UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
OFFICE OF THE SECRETARY

ATOMIC SAFETY AND LICENSING BOARD

In the Matter of

ENTERGY NUCLEAR OPERATIONS, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 & 3

Regarding the Renewal of Facility Operating Licenses
No. DPR-26 and No. DPR-64 for an Additional 20-year Period

USNRC Docket Nos.
50-247 & 50-286

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 currently the *Edward E. Hood Professor of Engineering* at Rensselaer Polytechnic Institute (RPI) in Troy, New York. I hold 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 PhD. in Mechanical Engineering from Stanford University. At RPI, I have served as both the Dean of Engineering and the Chairman of the Department of Nuclear Engineering & Science. 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 Chair of the Thermal-Hydraulics Division; the American Society of Mechanical Engineers (ASME), where I was Chair of the Nuclonics 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 was also the editor of the *Journal of Nuclear Engineering & Design*. 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), and the Electric Power Research Institute (EPRI). I am a member of the National Academy of Engineering (NAE), and have been elected Fellow of both the ANS and the ASME. Over the last 40 years, I have 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 available at: **http://www.rpi.edu/~laheyr/laheyvita.html**.

2. I am very familiar with the operation of, and safety analyses associated with, pressurized water nuclear reactors (PWRs), the type of reactor currently in operation at the Indian Point (IP) site in Buchanan, New York.

3. I have reviewed the license renewal application for the two Indian Point nuclear reactors that was submitted by Entergy Nuclear Operations, Inc. (Entergy). These reactors are known as Indian Point-2 (IP2) and Indian Point-3 (IP3). In my opinion, and as I explain more fully below, the United States Nuclear Regulatory Commission (USNRC) should grant a hearing on at least five aging management and safety issues: (1) embrittlement of the reactor pressure vessels and associated internals; (2) the potential for fatigue failure; (3) inadequate baseline inspections of IP2 and IP3; (4) the need for enhanced inspections because of an inadequate water/cement ratio in the containment structures; and, (5) the risk of a terrorist attack on the spent fuel storage pools.

4. As nuclear power plants exceed their normal design life of 40 years, key structural components degrade – they begin to wear out. To assure safe operation during a 20-year life extension of the IP reactors, it is imperative not to erode the original design safety factors in the

interest of keeping the plant running (e.g., by doing "best estimate" rather than licensing-type safety evaluations). Rather, one must ensure that the record reflects a thorough licensing analysis, which, unfortunately, the present application does not accomplish.

5. To ensure that aging systems, structures, and components do not become compromised during the term of an extended operating license, the USNRC has recognized the importance of developing and implementing an effective aging management program for such systems, structures and components. *See generally* 10 C.F.R. Part 54. Among other things, it is important to develop and implement an effective aging management program to ensure that the facility's systems, structures and components do not become compromised due to neutron-induced embrittlement or fatigue. Entergy's license renewal application (Appendix B - Aging Management Program and Activities), however, does not contain a commitment to the development and implementation of an effective aging management program to preclude failures due to embrittlement and fatigue.

Embrittlement of Reactor Pressure Vessels (RPVs) and Associated Internals

6. The first issue that the USNRC should conduct a hearing on for Entergy's relicensing application for IP2 and IP3 is the phenomena of embrittlement of the reactor pressure vessels (RPVs) and the associated internals at both plants. Embrittlement of the RPVs and their associated internals is one of the most important age-related phenomena that the USNRC must consider in its review of Entergy's relicensing application. Failure to carefully consider the effects of embrittlement could result in system/component failures which could lead to a meltdown of the core and a release of a significant amount of radiation subsequent to various accident scenarios. As explained in more detail below, some of the RPV structures and internal

components in IP2 and IP3 have serious embrittlement concerns which are not adequately addressed in Entergy's relicensing application.

7. Entergy currently operates two pressurized water nuclear reactors (PWRs) at Indian Point. PWRs have water (i.e., the primary coolant) under high pressure flowing through the core in which heat is generated by the fission process. This heat is absorbed by the coolant and then transferred from the coolant in the primary system to lower pressure water in the secondary system via a heat exchanger (i.e., a steam generator), which, in turn, generates steam on the secondary side. These Nuclear Steam Supply Systems (NSSS) are located inside the containment structure. After leaving the containment building, via main steam piping, the steam drives a turbine, which turns a generator to produce electrical power.

