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, Entergy Nuclear Indian Point 3, LLC, and Entergy Nuclear Operations, Inc.	DPR-26, DPR-64 February 13, 2015
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DECLARATION OF DR. RICHARD T. LAHEY, JR.

I, Richard T. Lahey, Jr., declare under penalty of perjury that the following is true and correct:

The purpose of this Declaration is to document some of my continuing technical concerns associated with the relicensing of Indian Point Unit 2 (IP2) and Indian Point Unit 3 (IP3), the two operating nuclear reactors in Buchanan, New York.

INTRODUCTION

1. I am currently the *Edward E. Hood Professor Emeritus of Engineering* at Rensselaer Polytechnic Institute (RPI) in Troy, New York. I have earned the following academic degrees: a B.S. in Marine Engineering from the United States Merchant Marine Academy, a M.S. in Mechanical Engineering from RPI, a M.E. in Engineering Mechanics from Columbia University, and a Ph.D. in Mechanical Engineering from Stanford University. While I was an active member of the faculty

at RPI, I served as the Dean of Engineering, the Chairman of the Department of Nuclear Engineering & Science and the Chair of Rensselaer's Faculty Council.

2. In addition, before coming to RPI, I was directly responsible for the nuclear reactor safety research & development (R&D) programs for the General Electric Company (GE). Moreover, I have extensive experience with both military (*i.e.*, naval) and commercial light water nuclear reactor (LWR) technology.

3. I am a member of various professional societies, including: the American Nuclear Society (ANS), where I was a member of the Board of Directors and Executive Committee, and was the founding Chair of the ANS Thermal-Hydraulics Division; the American Society of Mechanical Engineers (ASME), where I was Chair of the Nucleonics Heat Transfer Committee, K-13; the American Institute of Chemical Engineering (AIChE), where I was the Chair of the Energy Transport Field Committee; and the American Society of Engineering Educators (ASEE), where I was Chair of the Nuclear Engineering Division. I have also been an Editor of the international *Journal of Nuclear Engineering & Design*.

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

5. Over the last 50 years, I have published numerous books, monographs, chapters, articles, reports, and journal papers on nuclear engineering and nuclear reactor safety technology, and have received many honors and awards for my career accomplishments, including: the E.O. Lawrence Memorial Award of the Department of Energy (DOE), the Arthur Holly Compton Award of the ANS, the Glenn Seaborg Medal of the ANS, and the Donald Q. Kern Award of the AIChE. I am widely considered to be an expert in matters relating to the design, operations, safety and aging of nuclear power plants. My *Curricula Vitae*, which is Exhibit NYS-295 in this proceeding, is attached to this report and describes the details of my educational and professional background and qualifications (Attachment 1).

6. I am quite familiar with the type of pressurized water nuclear reactors (PWRs) at the Indian Point site in Buchanan, New York, and with the issues and developments in this relicensing proceeding. I previously submitted a November 30, 2007 declaration in support of the *Notice of Intention to Participate and Petition to Intervene* filed by the State of New York in this proceeding, an April 7, 2008 declaration in further support of the State's Supplemental Contention 26-A (Metal Fatigue), a September 8, 2010 declaration in support of the State's New and Amended Consolidated Contention Concerning Metal Fatigue (NYS-26B/RK-TC-1B), a September 15, 2010 declaration in support of the State's Additional Bases for Previously-Admitted Contention NYS-25 (Embrittlement of Reactor Pressure Vessels and Associated Internals), a September 30, 2011 declaration and a November 1, 2011 declaration in support of the State's Joint Contention NYS-

38/RK-TC-5, following the publication of the first Supplemental Safety Evaluation Report (SSER) for the proposed renewal of the operating licenses for the Indian Point nuclear facilities, and a December 20, 2011 report in support of the State's Contention 25 (NYS-25) and Consolidated Contention 26B (NYS-26B/RK-TC-1B). Furthermore, I have provided pre-filed direct testimony in support of the State's Contention 25 (NYS-25) and Consolidated Contention 26B (NYS-26B/RK-TC-1B) dated December 22, 2011, and in support of the State's Joint Consolidated Contention 38 (NYS-38/RK-TC-5) dated June 18, 2012, and have pre-filed rebuttal testimony in further support of Consolidated Contention NYS-26B dated June 29, 2012 and Joint Contention NYS-38/RK-TC-5 dated November 9, 2012.

