

In the Matter of:

Entergy Nuclear Operations, Inc.
(Indian Point Nuclear Generating Units 2 and 3)

NYS000567

Submitted: September 9, 2015



ASLBP #: 07-858-03-LR-BD01
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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.

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

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

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

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

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

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

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

21 A. Yes.

22 Q. Has Entergy's and the USNRC Staff's August pre-filed
23 testimony caused you to change the testimony and opinions that

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1 you have previously submitted in this proceeding in connection
2 with Contention NYS-25 and Contention NYS-26B/RK-TC-1B?

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

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

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

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

4 A. There are some data which show that irradiation
5 embrittlement can increase fatigue life in certain situations,
6 especially where high cycle, low amplitude, fatigue occurs.
7 However, there are other data which show that for low cycle,
8 high amplitude fatigue the effects of embrittlement decreases
9 fatigue life by reducing the number of cycles to failure (N_f).
10 As the USNRC Staff concedes, the data regarding the effects of
11 irradiation embrittlement on fatigue life are currently
12 incomplete and inconclusive. NRC000168, at A154; NRC000197, at
13 A196, A200. However, in the face of this uncertainty, Entergy
14 and the USNRC Staff simply assume that the effect can be
15 ignored, and that the Indian Point reactors can continue to be
16 operated until any synergistic degradation effects are directly
17 observed in the operating plants. NRC000197, at A204 ("If
18 synergistic effects of aging mechanisms were to occur, the
19 resulting degradation will likely be found in at least one plant
20 in the fleet.") This is exactly what I mean when I say that
21 Entergy and the USNRC Staff have taken a "wait-and-see"
22 approach. Unfortunately, such an approach could lead to a
23 component failure and potentially serious consequences that

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1 cannot be easily addressed after such a failure has occurred. A
2 much more prudent approach would be to apply an uncertainty
3 factor, or penalty, to the CUF_{en} calculations, and to repair or
4 replace components with very high projected CUF_{en} values. This
5 is a well-accepted engineering practice; indeed, the ASME Code
6 applies a similar uncertainty penalty to CUF_{en} calculations
7 (i.e., a factor of 2 on stress or 20 cycles) to account for
8 uncertainties in test data, which were obtained from tests on
9 small scale, polished metal samples in air, rather than on
10 actual industrial structures, components and fittings in a
11 reactor environment (e.g., in a PWR).

12 [REDACTED]
13 [REDACTED]
14 [REDACTED]

15 A. [REDACTED]
16 [REDACTED]

17 [REDACTED] The operative document for management of aging
18 degradation effects on RVI components at Indian Point is the
19 Revised and Amended RVI Plan, which relies on MRP-227-A and
20 which was the subject of the Second Supplemental Safety
21 Evaluation Report for Indian Point (NUREG-1930, Supplement 2)
22 (Exh. NYS000507). I remain concerned with the adequacy of the
23 Revised and Amended RVI Plan, because it relies entirely on

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1 inspections to determine the condition of RVI components. The
2 absence of detectable surface cracks does not necessarily mean
3 that the embrittled and fatigue-weakened structures, components
4 and fittings are not vulnerable to early (i.e., $CUF_{en} < 1.0$)
5 failures. As MRP-227-A concedes, inspections cannot determine
6 the existence and extent of embrittlement. Accordingly, the
7 Revised and Amended RVI Plan does not account for the
8 possibility that embrittled and fatigue-weakened RVI components
9 could be subject to a shock load which would cause them to fail
10 suddenly.