8. The RPV is the primary container that holds the core (i.e., the nuclear fuel); it also serves as a key part of the reactor coolant's pressure boundary. In addition, there is a pressurizer on the primary side that performs several functions. In particular, it maintains the operating pressure on the primary side of the nuclear reactor and compensates for variations in reactor coolant volume during load changes during reactor operation, as well as reactor heat-up and cool-down. The reactor coolant also moderates the neutrons produced in the core since a PWR will not function unless the neutrons are sufficiently moderated (i.e., slowed down due to collisions with the hydrogen molecules in the primary coolant).

9. Embrittlement refers to the change in the mechanical properties (or structure) of materials, such as metals, that can occur over time under the bombardment of neutrons. The degree of exposure to neutrons is normally expressed in terms of a "fluence" (i.e., the neutron flux times the duration of the irradiation process). The exposure to neutrons causes damage to metals and makes them more brittle so that they become more susceptible to failure due to cracking or fracture. For the relicensing of the two reactors at Indian Point, embrittlement of the

RPVs and their associated internals is an important age-related safety concern, particularly in the so-called "belt line" region of the RPV, which is the region that is the closest to the reactor core.

10. For a PWR to operate safely, the metals involved need to be sufficiently ductile, which means that they must be able to deform without experiencing failures. When metals, such as steel, experience a significant fluence, which happens to the materials in close proximity to the reactor core (e.g., the steel RPV and associated internals), the temperature required for them to maintain some ductility is increased as the metal is continually bombarded by a neutron flux. The temperature at which there is a marked change from ductile to non-ductile behavior is often called the "nil ductility temperature" (NDT). However, even for temperatures well above the NDT, the irradiated metals continue to be damaged and further embrittled due to the neutron bombardment. Indeed, the neutron damage will not be annealed out (i.e., be neutralized) unless the damaged metals are taken to temperatures that are well above PWR operating temperatures.

11. One important safety concern associated with embrittlement is the ability of the metals to withstand a thermal shock event. A thermal shock can occur in various ways, for example: (1) during a loss of coolant accident (LOCA), or, (2) during various anticipated transients without scram (ATWS). A particularly bad LOCA event is one in which there is a rapid depressurization of the secondary side which causes a reactor Scram (i.e., a rapid insertion of the control rods, which terminates the nuclear chain reaction) and rapid cooling of the primary coolant via the steam generators. This accident leads to a severe pressurized thermal shock of the RPV and the associated internals.

12. A degradation in ductility will adversely affect a PWR's ability to withstand pressurized thermal shock transients, and thus there is a threat to the integrity of the embrittled internal structures in the RPV and to the RPV itself. That is, various accidents can expose the embrittled RPV belt-line materials to significant pressurized thermal shocks. The resultant

stresses may then cause the RPV and/or RPV internals to fail structurally. If so, the ability to effectively cool the decay heat in the core may be lost.

13. Severe thermal shocks can also occur during a design basis accident (DBA) LOCA event (i.e., a breach of main coolant piping on the primary side), which rapidly depressurizes the primary side and leads to the injection of relatively cool emergency core coolant into the RPV (e.g., from the accumulators). As noted previously, this may lead to the failure of a highly embrittled RPV and/or the RPV internal structures, and thus the inability to subsequently cool the core.

14. Sections A.2.2 (Evaluation of Time-Limited Aging Analysis - Unit 2) and A.3.2 (Evaluation of Time-Limited Aging Analysis - Unit 3) of Appendix A of the license renewal application briefly mention thermal shock as it relates to the requirement of the licensee to perform an assessment of the projected values of reference temperature whenever a significant change occurs in the projected values of the adjusted reference temperature for pressurized thermal shock (RT_{PTS}). The license renewal application, however, does not indicate if the applicant performed any age-related accident analyses, or if it even took embrittlement into account when assessing the effect of these transient loads.