7. The factual statements and the expression of opinion in this report are based on, among other things, my best professional knowledge, my extensive professional experience in nuclear reactor technology, and my review of various recently-created documents including, but not limited to: the USNRC Staff's November 2014 second Supplemental Safety Evaluation Report (SSER2) for the proposed renewal of the operating licenses for the Indian Point facilities, the applicant's February 17, 2012 proposed amendment to its license renewal application (LRA) entitled, "Revised Reactor Vessel Internals Program and Inspection Plan" [NL-12-037], the applicants' responses to various requests for additional information (RAIs) from the USNRC, in particular those dated September 28, 2012 [NL-12-134], November 20, 2012 [NL-12-166], May 7, 2013 [NL-13-052], September 27, 2013 [NL-13-122], and August 5, 2014 [NL-14-093], various

other correspondences between the applicant and USNRC Staff, various documents recently-published by the USNRC, DOE, and EPRI, as well as many other related technical documents, including those referenced at the end of this report and in my previous ASLB submittals in this relicensing proceeding.

OVERVIEW

8. The two operating Indian Point reactors (IP2 and IP3) are among the older operating nuclear reactors in the United States. IP2 reached the end of its initial 40-year operating license term on September 28, 2013, and IP3 will reach the end of its initial operating license term on December 12, 2015. The applicant, Entergy Nuclear Operations, Inc., submitted a License Renewal Application (LRA) in 2007, seeking to extend the operation of IP2 and IP3 for an additional 20 years each. If this extension is approved and the plants operate for an additional 20 years, the Indian Point reactors will be among the first group of United States nuclear reactors allowed to operate out to 60 years. Considering the proximity of the Indian Point reactors to the New York City metropolitan area – the reactors are about 24 miles north of the New York City line – it is essential that the applicant minimize any safety risk posed by the continued operation of IP2 and IP3.

9. A nuclear power reactor 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. For example, Figure-1, below, identifies some of the different materials used for the components in a typical pressurized water nuclear reactor (PWR):

