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,	DPR-26, DPR-64
Entergy Nuclear Indian Point 3, LLC, and	
Entergy Nuclear Operations, Inc.	September 9, 2015

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PRE-FILED SUPPLEMENTAL REPLY WRITTEN TESTIMONY OF

Dr. RICHARD T. LAHEY, JR.

REGARDING CONTENTION NYS-25

On behalf of the State of New York (“NYS” or “the State”), the Office of the Attorney General hereby submits the following testimony by RICHARD T. LAHEY, JR., PhD. regarding Contention NYS-25.

Q. Please state your full name.

A. Richard T. Lahey, Jr.

Q. By whom are you employed and what is your position?

A. I am retired and am currently the Edward E. Hood Professor Emeritus of Engineering at Rensselaer Polytechnic Institute (RPI), which is located in Troy, New York.

Q. Have you previously summarized your educational and professional qualifications?

A. Yes, my education and professional qualifications and experience are described in my Curricula Vitae and previously filed testimony in this proceeding.

Q. I show you what has been marked as Exhibit ENT000616 and ENT000679. Do you recognize those documents?

A. Yes. They are copies of the pre-filed testimony of the witnesses for Entergy on Contentions NYS-25 and NYS-26B/RK-TC-1B that were submitted in August 2015.

Q. I show you what has been marked as Exhibit NRC000197 and NRC000168. Do you recognize those documents?

A. Yes. They are copies of the pre-filed testimony of NRC Staff witness that were submitted in August 2015. NRC000168 concerns Contention NYS-26B/RK-TC-1B, and NRC000197 concerns Contention NYS-25. (I note that portions of those two USNRC submissions also discuss Contention NYS-38/RK-TC-5, which I will discuss separately.)

Q. Have you had an opportunity to review ENT000616, ENT000679, NRC000168, and NRC000197?

A. Yes.

Q. Has Entergy's and the USNRC Staff's August pre-filed testimony caused you to change the testimony and opinions that you have previously submitted in this proceeding in connection with Contention NYS-25 and Contention NYS-26B/RK-TC-1B?

A. In general, no. Entergy and the USNRC Staff have failed to resolve the age-related safety concerns that I have raised throughout this relicensing proceeding. They continue to approach various aging mechanisms in silos, without addressing the potential synergistic interactions between multiple degradation mechanisms, and my related safety concerns.

Q. According to the USNRC Staff [redacted], the "Expanded Materials Degradation Assessment" (EMDA) (NYS000484A-B) and Light Water Reactor Sustainability (LWRS) Program (NYS000485) do not apply to IP2 and IP3 because they are associated with subsequent license renewal from 60 to 80 years. (NRC000168, at A176, A178; [redacted]). Do you agree?

A. Absolutely not. The EMDA and LWRS programs study aging degradation mechanisms that affect all licensed nuclear reactors. The effects studied in the EMDA and LWRS programs are also quite relevant to the safe extended operation of nuclear reactors out to 60 years. The size and cost of these research programs reveals the importance placed on studying the various PWR aging concerns that I have previously raised.

Q. Do you agree with [redacted] the USNRC Staff that irradiation embrittlement can increase a component's fatigue life?

A. There are some data which show that irradiation embrittlement can increase fatigue life in certain situations, especially where high cycle, low amplitude, fatigue occurs. However, there are other data which show that for low cycle, high amplitude fatigue the effects of embrittlement decreases fatigue life by reducing the number of cycles to failure (N_f). As the USNRC Staff concedes, the data regarding the effects of irradiation embrittlement on fatigue life are currently incomplete and inconclusive. NRC000168, at A154; NRC000197, at A196, A200. However, in the face of this uncertainty, Entergy and the USNRC Staff simply assume that the effect can be ignored, and that the Indian Point reactors can continue to be operated until any synergistic degradation effects are directly observed in the operating plants. NRC000197, at A204 ("If synergistic effects of aging mechanisms were to occur, the resulting degradation will likely be found in at least one plant in the fleet.") This is exactly what I mean when I say that Entergy and the USNRC Staff have taken a "wait-and-see" approach. Unfortunately, such an approach could lead to a component failure and potentially serious consequences that cannot be easily addressed after such a failure has occurred. A much more prudent approach would be to apply an uncertainty factor, or penalty, to the CUF_{en} calculations, and to repair or replace components with very high projected CUF_{en} values. This is a well-accepted engineering practice; indeed, the ASME Code applies a similar uncertainty penalty to CUF_{en} calculations (i.e., a factor of 2 on stress or 20 cycles) to account for uncertainties in test data, which were obtained from tests on small scale, polished metal samples in air, rather than on actual industrial structures, components and fittings in a reactor environment (e.g., in a PWR).