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

15 A. No. This is a perfect example of the type of "silo"
16 thinking that I am concerned about. According to the USNRC
17 Staff [REDACTED], embrittlement and the associated aging
18 effects can be managed through the inspection-based Revised and
19 Amended RVI Plan, while fatigue is managed through a separate
20 Fatigue Management Plan (FMP), and no further consideration of
21 the interaction between multiple aging mechanisms is necessary.
22 NRC000168, at A185. For example, the USNRC Staff argues that
23 the portions of my June 2015 testimony (NYS000530) relating to

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1 embrittlement are not relevant to the management of metal
2 fatigue, even though I have repeatedly argued that embrittlement
3 effects should be considered when assessing component fatigue
4 life. NRC000168, at A168, A170. As long as CUF_{en} is calculated
5 to be less than 1.0, [REDACTED] the USNRC Staff apparently
6 believe that no critical structures, components, and fittings
7 will exhibit fatigue-induced surface cracking and therefore the
8 effects of embrittlement need not be considered in the fatigue
9 calculation. NRC000168, at A153. Their approach, however,
10 fails to recognize certain basic realities. In particular, the
11 CUF for a structure, component or fitting is defined as the
12 number of fatigue cycles it is expected to experience (N)
13 divided by the number of cycles to failure (N_f). The N_f for a
14 new, ductile material can be significantly greater than the N_f
15 for a highly embrittled material; indeed, this is known to be
16 true for large amplitude, low cycle fatigue. See e.g. Kanaski,
17 et al., "Fatigue and Stress Corrosion Cracking Behaviors of
18 Irradiated Stainless Steels in PWR Primary Water," ICONE-5, at
19 2372 (May 1997) (Exh. NRC000177); Arai, et al., "Irradiation
20 Embrittlement of PWR Internals," Proceedings ASME/JSME 2d
21 International Nuclear Engineering Conference, Vol. 2, at 103
22 (1993) (Exh. NYS000564); Korth, G.E. & Harper, M.D., "Effects of
23 Neutron Radiation on the Fatigue and Creep/Fatigue Behavior of

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

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

15 A. Yes.

16 Q. How are you familiar with this Figure?

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

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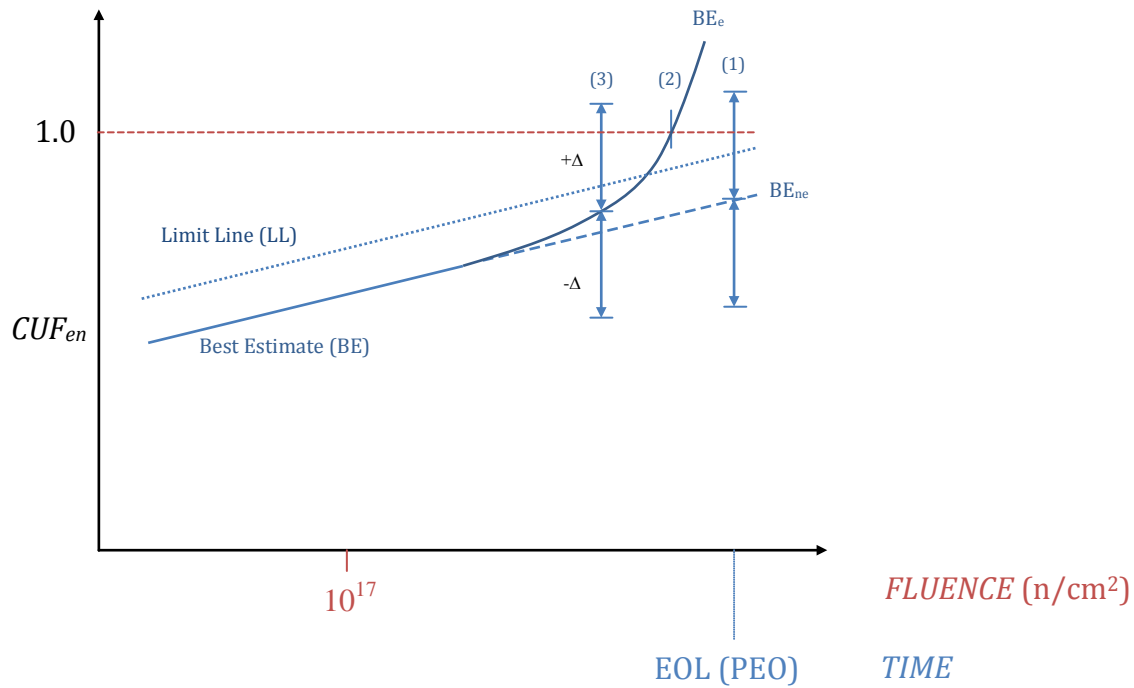


Figure 1: Comparison of Limit Line (LL) and Best Estimate (BE) predictions, with embrittlement (BE_e) and without embrittlement (BE_{ne}).