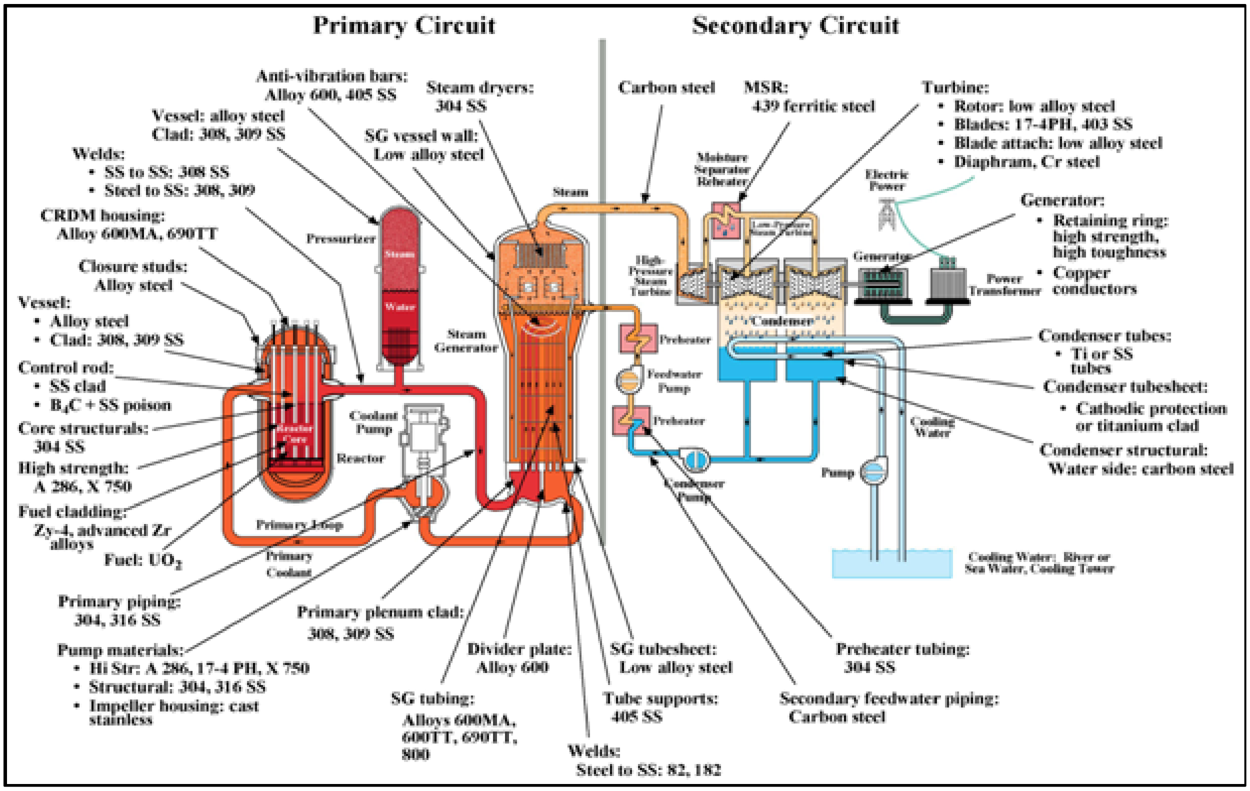

Figure 1. Source: DOE, Light Water Reactor Sustainability Program: Materials Aging and Degradation Technical Program Plan, at 2, Figure 1 (August 2014).

As a consequence of this wide variety of materials there are many different aging challenges which must be addressed to assure the health and safety of the American public when considering extended plant operations. [*See generally* 10 C.F.R. § 54.21(a), (c)].

10. 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 [Expanded Materials Degradation Assessment (EMDA);

NUREG/CR-7153, Vol. 3, at 4 tbl. 1.1 (October 2014)]. Extending the operation of a reactor beyond its 40-year design life will increase the challenges to the integrity of important LWR systems, structures, and components.

11. A central challenge for extended reactor operations is the substantial uncertainty posed by the interrelated and possibly synergistic effects of aging degradation on the multiple components in the primary and secondary systems of a nuclear power plant. Recognizing the seriousness of the challenge posed by the interrelated aging processes on important components, the federal government has embarked on a program known as the Light Water Reactor Sustainability Program which includes inquiry into whether the different materials and LWR components can continue to perform their intended function during the extended operation of a nuclear reactor. The Light Water Reactor Sustainability Program has identified numerous uncertainties with the interaction of various aging mechanisms on the many components of a nuclear reactor during periods of operation beyond their original, 40-year design life [DOE, Light Water Sustainability Program, Material Aging and Degradation Technical Program Plan (August 2014), at 1-6].

12. Thus, as previously noted, the DOE and USNRC, in conjunction with various national laboratories, have recently embarked on an ambitious R&D program to understand and resolve issues related to these interacting and synergistic effects [NUREG/CR-7153, Vol. 2, "Expanded Materials Degradation Assessment (EMDA), Aging of Core Internals and Piping Systems" (October 2014), at 1-5]. However, at this time the USNRC and DOE do not know the full extent or

severity of the interacting and synergistic degradation mechanisms [*Id.* at 3]. In particular, the consequences of the interaction of embrittlement, fatigue, and the corrosion-induced degradation of various reactor vessel internals (RVI), and safety-related components/systems during shock loads, remains unknown [*see, e.g.,* NUREG/CR-6909 Rev. 1 (March 2014 (draft)), at 11 ("it is not possible to quantify the impact of irradiation on the prediction of fatigue lives in PWR primary water environments compared to those in air.")]. This uncertainty is of course very significant since most of the available fatigue data which forms the basis of the ASME Boiler and Pressure Vessel design code (Section-III) is unirradiated small test sample data taken in air.

13. Also, at a recent briefing to the Metallurgy and Reactor Fuels Subcommittee of the Advisory Committee on Reactor Safeguards (ACRS), the USNRC staff recognized that "synergistic interactions" in light water reactor environments may enhance crack initiation and growth rate in steel components [Stevens, Gary L., Presentation to 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; *see also* NUREG/CR-6909 Rev. 1 (March 2014 (draft)), at 9].

14. Moreover, a recent report, prepared by Argonne National Laboratory for USNRC, acknowledges, with respect to cast austenitic stainless steels (CASS), that "a combined effect of thermal aging and irradiation embrittlement could reduce the fracture resistance even further to a level neither of these degradation

mechanisms can impart alone" [Chen, *et al.*, "Crack Growth Rate and Fracture Toughness Tests on Irradiated Cast Stainless Steels," NUREG/CR-7184, (Revised December 2014), at xv]. Indeed, the report recognizes that, "no data are available at present with regard to the combined effect of thermal aging and irradiation embrittlement" on CASS [*Id.*].

