UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

In re:	Docket Nos. 50-247LR and 50-286LR
License Renewal Application Submitted By	ASLB No. 07-858-03-LR-BD01
Entergy Indian Point 2, LLC, Entergy Indian Point 3, LLC, and Entergy Nuclear Operations, Inc.	DPR-26, DPR-64 September 15, 2010

DECLARATION OF DR. RICHARD T. LAHEY, JR.

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

1. I am the *Edward E. Hood Professor Emeritus of Engineering* at Rensselaer Polytechnic Institute (RPI) in Troy, New York, a member of the National Academy of Engineering (NAE), a Fellow of the American Nuclear Society (ANS) and the American Society of Mechanical Engineers (ASME), and an expert in matters relating to the operations, safety, and the aging of nuclear power plants. I have previously submitted a declaration in support of the Notice of Intention to Participate and Petition to Intervene filed by the State of New York in this proceeding on November 30, 2007, which sets forth my qualifications in detail. By way of summary, I have held various positions in the nuclear industry and academia, and served on numerous panels and committees for the U.S. Nuclear Regulatory Commission (USNRC), Idaho National Engineering Laboratory (INEL),

Oak Ridge National Laboratory (ORNL), Electric Power Research Institute (EPRI), National Aeronautics & Space Administration (NASA), and the National Research Council (NRC). I have also held various positions in the nuclear industry and academia, including Dean of Engineering and Chair of the Department of Nuclear Engineering & Science at RPI. I have also been the lead engineer and manager of various departments responsible for safety analyses, Heat Transfer Mechanisms and Core & Safety Development for the General Electric Company (GE), including both military (*i.e.*, Naval) and commercial nuclear reactors. Over the last 40 years, I have also published numerous books, monographs, chapters, articles, studies, reports, and journal papers on nuclear engineering and nuclear reactor safety technology, and most of these publications have been peer reviewed. My *curricula vitae*, which more fully describes my educational and professional background and qualifications, is attached to this declaration and is available at: http://www.rpi.edu/~laheyr/laheyvita.html.

2. The factual statements and the expression of opinion in this declaration are based on, among other things, my best professional knowledge, my extensive professional experience in nuclear reactor technology, and my review of Entergy's April 2007 License Renewal Application, Entergy's July 15, 2010 submission to the Atomic Safety and Licensing Board (conveying Entergy's July 14, 2010 License Renewal Application Amendment No. 9 (communication NL 10-063 to NRC Staff), and other documents referenced in this declaration.

3. This declaration documents my recommendations and concerns about Entergy's new July 14, 2010 aging management program discussed in NL-10-063 and the detection and management of the embrittlement and/or corrosion-induced cracking of important structures and fittings within the reactor pressure vessel (RPV), and the age-related safety issues for the Indian Point reactors. My recommendations and concerns are based on my extensive experience and expertise in the field of nuclear reactor thermal-hydraulics and safety. Moreover, this input is based on, and expands upon, many of the concerns that I raised in my prior ASLB declaration concerning the re-licensing of IP-2 & 3 .

4. As I stated in my initial November 2007 declaration on these issues in support of the State of New York's Contention 25, in my professional judgment the applicant failed to demonstrate that it had adequately accounted for the aging phenomena of embrittlement for components inside the reactor pressure vessels at Indian Point Unit 2 and Unit 3. My professional judgment has not fundamentally changed based upon Entergy's July 14, 2010 submission of License Renewal Application, No. 9 [NL 10-063].

The Indian Point Reactors

5. Entergy's Indian Point Units 2 & 3 are currently under consideration for 20-year life extensions beyond their original 40-year design life. If approved, these plants will be licensed for operational levels of about 48 effective full power

years (EFPY). These Westinghouse designed plants are 4-loop PWRs and they are currently[1] rated at power levels of 3,216.4 MW_t.

6. They are sited on the east bank of the Hudson River in Buchanan, NY, which is about 24 miles north of the New York City (NYC) border.[2] Because of their close proximity to a very highly populated area (*i.e.*, the NYC metropolitan area), which is also the world's leading financial center, it is vital that IP Units 2 & 3 fully and unambiguously meet all reasonable and applicable criteria for safe operation. This is particularly true when considering life extension, since, like metal fatigue failures, failures due to embrittlement are much more likely as the plants age.

