Exhibit NYSR00343
Revised: December 27, 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 December 22, 2011

STATE OF NEW YORK AND RIVERKEEPER, INC.
INITIAL STATEMENT OF POSITION
CONSOLIDATED CONTENTION NYS-26B/RK-TC-1B

Office of the Attorney General
for the State of New York
The Capitol
State Street
Albany, New York 12224

STATEMENT OF POSITION

PRELIMINARY STATEMENT

In accordance with 10 C.F.R. § 2.1207(a)(1) and the Atomic Safety and Licensing Board's Scheduling Order, dated June 7, 2011, and updated on October 7, 2011, the State of New York and Riverkeeper, Inc. (hereinafter "Petitioners") hereby submit the Initial Statement of Position on admitted New York State/Riverkeeper Consolidated Contention NYS-26B/RK-TC-1B. The contention is supported by the expert testimony of Dr. Richard T. Lahey and Dr. Joram Hopenfeld and numerous technical documents.

NYS - 26B/RK-TC-1B alleges:

> Entergy's License Renewal Application does not include an adequate plan to monitor and manage the effect of aging due to metal fatigue on key reactor components in violation of 10 C.F.R. § 54.21(c)(1)(iii).[1]

This contention challenges the adequacy of Entergy's reanalyses and its plan to manage the effects of metal fatigue during the requested extended license terms.[2] Entergy has failed to demonstrate that its AMP for metal fatigue is legally sufficient because:

1. The methodology to determine whether CUFen for any particular component is >1 - *i.e.* the WESTEMs computer program - is technically deficient;

[1] Entergy has conceded its response to the metal fatigue issue is to seek to demonstrate that it complies with 10 C.F.R. § 54.21(c)(1)(iii) and thus that it must demonstrate that "[t]he effects of aging on the intended function(s) will be adequately managed for the period of extended operation." *See* Memorandum And Order (Ruling on Motion for Summary Disposition of NYS-26/26A/Riverkeeper TC-1/1A (Metal Fatigue of Reactor Components) and Motion for Leave to File New Contention NYS-26B/Riverkeeper TC-1B), November 10, 2010 at 5.

[2] State of New York's and Riverkeeper's Motion for Leave to File a New and Amended Contention Concerning the August 9, 2010 Entergy Reanalysis of Metal Fatigue (Sept. 9, 2010).

2. The input values chosen by Entergy for its use of WESTEMs are not technically defensible and understate the extent of metal fatigue;

3. The range of components for which the CUFen calculations are proposed to be conducted is too narrow.

PROCEDURAL HISTORY

Entergy's LRA included the results of CUFen calculations that identified several components whose CUFen values were <1. LRA at Tables 4.3-13 and 4.3-14. Initially Entergy proposed to merely choose from several options, at some time after license renewal hearings were completed, how it would address these excess values. LRA at 4.3-20 to 4.3-23. However it eventually chose to redo the CUFen calculations in order to attempt to demonstrate that all relevant components would have CUFens <1. *See* Letter from Fred Dacimo, Entergy Nuclear Northeast, to U.S. Nuclear Regulatory Commission, NL-10-082 (Aug. 9, 2010) (ML102300504) (NYS000352) [hereinafter NL-10-082]; *see also* Letter from Kathryn M. Sutton and Paul M. Bessette, Counsel for Entergy Nuclear Operations, Inc., to Atomic Safety and Licensing Board (Aug. 10, 2010) (ML102310325) (NYS000352). This latter filing resulted in New York and Riverkeeper filing Contention NYS-26B/RK-TC-1B which was admitted by the Board. *Entergy Nuclear Operations, Inc.* (Indian Point Nuclear Generating Units 2 and 3), Memorandum and Order (Ruling on Motions for Summary Disposition of NYS-26/26A/RiverkeeperTC-1/1A (Metal Fatigue of Reactor Components) and Motion for Leave to File New Contention NYS-26B/Riverkeeper TC-1B)) (November 4, 2010) ("Summary Disposition Order") at 28-29. The Board also dismissed as moot Entergy's Motion for Summary Disposition of New York and

Riverkeeper's previous metal fatigue contentions. *Id.*

In its ruling on the admissibility of NYS-26B/RK-TC-1B the Board also addressed the applicability of the Commission's decision in *Entergy Nuclear Vermont Yankee, L.L.C., & Entergy Nuclear Operations, Inc.* (Vermont Yankee Nuclear Power Station), CLI-10-17, 72 NRC __ (July 8, 2010) where the Commission set forth various standards applicable to contentions related to metal fatigue contentions. The Board held that:

> While it is clear from the Commission's ruling in Vermont Yankee that an Applicant cannot be required to perform CUFen calculations in evaluating TLAAs for meeting Section 54.21(c)(1)(i) and (ii), this is not an issue here because Entergy volunteered to perform CUFen calculations in all instances and has used the results from these calculations in addressing Section 54.21(c)(1)(iii). See Vermont Yankee, CLI-10-17, 72 NRC at __ (slip op. at 48); NL-10-082, Attach. 1, Fatigue Monitoring Program Clarification at 1.

Summary Disposition Order at 13, n.58.

LEGAL STANDARDS AND REGULATORY FRAMEWORK

In Applicant's Motion for Summary Disposition of New York State Contentions 26/26A & Riverkeeper Technical Contentions 1/1A (Metal Fatigue of Reactor Components) filed August 10, 2010, Entergy made clear that the legal bases upon which it relies for its metal fatigue AMP is that it complies with NUREG-1801, Rev. 1, *Generic Aging Lessons Learned* ("GALL") and NUREG/CR-6260 (Feb. 2005)(NYS000146A-C), *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components* (Feb. 1995)(NYS000355). However, to prevail in its argument Entergy must demonstrate both that it does in fact comply with those requirements of those two guidance documents and that compliance with them is adequate to meet the requirements of 10 C.F.R. § 54.21(c)(1)(iii). As discussed below, Entergy

fails to accomplish either of these goals.

First, Entergy conducted a CUFen analysis of certain components as part of its initial LRA filing and found several components whose CUFen values were >1. LRA at Tables 4.3-13 and 4.3-14. When components are found not to comply with the acceptance criteria (i.e., CUFen >1), "corrective actions" must be taken, which "include a review of additional affected reactor coolant pressure boundary locations." GALL at X M-2 (NYS000146A-C); *see also* MRP-47, Revision 1, Electric Power Research Institute, *Materials Reliability Program:Guidelines.for Addressing Fatigue Environmental Effects in a License Renewal Application*, at 3-4 (2005) ("MRP-47")(NYS000350). Entergy failed to expand its analysis of relevant locations to the extent required by GALL.

Second, by committing to compliance with GALL, Entergy is committed to a metal fatigue AMP that "addresses the effects of the coolant environment on component fatigue life by assessing the impact of the reactor coolant environment on a sample of critical components for the plant" and is capable of "[m]aintaining the fatigue usage factor below the design code limit". GALL, Rev. 1, Vol. 2 at X M-1 (NYS000146A-C). However, it is not sufficient for Entergy to merely assert that its AMP methodology for calculating CUFen "addresses the effects of the coolant environment on component fatigue life" and will "[maintain] the fatigue usage factor below the design code limit. It must demonstrate that the methodology it has accepted will actually achieve those goals. Summary Disposition Order at 24.

New York and Riverkeeper offer the opinion of Drs. Lahey and Hopenfeld, as well as numerous documents, that demonstrate in several important respects how the AMP methodology

that Entergy intends to use for its AMP is deficient and is unable to achieve the GALL goals to which Entergy is committed.

SUPPORTING EVIDENCE

This section sets forth the evidence that supports NYS-26-B/RK-TC-1B. Intervenors are relying on the pre-filed testimony and reports of Dr. Richard T. Lahey and Dr. Joram Hopenfeld and numerous technical documents.[3] Because portions of the Reports of Drs. Lahey and Hopenfeld reference and discuss material alleged to be proprietary and in order to avoid creating two versions of this Statement of Position, this summary is general in many instances and does not discuss in detail the specific examples used in the Reports.