[redacted]

A. [redacted] The operative document for management of aging degradation effects on RVI components at Indian Point is the Revised and Amended RVI Plan, which relies on MRP-227-A and which was the subject of the Second Supplemental Safety Evaluation Report for Indian Point (NUREG-1930, Supplement 2) (Exh. NYS000507). I remain concerned with the adequacy of the Revised and Amended RVI Plan, because it relies entirely on inspections to determine the condition of RVI components. The absence of detectable surface cracks does not necessarily mean that the embrittled and fatigue-weakened structures, components and fittings are not vulnerable to early (i.e., CUF_{en} < 1.0) failures. As MRP-227-A concedes, inspections cannot determine the existence and extent of embrittlement. Accordingly, the Revised and Amended RVI Plan does not account for the possibility that embrittled and fatigue-weakened RVI components could be subject to a shock load which would cause them to fail suddenly.

Q. Do you agree with the USNRC Staff [redacted] that MRP-227-A inspections, coupled with environmentally assisted fatigue calculations resulting in a CUF_{en} of less than 1.0, are adequate to manage the effects of aging on RVI components?

A. No. This is a perfect example of the type of "silo" thinking that I am concerned about. According to the USNRC Staff [redacted], embrittlement and the associated aging effects can be managed through the inspection-based Revised and Amended RVI Plan, while fatigue is managed through a separate Fatigue Management Plan (FMP), and no further consideration of the interaction between multiple aging mechanisms is necessary. NRC000168, at A185. For example, the USNRC Staff argues that the portions of my June 2015 testimony (NYS000530) relating to embrittlement are not relevant to the management of metal fatigue, even though I have repeatedly argued that embrittlement effects should be considered when assessing component fatigue life. NRC000168, at A168, A170. As long as CUF_{en} is calculated to be less than 1.0, [redacted] the USNRC Staff apparently believe that no critical structures, components, and fittings will exhibit fatigue-induced surface cracking and therefore the effects of embrittlement need not be considered in the fatigue calculation. NRC000168, at A153. Their approach, however, fails to recognize certain basic realities. In particular, the CUF for a structure, component or fitting is defined as the number of fatigue cycles it is expected to experience (N) divided by the number of cycles to failure (N_f). The N_f for a new, ductile material can be significantly greater than the N_f for a highly embrittled material; indeed, this is known to be true for large amplitude, low cycle fatigue. See e.g. Kanaski, et al., "Fatigue and Stress Corrosion Cracking Behaviors of Irradiated Stainless Steels in PWR Primary Water," ICONE-5, at 2372 (May 1997) (Exh. NRC000177); Arai, et al., "Irradiation Embrittlement of PWR Internals," Proceedings ASME/JSME 2d International Nuclear Engineering Conference, Vol. 2, at 103 (1993) (Exh. NYS000564); Korth, G.E. & Harper, M.D., "Effects of Neutron Radiation on the Fatigue and Creep/Fatigue Behavior of

Type 308 Stainless Steel Weld Materials at Elevated Temperatures," Proceedings of the 7th International Symposium on the Effects of Radiation on Structural Materials, Gatlinburg, TN (June 1974) (Exh. RIV000152). Additionally, embrittled and fatigue-weakened structures may not be able to tolerate significant seismic and shock loads as well as fully ductile structures can. Thus, when the effects of embrittlement and fatigue are considered together, there is a real risk that components will fail before their calculated CUF_{en} value reaches unity.

Q. I show you a document marked as Exhibit [NYS000566] that is entitled, Figure 1: "Comparison of Limit Line and Best Estimate Predictions with Embrittlement and Without Embrittlement." Are you familiar with this Figure?

A. Yes.

Q. How are you familiar with this Figure?

I developed Figure-1 in connection with my review of Entergy's and the USNRC's Revised Statements of Position and Revised Testimony in this proceeding.

