NYS000300 (REDACTED PUBLIC VERSION)
Submitted: December 22, 2011

UNITED STATES
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

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 September 9, 2010

The State of New York provisionally designates the attached Declaration of Dr. Richard T. Lahey dated September 8, 2010 as containing Confidential Proprietary Information Subject to Nondisclosure Agreement

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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:

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 and is incorporated by reference.[1] By way of summary, I have held various positions in

[1] I also submitted a declaration in support of the State of New York's Supplemental Contention 26-A dated April 7, 2008, which is also incorporated by reference.

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), and the National Academy of Science (NAS). 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, and was lead engineer and manager of various departments responsible for heat transfer mechanisms and core and safety development for the General Electric Company (GE). 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 Submittal Regarding the Completion of Commitment-33 for Indian Point Units 2 and 3 (Aug. 10, 2010)("Environmental Fatigue Evaluations") that conveyed Entergy's NL-10-082 communication to NRC Staff, two Westinghouse Electric Company LLC environmental fatigue evaluations of Indian Point Unit 2 and Indian Point Unit 3 (received by the State on August 12, 2010), the Applicant's Motion for

Summary Disposition of New York State Contentions 26/26A and Riverkeeper Technical Contentions 1/1A (Metal Fatigue of Reactor Components)(Aug. 25, 2010), and other documents referenced in this declaration. This new declaration is based on, and expands upon, many of the concerns that I raised in my prior ASLB testimony concerning metal fatigue associated with the relicensing of Indian Point reactors Units 2 & 3 (or IP-2 & 3).

3. As I stated in my initial declaration on this issue in support of the State of New York's Contention 26, in my professional judgment, the applicant failed to demonstrate that it had adequately accounted for the aging phenomena of metal fatigue in Indian Point Unit 2 and Unit 3. My professional judgment has not fundamentally changed based upon Entergy's August 10, 2010 submission of their new Environmental Fatigue Evaluations.

4. 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 (W) designed plants are 4 loop PWRs currently[2] rated at 3,216.4 MW_t. They are sited on the Hudson River in Buchanan, NY, which is about 24 miles north of the New York City (NYC) border. Because of their close proximity to a very highly populated area (*i.e.*, NYC metropolitan area), which is also the world's leading financial center, it is vital that IP Units 2 & 3 fully and

[2] 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.

unambiguously meet all reasonable and applicable criteria for safe operation. This is particularly true when considering life extension, since fatigue failures are much more likely as the plants age.

5. The standard review plan of the USNRC for the license renewal applications of nuclear power plants is given in NUREG-1800, Rev. 1 (Standard Review Plan). This plan is a highly prescriptive process which allows little opportunity for the discovery of any new age-related safety concerns. However, I believe that all important safety concerns must be addressed to assure the health and safety of the American public during extended plant operations. The NRC Staff have also prepared a guidance document entitled the "Generic Aging Lessons Learned (GALL) Report," NUREG-1801, Rev. 1, in which Staff seeks to describe Aging Management Programs (AMP) for the extended operations of nuclear power plants. In the case of fatigue, Entergy has now submitted further environmental fatigue evaluations for both Indian Point Unit 2 and Unit 3, which were prepared by Westinghouse, and has thus elected to try and close out metal fatigue issues during the ASLB relicensing hearings. Unfortunately, as will be discussed subsequently, their new fatigue analyses are incomplete, inadequate and unacceptable.

6. While the USNRC's review process is fairly comprehensive, it fails to consider some very important age-related safety issues associated with the extended operation of a pressurized water nuclear reactor (PWR) power plant. For example, the fatigue of various highly irradiated (and embrittled) structures and

fittings within the reactor pressure vessels (RPV) due to operational and abnormal transients (*e.g.*, SCRAMs). In addition, the impact of thermal and pressure shock loads on the fatigue-weakened structures and fittings both outside and inside the RPV. Typical shock loads include those associated with the plant's design basis accident (DBA) loss of coolant accident (LOCA), various secondary side LOCAs and anticipated transients without SCRAM (ATWS) events. In my opinion this is an extremely serious deficiency in the USNRC's standard review plan for plant life extension as well as Entergy's aging management plan (AMP).