Fatigue is a very important age-related safety concern, particularly when a significant 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 a plant-specific aging management program (AMP) for the extended operation of a nuclear 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

[3] *See* Prefiled Written Testimony of Dr. Joram Hopenfeld regarding NYS-26-B/RK-TC-1B, Exh. RIV000034, Report of Dr. Joram Hopenfeld in support of NYS-26-B/RK-TC-1B, Exh. RIV000035, Pre-Filed Written Testimony Of Richard T. Lahey, Jr. Regarding Contention NYS-25, Supplemental Pre-Filed Written Testimony Of Richard T. Lahey, Jr. Regarding Contention NYS-26 and Report and Supplemental Report of Dr. Richard T. Lahey, Jr. (Exhibit NYS000296 and 000297) and all exhibits identified as relevant to this Contention.

significant cracking is expected occurs when CUF = 1.0, and one must 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 (NYS000357)] to degrade the fatigue life of immersed metal structures, components and fittings, the USNRC also requires that environmental corrections be applied to the calculated CUF, and it specifies the formulas/curves to be used for these corrections [*e.g.,* NUREG/CR-5704 (NYS000354); NUREG/CR-6583(NYS000356)]. Significantly, CUF_{en} >CUF and the environmentally-adjusted fatigue analyses must satisfy CUF_{en} < 1.0 during extended plant operations. These fatigue evaluations do not take into account that the structures, components and fittings being analyzed may have experienced significant corrosion and irradiation-induced embrittlement, and thus can experience early fatigue-induced failures.

In the original relicensing submittal for Indian Point 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 several important structures and components will significantly exceed the environmentally-adjusted CUF_{en} = 1.0 criterion during the proposed extended operations period. LRA at Tables 4.3-13 and 4.3-14. In particular, the pressurizer surge line and nozzle, and the reactor coolant system charging system nozzle (on the primary side), and 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 (CRD) 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]). *Id*. Once CUF_{en}

violations of this type are found, Entergy was expected [GALL, Rev.1, Vol.2, pg.X M-2 (NYS000146A-C); EPRI, MRP-47at 3-4 (NYS000350)] to also do fatigue analyses for other potentially limiting reactor structures, components and fittings. However, a systematic fatigue analysis of other structures, components and fittings was apparently not done.

Nevertheless, in order to perform more mechanistic, but less conservative, fatigue evaluations, Entergy contracted with Westinghouse to redo the ex-vessel fatigue analyses for IP-2 and IP-3. These results were reported separately [WCAP-17199-P / WCAP-17200-P, "Environmental Fatigue Evaluation for Indian Point Unit 2/3" (June 2010)(NYS000361-NYS000362)]. Rather than performing standard ASME code evaluations, as Entergy had done before, these calculations were done using WESTEMS, a proprietary computer code of Westinghouse. Subsequent to submitting these calculations to the ASLB, the ASLB directed Entergy to release the WESTEMS code manuals to New York for review. The parts of the manuals transmitted to New York contained assumptions and models (particularly for the thermal-hydraulics models) used by Westinghouse in WESTEMS.

The new CUF_{en} results filed with the ASLB by Entergy [NL-10-82] show that the previously most limiting CUF_{en} were reduced by more than an order of magnitude (*e.g.*, the results for the pressurizer surge line piping and the RCS piping charging system nozzle), which is a very significant reduction, and one that must be very carefully verified since it significantly reduces the design safety margins implicit in the original ASME code evaluations. 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. In particular, for the IP-2 RHR line, CUF_{en} = 0.9434, and for the IP-3 RHR line, CUFen = 0.9961. Thus, virtually any error in these results could lead to a violation of the USNRC's CUFen = 1.0 limit.

Report and Testimony of Dr. Richard T. Lahey

Entergy relies on the WESTEMS computer code and its analytical results to demonstrate that it has a legally adequate AMP for metal fatigue. However, no error analysis of the WESTEMS results was generated by Westinghouse nor provided to the Board or New York 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. One would normally expect to see a detailed 'propagation-of-error' type of analysis [*see e.g.*, Vardeman & Jobe, "Basic Engineering Data Collection and Analysis," Duxbury, pp. 310-311 (2001) (NYS000347)] to determine the overall uncertainty in the CUFen results given by WESTINGHOUSE It is well known that all engineering analyses are based on imperfect mathematical models of reality and various code user assumptions which inherently involve some level of error. In addition, as the USNRC Staff confirmed in the SSER [NUREG-1930, Supplement 1 at 4-2 (NYS000326A-F)], WESTEMS permits the code user to make assumptions and interventions that can affect the outcome. As a consequence, without a well-documented error analysis, the accuracy of Entergy's new fatigue results are quite uncertain and thus can not be used to establish the integrity of the structures, components or fittings being analyzed. What is clear is that there are many possible sources of error in the results that Entergy (and Westinghouse) have provided to the ASLB and the parties.