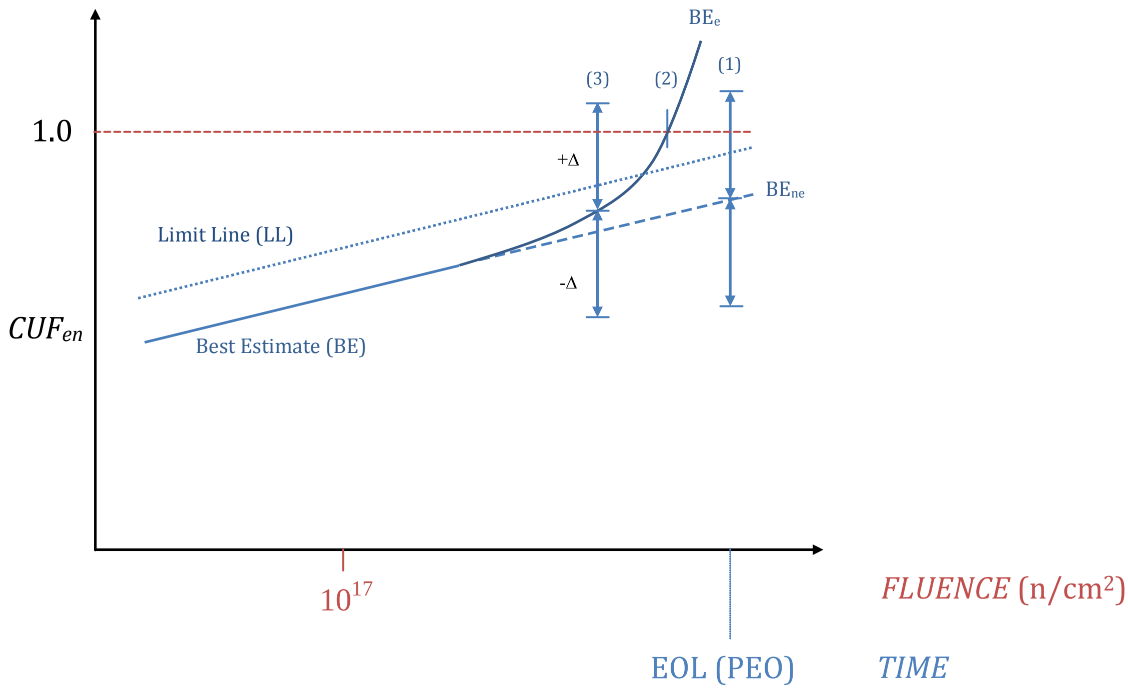

Figure 1: Comparison of Limit Line (LL) and Best Estimate (BE) predictions, with embrittlement (BE_e) and without embrittlement (BE_{ne}).

Note:
(1) BE_{ne} predicts possible failure at end of life (EOL).
(2) BE_e predicts failure (CUF_{en} = 1.0) before end of life.
(3) BE_e predicts possible failure well before end of life.

Q. Why did you prepare this Figure?

A. I prepared this Figure to visually illustrate some of the concerns that I have set forth in my testimony in this proceeding. Specifically, this Figure shows three important things: (1) a typical WESTEMS™ "Limit Line" prediction, (2) a "Best Estimate" prediction of CUF_{en}, the cumulative usage factor under reactor operating conditions, and the associated uncertainty interval (Δ), and, (3) how the neutron fluence-induced embrittlement of a RVI structure, component, or fitting can dramatically increase the possibility of a fatigue failure, as represented by the cumulative usage factor under PWR operating conditions, CUF_{en} , being equal to unity. This figure applies to RVI structures, components, and fittings made of stainless steel or other materials. In summary, in Figure-1, I have illustrated two different approaches for calculating a time-dependent CUF_{en}, namely the "Limit Line" and the "Best Estimate" calculation. For the "Best Estimate" line, I also illustrate typical uncertainty intervals, delta (Δ), and the effects of fatigue on embrittled (e) and on non-embrittled (ne) RVIs.

