did not evaluate how the failure of highly fatigued and embrittled components – some of which may have failed entirely – would respond to an unexpected shock load, or whether the core would maintain a coolable geometry in the event that such a shock load caused multiple components or populations of components to fail. Revised Lahey PFT,

at 53-54 (Exh. NYS000482); February 2015 Lahey Declaration, at ¶¶25, 28 (Exh. NYS000483).

25. Entergy intends to rely extensively on Visual (VT-3) Examination, NL-12-037, Attachment 2, at 37-51 (Exh. NYS000496); MRP-227-A, at 4-3 to 4-4 (Exh. NRC000114A-F), although there is substantial evidence that this inspection methodology is often subjective and ineffectual. February 2015 Lahey Declaration, at ¶32 (Exh. NYS000483); Tregoning, R.L., "Reasons for Non-concurrence on Draft Safety Evaluation for the EPRI Topical Report (TR) Materials Reliability Program (MRP) Report 1016596 (MRP-227), Revision 0, Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines" (March 22, 2011) (Exh. NYS000508); Case, Michael J., Comments for the Document Sponsor to Consider Pertaining to Non-concurrence on Draft Safety Evaluation for EPRI Topical Report (TR) Materials Reliability Program (MRP) Report 1016596 (MRP-227), Revision 0, Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines, RES/DE (March 22, 2011) (Exh. NYS000509).

26. The Amended and Revised RVI Plan, as well as the NRC review in the SSER2, generally consider various age-related degradation mechanisms in "silos" without considering their interaction and synergism. Revised Lahey PFT, at 14-15 (Exh. NYS000482); February 2015 Lahey Declaration, at ¶ 16 (Exh. NYS000483).

ARGUMENT

ENTERGY HAS NOT DEMONSTRATED THAT THE EFFECTS OF AGING ON CERTAIN INTENDED FUNCTIONS WILL BE ADEQUATELY MANAGED FOR THE PERIOD OF EXTENDED OPERATION.

Pursuant to 10 C.F.R. § 54.33(a), all license renewals require the holder of the license to be subject to 10 C.F.R. § 50.54 which includes, *inter alia*, the obligation to comply with "all rules, regulations, and orders of the Commission." *Id.* § 50.54(h). Among those rules and regulations is 10 C.F.R. § 50.46(b)(4), which requires that following a LOCA "[c]alculated changes in core geometry shall be such that the core remains amenable to cooling." The obligation to maintain a coolable core geometry following a DBA LOCA has been described as one of the requirements that is "fundamental to providing reasonable assurance that [a proposed action] will not endanger the health and safety of the public." *Duke Energy Corporation* (Catawba Nuclear Station, Units 1 and 2) 60 N.R.C. 713, 724, LBP-04-32 (2004).

The purpose of an AMP is to assure that the "effects of aging will be adequately managed so that [certain] intended function(s) will be maintained consistent with the CLB for the period of extended operation." 10 C.F.R. § 54.21(a)(3). Maintaining a coolable core geometry is one of the functions that must be maintained throughout the period of extended operation. 10 C.F.R. §§ 54.33(a), 54.4(a)(i), (ii), (iii) and 54.4(b). An applicant must demonstrate that:

> there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the CLB . . . [including] managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under §54.21(a)(1).

10 C.F.R. § 54.29(a)(1). Here, Entergy must prove that its AMP for components whose continued integrity is essential to maintaining a coolable core geometry will achieve the regulation's objective. Entergy has failed to meet that obligation.

I. **Entergy's Revised and Amended RVI Plan Does Not Assure That the Effects of Aging on RVI Components Will Be Adequately Managed.**