7. Fatigue is a very important age-related safety concern, particularly when plant life extension is being considered. In fact, it is one of the primary things that must be considered when doing a time-limited aging analysis (TLAA) or developing an aging management program for the extended operation of a power reactor. A common figure of merit used in the American Society of Mechanical Engineers (ASME) code [Section-III] to appraise the possibility of fatigue failure is the cumulative usage factor (CUF), which is the ratio of the number of cycles experienced divided by the number of allowable cycles. The maximum number of cycles which can be experienced by a structure or component before cracking is expected occurs when CUF = 1.0, and one should have CUF < 1.0 during the period of plant operation. In addition, since the high pressure/temperature primary coolant is known [*e.g.*, NUREG/CR-6909] to degrade the fatigue life of immersed metal structures and components, the USNRC also requires that environmental corrections be applied to the calculated CUF, and it specifies formulas/curves to be

used for these corrections [*e.g.*, NUREG/CR-5704; NUREG/CR-6583]. Moreover, the environmentally-adjusted fatigue analyses must satisfy $CUF_{en} < 1.0$ during extended plant operations.

8. In the original relicensing submittal for IP Units 2 & 3, Entergy analyzed typical limiting PWR structures and fittings using some of those given in NUREG/CR-6260 [pg. 5-62], and this analysis showed that some important structures and components will significantly exceed the environmentally-adjusted $CUF_{en} = 1.0$ criterion during the proposed extended operations period. In particular, the pressurizer surge line and nozzle (on the primary side), the reactor coolant system charging system nozzle, the steam generator main feed water nozzles and tube/tube-sheet welds (on the secondary side), and the upper joint canopy of the IP-2 control rod drive mechanisms, all had unacceptably high CUF_{en} (*e.g.*, $CUF_{en} > 9.0$ for the IP-2 and IP-3 pressurizer surge lines and $CUF_{en} > 15$ for the IP-2 RCS charging system nozzle [LRA-Section 4]). In my opinion, if these results can not be conclusively shown to be invalid by doing more detailed fatigue analyses, the deficient components should be replaced/repaired <u>prior</u> to extended operations; indeed, it would be the only responsible thing to do since the last thing one wants is to induce a primary or secondary side LOCA due to a fatigue failure. In any event, once CUF_{en} violations are found Entergy was expected [NUREG-1801, Rev. 1, Vol. 2, pg. X M-2; EPRI, MRP-47; "Guidelines for Addressing Fatigue Environmental Effects in a Licensing Renewal Application," pg. 3-4 (2005)] to also

do fatigue analyses for other important reactor structures and fittings. However, this was not done.

9. Anyway, in order to perform more mechanistic, and presumably less conservative, fatigue evaluations, Entergy contracted with Westinghouse (W) to redo the fatigue analyses for IP-2 and IP-3. These results were reported separately [WCAP-17199-P, "Environmental Fatigue Evaluation for Indian Point Unit 2" (June 2010) & WCAP-17200-P, "Environmental Fatigue Evaluation for Indian Point Unit 3" (June 2010)]. The calculations were done using WESTEMS, a proprietary computer code of W; however the full documentation for this code was not provided to me for review.[3] Without being able to review the WESTEMS code manuals, in which the detailed assumptions and models (particularly for the thermal-hydraulics) used by W are presumably given, it is not possible to fully understand and critique the validity of Entergy's new CUF_{en} results.

10. The new CUF_{en} results filed with the ASLB by Entergy ["Applicant's Motion for Summary Disposition of NYS Contentions 26/26A & Riverkeeper Technical Contentions 1/1A (Metal Fatigue of Reactor Components)" (August 25, 2010)] show that the previously most limiting CUF_{en} were reduced by more than an order of magnitude (*i.e.*, the results for the pressurizer surge line piping and RCS piping charging system nozzle), which is an astonishing change, and one that must

[3] Two brief proprietary excerpts of the WESTEMS computer code manual were provided to the State of New York on Friday evening, September 3, 2010. I reviewed these brief excerpts, but they did not shed light on the thermal-hydraulics models employed in the WESTEMS code.

be very carefully reviewed and verified since it significantly erodes safety margin. Additionally, for the first time, limiting fatigue analysis results were given for the residual heat removal (RHR) system piping and nozzles, and the results for these components were very close to the unity limit. That is, for the IP-2 RHR line, CUF_{en} = **0.9434** and for the IP-3 RHR line, CUF_{en} = **0.9961**. Thus, almost any reasonable error in these results could lead to a violation of the USNRC's CUF_{en} = 1.0 limit.

11. Unfortunately an error analysis was not made available by either Entergy or Westinghouse, nor were any results provided showing that the computational results exhibited nodal convergence, or how they were bench-marked against representative experimental data and/or analytical solutions. Normally, one would expect to see a detailed "propagation-of-error" type of analysis [*e.g.*, Vardeman & Jobe, "Basic Engineering Data Collection and Analysis," Duxbury, pp. 310-311 (2001)] to determine the overall uncertainty in the CUF_{en} results given by W. Indeed, all engineering analyses are based on mathematical models of reality and assumptions which inherently involve some level of error. As a consequence, without a well documented error analysis, the accuracy of Entergy's and Westinghouse's new fatigue results are quite uncertain. What is clear, however, is that there are many possible sources of error in these results. *For example*:

(i) [redacted]

[redacted]

(ii) [redacted]

(iii) [redacted]

(iv) [redacted]

(v) [redacted]

In any event, in my opinion, the accuracy of W's new fatigue evaluations, certainly those that are close to CUFen = 1.0, are quite uncertain and this uncertainty must be quantified with a detailed error analysis.