15. In fact, multiple recent documents confirm that the USNRC does not fully understand the interrelationship between embrittlement, high or low cycle fatigue, and shock loads for highly fatigued and/or embrittled components made of CASS, non-cast stainless steels, or other alloys. For example, a recent paper presented at an MPA Seminar in Stuttgart, Germany confirms that, at present, USNRC staff does not have a clear solution to the challenges posed by synergistic age-related degradation mechanisms. [Stevens, *et al.*, (October 2014), at 9-10]. Instead of mandating a sure solution to this important age-related safety problem (*i.e.*, repairing or replacing the degraded systems ,structures and fittings) the Stuttgart presentation (as well as other USNRC publications) shows that, by default, the USNRC simply proposes the continued use of the environmental factor method for fatigue life for evaluating the *in situ* degradation of structures and components [*Id.* at 10], a method which is not necessarily conservative and one that certainly does not address all the synergistic effects that New York State is concerned about. This was done even though a recent draft report on the "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials," prepared by Argonne National Laboratory (ANL) and USNRC Staff, recognizes the

"inconclusive" nature of existing data on the synergistic effects of irradiation and fatigue, and other aging mechanisms in LWR environments, and concludes that "additional fatigue data on reactor structural materials irradiated under LWR operating conditions are needed." [NUREG/CR-6909, Rev. 1 (March 2014 (draft)), at 11]. Furthermore, during a "Briefing on Subsequent License Renewal" to the USNRC, the USNRC's Chief of the Corrosion and Metallurgy Branch, Dr. Mirela Gravila, testified that the Piping and Core Internals Panel had recognized "significant gaps" in technical knowledge with respect to the effects of irradiation-induced degradation on the RVI components [Trans. of Briefing on Subsequent License Renewal, at 77 (May 2014)].

DISCUSSION

16. At the beginning of this license renewal proceeding there was apparently little or no recognition on the part of the USNRC of the simultaneous and interrelated synergistic effects of the various age-related degradation mechanisms. Also, at the time the Indian Point LRA was originally submitted, the Generic Aging Lessons Learned (GALL) guidance document developed by the nuclear industry and USNRC did not even recognize the aging challenges to reactor vessel internal (RVI) components. Anyway, the evaluations done of the various age-related degradation mechanisms, and their safety significance, were done separately in "silos" (*i.e.,* there was no consideration of their interaction and synergism). It should be noted that the State of New York identified early-on in the Indian Point relicensing proceedings the challenges and safety significance posed by

this approach, which did not consider interacting and synergistic effects. As discussed previously, the DOE, through its system of national laboratories, has recently acknowledged these challenges and the complexity posed by such interrelated and synergistic degradation mechanisms. A DOE demonstrative figure (Figure-2, below) depicts some, but not all (*e.g.,* not explicitly including fatigue) of the interactions among component materials, environments, and stresses in an operating LWR nuclear power plant:

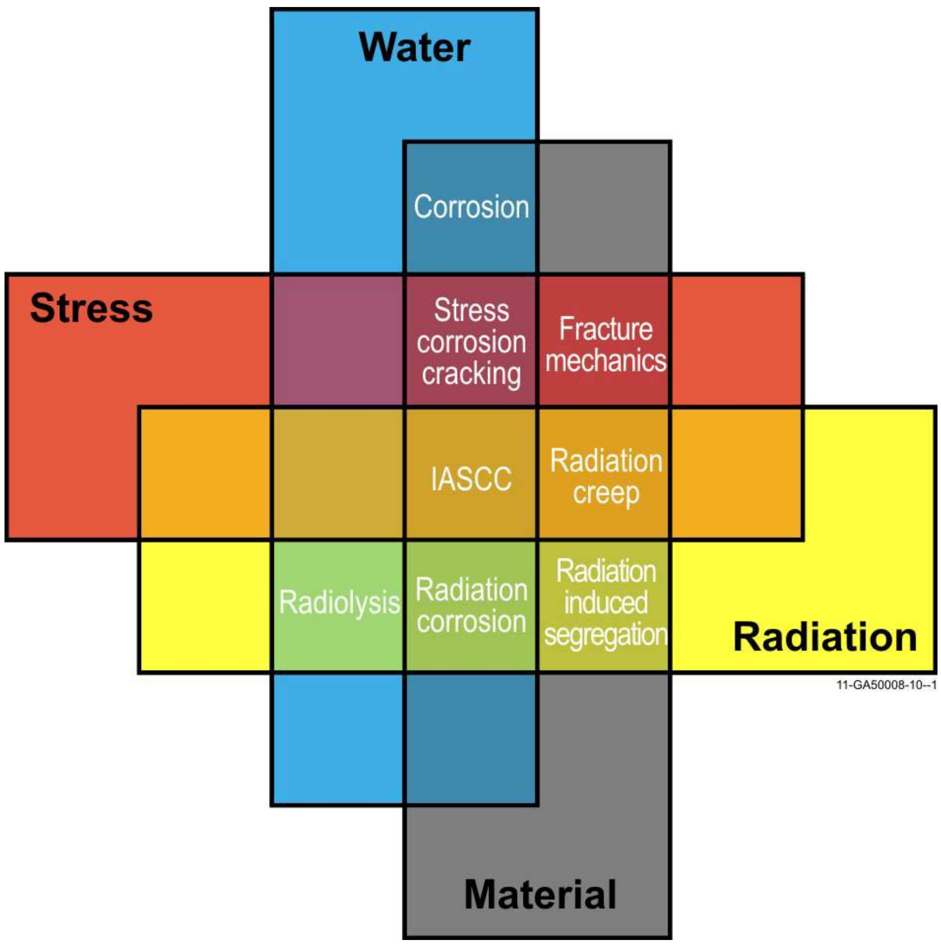

Figure 2. Source: DOE, Light Water Reactor Sustainability Program: Materials Aging and Degradation Technical Program Plan, at 5, Figure 2 (August 2014).

17. Although the federal government has now recognized the reality of these synergistic and interacting degradation mechanisms, the uncertainty and challenges posed by them have not been resolved. A good example of an unresolved technical concern is the interaction of fatigue, corrosion and shock loads for various highly embrittled (*i.e.,* radiation–induced) Reactor Vessel Internals (RVIs) – important reactor components that were not originally considered in the license renewal application (LRA) for the Indian Point facilities. Fortunately, during the course of these ASLB hearings on Indian Point the USNRC has now recognized and highlighted the importance of RVIs [*see generally* USNRC Report, "Final Interim Guidance LR-ISG-2011-04 Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors," NRC-ISG-2011-04 (May 28, 2013)] and has given applicants for license renewal *guidance* on how to address RVI aging.

18. Unfortunately, for now anyway, the applicant has been simply directed to comply with industry (EPRI) guidance [MRP-227-A] concerning *inspection* of the various RVI structures, fittings and components. Dependence on inspection alone, however, does not provide a solution to many of the uncertainties and technical challenges surrounding RVI aging and degradation. To begin with, depending on the type of component, inspection may not be possible for the entire component, or for the entire set of such components, given the location of the component(s) and their possible inaccessibility. For example, a visual inspection of the external head of a bolt does not necessarily provide insight into the integrity of the remainder of

the bolt which is not visible. Also, not all of the core support structures are accessible for inspection, and so surrogate structures have been chosen by Entergy to assess age-related degradation mechanisms. For example, the girth weld of the core barrel has been proposed by the applicant as a leading indicator for irradiation-induced embrittlement (IE) and irradiation-assisted stress corrosion cracking (IASCC) of the core support column caps [Dacimo, NL-14-093 (August 5, 2014), at 2-3], even though these components are very different and they may be exposed to different degradation mechanisms and shock loads.

19. Moreover, an inspection focused on one type of age-related degradation mechanism does not necessarily work for another ongoing degradation process that is affecting the same component, and the effect of shock loads on the integrity of various RVIs and piping systems is not addressed by inspections. The applicant's proposed approach to RVI aging management – as set out in NL-12-037 and in subsequent communications with the USNRC, and as condoned by the USNRC Staff's SSER2 and EPRI's MRP-227-A – is an inspection-based approach which 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 such weakened components could nonetheless fail as a result of an abnormal thermal or pressure shock load. In short, many of New York's main concerns about the cumulative and ongoing synergistic aging effects have simply been ignored by Entergy and the USNRC, and are not adequately addressed by MRP-227-A.