Entergy's Revised and Amended RVI Plan does not assure that the effects of aging on RPV internals will be adequately managed to maintain intended functions during Indian Point's period of extended operation. The continued integrity of the RVI components is essential to ensuring that the core maintains a coolable geometry. However, RVI components operate in a harsh environment, and are subject to a variety of aging mechanisms. February 2015 Lahey Declaration, at ¶10 (Exh. NYS000483); *see* Figure 3, *supra*. Some of these aging mechanisms are known to work synergistically, meaning the cumulative effect of the simultaneous aging effects is greater than would be expected if the effects were considered individually and then added together. For example, irradiation embrittlement is known to decrease the resistance of RVI component materials to crack propagation. Revised Lahey PFT, at 18-19 (Exh. NYS000482) (and sources cited therein). While NRC regulations anticipate that some irradiation assisted cracking could occur in the RPV at neutron fluence levels of $1x10^{17}$ n/cm^2, 10 C.F.R. Part 50, Appendices G & H; RIS 2014-11 (Exh. NYS000494), the RPV internals are subject to fluence levels ten-thousand to a million times greater (10^{21} to 10^{23} n/cm^2) by the end of a period of extended operation. Revised Lahey PFT at 27-28 (Exh. NYS000482); LWRS (August 2014), at 18 (Exh. NYS000485); *see* Figure 4, *supra*. The extent and severity of the cumulative aging effects on RVI components caused by this intense irradiation, when coupled with fatigue and corrosive aging effects at work in the RVI, is largely unknown. Revised Lahey PFT, at 17-18, 19-21 (Exh. NYS000482); NUREG/CR-6909 Rev. 1 (March 2014 (draft)), at 11 (Exh. NYS000490); Chen, *et al.*, "Crack Growth Rate and Fracture Toughness Tests on Irradiated Cast Stainless Steels," NUREG/CR-7184, (Revised December 2014), at xv (Exh. NYS000488A-B); Trans. of Briefing on Subsequent License Renewal, at 77 (May 2014) (Exh. NYS000492); NUREG/CR-7153, Vol. 2: Aging of Core Internals and Piping Systems, at 181, 187, 210-211 (Exh. NYS000484A-B); Stevens, et al., (October 2014) at 9-10 (Exh. NYS000486).

Rather than taking proactive steps to repair or replace aging RVI components, Entergy has proposed to rely on a "wait and see" approach to aging management that relies on detection of cracks or other aging effects prior to part repair or replacement. In some cases, Entergy intends to wait for components to fail entirely before taking any preventative action. By relying on the detection of signs of degradation prior to repairing or replacing a part, Entergy fails to account for the possibility that heavily embrittled and fatigued RVI components may not have signs of degradation that can be detected by an inspection, but may nonetheless fail as a result of an abnormal seismic, thermal or pressure shock load. Revised Lahey PFT, at 39-40, 54 (Exh. NYS000482); February 2015 Lahey Declaration, ¶ 19 (Exh. NYS000483); *see* Transcript of Advisory Committee on Reactor Safeguards Meeting, Plan License Renewal Subcommittee, at 209-210 (April 23, 2015) (Exh. NYS000526) (in which a member of subcommittee observes that embrittled core components could fail during a seismic event). Furthermore, Entergy fails to account for the possible "chain reaction" that may occur if multiple highly embrittled and fatigued components fail simultaneously when subjected to a sudden shock load.

One example of the applicant's "wait-and-see" approach is its aging management plan for baffle former bolts. Although the applicant acknowledges that "cracking of baffle former bolts is recognized as a potential issue for the Indian Point units," Attachment 1 to NL-12-037, at 8 (Exh. NYS000496), the applicant does not propose to replace those bolts, only to continue monitoring them, Attachment 2 to NL-12-037, at 40, tbl. 5-2 (Exh. NYS000496); *see* Revised Lahey PFT, at 55-56 (Exh. NYS000482); February 2015 Lahey Declaration, ¶¶27-28 (Exh. NYS000483). In fact, the applicant has not yet developed inspection acceptance criteria for baffle former bolts in either IP2 or IP3. SSER2, at 3-20 (Exh. NYS000507). Instead, the applicant has agreed to develop a technical justification including acceptance criteria for baffle former bolts sometime prior to the first round of inspections, which might not occur until 2019 for IP2 and 2021 for IP3. SSER2, at 3-20 (Exh. NYS000507); Response to RAI 5, Attachment 1 to NL-12-089, at 11 (Exh. NYS000497).