Q. Please describe what is represented by the horizontal x-axis (i.e., the abscissa) in Figure 1.

A. Since fluence = φt (where φ is the fast neutron flux (n/cm^2-s) and t is the time of operation of the reactor) the x-axis tracks <u>both</u> the fluence (φ) and the time (t). As time passes, each RVI reaches the reactor's End of Life (EOL) for the Period of Extended Operation (PEO). Also, as we move from left-to-right along the x-axis (i.e., as the time of reactor operation increases), we see an increase in the number of fatigue cycles (N) that the various RVIs (one of which has been chosen to be displayed in Figure-1) have been subjected to.

Q. Please describe what is represented by the vertical y-axis (i.e., the ordinate) in Figure 1.

A. The y-axis displays CUF_{en}, the cumulative usage factor, considering PWR environmental conditions. At a nuclear power plant, the maximum number of fatigue cycles that can be experienced by any structure, component or fitting must always result in a CUF_{en} of less than 1.0. That is, the number of actual fatigue cycles (N) should always be less than the number of allowable cycles (N_f) to avoid failure of the RVI.

Q. In Figure-1 what does the “Limit Line” represent?

A. The “Limit Line” represents [redacted] as a supposedly-conservative [redacted] calculation of CUF_{en} for a hypothetical RVI component or structure. [redacted] The “Limit Line” shown in Figure-1 includes Entergy’s implicit assumption that a stainless steel RVI structure, component, or fitting remains perfectly ductile, and thus shows that the CUF_{en} for a RVI will increase with an increasing number of fatigue cycles, but that it is not affected by the fluence. The “Limit Line”, therefore, uniformly increases in CUF_{en} even after the fluence exceeds 10^{17} n/cm^2, where significant embrittlement begins to develop (i.e., it is assumed that the number of cycles to failure are not a function of fluence). Figure-1 shows that, as the RVI component approaches the end of life (EOL) for the period of extended operation (PEO), [redacted] "Limit Line" will approach (but still be below) CUF_{en} = 1.0. This is depicted as location (1).

Q. In Figure-1 what does the "Best Estimate" line represent?

A. The "Best Estimate" plot depicts what would be, for example, a WESTEMS prediction of CUF_{en} in which conservative assumptions are made and my concerns [redacted] have been addressed. In addition, the "Best Estimate" plot in Figure-1 also includes a prediction of a "propagation of errors" type analysis (i.e., an uncertainty analysis), which can, and should, be done.

Q. What does a "propagation of errors" analysis include?

A. This type of analysis considers important parameters that have some uncertainty associated with them; for example, the coarseness of the computational mesh, uncertainties in F_{en} , the "best estimate" local heat transfer coefficients, etc.; uncertainties which, in turn, impact the RVI's actual fatigue life (i.e., CUF_{en}). The net uncertainty intervals (Δ) also account for uncertainties in the number of transients (including, for example, seismic events), and synergistic effects of radiation (i.e., the fluence) and stress corrosion effects on metal fatigue. Accounting for the uncertainty in these parameters allows one to estimate the overall uncertainty (Δ) of the "Best Estimate" CUF_{en} prediction, so that we can see if the [redacted] "Limit Line" predictions are indeed sufficiently conservative. That is, comparing the "Best Estimate" results, plus Δ, with the "Limit Line" predictions reveals whether the "Limit Line" approach [redacted] is adequate or not.

Q. Can you indicate how one might perform a "propagation of error" analysis?

A. Yes, uncertainty analyses can be performed in a number of ways. One of the most commonly used methods by engineers is the method proposed by Kline & McClintock [ASME J. Mechanical Engineering, Vol.75, No.1, 3-8, Jan. 1953] (Exh. NYS000514). For the case in which the "Best Estimate" value of CUF_{en} involves "N" parameters, each having some uncertainty associated with them (e.g., the Dittus-Boelter convective heat transfer, [redacted] has an inherent uncertainty of about ±25%), the net uncertainty (Δ) is given by:

$$\Delta = \pm \left[\sum_{i=1}^{N} \left(\frac{\partial CUF_{en}}{\partial x_i}\right)^2 \Delta x_i^2\right]^{1/2}$$

This approach is widely used by engineers since it is relatively easy to evaluate and it gives an acceptable approximation of the "error bars" (Δ) in a "Best Estimate" evaluation.