7. The USNRC Staff have prepared a guidance document entitled the "Generic Aging Lessons Learned (GALL) Report," NUREG-1801, Rev. 1 (2005), in which Staff seeks to describe various Aging Management Programs (AMP) for the extended operations of nuclear power plants. That USNRC document does not specifically describe aging management programs for the embrittlement of internal components within the reactor pressure vessel (RPV), including, but not limited to, the: control rods and their associated guide tubes, assemblies, and seal welds, and many important in-core structures and fittings which will be discussed

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[1] The USNRC approved a stretch power increase of 3.26% for IP-2 in 2004 and a 4.85% increase for IP-3 in 2005; IP-2 and IP-3 also received 1.4% power uprates in 2003 and 2002, respectively.

[2] By way of additional reference, the Indian Point reactors are approximately 37 miles north of Wall Street in lower Manhattan, 3 miles southwest of Peekskill, 5 miles northeast of Haverstraw, 16 miles southeast of Newburgh, 17 miles northwest of White Plains, 23 miles northwest of Greenwich, Connecticut, 37 miles west of Bridgeport, Connecticut, and 37-39 miles north-northeast of Jersey City and Newark, New Jersey.

subsequently. *See* GALL, Chapter XI (Aging Management Programs); *see also* Entergy NL-10-063, at pg. 84 ("Revision 1 of NUREG-1801 includes no aging management program description for PWR reactor vessel internals."). Although the USNRC Staff did not include an aging management program to address reactor vessel internal components in GALL, I believe that all important safety concerns must be addressed to assure the health and safety of the American public during extended plant operations, and that the safety review for the requested licenses for extended operations of the two Indian Point reactors should include an analysis of the embrittlement of components inside the reactor pressure vessels and the implementation of a meaningful program to manage the embrittlement of such components during periods of extended reactor operation.

Embrittlement Phenomena

8. As previously discussed in my initial November 2007 declaration (¶¶ 6-18), one of the key age-related phenomena that must be considered in Entergy's License Renewal Application (LRA) is the embrittlement of the reactor pressure vessel's (RPV's) internal metal structures and fittings, which occurs due to the extended irradiation (*i.e.*, the neutron fluence, which is the neutron flux times the duration of the irradiation process) that will be experienced by these metal components, particularly those located within the so-called "belt line" region of the RPV (*i.e.*, the region of the RPV that is closest to the core) where the neutron flux is the highest. In addition, the reactor vessel internals may experience flow/thermal-transient-induced fatigue degradation, as well as embrittlement-induced

degradation due to various radiation damage mechanisms [Was, 2007], including damage due to void swelling which may occur because of transmutation and other effects [Was, 2007; NUREG/CR-6897; Barnes, 1964]. Also, some in-core components may experience irradiation assisted stress corrosion cracking (IASCC) [WCAP-14577, Rev.1-A, pgs,3-6 & 3-8, 2001], and/or primary water enhanced stress corrosion cracking (PWSCC) [WCAP-14577, Rev. 1-A, pgs. 3-4 & 3-5, 2001] due to prolonged exposure to the high temperature (*i.e.*, T > 400°F) borated primary coolant. In addition, cast austenitic stainless steel reactor components (*e.g.*, some reactor piping/fittings, pump casings, pressurizer spray heads, etc.) and various reactor pressure vessel (RPV) internals (*e.g.*, the upper mixing vanes, and the upper/lower core assemblies and support columns) are composed of a duplex stainless steel which contains both austenitic and ferretic phases, and are thus subject to embrittlement due to thermal aging [WCAP-14577, Rev.1-A, pgs. 3-12 & 3-13, 2001; EPRI Report TR-106092, 1997; NUREG/CR-4513, Rev.1]. Moreover, the heat affected zone (HAZ) of stainless steel and nickel alloy weldments may be more sensitive to embrittlement mechanisms than the base metals being joined [Hawthorne et al., 1986; NUREG/CR-6960; Carey, 2006].

9. In any event, embrittlement causes metals to lose ductility and become more susceptible to failures due to cracking or fracture. Also, reactor operations may be restricted by embrittlement since the temperature at which the embrittled metal structures and fittings change from non-ductile to ductile behavior (*i.e.*, the so-called nil ductility temperature, NDT) will increase as the reactor operates over

time and ages. Conversely, during reactor operations, the temperature at which the embrittled metal structures and fittings change from ductile to non-ductile behavior will increase. Significantly, this phenomenon implies that embrittled RPV internals will become progressively more vulnerable to failure due to thermal shocks as reactor operations continue. Obviously, irradiation damage is a serious age-related phenomena, and one that will not be annealed-out (*i.e.*, be healed) during reactor operations since PWR operating temperatures are too low for this to occur.