12. It is also significant to note that in-core fatigue failures of irradiated baffle-to-former bolts have been observed in operating PWRs [*e.g.*, WCAP-14577, Rev. 1, "License Renewal Evaluation: Aging Renewal Evaluation: Aging Management of Reactor Internals," pg. 2-29 (Oct. 2000); USNRC Staff Report, "Final Safety Evaluation of by the Office of Nuclear Reactor Regulation Concerning Westinghouse Owners Group Report, WCAP-14575, Revision 1, License Renewal Evaluation: Aging Management for Class 1 Piping and Associated Pressure Boundary Components, Project No. 686," (Nov. 8, 2000)] and B&W designed PWRs have had fatigue-induced failures of various in-core components even when CUF <

1.0 (presumably due to undetected manufacturing flaws) [Entergy Email: Esquillo to Stuard et al., Subject: " Section XI – Cracking" (8/30/06)]. Moreover, the possible effect of fatigue on the failure of in-core components was apparently known to Entergy [Entergy Email: Batch to Finnin, Subject: "Need to Evaluate High Cycle Fatigue to IPEC Baffle Bolts?" (12/28/06)]. Unlike postulated nuclear reactor accidents, the fatigue failures of in-core bolts are actuarial events that have happened and will likely happen again for sufficiently stressed materials. Moreover, it is not possible to inspect (*e.g.*, using UT) all the bolts within a RPV, and the nuclear industry has recommended [EPRI Report, MRP-228; "Materials Reliability Program: Inspection Standard for PWR Internals," (July 2009)] that an analysis be done to support continued operations if bolt failures are found during in-core non-destructive evaluations (NDE). However, it appears that these analyses will not take into account the various accident-induced pressure/thermal shock loads within the RPV, such as those due to a DBA LOCA. In this regard it is important to note that, unlike for the primary piping system, in-core DBA LOCA loads were not affected by the leak-before-break (LLB) rulings of the USNRC [NUREG/CR-4572; NUREG/CR-1061, Vol. 3; 10 C.R.F. Part 50, Appendix-A]. In any event, I believe that not doing an adequate safety analysis is totally unacceptable since further shock-load-induced bolting failures may lead to a blocked or distorted core geometry which, in turn, may not allow the ability to cool the core and can lead to core melting.

13. Like all mechanical systems, as nuclear power plants exceed their original design life (*i.e.*, 40 years) they begin to wear out and thus, to assure safe operation during plant life extension, it is important not to erode the original design-basis safety factors in the interest of keeping the plants running. In particular, in addition to the previously discussed bolting fatigue failure concerns, many other highly irradiated in-core structures and fittings (*e.g.*, core baffles, formers, etc.) will be subjected to some of the same (and even more) fatigue-inducing transients as those which effect the components that are external to the RPV (*e.g.*, those that were analyzed by W). However, no fatigue analysis of these important in-core components was done or provided and there was apparently no recognition of the importance of DBA LOCA, secondary side LOCA and ATWS loads on the integrity of these structures. As for in-core bolting, I believe that not doing a proper fatigue and safety analysis of these in-core structures and fittings is completely unacceptable since the shock-load-induced failure of in-core components may lead to a distorted core geometry, which may, in turn, not allow the ability to cool the core and result in core melting.

14. In summary, there are important age-related safety issues associated with the operation of IP Units 2 & 3 during a proposed 20-year life extension. In particular, there is a need to properly analyze and/or replace/repair components and structures which may reach or exceed their fatigue life prior to the end of extended plant operations. In my opinion, the revised fatigue analyses done by W for Entergy are not sufficient to allow the closure of Commitent-33 for IP-2 & IP-3. That is,

while a re-analysis of fatigue was performed for Entergy by W, it was not possible to thoroughly review the details of the models and assumptions used in these fatigue evaluations and there was no accompanying error analysis. Thus, the accuracy and uncertainty of these calculations (several of which were very close to $CUF_{en} = 1.0$) is unclear. Moreover, there were no fatigue evaluations done for various important irradiated and embrittled structures and fittings within the RPV, nor were was there any analyses presented showing the effect of various thermal/pressure shock loads on the limiting fatigued structures both within and outside the RPV. Thus, without a more complete fatigue and safety analysis (including a detailed error analysis) there is no valid technical basis on which to claim that the aging phenomena associated with metal fatigue has been adequately addressed by Entergy.