20. To date, the applicant, other licensees, and the USNRC have approached this issue without taking into account the simultaneous synergistic degradation forces at work on a particular component or material and without taking into account how a degraded component would respond in a non-steady state condition, such as the shock loads associated with a loss of coolant accident (LOCA) or SCRAM event. Examples include assuming that the core geometry remains intact and that the core internals, also known as the reactor pressure vessel internals (RVIs), do not deform or relocate, implicitly assuming in safety analyses that the in-core components and core geometry remain coolable.

21. Additionally, the applicant's evaluation of the fatigue life of the limiting reactor systems, structures and RVI components is inadequate. The USNRC has recently proposed to require all applicants for license renewal to evaluate the fatigue life of limiting components beyond those originally specified in NUREG/CR-6260, and to evaluate the effect of reactor coolant environment on the fatigue life of both external *and* internal (*i.e.,* RVIs) structures and systems [79 Fed. Reg. 69,884 (November 24, 2014)]. In this proceeding, the applicant agreed, in Commitment 49, to calculate the cumulative usage factors, adjusted for environmental degradation, (CUF_{en}) [Dacimo, Fred, Entergy, letter to Document Control Desk, USNRC, "Reply to Request for Additional Information Regarding the License Renewal Application," NL-13-122 (September 27, 2013), at 20]. These CUF_{en} calculations were done by Westinghouse, using their proprietary WESTEMS computer program, and were submitted to Entergy [CN-PAFM-12-35, Rev. 1, at 40-

43; CN-PAFM-13-32 (2013), at 7-9]. The initial analysis reported various CUF_{en} values that [redacted]. Subsequent results indicate that some of these CUF_{en} values are *extremely close* to the threshold for fatigue-induced crack initiation (*i.e.,* $CUF_{en} = 1.0$). For example, [redacted].[1] However, no "error analysis" of these results was presented even though a minor error could cause CUF_{en} to exceed unity. This is totally unacceptable since, as I have previously shown [Lahey Declaration in Support of NYS-26B/RK-TC-1B (September 8, 2010), at ¶ 11], there are many sources of modeling error in the calculations of WESTEMS which can affect these results. Numerous other components have CUF_{en} values [redacted].

22. Moreover, there has been no discussion of the effect of possible shock loads on the integrity of such severely fatigue-weakened structures. For example, Westinghouse has reported that the CUF_{en} for [redacted]

[1] [redacted]

[redacted] Even assuming this CUF_{en} calculation is accurate, it does not account for the possibility that the highly fatigued – but not yet having visible surface cracks – component may be exposed to an unexpected shock load that could cause it to fail. This is a good example of the type of "silo thinking" (*i.e.*, the fatigue and safety analyses are treated entirely separately) that NYS is concerned about.

23. While some age-related safety issues might eventually be resolved analytically or experimentally, in many cases it appears that the easiest and most cost-effective way to resolve them is to simply repair or replace the degraded structures, components and fittings, and this approach is what NYS has been proposing for some time (particularly for the degraded RVIs). In any event the applicant's "Revised Reactor Vessel Internals Program and Inspection Plan" (NL-12-037) and associated Commitment 49 (NL-13-122), which the USNRC Staff evaluated and approved in SSER2, do not resolve New York State's concerns over simultaneous and synergistic age-related degradation mechanisms that may affect various RVI components and structures.

24. The applicant continues to evaluate the various aging effects in "silos," in which an individual component, structure or fitting is evaluated for one degradation effect independently of other, synergistic aging mechanisms. As I have

repeatedly noted in this proceeding, the applicant has failed to evaluate the combined and synergistic effects of the various interacting degradation mechanisms. [*E.g.* Lahey Pre-filed Testimony Regarding Contention NYS-25 (December 22, 2011), at 12-13; Lahey Report in Support of Contention NYS-25 and Consolidated Contention NYS-26B/RK-TC-1B (December 20, 2011), at ¶ 19; Lahey Declaration in Support of Additional Bases for Contention NYS-25 (September 15, 2010), at ¶10].