10. How the rather complex metal degradation mechanisms associated with fatigue and irradiation interact is still an area of active research, but it is known that the radiation-induced damage on reactor vessel internals can be extensive, since they may experience a neutron fluence of up to 10^{23} n/cm^2 at neutron energy (E) levels of $E \geq 1$ MeV (i.e., $\geq$100 dpa)[3] [Was, 2007; Robinson, 2008] by the end of life (EOL) for extended operations. Indeed, the EOL Charpy impact Upper Shelf Energy (USE) for some thermally-aged cast stainless steel in-core components could be as low as 28 ft-lb$_f$ [WCAP-14577, pg. 3-13, 2001], which is well below the acceptable ASME code-specified minimum of 50 ft-lb$_f$, and even the 43 ft-lb$_f$ variance proposed by Westinghouse [WCAP-13587, Rev. 1, 1993], and endorsed by the ACRS [ACRS Letter, 9/23/09], as being acceptable for an Indian Point reactor pressure vessel (RPV) at the EOL.

[3] Displacements per atom (dpa) is a measure of radiation damage to a material.

11. Given that a variance from the applicable ASME Code is needed for the RPV itself, it is noteworthy that the RPV's inner wall experiences much less fluence than many of the in-core metal components that are inside the RPV and in closer proximity to the core and fuel rods. That is, the Indian Point RPVs are expected to experience ~1.9 10^{19} n/cm^2 by the end of extended operations [Entergy Letter, 3/8/10], which, as noted previously, is much less than the ~10^{23} n/cm^2 fluences that may be experienced by some RPV internals. The obvious conclusion is that RPV internal components will be significantly embrittled during the period of extended operations of IP-2 and IP-3; and much more so than the RPV inner wall. In-core components which are particularly vulnerable include the: core baffle, intermediate shells, former plates and bolts (particularly the re-entrant corners), and including the baffle-to-baffle bolt locations, the core barrel-to-former bolt locations and baffle-to-former bolt locations, core barrel (and its welds), lower core plate and support structures, clevis bolts, fuel alignment pins, thermal shield, and the lower support column and mixer. As discussed below, such in-vessel components also include the control rods and their associated guide tubes, plates, and welds.

12. Entergy acknowledges that, "PWR internals aging degradation has been observed in European PWRs, specifically with regard to cracking of baffle-former bolting." NL-10-063, at pg. 89. Indeed, EPRI has stated that, "considerable amount of PWR internals aging degradation has been observed in European PWRs." EPRI MRP-227, at A-4. Entergy also states: "As with other U.S. commercial PWR

plants, cracking of baffle former bolts is recognized as a potential issue for the [Indian Point] units." NL-10-063, at pg. 89. Moreover, material degradation has also been observed in control rod guide tube alignment (split) pins. EPRI MRP-227, at A-4.

13. As noted previously, any degradation in ductility will adversely effect the possible pressure-temperature (p-T) operating conditions (*i.e.*, there will be an increase in the nil ductility temperature, NDT). Also, it will adversely affect the ability of embrittled in-core components to withstand thermal shock transients and the decompression shock loads associated with a postulated design basis accident (DBA) loss of coolant accident (LOCA). Moreover, the metal structures and fittings within the RPV are subjected to many of the same transients (*e.g.*, a SCRAM - a rapid insertion of the control rods causing a rapid decrease in the core's power level), which are known to cause fatigue-induced degradation of the primary side piping, nozzles and structures [*see, e.g.*, my Sept. 8, 2010 ASLB Declaration on metal fatigue]. Hence, fatigue will also degrade the strength and ductility of many of the metal structures and fittings within the RPV, but virtually no fatigue analyses of this type have been presented by Entergy in their application for the extended operations of IP-2 and IP-3 [*see, e.g.*, my Sept. 8, 2010 ASLB Declaration on metal fatigue]. Entergy's recently-submitted aging management program amendment [NL-10-063], does not call for an analysis of the synergistic impacts of these different aging effects; I believe that this is a very serious omission and that

this deficiency should be corrected during the ASLB's re-licensing hearings for Indian Point Units 2 & 3.