(The additional Supplemental Report includes a discussion that Entergy and Westinghouse may consider as falling within the scope of the Confidentiality Order in this proceeding and is submitted separately in support of Contention NYS-26B/RK-TC-1B).

A consideration of some of the Indian Point reactor components that were analyzed for Entergy by Westinghouse underscores the importance of the concerns raised in this Contention. The SMiRT-19 paper [Cranford & Gary-W, 8/07 (NYS000360)] notes that a PWR's residual heat removal (RHR) system shares nozzles and piping with the plant's emergency core cooling system (ECCS). In particular, in the fatigue analysis for the ECCS accumulators (which passively inject ECC water into the cold leg of the primary system in the event of a LOCA) and the RHR system (which is normally used during each plant shut-down) one must combine the fatigue usage of both systems for their common components (i.e., the nozzle at the penetration into the cold leg) to obtain the resultant CUF_{en}. This design feature implies that a RHR/accumulator nozzle failure during a LOCA may breach the path for accumulator water injection into the RPVs downcomer region, thus preventing ECC water from reaching the core to mitigate core melting. This is obviously one of the most serious primary system boundary failures that can occur in a PWR. Hence, it is very troubling that the largest CUF_{en} calculated for the various components analyzed by Westinghouse were for this critical component. That is, CUF_{en} = 0.9961 for IP-3 RHR [NL-10-082, Table 4.3-14], and CUF_{en} = 0.9434 for IP-2 RHR [NL-10-082, Table 4.3-13 (NYS000352)], where both of these values were nearly equal to the CUF_{en} limit of 1.0. It is also significant to note that even though these were transient evaluations, in which the flow and thus convective heat transfer coefficient varied with time, a

constant heat transfer coefficient was specified by the user and some model modifications were apparently required to be made to obtain CUF_{en} < 1.0. In addition, a very simple, steady-state model was used to correct for the effect of the thermal sleeve, and unlike the thermal stress models, which showed an angular dependence at the nozzle's weld, the thermal sleeve model which was used had no angular dependence at all (i.e., it was a potentially non-conservative modeling assumption). Other reactor components analyzed using WESTEMS also had disturbingly large values for CUFen. [NL-10-082, Tables 4.3-13, 4.3-14 (NYS000352).

There is a lot of "engineering judgment" implicit in the CUFen results, and, since an error analysis has not been done to bound the uncertainty, and many results are disturbingly close to the CUFen = 1.0 limit, the results produced are not a reliable basis to assure the safety of IP-2 and IP-3 during extended plant operations. Indeed, these results are quite uncertain and this uncertainty should be quantified by doing parametric runs and a detailed error analysis. Moreover, because the effect of various shock loads on the failure of these fatigue-weakened components, structures, and fittings has not been considered, it is unclear that the health and safety of the American public is being adequately protected. As previously noted, these results are quite uncertain and this uncertainty needs to be quantified by doing a detailed error analysis. Earlier this year, the USNRC Staff required Entergy to disclose and make clear those user interventions that will be used in future WESTEMS analysis of IP-2 and IP-3 [SSER, NUREG-1930, Supplement 1 at 4-2 NYS000326A-F)], but this disclosure was not required for the results that Entergy has already submitted to the ASLB for IP-2 and IP-3. This information is needed in order to do a proper review of these important results.

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) (NYS000324); USNRC Staff Report, "Final Safety Evaluation 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 Associate Pressure Boundary Components, Project No. 686," (Nov. 8, 2000) NYS000353)], and B&W designed PWRs have had fatigue-induced failures of various in-core components even when CUF < 1.0 (perhaps due to undetected manufacturing flaws) [Entergy Email: Esquillo to Stuard et al., Subject: "Section XI - Cracking" (8/30/06) NYS000316)]. Significantly, the possible effect of fatigue on the failure of RPV internals was apparently well known to Entergy [Entergy Email: Batch to Finnin, Subject: "Need to Evaluate High Cycle Fatigue to IPEC Baffle Bolts?" (12/28/06) (NYS000315)]. Moreover, unlike postulated nuclear reactor accidents, the fatigue failures of in-core bolts are actual events that have happened and will likely happen again for sufficiently stressed materials. It is not possible to inspect (*e.g.*, using UT techniques) all the bolts within a RPV, and thus the nuclear industry has recommended [EPRI Report, MRP-228; "Materials Reliability Program: Inspection Standard for PWR Internals," (July 2009) (NYS000323)] 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 do not take into account the possibility of various accident-induced pressure and/or thermal shock loads within the RPV, such as those due to a DBA LOCA. Thus, the number of intact bolts which might be adequate

for normal operations may be totally inadequate to accommodate shock loads during accidents. Not doing a realistic safety analysis is totally unacceptable since shock-load-induced bolting failures may lead to a blocked or distorted core geometry which, in turn, may not allow the ability to adequately cool the core and can lead to core melting.