25. Generally, the applicant continues to approach the problem of synergistic aging effects on RVI components through "condition monitoring" (*i.e.,* inspections per MRP-227-A) rather than a comprehensive approach which includes detailed analyses and/or "preventative actions" (*i.e.,* repair and replacement) ["Revised Reactor Vessel Internals Program and Inspection Plan," Attachment 1 to NL-12-037, at 5]. This approach means that aging effects and degradation will not be addressed until cracks or other degradation mechanisms (*e.g.,* wear) have been directly observed ["Revised Reactor Vessel Internals Program and Inspection Plan," Attachment 1 to NL-12-037, at 5]. In short, component degradation will be addressed only after it occurs. The applicant incorrectly concludes that preventative actions, such as component replacement, are not required for most RVI components because cracking or other flaws can be detected before the failure of a component affects the safe operation of the reactor. Apparently, this is because the applicant's approach generally assumes that IP2 and IP3 will continually operate during the 20-year period of extended operation within normal "steady state"

parameters, and ignores the possibility that significantly fatigued, embrittled and corrosion-weakened, or otherwise degraded, RVI components, structures or fittings might be exposed to various shock loads which can cause them to deform or relocate such that core cooling is seriously degraded. In fact, the applicant's reactor safety analyses implicitly assume that the reactor core will maintain a coolable geometry, notwithstanding the degradation and possible deformation or relocation of various RVI components and the possible flow blockages and degraded core cooling which may occur, even during emergency core cooling system (ECCS) operation.

26. The applicant's "Revised Reactor Vessels Internals Program" acknowledges that other PWRs have experienced material degradation and failure of multiple RVI components, including cracking of baffle former bolting, cracking in other important bolting, wear in thimble tubes, and potential wear in control rod guide tube guide plates [Attachment 1 to NL-12-037, at 8]. The applicant *has* committed to replace one affected IP2 component – the degraded guide tube support pins (split pins) – by 2016 [SSER2, at 3-36; "Revised Reactor Vessel Internals Program and Inspection Plan," Attachment 1 to NL-12-037, at 8; Commitment 50, Attachment 1 to NL-13-122, at 7]. Interestingly, the applicant has agreed to replace the IP2 split pins, even though they were replaced once already in 1995, and even though the applicant claims that the failure of a split pin would not compromise reactor vessel functions [Response to RAI 16, Attachment 2 to NL-12-166, at 1]. However, for many other affected RVI components, the applicant proposes a "wait-and-see" approach.

27. For example, the applicant acknowledges that "cracking of baffle former bolts is recognized as a potential issue for the Indian Point units" ["Revised Reactor Vessel Internals Program," Attachment 1 to NL-12-037, at 8], but the applicant does not propose to replace those bolts, only to continue monitoring them ["Revised Reactor Vessel Internals Inspection Plan," Attachment 2 to NL-12-037, at 40, tbl. 5-2]. In fact, the applicant has not yet developed inspection acceptance criteria for baffle former bolts in either IP2 or IP3 [SSER, at 3-20]. Instead, the application 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; Response to RAI 5, Attachment 1 to NL-12-089, at 11].

28. 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]. 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]. 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]. 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 ["Revised Reactor Vessel Internals Inspection Plan," Attachment 2 to NL-12-037, tbl. 5-4, at 51]. 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]. 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. Moreover, the applicant fails to consider the possibility that a shock load (*e.g.*, due to LOCA) may cause the sudden failure of the remaining intact clevis bolts, which, in turn, may lead to an uncoolable core geometry. 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, an approach which is *totally unacceptable* in my professional engineering judgment, and is contrary to the requirements of 10 C.F.R. § 54.21.

29. The applicant's approach to analyzing the lower support structures' functionality and fracture toughness is similarly flawed. [Response to RAI-11-A,

Attachment 1 to NL-13-052, at 1-4]. 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, based on the lack of documented fractures of core support columns, that "only a limited number of columns could actually contain flaws of significant size." The applicant further assumed that the columns would be subject to "nominal normal operating stresses" [SSER2, at 3-43]. 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." 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 [NUREG/CR-7184, at xv (March, 2014] which showed the extreme sensitivity of crack growth rate and fracture toughness to irradiation. 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.

30. That is, 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 [Chen, *et al.*, at xv ("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; Stevens, *et al.*, (October 2014) at 9-10]. 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 [NUREG/CR-7153, Vol. 2: Aging of Core Internals and Piping Systems, at 161-188]. Indeed, the EMDA report prepared by the USNRC 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" [NUREG/CR-7153, Vol. 2: Aging of Core Internals and Piping Systems, at 183].