14. In addition, severe pressurized thermal shocks can occur during postulated accidents which may rapidly depressurize the secondary side of the reactor system and cause a SCRAM. While pressured thermal shock of the reactor pressure vessel (RPV) itself was discussed in Entergy's re-licensing applications, there was no indication of what new accident analysis was done (if any) in which both embrittlement and fatigue were explicitly taken into account when assessing the effect of the accident-induced transient loads on RPV internals. This is quite important since thermal shock may cause highly embrittled and fatigued in-core components to fail, perhaps leading to an uncoolable core geometry and core melt. In my opinion, one of the most serious omissions from the USNRC's GALL Report and Standard Review Plan is that there was no mention at all of how highly embrittled and fatigued internal RPV structures and fittings will respond to the severe transient decompression shock loads associated with a DBA LOCA and the subsequent thermal shock loads associated with the discharge into the primary side of the reactor of relatively cold emergency core coolant (ECC) from the accumulators. It is well known [*e.g.*, Tong & Weisman, pgs. 147-149, 1970] that a strong decompression shock wave, created during the subcooled blowdown phase of a DBA LOCA, can cause significant transient pressure differentials across various internal RPV structures. Detailed experiments (*e.g.*, LOFT) and analyses have

shown that, when ductile, these in-vessel metal structures are not likely to fail or deform to the point where a coolable geometry can not be maintained for the core. In contrast, no such experiments and analyses have been presented by Entergy to justify that highly embrittled and fatigued in-vessel components will not fail and that a coolable core geometry will be maintained subsequent to a DBA LOCA. This is a very serious and, in my opinion, a totally unacceptable omission since brittle and fatigue-weakened structures are known not to tolerate shock loads well (*e.g.*, they may break loose or fracture) and, if a coolable geometry of the core is not maintained, it can melt, releasing a significant amount of radiation and possibly causing a breach of the lower head of the RPV. It is incumbent on Entergy to prove this will not happen, since Federal regulation, 10 C.F.R. § 54.4(a), clearly states that reactor operators must: "provide the capability to shut down the reactor and maintain it in a safe shut-down condition." It is also important to stress that while the USNRC Staff noted in their review of the Safety Evaluation Reports (SERs) for IP-2 & 3 that, "...if certain reactor vessel internals failed, they could potentially inhibit core coolability during an accident." [SER, Dockets 50-247 and 50-286, pg. 2-40, (Nov. 2009)], their primary focus was on the reactor's sample tubing systems, and did not encompass the more critical RPV internal components such as those listed previously in paragraph 11. Also, the industry programs, which Entergy has proposed to follow under the AMP for RPV internals, are mute on the serious age-related safety concern of the coolability of PWR cores subsequent to an accident-induced failure of highly embrittled and fatigued PRV internals. Unfortunately, the

new aging management plan submitted by Entergy [NL-10-063] does not address or manage the synergistic aging effects of embrittlement and fatigue on RPV internals and the impact of accident-induced shock loads on these components..

15. In summary, I believe the USNRC has made a major error in not highlighting the above age-related safety issues in the GALL Report [NUREG-1801. Rev. 1] and the Standard Review Plan [NUREG-1800, Rev. 1]. Perhaps this is because of "stove piping" of the safety evaluations and the various AMP issues, in which each is discussed and analyzed separately, and thus the integrated and synergistic effect of accident-induced shock loads on highly embrittled and fatigued RPV internals was not considered. In fact, the aging phenomena of embrittlement and fatigue acting together and in concert with one another has apparently not been considered. In addition, there has apparently been some confusion associated with the USNRC's leak- before-break (LBB) ruling [NUREG/CR-4572, NUREG-1061, Vol. 3, 10 CFR 50, Appendix-A], in which the USNRC's the rules were changed for some of the dynamic ex-vessel LOCA loads (*i.e.*, for the pipe whip and jet loads) associated with the design basis accident (DBA). In particular, the USNRC now allows reactor operators to not use the ex-vessel loads associated with a double-ended pipe break if they can show that the primary side piping would be expected to leak well before it breaks. It is significant to note that the LBB ruling does not apply to the in-vessel DBA LOCA decompression and thermal shock loads. Unfortunately, the implications of this ruling have apparently been misunderstood by many in the nuclear industry, and it appears to have led USNRC staff to not be

overly concerned about the effect of DBA LOCA decompression loads, and emergency core cooling system (ECCS) or secondary side LOCA induced thermal shock loads, on highly embrittled and fatigued metal components within the RPV. As a consequence, it appears that this significant safety issue has been totally overlooked in the Standard Review Plan for Licensing Renewal Applications (LRAs).

Integrity of Control Rods, Guide Tubes, and Plates

16. Any aging management program concerning the embrittlement of reactor pressure vessel internals should include control rods and their associated guide tubes, plates, and welds within the scope of such program. The control rods and their associated guide tubes, plates, and welds are also very important RPV internals and their integrity is an extremely important safety concern. They are located in the core region of the RPV, and are inserted into the RPV through the upper head via so-called stub tubes. Their function is to absorb excess fission neutrons (*i.e.*, those not need to achieve a chain reaction) so that the power level of a reactor can be controlled.