15. Even more significantly, Entergy's failure to discuss how embrittled RPVs and RPV internal structures and components would respond to the highly transient severe decompression shock loads associated with a DBA LOCA is a very serious omission from its relicensing application. A decompression shock, created during the subcooled decompression phase of a DBA LOCA, can create significant transient pressure differentials across, at least, the following embrittled metal internal RPV structures: the core barrel, particularly in the "belt line" region of the reactor core; the thermal shield; the baffle plates and formers (and the loads on the associated bolts); and intermediate shells in the core. Detailed experiments (e.g., the Loss of Fluid Tests

(LOFT) at the Idaho National Engineering Laboratory (INEL)) and analyses have shown that when sufficiently ductile, internal structures of this type are not likely to deform to the point where a coolable geometry can not be maintained for the core during the reflood phase of a DBA LOCA.

16. In contrast, Entergy has not presented 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 DBA LOCA. *See, e.g.*, LRA §§ A.2.2, A.3.2. This is a serious and unacceptable omission by Entergy because embrittled structures are known not to tolerate shock loads well (i.e., they can fracture when exposed to high strain rate shock loads). 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, which would represent a serious challenge to the integrity of the containment structure.

17. A commonly used method to assess the effect of neutron damage on metal structures is to test in-core samples using a "Charpy test." In this simple impact test, the toughness of the material is determined and any degradation in toughness is a measure of radiation damage.

18. The tests of some of the RPV structures and components in IP2 and IP3 raise serious embrittlement concerns. *See* LRA, §§ A.2.2.1.3, A.3.2.1.3. Specifically, Entergy states the following in A.2.2.1.3:

> The predictions for percent drop in CVUSE at 48 EFPY are based on chemistry data, unirradiated CVUSE data, and 1/4 T fluence values. The projected 48 EFPY peak beltline fluence level was applied to all beltline materials with the exception of axial welds. Based on surveillance data, peak fluence levels at the beltline axial welds is based on the expected fluence at the 30 degree azimuth position.
>
> One intermediate shell plate (B2002-3) and one lower shell plate (B2003-1) have projected upper shelf energy levels that fall below 50 ft-lb during the period of extended operation. All remaining plate and weld beltline materials meet the requirement of exceed 50 ft-lb at 48 EFPY.
>
> An equivalent margins analysis performed in WCAP-13587, Rev. 1, demonstrated that the minimum acceptable USE for reactor vessel plate material in four-loop plants is 43 ft-lbs.

> In the safety assessment of WCAP-13587, the USNRC concluded the report demonstrated margins of safety equivalent to those of the ASME code for beltline plate and forging materials. The USE values are therefore acceptable since the lowest projected USE level for the beltline plate material through the period of extended operation of 47.4 ft-lb for intermediate shell plate B2002-3 is above the 43 ft-lbs minimum acceptable USE for four-loop plants determined in WCAP-13587 Rev. 1.

Thus, according to Entergy, based on Charpy tests of in-core samples, several in-core shells will not meet the upper shelf energy acceptance criterion of 50 ft-lb during the proposed relicensing period. Moreover, RPV internals in IP3 imply operational limits for extended life operations due to the high NDT associated with the predicted irradiation-induced embrittlement. Thus, irradiation-induced embrittlement is a significant issue which must be more thoroughly considered, particularly when evaluating various LOCA events, before any decision is made on the relicensing of IP2 and IP3.

Potential for Fatigue Failure

19. A second issue on which the USNRC should conduct a hearing on Entergy's relicensing application for IP2 and IP3 is the age-related criterion of fatigue. Fatigue is one of the primary considerations when conducting a time limited aging analysis (TLAA) as part of the USNRC's General Design Criteria (GDC) for nuclear power plants. Fatigue of various components in a nuclear reactor can result in pipe ruptures, component failures, and the migration of loose pieces of metal through the reactor system, which can interfere with the safe operation of a nuclear plant. The ultimate concern about fatigue is the increased potential for a primary or secondary side LOCA, or other safety-related events, such as a control rod ejection.