31. The applicant and USNRC 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; Response to RAI-11C, Attachment 1 to NL-14-093, at 1-4; Response to RAI-11B, Attachment 1 to NL-13-122, at 2-4; Response to RAI-11A, Attachment 1 to NL-13-052, at 1-3; Response

to RAI-11, Attachment 1 to NL-12-134, at 11-12]. 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). Thus, the support columns would not be a "bounding" component in a "bounding" location subject to "bounding" degradation mechanisms.

32. Furthermore,, to the extent that the applicant proposes to rely on visual (VT-3) inspection techniques for many RVI internals, the significant shortcomings of this technique to detect material cracking, degradation or wear prior to failure have been noted by USNRC staff [Tregoning, at 2-3; Case, at 1], and 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].

33. I would also like to add a note about safety margins. As reactors and their constituent components age, it becomes more important to preserve, rather than erode, operational safety margins. As discussed above, uncertainties exist, and accidents or unanticipated events can occur, and calculational/modeling mistakes are possible. For example, USNRC only recently became aware that certain methodologies prescribed in its NUREG-0800 Branch Technical Position (BTP) 5-3 for estimating the initial fracture toughness of reactor vessel materials may be non-conservative. [*See, e.g.,* Troyer and Devan (2014); Salas (2014); Kirk

and Sheng, USNRC (2014); *see also* Pressurized Water Reactor Owners Group, BTP 5-3 Industry Issue (proprietary)]. Various nuclear plants that received their construction permits before August 1973 relied on BTP 5-3 by to estimate reference temperature (RT_{NDT}) and upper shelf energy (USE) values in order to demonstrate compliance with ASME Code and USNRC margins for reactor pressure vessel integrity. Because RT_{NDT} and USE values serve as starting points for determining pressure-temperature (PT) heatup / cooldown curves, the consequences of this recent revelation could be significant, and may impact IP2 or IP3's reactor pressure vessel and its fittings integrity and plant operational limitations. Since unexpected errors of this type do occur, maintaining safety margins via not operating the plant too close to CUF_{en} and repair or replacement of aging parts prior to the end of the plant's design life (particularly for RVIs) would help to guard against potentially adverse impacts due to precisely this type of unexpected non-conservatism in flawed safety evaluations.

CONCLUSION

34. The applicant's "Revised Reactor Vessel Internals Program and Inspection Plan" as presented in NL-12-037, and as subsequently modified, is inadequate to support the proposed extended operation of both IP2 and IP3 for an additional 20 years. Neither the USNRC nor the applicant has adequately addressed our most significant concerns about the *interacting and synergistic effect* of the various age-related degradation mechanisms, and the effect of various possible *shock loads* on highly degraded components (and the implications for core

cooling). MRP-227-A, likewise, does not adequately address these issues. Indeed, everything continues to be done in "silos." Moreover, rather than the pre-emptive repair or replacement of degraded structures, components and fittings, the applicant proposes to manage the considerable uncertainty presented by the various synergistic aging effects on nuclear reactor operation beyond 40 years by operating with either greatly reduced safety margins (*e.g.,* accepting very large CUF_{en}), or simply waiting for failures or detectable degradation to occur (*e.g.*, by relying on RVI inspections pursuant to MRP-227-A to "manage" the aging issues and concerns). The applicant's approach implicitly assumes that: (1) synergistic aging effects will not be greater than the sum of their individual "siloed" effects; (2) It is acceptable to operate the reactors with significantly reduced safety margins; (3) The reactor core geometry (and, thus, core coolability) will not be affected by RVI component degradation and/or failure; and, (4) IP2 and IP3 will continuously operate within normal parameters and will experience no unexpected shock loads (*e.g.,* a secondary side LOCA), which may cause failures of a degraded external or internal (RVI) systems, structures, components or fittings. Considering the possibly catastrophic effects if even one of these assumptions proves incorrect, I believe that the applicant has failed to establish an acceptable Aging Management Program (AMP) so that IP2 and IP3 can be safely operated for an additional 20 years.

Pursuant to 28 U.S.C. § 1746, I declare under penalty of perjury that the foregoing is true and correct.

February 12, 2015
Saint Augustine, Florida

R. T. Lahey

Dr. Richard T. Lahey, Jr.
The Edward E. Hood Professor Emeritus of Engineering
Rensselaer Polytechnic Institute, Troy, NY

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