Q. In Figure-1, the "Best Estimate" line is below the "Limit Line." Does this mean that everything is OK?

A. Not necessarily. As shown in Figure-1, a "Best Estimate" prediction <u>plus</u> the uncertainty (Δ) may predict possible RVI structure, components or fitting failures (i.e., CUF_{en} = 1.0), sooner than the "Limit Line" would. Moreover, the effect of embrittlement can make the situation much worse. The problem is that no one knows how much conservatism, if any, is in the [redacted] "Limit Line". As a consequence, we can have no confidence that "Limit Line" results near CUF_{en} = 1.0, are sufficiently conservative to bound all the possible uncertainties, as illustrated in Figure-1 by Δ.

Q. In Figure-1, the "Best Estimate" line is shown for the case in which there is no effect of embrittlement on fatigue, BE_{ne}, [redacted] <u>and</u> the case in which embrittlement degrades fatigue life, BE_e. Can you explain what these lines mean?

A. Yes, the plot depicts two "Best Estimate" predictions for a hypothetical RVI component: one under conditions of no embrittlement (BE_{ne}), for which the "Best Estimate" CUF_{en} line continues to increase as the fatigue-inducing cycles (N) accumulate with time, and the other "Best Estimate" prediction (BE_e) which includes the effect of embrittlement on the (now time-dependent) maximum allowable cycles (N_f). For the latter case, increases in fluence result in more embrittlement and thus a more rapid increase in CUF_{en} than in the former case. This is because after a fluence of about 10^{17} n/cm^2, significant irradiation-induced damage and embrittlement begin to occur. Embrittlement results in the loss of facture toughness and the loss of ductility. Even though the data taken to date have been inconclusive, there is ample evidence that this embrittlement may reduce the number of fatigue cycles to failure (N_f) and thus increase the corresponding CUF_{en}, which is defined as $CUF_{en} = \frac{N}{N_f} F_{en}$.

Q. What does location (1)in Figure-1 show?

A. Location (1) shows that, even if embrittlement is not considered, RVI fatigue failure may occur by the end of life for the period of extended operation. This is because the sum of a

"Best Estimate" prediction plus the uncertainty interval (Δ) associated with this prediction would exceed CUF_{en} = 1.0. In this case the "Limit Line" result is clearly non-conservative.

Q. May I now direct your attention to location (2) in Figure-1. What does this location show?

A. Location (2) shows that embrittlement can result in the physical failure of the RVI well before the end of life for the period of extended operation.

Q. May I direct your attention to location (3) in Figure-1. What does this location show?

A. At location (3), the "Best Estimate" plus uncertainty (Δ) for the case considering embrittlement (BE_e) predicts possible failure of the hypothetical RVI component well before the end of life for the period of extended operation, even though the "Limit Line" prediction indicates substantial margin to failure at this point. That is, at location (3) the CUF_{en} for the "Best Estimate" line, accounting for embrittlement and for all uncertainties (i.e., BE_e + Δ), exceeds unity. Moreover, as I have stated in my previous testimony, highly embrittled RVIs may fail even sooner than by fatigue alone if significant seismic or shock loads occur. That is, fatigue-weakened and embrittled structures cannot tolerate large impulsive seismic and shock loads like fully ductile structures can. Moreover, other synergistic aging effects, such as the thermal embrittlement(TE) of some CASS RVIs (the upper part of the core support columns) would may move the BE_e curve to the left of its position shown in Figure-1 (i.e., the RVI would be predicted to reach failure, CUF_{en} = 1.0, even sooner than the cases (2) and (3) in Figure-1).

Q. Can any conclusions about the “Limit Line” and “Best Estimate” lines be drawn from Figure 1?

A. Yes. Because of Entergy’s failure to explicitly account for uncertainties, and for the effect of embrittlement on fatigued RVIs, we can have no confidence in the “Limit Line” type of CUF_{en} predictions [redacted]

Q. In response to Dr. Hopenfeld’s testimony, the USNRC Staff stated that “CUF or CUF_{en} analyses are not required for safety assessments of DBA events . . . because CUF and CUF_{en}, which are indicators of possible fatigue crack initiation, are not a significant contributor to safety during DBA events." NRC000168, at A145. How do you respond?