20. A common figure of merit used to appraise the possibility of fatigue failure is the cumulative usage factor (CUF), which is the ratio of the number of cycles experienced by a structure or component divided by the number of allowable cycles for that structure or component. At a nuclear power plant, the maximum number of cycles that should be experienced

by any structure or component should always result in a CUF of less than 1.0. In other words, the number of actual cycles experienced should always be less that the number of allowable cycles.

21. Entergy's relicensing application demonstrates that several important structures and components in IP2 and IP3 are near, or will exceed during the proposed relicensing period, the CUF = 1.0 criterion. Entergy provides data in Tables 4.3-13 (IP2) and 4.3-14 (IP3) of its relicensing application indicating that some key reactor components will have a greater potential for failure due to metal fatigue before the years 2033 and 2035 (i.e., during the proposed period of extended plant operation for IP2 and IP3, respectively). In particular, according to Entergy, the following reactor components all have unacceptably high CUF: (1) the pressurizer surge line piping (on the primary side- IP2 and IP3), (2) the reactor coolant system (RCS) piping charging system nozzle (IP2), and, (3) the pressurizer surge line nozzle (IP3). These exceedences are shown in the following table:

Component	**Plant**	**Environmentally Adjusted CUF (Entergy's data)**	**Amount of exceedence of 1.0 CUF criterion**
Pressurizer surge line piping	IP2	9.21	nearly 10 times
Pressurizer surge line piping	IP3	9.21	nearly 10 times
RCS piping charging system nozzle	IP2	15.20	over 15 times
Pressurizer surge line nozzle	IP3	2.35	more than double

22. Entergy has failed to adequately explain or commit to how it will address these exceedences. It claims that it will: (1) "refine the fatigue analyses," that is, rework the calculations, (2) conduct an inspection program, and, (3) repair or replace the components that exceed acceptable limits. In my opinion, the third option – repair or replace the components that

exceed the 1.0 criterion – is the only option that adequately addresses this important aging issue and adequately protects the public.

Inadequate Baseline Inspections of IP2 and IP3

23. A third issue on which the USNRC should conduct a hearing for Entergy's relicensing application for IP2 and IP3 concerns Entergy's failure to provide any meaningful inspection data or comprehensive inspection program for the proposed life extensions.

24. As part of the relicensing review, and prior to the commencement of any extended operations, the USNRC 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 at least 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.

25. Conducting baseline inspections of IP2 and IP3 is critical to the aging analysis required by the USNRC. 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 USNRC, 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. If the answer to that question – whether they can continue to operate for 20 more years – is yes, then the baseline inspections are not wasted since they provide valuable data with which to assess the performance of these already aging plants as they continue to operate and age for 20 more years beyond their original design life (40 years). Conducting a baseline inspection for the license extension of a nuclear power plant should be considered to be

routine, sound engineering practice since it will establish the state of the reactor facility/systems/structures/components at the end of their design life and indicate any degradation which may have occurred over the lives of these plants. The failure to conduct thorough baseline inspections prior to life extension is reckless and runs counter to rudimentary engineering practice.

26. The inspection program that Entergy proposes in the license renewal application is vague and ill-defined. Throughout Appendix B - Aging Management Programs and Activities of the license renewal applications, indicates that Entergy, for most of the facility components, will "participate in the industry programs for investigating and managing aging effects on reactor internals and evaluate and implement the results of the industry programs as applicable to the reactor internals." Without further detail, there is no way for the State to determine if these industry inspection programs are sufficient. Omitting details does this rise to the level of sound engineering practice. This vague "to-be-determined-later" proposal postpones Entergy's commitment until after all intervenors are scheduled to file their contentions. This deferral could also remove the USNRC's review of any contingent or theoretical monitoring programs from the public license renewal proceeding.

27. In Appendix B, Entergy also proposes to begin its vague and ill-defined inspection process no later than twenty-four months before it starts its extended operations at IP2 and IP3. That hardly provides data for the USNRC, or anyone else, to review during the relicensing process. These plants were not designed to operate for 60 years; they were designed to operate for 40 years. The USNRC and any intervenors should be able to evaluate now, prior to any USNRC decision on relicensing, the proposed inspections and the inspection methods to be used to assess their adequacy to provide the necessary baseline information on the condition of IP2 and IP3 at the end of their 40-year design lives.