Another example of the applicant's "wait-and-see" approach for the RVIs is the applicant's proposal for managing aging effects on the clevis insert bolts. SSER2, at 3-23 to 3-26 (Exh. NYS000507). Like the split pins that the applicant is replacing for the second time, clevis insert bolts are susceptible to primary water stress corrosion cracking (PWSCC). MRP-227-A, Appendix A, at A-2 (Exh. NRC000114A-F). Failures of clevis insert bolts, apparently caused by PWSCC, were detected at a Westinghouse-designed reactor in 2010. Out of 48 clevis bolts in this reactor, 29 were partially or completely fractured but only 7 of those damaged bolts were visually detected as having failed. SSER2, at 3-25 (Exh. NYS000507). Despite this high rate of failure (60% of the total bolts were damaged) and low rate of visual detection (only 24% of the damaged bolts were detected), the applicant proposes to manage the aging degradation of clevis insert bolts with visual (VT-3) inspections rather than pre-emptive replacement. Attachment 2 to NL-12-037, tbl. 5-4, at 51 (Exh. NYS000496).

The applicant apparently acknowledges that visual inspections will not detect the majority of clevis bolt cracks prior to failure, but justifies this approach on the grounds that "crack detection prior to bolt failure is not required due to design redundancy." Response to RAI 17, Attachment 1 to NL-13-122, at 8 (Exh. NYS000502). In fact, the applicant appears to suggest that the failure of multiple clevis insert bolts will not seriously affect the *steady state* operation of the reactor. The applicant then analyzes the effect of clevis bolt failures on various other components. The applicant's analysis of the effects of clevis bolt failures assumes that all other components will be functioning according to their design specifications, and does not consider the fact that the other components may also be undergoing degradation from various interacting aging mechanisms. Revised Lahey PFT, at 57-58 (Exh. NYS000482). Moreover, the applicant fails to consider the possibility that a shock load (*e.g.*, due to seismic event or LOCA) may cause the sudden failure of the remaining intact clevis bolts, which, in turn, may lead to an uncoolable core geometry. *Id.* at 58. In short, rather than taking proactive steps to replace clevis bolts prior to failure, the applicant proposes to wait for clevis bolt failures to occur before taking steps to address the problem.

The applicant's approach to analyzing the lower support structures' functionality and fracture toughness is similarly flawed. Revised Lahey PFT, at 58-60; *see* Response to RAI-11-A, Attachment 1 to NL-13-052, at 1-4 (Exh. NYS000501). For example, the applicant noted that irradiation embrittlement effects would only be significant in the presence of pre-existing flaws or service induced defects, together with a stress level capable of crack propagation. In its analysis, the applicant assumed that the columns would be subject to "nominal normal operating stresses." SSER2, at 3-43 (Exh. NYS000507). When NRC staff inquired about the most recent visual inspections of the core support structures, the applicant acknowledged that the CASS support column caps were inaccessible to inspection and that VT-3 visual inspection offered "no meaningful information regarding the structural integrity of the columns." *Id.* at 3-44. Under these circumstances, the applicant's conclusion that irradiation-induced cracking of core support columns is "unlikely" represents wishful thinking and is contrary to recent studies, Chen, et al., NUREG/CR-7184, at xv (March 2014) (Exh. NYS000488A-B), which showed the extreme sensitivity of crack growth rate and fracture toughness to irradiation. Revised Lahey PFT, at 59 (Exh. NYS000482). Moreover, it ignores the fact that these and other non-CASS RVI structures and components undergo a range of aging degradation mechanisms simultaneously under steady and non-steady state conditions, and that embrittlement or susceptibility to fracture simply cannot be adequately detected using currently available inspection techniques. *Id.*