A. Again, this is a perfect example of "silo" thinking. The USNRC Staff refuses to assess the risk that a highly fatigued system, component or fitting which has been fatigue-weakened and embrittled as a result of neutron fluence, for example, could fail, relocate, and degrade core cooling, when it is subjected to a DBA LOCA or some other significant shock load. This approach ignores the reality of a reactor environment, where multiple aging mechanisms act simultaneously on RVIs and other components, and that this age-related degradation needs to be taken into account in plant safety analyses.

Q. [redacted]

A. [redacted] As I have stated above, ignoring the effects of accident loads on aging RVI components is not at all appropriate. While it is probably true that fatigued and embrittled RVI components may operate normally during steady-state operations, they can fail suddenly and catastrophically when subjected to a significant shock load. Without considering accident-induced loads, Entergy cannot provide adequate assurance that the RVIs will continue to perform their functions during the period of extended operation (PEO).

Q.   Have there been RVI component failures at nuclear reactors in the past?

A.   Yes. Baffle former bolts and clevis insert bolts have failed at several PWRs in the past, clearly demonstrating that inspections alone will not detect aging effects <u>prior</u> to component failure.

Q.   With respect to the clevis insert bolt failures, do you agree with the USNRC Staff's testimony that "[t]he failed bolts were detected visually" (A290)?

A.   Partially. According the SSER2 at 3-25 (Exh. NYS000507), only 7 out of 29 damaged bolts, or about 24%, were detected visually. The vast majority were not detected visually. Nonetheless, Entergy has proposed to conduct only visual inspections of clevis insert bolts. Exh. NYS000496, at 51.

Q.   [redacted]

[redacted]

A. [redacted] the RVI AMP does not consider the reality of conditions within the reactor. All of the components in and around the clevis insert bolts are also undergoing a range of aging mechanisms which may affect their functionality or their ability to withstand a sudden shock load. [redacted]

Q. [redacted]

A. [redacted] I do not believe that this is a safe or responsible approach to aging management. Although, due to baffle bolt redundancy, the plant might be able to function properly during steady-state operations, when a large percentage of the incore bolting has failed, a significant shock load could cause many of the remaining bolts to suddenly fail, resulting in the relocation of core components and the possible loss of a coolable core geometry.

Q. When discussing aging management of the baffle-former bolts, the USNRC Staff describes the number of baffle-former bolts in "[t]hree-loop Westinghouse design PWRs like IP2 and IP3[.]" NRC000197, at A243. Is it accurate to say that IP2 and IP3 are three-loop PWRs?

A. No. IP2 and IP3 are actually four-loop Westinghouse designed PWRs. It is not clear why the USNRC Staff believes the IP2 and IP3 reactors use a three-loop design. Perhaps this reference is a "cut and paste" remnant from another proceeding for a different facility that USNRC Staff has reused here.

Q. [redacted]

A. [redacted] Virtually all metals (embrittled or not), experience a decrease in ductility as the temperature decreases. However, irradiated carbon steel undergoes a very sharp decrease in ductility at a fluence-dependent temperature commonly called the nil ductility temperatures (NDT), while stainless steel does not. Nevertheless, as I have said before, sufficiently strong shock loads can lead to failure of highly fatigue-weakened and embrittled stainless steel RVIs.

Q. [redacted]

A. [redacted]

[redacted] The USNRC Staff concedes that "[g]iven the variability in assumptions made by different analysts, it is difficult to explicitly quantify the exact overall safety margin present in fatigue calculations." NRC000168, at A210. [redacted] However, given the critical siting of the Indian Point plants in the New York metropolitan area, it is especially appropriate to identify and <u>verify</u> the assumptions and calculations. Indeed, as distilled by the concept of "trust but verify," verification precedes trust.

Q. Do you agree with [redacted] the USNRC Staff that a "propagation of errors" analysis (i.e., an uncertainty analysis), is not needed for EAF calculations?

A. No. [redacted] the USNRC Staff claim that an uncertainty analysis is not necessary because the EAF calculation is "deterministic" and contains adequate conservatisms. NRC000168, at A171; [redacted]

[redacted]

Q. So far you have been talking about reactor vessel internals (RVIs). Do you have some similar concerns about the fatigue evaluations for primary pressure boundary components and fittings?