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.

1 Q. Why did you prepare this Figure?

2 A. I prepared this Figure to visually illustrate some of

3 the concerns that I have set forth in my testimony in this

4 proceeding. Specifically, this Figure shows three important

5 things: (1) a typical WESTEMS™ "Limit Line" prediction, (2) a

6 "Best Estimate" prediction of CUF_{en}, the cumulative usage factor

7 under reactor operating conditions, and the associated

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1 uncertainty interval (Δ), and, (3) how the neutron fluence-
2 induced embrittlement of a RVI structure, component, or fitting
3 can dramatically increase the possibility of a fatigue failure,
4 as represented by the cumulative usage factor under PWR
5 operating conditions, CUF_{en} , being equal to unity. This figure
6 applies to RVI structures, components, and fittings made of
7 stainless steel or other materials. In summary, in Figure-1, I
8 have illustrated two different approaches for calculating a
9 time-dependent CUF_{en} , namely the "Limit Line" and the "Best
10 Estimate" calculation. For the "Best Estimate" line, I also
11 illustrate typical uncertainty intervals, delta (Δ), and the
12 effects of fatigue on embrittled (e) and on non-embrittled (ne)
13 RVIs.

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

16 A. Since fluence = ϕt (where ϕ is the fast neutron flux
17 (n/cm^2-s) and t is the time of operation of the reactor) the x-
18 axis tracks both the fluence (ϕ) and the time (t). As time
19 passes, each RVI reaches the reactor's End of Life (EOL) for the
20 Period of Extended Operation (PEO). Also, as we move from left-
21 to-right along the x-axis (i.e., as the time of reactor
22 operation increases), we see an increase in the number of

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1 fatigue cycles (N) that the various RVIs (one of which has been
2 chosen to be displayed in Figure-1) have been subjected to.

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

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

12 Q. In Figure-1 what does the "Limit Line" represent?

13 A. The "Limit Line" represents [REDACTED]
14 [REDACTED] as a supposedly-conservative [REDACTED]
15 calculation of CUF_{en} for a hypothetical RVI component or
16 structure. [REDACTED]

17 [REDACTED]
18 [REDACTED] The "Limit Line" shown in Figure-1 includes
19 Entergy's implicit assumption that a stainless steel RVI
20 structure, component, or fitting remains perfectly ductile, and
21 thus shows that the CUF_{en} for a RVI will increase with an
22 increasing number of fatigue cycles, but that it is not affected
23 by the fluence. The "Limit Line", therefore, uniformly

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1 increases in CUF_{en} even after the fluence exceeds 10^{17} n/cm²,
2 where significant embrittlement begins to develop (i.e., it is
3 assumed that the number of cycles to failure are not a function
4 of fluence). Figure-1 shows that, as the RVI component
5 approaches the end of life (EOL) for the period of extended
6 operation (PEO), [REDACTED] "Limit Line" will approach
7 (but still be below) $CUF_{en} = 1.0$. This is depicted as location
8 (1).