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

September 8, 2010
Troy, New York

R. T. Lahey

Dr. Richard T. Lahey, Jr.

Referenced Documents

American Society of Mechanical Engineers (ASME) code, Section-III

Cengel & Turner, "Fundamentals of Thermal-Fluid Sciences," McGraw-Hill, (2001)

Electric Power Research Institute (EPRI), MRP-47; "Guidelines for Addressing Fatigue Environmental Effects in a Licensing Renewal Application," (2005)

Electric Power Research Institute (EPRI) Report, MRP-228; "Materials Reliability Program: Inspection Standard for PWR Internals," (July 2009)

Entergy, License Renewal Application, Section 4

Entergy, License Renewal Application, Amendment 2

Entergy's Submittal Regarding the Completion of Commitment-33 for Indian Point Units 2 and 3 (Aug. 10, 2010),

> *conveying* Entergy's NL-10-082 communication to NRC Staff (Aug. 9, 2010)

Entergy's Motion for Summary Disposition of New York State Contentions 26/26A and Riverkeeper Technical Contentions 1/1A (Metal Fatigue of Reactor Components)(Aug. 25, 2010)

Entergy Email: Esquillo to Stuard et al., Subject: "Section XI – Cracking" (8/30/06) and email string:

> Friday; June 16, 2006; 10:25 AM; From: Mark A. Rinckel; To: Ron Finnin; Cc: acox@entergy.com, Michael D. Stroud, Virglio M. Esquillo, and Stan Batch; Subject: Section XI—Cracking
>
> Wednesday; August 30, 2006; 9:33 AM; From: Virgilio M. Esquilla; To: William L. Stuard, Mark L. Warren, Carole L. Naugle, and Kenneth R. Allison; Subject: FW: Section XI—Cracking
>
> Friday; December 8, 2006; 9:16 AM; From: Kenneth R. Allison; To: William L. Stuard; Subject: FW: Section XI--Cracking; Attach: Section XI-Standards.pdf

Entergy Email: Batch to Finnin, Subject: "Need to Evaluate High Cycle Fatigue to IPEC Baffle Bolts?" (12/28/06) and email string :

Friday; December 28, 2006; 1:58 PM; From: Stan Batch; To: Ron Finnin; Cc: Don Fronabarger, Ted S. Ivy; Subject: need to evaluate high cycle fatigue for IPEC baffle bolts?

Friday; January 12, 2007; 10:14 AM; From: Stan Batch; To: Walter Wittich and Nelson F. Azevedo; Cc: Ron Finnin and Don Fronabarger; Subject: need to evaluate high cycle fatigue for IPEC baffle bolts?

Kreith, "Principles of Heat Transfer," *Int. Text Book Co.*, (1961)

Lahey, R. T., Declaration in Support of Notice of Intention to Participate and Petition to Intervene filed by the State of New York in Indian Point license renewal proceeding on November 30, 2007

Lahey, R. T., Declaration in support of the State of New York's Supplemental Contention 26-A in Indian Point license renewal proceeding, dated April 7, 2008

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NRC Staff Report, "Final Safety Evaluation of by the Office of Nuclear Reactor Regulation Concerning Westinghouse Owners Group Report, WCAP-14575, Revision 1, License Renewal Evaluation: Aging Management for Class 1 Piping and Associated Pressure Boundary Components, Project No. 686," (Nov. 8, 2000)

NUREG/CR-6909

NUREG/CR-5704

NUREG/CR-6583

NUREG/CR-6260

NUREG/CR-4572

NUREG/CR-1061, Vol. 3

Westinghouse, WCAP-17149-P, Rev. 1, "Evaluation of Pressurizer Insurge/Outsurge Transients for Indian Point Unit 2," IPECPROP00056663 (July 2010)

Westinghouse, WCAP-17162-P, Rev. 1, "Evaluation of Pressurizer Insurge/Outsurge Transients for Indian Point Unit 3," IPECPROP00056717 (July 2010)

Westinghouse, WCAP-17199-P, "Environmental Fatigue Evaluation for Indian Point Unit 2," (June 2010)

Westinghouse, WCAP-17200-P, "Environmental Fatigue Evaluation for Indian Point Unit 3," (June 2010)

Westinghouse, WCAP-14577, Rev. 1, "License Renewal Evaluation: Aging Renewal Evaluation: Aging Management of Reactor Internals," (Oct. 2000)

Westinghouse, WESTEMS computer code manual (brief excerpts)

Vardeman & Jobe, "Basic Engineering Data Collection and Analysis," Duxbury, (2001)