A. Yes, except for the fact that radiation-induced embrittlement (IE) is not normally an issue, virtually all of my previously discussed concerns are the same. [redacted]

[redacted]

Q. [redacted] the USNRC Staff state that Entergy will compare the actual transients experienced at Indian Point with the transients anticipated in the EAF calculations. Does this approach address your concerns?

A. No. The number of fatigue cycles is obviously important; however, my principal concern has never been that components will experience a greater number of transient cycles than anticipated in the EAF calculation. In contrast, my real concern for the integrity of primary pressure boundary components is that a significant shock load – caused by a seismic, LOCA, or other event – could cause the fatigue-weakened component to suddenly fail well before their associated CUF_{en} value reaches 1.0. [redacted]

A. [redacted] In my opinion, an observable surface crack of 3mm or greater, which would be expected when CUF_{en} reaches 1.0, is quite 'significant'. However, microscopic cracks exist and propagate within the metal structures, components and fittings even when $CUF_{en} < 1.0$. These microscopic cracks cannot be detected by non-destructive testing (NDT), but they can weaken the components and make them more susceptible to failures during a significant seismic or pressure/thermal shock load event, especially when coupled with embrittlement due various thermal embrittlement or corrosion mechanisms.

[redacted]

A. [redacted] what I mean by not reducing design and safety margins as the plant ages is that component fatigue life calculations should retain the original design conservatisms and that the components should be repaired or replaced if they exceed acceptable design margins during the period of extended operations. [redacted]

[redacted] In my opinion, removing modeling conservatisms as IP2 and IP3 exceed 40 years of operation, and the components become more and more degraded, is both irresponsible and dangerous.

Q. [redacted]

[redacted] [redacted]

[redacted] [redacted]

[redacted]

Q. According to the USNRC Staff, aging mechanisms acting on the core support columns are "properly assessed from the normal, steady-state operating conditions." NRC000197, at A306. How do you respond?

A. This approach does not account for seismic or LOCA-type shock loads, which could cause the columns to fail.

Q. I show you what has been marked as Exhibit NYS000563. Do you recognize it?

A. Yes, it is an excerpt from an USNRC training manual, entitled "Reactor Concepts Manual – PWR Systems." It was prepared by the USNRC Technical Training Center.

Q. What information did you draw from this USNRC document?

A. This document identifies the relative location of various components in a 4-loop, Westinghouse, pressurized water reactor (PWR). On page 4-25, it shows that one of the accumulators uses the same nozzle as the residual heat removal (RHR) system, the low pressure coolant injection (LPCI), and the intermediate pressure coolant injection (IPCI) safety systems. In contrast, the high pressure coolant injection (HPCI) system has a separate connection on the same cold leg as the RHR/Accumulator nozzle.

Q. Why is that relevant?

A. [redacted] the components could fail

if subjected to a significant seismic event or shock load. As I have pointed out in some of my previous ASLB testimony, the failure of this important primary pressure boundary structure could lead to a LOCA event in which the Accumulator, LPCI and IPCI systems are not be able to inject water into the reactor's cold leg, and thus into the core, which, in turn, could lead to core melting.

Q. Do you have particular concerns with respect to any other components?

A. Yes. [redacted]

[redacted] the USNRC Staff has indicated that if transients accumulate “at a rate greater than the rate assumed in the fatigue calculation,” Entergy would be permitted to conduct yet another “more refined analysis.” NRC000168, at A106. In short, there is apparently no limit to the number of times CUF_{en} can be recalculated to obtain a CUF_{en} result less than unity, and there is no standard which defines the amount of conservatism that must be retained in these calculations.

Q. Does this complete your testimony?

A. Yes, it does. I do, however, reserve the right to supplement my testimony if new information is disclosed or introduced.

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DECLARATION OF RICHARD T. LAHEY, JR.

I, Richard T. Lahey, Jr., do hereby declare under penalty of perjury that my statements in the foregoing testimony and my statement of professional qualifications are true and correct to the best of my knowledge and belief.

Executed in Accord with 10 C.F.R. § 2.304(d)

Dr. Richard T. Lahey, Jr.

The Edward E. Hood Professor Emeritus of Engineering

Rensselaer Polytechnic Institute, Troy, NY 12180

(518) 495-3884, laheyr@rpi.edu