By merely relying on MRP 227-A for its aging management plan, the applicant has ignored the large uncertainties that exist with respect to the effects of irradiation-induced aging phenomena. Revised Lahey PFT, at 60-61 (Exh. NYS000482); *see* Chen, et al., NUREG/CR-7184, at xv (Exh. NYS000484A-B) ("no data are available at present with regard to the combined effect of thermal aging and irradiation embrittlement" on CASS); *see also* NUREG/CR-7153, Vol. 2: Aging of Core Internals and Piping Systems, at 181, 187, 210-211 (Exh. NYS000484A-B); Stevens, et al., (October 2014) at 9-10 (Exh. NYS000486). While the applicant's Thermal Aging and Neutron Irradiation of Cast Austenitic and Stainless Steel (CASS) program generally recognizes the potential adverse synergistic effects of elevated coolant temperature and irradiation on the fracture toughness of CASS materials, a broader recognition of this principle is needed by the applicant, since RVI components made from non-cast stainless steel will also experience the combined effects of irradiation-induced embrittlement, corrosion, and other aging mechanisms. Revised Lahey PFT, at 61 (Exh. NYS000482); NUREG/CR-7153, Vol. 2: Aging of Core Internals and Piping Systems, at 161-188 (Exh. NYS000484A-B). Indeed, the Expanded Materials Degradation Assessment (EMDA) report prepared by the NRC and DOE specifically notes that "the concept of a threshold fluence . . . is scientifically misleading" and that irradiation-assisted stress corrosion cracking "initiation and growth must be understood in terms of the interdependent effects of many parameters." *Id.* at 183.

The applicant and NRC Staff have devoted significant time addressing thermal embrittlement (TE) and irradiation-induced embrittlement (IE) effects on the CASS support columns. SSER2, at 3-40 to 3-47 (Exh. NYS000507); Response to RAI-11C, Attachment 1 to NL-14-093, at 1-4 (Exh. NYS000505); Response to RAI-11B, Attachment 1 to NL-13-122, at 2-4 (Exh. NYS000502); Response to RAI-11A, Attachment 1 to NL-13-052, at 1-3 (Exh. NYS000501); Response to RAI-11, Attachment 1 to NL-12-134, at 11-12 (Exh. NYS000498). In contrast, the applicant has failed to evaluate the synergistic mechanisms that operate on other important and vulnerable RVI components, such as the core baffles, baffle bolts, and formers. Compared to the baffles, baffle bolts, and formers, the core support columns are located in an area of the reactor pressure vessel which is subject to less radiation fluence (and thus embrittlement). Revised Lahey PFT, at 61 (Exh. NYS000482); February 2015 Lahey Declaration, ¶ 31 (Exh. NYS000483). By failing to consider the synergistic effects of embrittlement on these other components, the Amended and Revised RVI Plan fails to provide a reasonable assurance that the effects of aging will be adequately managed during Indian Point's period of extended operation. *See* 10 C.F.R. § 54.21(a)(3).

The Revised and Amended RVI Plan is also inadequate because it relies substantially on Visual (VT-3) Examinations, *see* NL-12-037, Attachment 2, at 37-51 (Exh. NYS000496); MRP-227-A, at 4-3 to 4-4 (Exh. NRC000114A-F), which have significant shortcomings in their ability to detect material cracking, degradation or wear prior to component failure. Lahey Revsied PFT, at 61-62 (Exh. NYS000482); February 2015 Lahey Declaration, ¶32 (Exh. NYS000483). The problem is that the MRP-227-approved VT-3 methodology is incapable of detecting substantial cracks in RPVIs that do not exhibit as gross deformation of the component, and that use of other methodologies such as UT, EV-1 and VT-1, also recommended by MRP-227-A, are more reliable and better able to detect stress corrosion cracking. Tregoning, Reasons for Non-concurrence (Exh. NYS000508); Case, Comments for the Document Sponsor to Consider Pertaining to Non-Concurrence (Exh. NYS000509). The inadequacy of VT-3 inspections is illustrated by the visual detection of only 7 out of 29 fractured clevis insert bolts at a Westinghouse PWR in 2010. SSER2, at 3-25 (Exh. NYS000507). Even MRP-227-A recognizes that EV-1 and VT-1 have detection capabilities that are superior to VT-3. MRP-227-A at 4-4 (Exh. NRC000114A-F) ("Unlike the detection of general degradation conditions by visual (VT-3) examination, visual (VT-1) and enhanced visual (EVT-1) examinations are conducted to detect discontinuities and imperfections on the surface of components, including such conditions as cracks, wear, corrosion, or erosion. Specifically, VT-1 is used for the detection of surface discontinuities such as gaps, while EVT-1 is used for the detection of surface breaking flaws").