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

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

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

21 A. This type of analysis considers important parameters
22 that have some uncertainty associated with them; for example,
23 the coarseness of the computational mesh, uncertainties in F_{en} ,

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1 the "best estimate" local heat transfer coefficients, etc.;
2 uncertainties which, in turn, impact the RVI's actual fatigue
3 life (i.e., CUF_{en}). The net uncertainty intervals (Δ) also
4 account for uncertainties in the number of transients
5 (including, for example, seismic events), and synergistic
6 effects of radiation (i.e., the fluence) and stress corrosion
7 effects on metal fatigue. Accounting for the uncertainty in
8 these parameters allows one to estimate the overall uncertainty
9 (Δ) of the "Best Estimate" CUF_{en} prediction, so that we can see
10 if the [REDACTED] "Limit Line" predictions are indeed
11 sufficiently conservative. That is, comparing the "Best
12 Estimate" results, plus Δ , with the "Limit Line" predictions
13 reveals whether the "Limit Line" approach [REDACTED]
14 [REDACTED] is adequate or
15 not.

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

18 A. Yes, uncertainty analyses can be performed in a number
19 of ways. One of the most commonly used methods by engineers is
20 the method proposed by Kline & McClintock [ASME J. Mechanical
21 Engineering, Vol.75, No.1, 3-8, Jan. 1953] (Exh. NYS000514).
22 For the case in which the "Best Estimate" value of CUF_{en} involves
23 "N" parameters, each having some uncertainty associated with

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1 them (e.g., the Dittus-Boelter convective heat transfer, [REDACTED]
2 [REDACTED] has an inherent uncertainty of about $\pm 25\%$),
3 the net uncertainty (Δ) is given by:

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

4
5
6 This approach is widely used by engineers since it is relatively
7 easy to evaluate and it gives an acceptable approximation of the
8 "error bars" (Δ) in a "Best Estimate" evaluation.

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

11 A. Not necessarily. As shown in Figure-1, a "Best
12 Estimate" prediction plus the uncertainty (Δ) may predict
13 possible RVI structure, components or fitting failures (i.e.,
14 $CUF_{en} = 1.0$), sooner than the "Limit Line" would. Moreover, the
15 effect of embrittlement can make the situation much worse. The
16 problem is that no one knows how much conservatism, if any, is
17 in the [REDACTED] "Limit Line". As a consequence, we can have no
18 confidence that "Limit Line" results near $CUF_{en} = 1.0$, are
19 sufficiently conservative to bound all the possible
20 uncertainties, as illustrated in Figure-1 by Δ .

21 Q. In Figure-1, the "Best Estimate" line is shown for the
22 case in which there is no effect of embrittlement on fatigue,
23 BE_{ne} , [REDACTED] and the case in

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1 which embrittlement degrades fatigue life, BE_e . Can you explain
2 what these lines mean?

3 A. Yes, the plot depicts two "Best Estimate" predictions
4 for a hypothetical RVI component: one under conditions of no
5 embrittlement (BE_{ne}), for which the "Best Estimate" CUF_{en} line
6 continues to increase as the fatigue-inducing cycles (N)
7 accumulate with time, and the other "Best Estimate" prediction
8 (BE_e) which includes the effect of embrittlement on the (now
9 time-dependent) maximum allowable cycles (N_f). For the latter
10 case, increases in fluence result in more embrittlement and thus
11 a more rapid increase in CUF_{en} than in the former case. This is
12 because after a fluence of about 10^{17} n/cm², significant
13 irradiation-induced damage and embrittlement begin to occur.
14 Embrittlement results in the loss of fracture toughness and the
15 loss of ductility. Even though the data taken to date have been
16 inconclusive, there is ample evidence that this embrittlement
17 may reduce the number of fatigue cycles to failure (N_f) and thus
18 increase the corresponding CUF_{en} , which is defined as $CUF_{en} = \frac{N}{N_f} F_{en}$.