Water/Cement Ratio of Containment Structures

28. A fourth issue on which the USNRC should conduct a hearing on Entergy's license renewal application for IP2 and IP3 concerns Entergy's lack of sufficient or meaningful commitments to monitor the integrity of the containment structures. *See* Appendix B, section B.1.7 for a discussion of the Containment Leak rate Program and section B.1.8 for a discussion of the Containment Inservice inspection. Containment structures are steel-lined, reinforced concrete structures which enclose the nuclear reactor systems. They are required to prevent the escape of radiation in the event of an accident and are part of the "defense in depth" safety design for nuclear reactors in the United States. Significant concerns exist regarding the continued integrity of the containment due to the water/cement ratio in the IP2 and IP3 containment structures and the proposed aging management and monitoring of those structures during any license renewal term.

29. The water/cement ratio is a measure of the density and strength of the concrete used in the containment structures; the lower the ratio, the stronger the concrete. The USNRC has established – after IP2 and IP3 were constructed – that an acceptable water/cement ratio is in the range of 0.35 to 0.45. When the water/cement ratio at a plant exceeds this range, the plant operator is normally expected to conduct enhanced inspections. Enhanced inspections would confirm the integrity of concrete in the containment structures during another 20 years of plant operation. If enhanced inspections reveal a structural problem, the USNRC can either order corrective measures or determine that the plant can no longer operate safely. So even though the more stringent water/cement ratio would not apply here, the concern remains the same, and Entergy should conduct enhanced inspections.

30. According to Entergy, the water/cement ratio for the containment structures at IP2 and IP3 is within the ratio of up to 0.57. LRA, p.3.5-6, § 3.5.2.2.1.1. This ratio is larger than the USNRC's acceptable range of 0.35 to 0.45. *See* NUREG 1801. This discrepancy requires

enhanced inspections of the containment structures throughout any license renewal term. In other words, because of this construction deficiency, the Indian Point containment structures require a more thorough and frequent monitoring protocol than would a facility that met the required water/cement ratio. Entergy, however, has not proposed to do enhanced inspections in its relicensing application. *See generally* LRA. Thus, the USNRC should require Entergy to perform enhanced inspections of the IP2 and IP3 containment structures, or to conclusively prove that these inspections are not required.

Terrorist Attacks on Spent Fuel Pools

31. A fifth issue on which the USNRC should conduct a hearing on Entergy's relicensing application for IP2 and IP3 is the safety of the storage of spent fuel, and the consequences of a terrorist attack on the spent fuel pools at all three of the Indian Point reactors. This issue arises from the USNRC's severe accident mitigation analysis (SAMA) and its review of environmental impacts under the National Environmental Policy Act (NEPA). A terrorist attack on the spent fuel pools could result in radiation releases that could cause significant adverse environmental and health effects and property damage in one of the most populated areas of the country – the New York metropolitan area.

32. All three Indian Point plants have spent fuel pools outside their containment buildings that contain large quantities of radioactive material. After it is used in nuclear reactors to generate energy, spent nuclear fuel remains extremely radioactive. To protect workers, facilities, and neighboring communities, most nuclear power plants in the nation have constructed large swimming-pool-like structures in which the spent fuel was to be stored temporarily until it cooled sufficiently to allow its transfer to a final disposal site in the United States. Because no final disposal site has yet been developed, the spent fuel has remained for decades in these temporary

storage pools. The storage pools are susceptible to fire and radiological releases in the event the pools drain.