However, neither MRP-227-A nor the Revised and Amended RVI Plan give any rational justification for the widespread approval of the use of VT-3 for such critical components as baffle-former bolt assemblies. Reliance on the assumption that there will be a grossly visible defect before the component reaches critical degradation is a hypothesis that has not been tested and as to which Entergy offers no supporting evidence.

MRP-227-A also contains text that raises question about how Entergy will inspect for and manage RPVI embrittlement. Footnote 1 to MRP-227-A Table 3-3 states as follows:

> The significance of thermal and irradiation embrittlement is directly related to the probability of a flaw existing in the component. *There are no recommendations for inspection to determine embrittlement level because these mechanisms cannot be directly observed*. However, potential embrittlement must be considered in flaw tolerance evaluations.

MRP-227-A, at 3-23, Table 3-3, "Final Disposition of Category B and C Westinghouse internals," note 1 (emphasis added) (Exh. NRC000114A-F). The recognition that embrittlement mechanisms "cannot be directly observed" further supports the concerns about an AMP that relies so heavily on visual exams.

In summary, Entergy recognizes that it is obligated to provide an AMP for RPVIs. However, the AMP offered is deficient in several fundamental ways which make it impossible for Entergy to demonstrate the adequacy of the RPVI AMP to assure the coolability of the core geometry following accidents, such as a DBA LOCA or seismic event. Thus, the LRA must be denied. *See* 10 C.F.R. §§ 54.21(a)(3), 54.29(a)(1).

II. <u>Entergy's Evaluation of the Fatigue Life of Limiting Reactor Systems, Structures and RVI Components Is Inadequate.</u>

Additionally, the applicant's evaluation of the fatigue life of the limiting reactor systems, structures and RVI components is inadequate. In this proceeding, the applicant agreed, in Commitments 33, 43 and 49, to calculate the cumulative usage factors, adjusted for environmental degradation (CUF_{en}) for external and internal components. Dacimo, Fred, Entergy, letter to Document Control Desk, NRC, "Reply to Request for Additional Information Regarding the License Renewal Application," NL-13-122 (September 27, 2013), at 20 (Exh. NYS000502). Under this approach, "crack initiation is assumed to have started in a structural component when the fatigue usage factor at the point of the component reaches the value of 1, the design limit on fatigue." Generic Aging Lessons Learned (GALL) Report, NUREG-1801, Rev. 2 (2010), X.M1-1 (Exh. NYS000147A-D). Accordingly, Entergy has committed to take corrective action if a component's CUF_{en} value exceeds 1.0. *See* LRA Commitment 49, Attachment 1 to Letter from Fred Dacimo to NRC Document Control Desk, NL-13-052 (May 7, 2013), at 9 (Exh. NYS000501) (ML13142A202); LRA Commitment 33, Attachment 2 to NL-13-052, at 15 (Exh. NYS000501).

[redacted]

[redacted]

[redacted]

Additionally, the calculated CUF_{en} values for internals are non-conservative, because they do not account for the synergistic effects of embrittlement. Revised Lahey PFT, at 63-64 (Exh. NYS000482). The allowable cycles to failure for components used in the CUF_{en}

calculation are determined from small scale experiments using metal test samples which are exposed to simulated reactor coolant environments. *Id.* at 64. However, the fatigue experiments do not use highly embrittled metal test samples, which have a lower expected fatigue life than non-embrittled (ductile) materials. *Id.*

Moreover, there has been no discussion of the effect of possible shock loads on the integrity of such severely fatigue-weakened structures. [redacted] Even assuming this CUF_{en} calculation is accurate, it does not account for the possibility that a component which is highly fatigued but does not yet have visible surface cracks may be exposed to an unexpected shock load or seismic event that could cause it to fail. Revised Lahey PFT, at 68-69 (Exh. NYS000482). This concern prompted a member of the ACRS Plant License Renewal Subcommittee to observe that "you don't even need to have a crack if these [core support] columns are really brittle and they have an earthquake loading" and that it "would be really bad day if you dislodge the core during an earthquake because these support columns are embrittled." Transscript of Advisory Committee on Reactor Safeguards Meeting, Plant License Renewal Subcommittee, at 209-211 (April 23, 2015) (Exh. NYS000526).[12] The ACRS observation about