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

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

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

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

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

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

11 A. At location (3), the "Best Estimate" plus uncertainty
12 (Δ) for the case considering embrittlement (BE_e) predicts
13 possible failure of the hypothetical RVI component well before
14 the end of life for the period of extended operation, even
15 though the "Limit Line" prediction indicates substantial margin
16 to failure at this point. That is, at location (3) the CUF_{en} for
17 the "Best Estimate" line, accounting for embrittlement and for
18 all uncertainties (i.e., $BE_e + \Delta$), exceeds unity. Moreover, as I
19 have stated in my previous testimony, highly embrittled RVIs may
20 fail even sooner than by fatigue alone if significant seismic or
21 shock loads occur. That is, fatigue-weakened and embrittled
22 structures cannot tolerate large impulsive seismic and shock
23 loads like fully ductile structures can. Moreover, other

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1 synergistic aging effects, such as the thermal embrittlement (TE)
2 of some CASS RVIs (the upper part of the core support columns)
3 would may move the BE_e curve to the left of its position shown in
4 Figure-1 (i.e., the RVI would be predicted to reach failure,
5 $CUF_{en} = 1.0$, even sooner than the cases (2) and (3) in Figure-1).

6 Q. Can any conclusions about the "Limit Line" and "Best
7 Estimate" lines be drawn from Figure 1?

8 A. Yes. Because of Entergy's failure to explicitly
9 account for uncertainties, and for the effect of embrittlement
10 on fatigued RVIs, we can have no confidence in the "Limit Line"
11 type of CUF_{en} predictions [REDACTED]

12 [REDACTED]
13 [REDACTED]
14 [REDACTED]
15 [REDACTED]
16 [REDACTED]
17 [REDACTED]
18 [REDACTED]
19 [REDACTED]

20 Q. In response to Dr. Hopenfeld's testimony, the USNRC
21 Staff stated that "CUF or CUF_{en} analyses are not required for
22 safety assessments of DBA events . . . because CUF and CUF_{en} ,
23 which are indicators of possible fatigue crack initiation, are

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1 not a significant contributor to safety during DBA events."

2 NRC000168, at A145. How do you respond?

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

13 Q. [REDACTED]
14 [REDACTED]
15 [REDACTED]
16 [REDACTED]
17 [REDACTED]
18 [REDACTED]
19 [REDACTED]

20 A. [REDACTED] As I have stated above, ignoring the
21 effects of accident loads on aging RVI components is not at all
22 appropriate. While it is probably true that fatigued and
23 embrittled RVI components may operate normally during steady-

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1 state operations, they can fail suddenly and catastrophically
2 when subjected to a significant shock load. Without considering
3 accident-induced loads, Entergy cannot provide adequate
4 assurance that the RVIs will continue to perform their functions
5 during the period of extended operation (PEO).

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

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

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

15 A. Partially. According the SSER2 at 3-25 (Exh.
16 NYS000507), only 7 out of 29 damaged bolts, or about 24%, were
17 detected visually. The vast majority were not detected
18 visually. Nonetheless, Entergy has proposed to conduct only
19 visual inspections of clevis insert bolts. Exh. NYS000496, at
20 51.

21 Q. [REDACTED]
22 [REDACTED]
23 [REDACTED]

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1

[REDACTED]

2

[REDACTED]

3

[REDACTED]

4

A. [REDACTED] the RVI

5

AMP does not consider the reality of conditions within the

6

reactor. All of the components in and around the clevis insert

7

bolts are also undergoing a range of aging mechanisms which may

8

affect their functionality or their ability to withstand a

9

sudden shock load. [REDACTED]

10

[REDACTED]

11

[REDACTED]

12

[REDACTED]

13

[REDACTED]

14

[REDACTED]

15

[REDACTED]

16

[REDACTED]

17

Q. [REDACTED]

18

[REDACTED]

19

[REDACTED]

20

[REDACTED]

21

A. [REDACTED] I do not believe that this is a safe or

22

responsible approach to aging management. Although, due to

23

baffle bolt redundancy, the plant might be able to function

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1 properly during steady-state operations, when a large percentage
2 of the incore bolting has failed, a significant shock load could
3 cause many of the remaining bolts to suddenly fail, resulting in
4 the relocation of core components and the possible loss of a
5 coolable core geometry.