33. I served as a member of a committee that conducted a study under the auspices of the National Research Council (NRC) of the National Academy of Sciences (NAS), which reviewed the safety and security of spent nuclear fuel storage. The committee was officially called the "Committee on the Safety and Security of Commercial Spent Nuclear Fuel Storage of the Board of Radioactive Waste Management," and it reported directly to the United States Congress. In 2005, the National Research Council published both public and classified reports of the Committee's study, which I co-authored. The public report, "Safety and Security of Commercial Spent Nuclear Fuel Storage," is attached as **Exhibit A**. (National Research Council of the National Academies, *Safety and Security of Commercial Spent Nuclear Fuel Storage: Public Report*, (copyright 2006) (hereinafter called the *NAS Study*). I understand that the State of New York will request that the ASLB be allowed to review the National Research Council's confidential report during the license renewal proceedings. In any event, my colleagues on the National Research Council Committee and I studied various possible terrorist attack scenarios, and we concluded that spent fuel pools, such as those at Indian Point, are indeed vulnerable to such attacks.

34. Regarding the potential of an attack on the three spent fuel pools at the Indian Point plants, the Generic EIS state that "if such an event were to occur, the resultant damage and radiological release would be no worse than expected from internal events." *See* USNRC's Generic Environmental Impact Statement (NUREG-1437 Vol. 1): § 5.3.3.1 (Review of Existing Impact Assessments). The Generic EIS conclusion may be true for a terrorist attack on or within the primary containment, but it is most certainly not true for a terrorist attack on any or all of the three spent fuel pools at Indian Point. Indeed, far more radioactivity is present in the spent fuel

located in the three spent fuel storage pools at Indian Point than there is in the active core of the two nuclear reactors.

35. Additionally, the spent fuel pools are not enclosed by a leak-tight containment structure. Rather, they are surrounded by only a confinement building, which is not a leak-tight containment structure. Thus, if a terrorist attack leads to pool drainage and a propagating zirconium fire, much of the radioactive inventory in the spent fuel could be released to the environment. The resulting plume of radiation released into the atmosphere can result in significant adverse environmental and health effects and property damage in and around the Indian Point plants, including New York City (NYC), and the immediate portions of northern New Jersey and southwestern Connecticut. Approximately twenty million people reside or work within a 50-mile radius of NYC. At risk, too, are trillions of dollars of property in the tri-state region and, of course, the financial capital of the world (NYC) could be seriously disrupted.

36. The NAS Study made several recommendations for mitigation, including the rearrangement of the spent fuel in the storage pools and spray cooling. Entergy has not indicated in its relicensing application that it has adopted these mitigation measures for any of the spent fuel pools at Indian Point. Although Entergy is apparently moving some of its spent fuel from the spent fuel pools to dry cask storage, that will not completely mitigate the threat outlined above since the most highly radioactive fuel generates the most decay heat and thus must remain in the spent fuel pools. In other words, the two active reactors will continually generate more spent fuel during the proposed renewal period, and because of its decay heat and radioactivity, this spent fuel must remain in the spent fuel pools for some time before it can be moved to dry cask storage (i.e., the natural convective cooling by air in dry cask storage can not keep this fuel cool enough).

37. Moreover, the movement of some spent fuel from the Unit 2 and Unit 3 spent fuel pools to an on-site dry cask storage area will not significantly reduce the density of spent fuel

inside those units' pools because the two reactors will continually generate more spent fuel during any renewal period. Nor does the LRA evaluate these mitigation alternatives.

38. Finally, given the proximity of these plants to New York City, the potential health, environmental, and financial impacts are very significant – much more so than at any other nuclear power plant in the nation. Thus, Entergy's application for license extension must consider all reasonable severe accident mitigation alternatives (SAMA) concerning a terrorist attack on the spent fuel pools.

Conclusion

39. In summary, these five aging and safety related issues – (1) embrittlement of the reactor pressure vessels and associated internals; (2) the potential for fatigue failure; (3) inadequate baseline inspections of IP2 and IP3; (4) the need for enhanced inspections because of inadequate water/cement ratio in the containment structures; and, (5) the risk of a terrorist attack on the spent fuel pools – all demonstrate that IP2 and IP3 have significant aging and safety related issues that need to be addressed in the context of this relicensing proceeding. The applicant has glossed over many of these issues, to the extent that it has addressed them at all. The USNRC should conduct a very rigorous inquiry by accepting these contentions and holding a hearing on each of them.

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

Dated: November ___, 2007
Troy, New York

R. T. Lahey

Richard T. Lahey, Jr.