[12] The transcript of the ACRS License Renewal subcommittee meeting at which these comments were made is replete with apparent typographic errors, in which the letters "B" or "C"

external seismic forces takes on added importance given the recent acknowledgment by Entergy and NRC Staff that the potential seismic hazard curves for the Indian Point site are higher than the seismic spectra developed in the 1970s during the proceedings concerning the initial operating licenses. NRC, Slides, "Near-term Task Force, Recommendation 2.1 Seismic Hazard Evaluation: Entergy," at 6-7 (June 19, 2014) (Exh. NYS000528) (charts depicting initial safe shut down earthquake [SSE] and updated ground motion response spectra [GMRS] produced by Entergy and Staff in 2014 for the Indian Point facilities); *see* Figures 6 & 7, *infra*. The failure to consider such factors in the analysis of component fatigue is a good example of the type of "silo thinking" (i.e., the fatigue and safety analyses are treated entirely separately) that the State is concerned about.

In short, Entergy's fatigue calculations for RVI and other components do not ensure that those components will not fail as a result of the combined effects of fatigue and embrittlement, especially if they are exposed to a shock load. Rather, various operational and accident-induced shock loads could cause failures well before the 1.0 fatigue limit is reached. Revised Lahey PFT, at 68-69 (Exh. NYS000482). Considering the importance of RVI integrity to maintaining a coolable core geometry, the inadequacy of Entergy's fatigue calculations means that Entergy has failed to establish that the effects of aging on the RVI components will be adequately managed for the period of extended operation. For this reason, the LRA should be denied. *See* 10 C.F.R. § 54.21(a)(3).

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apparently take the place of substantive comments, some of which appear to be directly relevant to the State's Contention NYS-25. *See, e.g.*, Hearing Trans, at 206-215 (Exh. NYS000526). The State reserves the right to submit supplemental briefing if and when a clean and complete copy of the hearing transcript becomes available.

III. Entergy's Proposed RVI AMP and FMP Impermissibly Erode Important Safety Margins, and Fail to Account for Uncertainties Inherent in All Modeling

Entergy's approach to license renewal relies heavily on eroding safety margins that have been in place since Indian Point was constructed, even as the facility reaches the end of its planned period of operation and into its period of extended operation. With respect to the heavily fatigued and embrittled RVIs, Entergy proposes to run them until they fail, with the hope they will not affect the coolability of the in-core geometry when they do fail. [redacted] This approach does not provide a reasonable assurance that the facility will continue to operate safely during its period of extended operation.

The importance of maintaining a safety margin when operating a nuclear power plant was highlighted recently, when a significant modeling calculation was revealed to be non-conservative. The non-conservatism specifically affects a calculation referred to as "Branch Technical Position 5-3" (BTP 5-3), which is used to evaluate the unirradiated nil ductility transition reference temperature (RT_{NDT}) and upper shelf energy (USE) for reactor components outside of the RPV beltline, especially in older reactors where detailed test data were not available. Troyer, et al., "An Assessment of Branch Technical Position 5-3 to Determine Unirradiated RTNDT for SA-508 Cl.2 Forgings," Paper No. PVP2014-28897, Proceedings of the ASME 2014 Pressure Vessels and Piping Conference, Anaheim, California (July 20-24, 2014) (Exh. NYS000516). The BTP 5-3 equation consists of several components known as "positions." NRC, Slides, "Assessment of BTP 5-3 Protocols to Estimate RTNDT(u) and USE (June 4, 2014), at 4 (Exh. NYS000518). In 2014, AREVA Inc. published a paper and sent a letter to NRC noting that some of the positions of BTP 5-3 appeared to be non-conservative when compared to actual test results. Troyer, et al. (Exh. NYS000516); Letter from Pedro Salas, Regulatory Affairs Director, AREVA, to NRC, regarding Potential Non-Conservatism in Branch Technical Position 5-3 (January 30, 2014) (Exh. NYS000517). NRC Staff then compared the BTP 5-3 predictions to actual test data, and determined that BTP 5-3 was non-conservative in all positions – in some positions, more than 90% of the test data were non-conservative when compared to the BTP 5-3 predictions. NRC, Slides, at 27 (Exh. NYS000518).