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

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

16 Q. [REDACTED]
17 [REDACTED]
18 [REDACTED]
19 [REDACTED]
20 [REDACTED]
21 [REDACTED]
22 [REDACTED]

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

9 Q. [REDACTED]
10 [REDACTED]
11 [REDACTED]

12 A. [REDACTED]
13 [REDACTED]
14 [REDACTED]
15 [REDACTED]
16 [REDACTED]
17 [REDACTED]
18 [REDACTED]
19 [REDACTED]
20 [REDACTED]
21 [REDACTED]
22 [REDACTED]
23 [REDACTED]

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[REDACTED]

[REDACTED]

[REDACTED]

[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]

[REDACTED]

[REDACTED] However, given the critical siting of the Indian Point plants in the New York metropolitan area, it is especially appropriate to identify and verify 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]

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1 [REDACTED]

2 [REDACTED]

3 [REDACTED]

4 [REDACTED]

5 [REDACTED]

6 [REDACTED]

7 [REDACTED]

8 [REDACTED]

9 [REDACTED]

10 [REDACTED]

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

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

18 [REDACTED]

19 [REDACTED]

20 [REDACTED]

21 [REDACTED]

22 [REDACTED]

23 [REDACTED]

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[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]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

[REDACTED]

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1 A. [REDACTED] In my opinion, an observable surface crack of
2 3mm or greater, which would be expected when CUF_{en} reaches 1.0,
3 is quite 'significant'. However, microscopic cracks exist and
4 propagate within the metal structures, components and fittings
5 even when $CUF_{en} < 1.0$. These microscopic cracks cannot be
6 detected by non-destructive testing (NDT), but they can weaken
7 the components and make them more susceptible to failures during
8 a significant seismic or pressure/thermal shock load event,
9 especially when coupled with embrittlement due various thermal
10 embrittlement or corrosion mechanisms.

11 [REDACTED]
12 [REDACTED]
13 [REDACTED]
14 [REDACTED]
15 [REDACTED]

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

22 [REDACTED]
23 [REDACTED]

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

5 Q. [REDACTED]
6 [REDACTED]
7 [REDACTED]
8 [REDACTED]
9 [REDACTED]
10 [REDACTED]
11 [REDACTED]
12 [REDACTED]
13 [REDACTED]
14 [REDACTED]
15 [REDACTED]
16 [REDACTED]
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[REDACTED]

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

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

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

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

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

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

17 Q. Why is that relevant?

18 A. [REDACTED]
19 [REDACTED]
20 [REDACTED]
21 [REDACTED]
22 [REDACTED]

23 [REDACTED] the components could fail

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

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

10 A. Yes. [REDACTED]
11 [REDACTED]
12 [REDACTED]
13 [REDACTED]
14 [REDACTED]
15 [REDACTED]
16 [REDACTED]
17 [REDACTED]
18 [REDACTED]
19 [REDACTED]
20 [REDACTED]
21 [REDACTED]
22 [REDACTED]
23 [REDACTED]

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[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?

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1 A. Yes, it does. I do, however, reserve the right to
2 supplement my testimony if new information is disclosed or
3 introduced.
4

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1 UNITED STATES

2 NUCLEAR REGULATORY COMMISSION

3 BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

4 -----X

5 In re: Docket Nos. 50-247-LR; 50-286-LR

6 License Renewal Application Submitted by ASLBP No. 07-858-03-LR-BD01

7 Entergy Nuclear Indian Point 2, LLC, DPR-26, DPR-64

8 Entergy Nuclear Indian Point 3, LLC, and

9 Entergy Nuclear Operations, Inc. September 9, 2015

10 -----X

11 **DECLARATION OF RICHARD T. LAHEY, JR.**

12 I, Richard T. Lahey, Jr., do hereby declare under penalty

13 of perjury that my statements in the foregoing testimony and my

14 statement of professional qualifications are true and correct to

15 the best of my knowledge and belief.

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



17 _____

18

19 Dr. Richard T. Lahey, Jr.

20 The Edward E. Hood Professor Emeritus of Engineering

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