17. With respect to control rods and their associated guide tubes and plates, of particular concern is the significant and reoccurring stress corrosion cracking that has been observed in the J-groove seal welds on the control rod drive (CRD) stub tube penetrations of the upper head of PWR RPV's. By way of example, according to USNRC documents, earlier this year the operator of the Davis-Besse reactor, "found evidence boric acid deposits and indications of primary water stress corrosion cracking in their nozzles and welds." NRC Staff, Division of Component

Integrity, NRC Perspectives on PWR Materials Issues, at 7 (June 2010) ML101520577. According to the USNRC, "the timing and extent of cracking was unexpected." *Id.*; *see also* NRC News, III-10-123 (May 26, 2010). It is significant to note that this type of leakage had been found earlier (*i.e.*, in 2002) at Davis Besse, and it nearly resulted in a major LOCA due to a massive corrosion-induced failure of the upper RPV head. In fact, the stress corrosion cracking of these type welds is widely considered to be one of the biggest challenges currently facing operating PWRs [NEI 03-08 [Addenda], at D-5 (June 2009)].

18. In addition, because of geometric considerations, many PWRs (including IP-2 and IP-3) can not meet the USNRC's required minimum coverage for the non-destructive testing (NDT) of these important J-groove welds [Walpole, 2009], and thus the integrity of these stub tube welds can not be confirmed.. It appears that to help address this chronic problem Entergy has ordered two new RPV heads [Telecon-USNRC/Entergy Report, March 18, 2008], but they have not yet been scheduled for installation at Indian Point [Telecon-USNRC/Entergy, March 18, 2008]]. In any event, unlike the rather superficial treatment given this important safety concern by Entergy [NL-10-063], I believe that a tangible, enforceable, and viable aging management program must developed and implemented before re-licensing the Indian Point reactor plants for extended operations since the integrity of these welds must be assured. If not, due to the leakage of borated primary coolant through cracked welds, there can be aggressive corrosion and wasting of the unclad outer surface of the upper head of the RPV

(such as the serious event that occurred at Davis-Besse and was identified in 2002). Worse yet, there might be an inadvertent control rod ejection (*e.g.*, due to a massive failure of the welds in the upper RPV head), which could cause a relatively major reactivity excursion, leading to core melting and significant radiation releases.

Baseline Inspections

19. With respect to Entergy's proposal to conduct baseline examinations of the RPV internals, I note that I previously called on Entergy to conduct such examinations and for NRC Staff to require the conduct of such examinations before entering the period of extended operations. In particular, in my initial November 2007 declaration in support of the State of New York's motion to intervene in this proceeding, I stated:

> As part of the relicensing review, and prior to the commencement of any extended operations, the NRC should require Entergy to conduct a thorough baseline inspection of both IP2 and IP3. These inspections should involve both visual and physical characterization and the non-destructive testing (NDT) of structures and components, including but not limited to the RPV, the RPV heads/fittings, the control rod drive mechanisms and associated RPV penetrations, most RPV internal hardware, and all key welds and fittings in the primary and secondary systems of the reactors.
>
> ....Thorough baseline inspections should examine the changes that the plants' systems, structures, and components have experienced during the first three and a half decades of operation. Without proper inspections, the NRC, the applicant, and the public will not have the necessary information to assess whether these plants are in any condition to continue to operate for an additional 20 years.... Routine, sound engineering practice requires a thorough baseline inspection for the license extension of a nuclear power plant to establish the state of the reactor facility, systems, structures and components at the end of their design life and disclose degradation which may have occurred. The failure to conduct thorough baseline inspections prior to life

extension is reckless and runs counter to rudimentary engineering practice.

See November 2007 Declaration of Richard Lahey, at ¶¶ 24, 25; *see also* State of New York Notice of Intention to Participate and Petition to Intervene, at pgs. 217-220, State of New York Contention 23 (Baseline Inspections).[4] While Entergy now seems to have embraced the concept of baseline inspections for RPV internals, the text of the proposed aging management plan as set forth in NL-10-063 lacks sufficient details to know when the baseline inspections of the RPV and its internals will begin or when they will be completed. In my opinion, this should occur before the onset of extended operations.

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[4] Both Entergy and the USNRC Staff opposed this proposal and the State's Contention 23.

20. For the all reasons given above, I do not believe the Entergy's July 15, 2010 communication to the Board [NL-10-063] concerning a new AMP for RPV internals is adequate to address the new safety concerns and technical issues that I have raised herein.

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

September 15, 2010
Troy, New York

R. T. Lahey

Dr. Richard T. Lahey, Jr.

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