As for all mechanical systems, when 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 in-core bolting fatigue failure concerns, many other highly irradiated in-core structures, components and fittings (e.g., core baffles, formers, etc.) and welds (e.g., on thermal shields [Westinghouse Owners Group Report, March 2001 (NYS000341)]) will be subjected to some of the same (and even more) fatigue-inducing transients as those which effect the structures, components and fittings that are external to the RPV (e.g., those piping systems, components and fittings that were analyzed by W). However, no fatigue analysis of these important RPV internals was provided by Entergy, and there was apparently no recognition of the importance of a DBA LOCA, secondary side LOCA and ATWS loads on the integrity of these structures, components and fittings. As for in-core bolting, not doing a proper fatigue and safety analysis of these embrittled and fatigued RPV internal structures, components and fittings is completely unacceptable since shock-load-induced failures of RPV internals may lead to a distorted or blocked core geometry, which may, in turn, not allow the ability to adequately cool the core and result in core melting.

As noted in Dr. Lahey's Report in support of Contention 25 and 26 and his Prefiled

Testimony for both Contentions, *see* NYS000296/NYS000343, NYS000294/NYS000344, respectively, there are a number of degradation mechanisms which impact the integrity of critical components inside the reactor pressure vessel and within the reactor coolant pressure boundary. Some of these components, like the reactor vessel internals, are subjected to metal fatigue, embrittlement and stress corrosion cracking. However, in developing the details of AMPs for various components inside the reactor pressure vessel and within the reactor coolant pressure boundary, neither Entergy, NRC Staff nor industry consultants, have considered the synergistic effects of all these mechanisms. Thus, the criteria for determining when a component is unacceptable considers metal fatigue separately from embrittlement and separately from stress corrosion cracking thus creating the real possibility that a critical component could be simultaneously close to, but over the acceptance in two or more of these categories and no corrective action would be taken. Given the vital importance of these components, that possibility is unacceptable and fails to provide the assurance that critical components will not fail in the event of a severe shock such as will occur if there is a DBA LOCA.

Testimony and Report of Dr. Joram Hopenfeld

As described in additional detail in his report and testimony, Dr. Hopenfeld describes the aging phenomenon of metal fatigue, and why it poses a potential safety risk for nuclear plant operations if not managed appropriately. Dr. Hopenfeld concludes that Entergy has failed to demonstrate that it will adequately and effectively manage the serious aging mechanism of metal fatigue throughout the proposed extended licensing terms at Indian Point. Entergy's "refined" reanalyses of environmental fatigue factors are flawed and inaccurate, and fail to demonstrate

that certain components will remain within allowable acceptance criteria for metal fatigue. After reviewing Entergy's LRA and relevant Amendments, Entergy's "refined" " Environmental Fatigue Evaluations for Indian Point Units 2 and 3, RAIs from NRC Staff and corresponding responses by Entergy, numerous documents disclosed by Entergy as relevant to Intervenors' metal fatigue contention, the NRC Staff's Safety Evaluation Report, and other information, Dr. Hopenfeld's analysis shows that many key components may become susceptible to metal fatigue and pose safety risks during the proposed periods of extended operation. In light of this information, Entergy should have, but did not, expand the scope of components to be assessed for metal fatigue. Finally, Entergy has otherwise not provided sufficient details concerning its Aging Management Program ("AMP") to ensure that the degradation effects of metal fatigue would be adequately handled during the proposed period of extended operation.

1. Entergy's "refined" reanalyses of environmental fatigue factors are flawed and inaccurate, and fail to demonstrate that certain components will remain within allowable acceptance criteria for metal fatigue.

In order to assess the susceptibility of plant components to metal fatigue, the individual usage factor in air is multiplied by a corresponding environmental correction factor, "Fen." Fen is the ratio of the fatigue life in air at room temperature to the fatigue life in water at the local temperature. The environmentally corrected CUF is expressed as CUFen.