Summary on Part I – Technical Evaluation

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- Positions 1.1(3) and 1.2
  - Results of the two studies are similar
  - Staff analysis confirms non-conservatism
- Position 1.1(4)
  - EG&G report demonstrates position is non-conservative
  - Awaiting NDTT data from Archives to complete staff assessment

Position of BPT 5-3		Forging Non-Conservative Prediction Rate		Plate Non-Conservative Prediction Rate	
		EG&G Data	Raw Data	EG&G Data	Raw Data
1.1(3)	(a) TRANS = 0.65×LONG	43%	48%	33%	19%
	(b) $T_{C(TRANS)} = T_{C(LONG)} + 20\ ^{\circ}F$	50%	57%	70%	63%
1.1(4)	$RT_{NDT} = T_{45(LONG)}$	93%	TBD	38%	TBD
	$RT_{NDT} = T_{30(LONG)} + 20\ ^{\circ}F$	93%	TBD	38%	TBD
1.2	$USE_{TRANS} = 0.65 \times USE_{LONG}$	14%	33%	20%	13%

Figure 5. Non-Conservatism documented by NRC Staff in all BTP 5-3 positions. Source: NRC, Slides, "Assessment of BTP 5-3 Protocols to Estimate RTNDT(u) and USE (June 4, 2014), at 27 (Exh. NYS000518).

[redacted]

The precise scope of the impact of the BTP 5-3 non-conservatism on IP2 and IP3 is not yet known. Indeed, a portion of the IP3 RPV was previously found to exceed the PTS screening criteria, and it is not clear whether that RPV portion or other IP3 RPV components are affected by the BTP 5-3 non-conservatism. Revised Lahey PFT, at 44. Additionally, a recent American Society of Mechanical Engineers (ASME) Pressure Vessels & Piping Conference, NRC staff also highlighted newly-identified non-conservatisms in sections of the ASME code regarding fracture toughness applicable to nuclear reactor operations. Revised Lahey PFT, at 75; Kirk, M. et al., "Assessment of Fracture Toughness Models for Ferritc Steels Used in Section XI of the ASME Code Relative to Current Data-Based Model," PVP 2014-28540 (Exh. NYS000520).

Another recent development demonstrating the importance of safety margins is the acknowledgment by Entergy and NRC Staff that the potential seismic hazard curves for the Indian Point site are higher than the seismic spectra developed in the 1970s during the proceedings concerning the initial operating licenses. NRC, Slides, "Near-term Task Force, Recommendation 2.1 Seismic Hazard Evaluation: Entergy," at 6-7 (June 19, 2014), (Exh.NYS000528). During a June 19, 2014 public meeting, NRC Staff's presentation included charts depicting the initial safe shut down earthquake (SSE) spectra in blue and the updated ground motion response spectra (GMRS) produced by Entergy and Staff in 2014 in red and green (respectively) for the Indian Point facilities. *See* Figures 6 and 7.

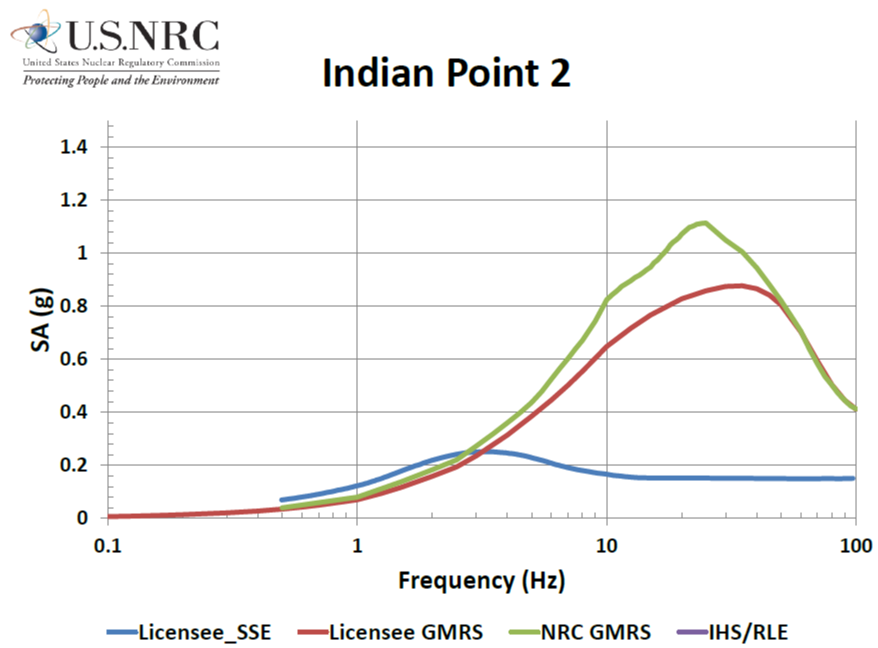

Figure 6. Chart depicting safe shutdown earthquake (SSE) spectra and updated ground motion response spectra (GMRS) for Indian Point 2. Source: NRC, Slides, "Near-term Task Force, Recommendation 2.1 Seismic Hazard Evaluation: Entergy" (June 19, 2014), at 6 (Exh. NYS000528).

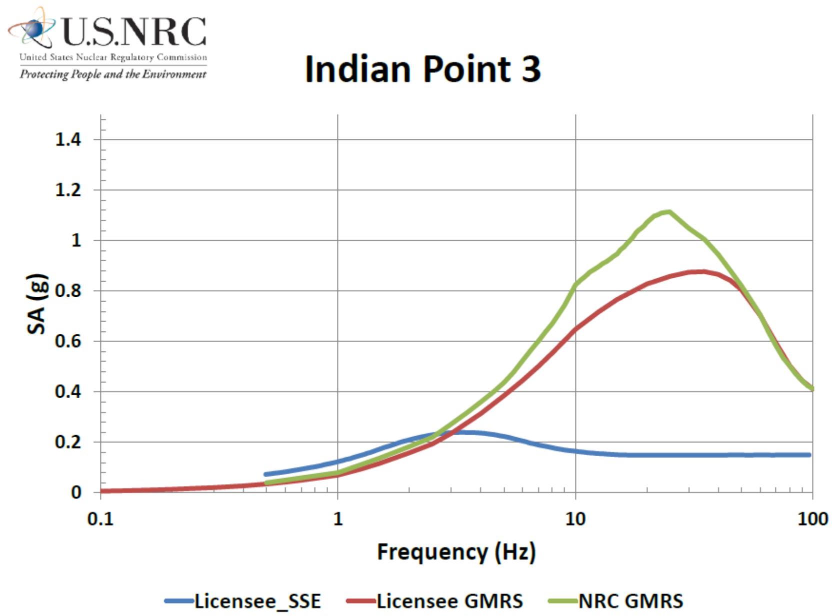

Figure 7. Chart depicting safe shutdown earthquake (SSE) spectra and updated ground motion response spectra (GMRS) for Indian Point 3. Source: NRC, Slides, "Near-term Task Force, Recommendation 2.1 Seismic Hazard Evaluation: Entergy" (June 19, 2014), at 7 (Exh. NYS000528).

For both Indian Point Unit 2 and Indian Point Unit 3, the recent updated seismic spectra exceed the 1970s era seismic spectra.

The recent revelations regarding the BTP 5-3 and ASME non-conservatisms and revised seismic spectra highlight the importance of maintaining safety margins when operating a nuclear facility. Calculations and predictions – however well-established – are nonetheless imperfect, and in some cases can prove to be inadequate to ensure a plant's safe operation. Safety margins help ensure that errors or non-conservatisms are not discovered through catastrophic component failure. Entergy, by systematically removing conservatisms and proposing to operate various components to the brink of predicted failure, does not provide an adequate assurance that Indian Point will continue to operate safely during its period of extended operations. *See* 10 C.F.R. §

54.21(a)(3).

CONCLUSION

For the above reasons Entergy's application to renew the operating licenses for Indian Point Unit 2 and Indian Point Unit 3 should be denied.

Respectfully submitted,

Dated: June 9, 2015

Signed (electronically) by

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