Dr. Hopenfeld reviewed various Entergy documents that described the "refined" CUFen analysis conducted by Westinghouse for Entergy.[4] Dr. Hopenfeld found there is a wide margin

[4] Exh. RIV000035, 4-21. Westinghouse, "EnvFat User's Manual Version 1," May 2009, at § 2 (IPECPROP00056785), Exh. RIV000036. Environmental Fatigue Evaluation for Indian

of error in the calculations, due to Entergy's failure to adequately address many critical underlying assumptions that would be a part of a proper fatigue analysis. Entergy's new calculations have likely grossly under-predicted the CUFen values for the components evaluated for the following reasons.

i. The calculations fail to properly adjust laboratory data to account for the actual reactor environment in the calculation of the Fen factors to apply.

Due to significant differences that exist between the laboratory and reactor environment, there are numerous uncertainties in applying the Fen equations to actual reactor components. In NUREG/CR-6909, Effect of LWR Coolant Environment on Fatigue Life of Reactor Materials , Argonne National Laboratories ("ANL")[5] identifies numerous such uncertainties, which include such things as material composition, component size and geometry, loading history, strain rate, mean stress, water chemistry, dissolved oxygen levels, temperature, and flow rate.[6] Such uncertainties can have a significant effect upon fatigue life and ignoring them will result in underestimated CUFen calculations. For example, variations of temperature when temperature is below 150°C can reduce fatigue life by a factor of two, and increased water conductivity due to the presence of trace anionic impurities in the coolant, which has already been documented to cause stress corrosion cracking at several nuclear plants, may decrease fatigue life of austenitic stainless steels. In addition, surface temperature fluctuations and non-uniform temperature

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Point Unit 2, WCAP-17199-P, Revision 0 (Westinghouse, June 2010), IPECPROP00056486, Exh. NYS000361, Environmental Fatigue Evaluation for Indian Point Unit 3, WCAP-17200-P, Revision 0 (Westinghouse, June 2010), IPECPROP00056577, Exh. NYS000362.

[5] Exh. NYS000357

[6] *See* NUREG/CR-6909 at 72.

distributions during stratification can increase the potential for crack initiation and growth, thereby reducing fatigue life. So, to appropriately apply the Fen equations to actual reactor components, the user must consider all of the relevant uncertainties, and the results must be adjusted to account for the varying parameters. In NUREG/CR-6909, ANL further specifies that appropriate bounding Fen values of 12 for stainless steel and 17 for carbon and low alloy steel to account for the numerous uncertainties in using the Fen equations.[7] These bounding Fen factors are not necessarily conservative, and it is reasonable to expect even higher Fen values in the actual reactor environment, especially for those components that experience stratified flows and thermal striping.

Dr. Hopenfeld's testimony describes how Entergy's "refined" EAF analyses have not adequately evaluated the numerous uncertainties associated with determining acceptable Fen values, or, in the alterative, applied bounding Fen values to conservatively ensure that such uncertainties are accounted for. Entergy's calculation of fatigue life for selected components would be significantly affected if all relevant uncertainties were actually considered. Entergy's failure to properly account for the numerous uncertainties inherent in determining the appropriate Fen value from using equations derived from laboratory tests has resulted in calculations that underestimate the CUFen for the analyzed components. The use of the bounding recommended by ANL, which represent far more realistic values than most of those calculated and used by Entergy, would increase the CUFen values beyond unity for a number of the components

[7] *See* NUREG/CR-6909 at iii, 3.

analyzed, as Dr. Hopenfeld calculated in his report.[8]

ii. The calculations use incorrect values for dissolved oxygen ("DO") levels in the calculation of the Fen factors to apply.

One of the largest uncertainties in determining appropriate Fen values is the concentration of dissolved oxygen ("DO") in the water at the surface of each component during the transient. The Fen varies exponentially with the DO level. For example, an increase in oxygen concentration by a factor of four, in comparison to steady state values, would increase the Fen by a factor of 55. The value of Fen is, therefore, sensitive to the uncertainties in DO concentrations. For example, the equations for determining Fen were experimentally derived under conditions where the temperature and DO at the surface of the specimen were known. In contrast, in a reactor plant, the DO in many cases is unknown. The difficulty of determining DO levels during transients is well described by EPRI in its guidelines for addressing fatigue in a LRA.[9] This is particularly true during startup and shutdown transients. Data of the Electrical Power Research Institute (EPRI) on actual oxygen concentrations in a Boiling Water Reactor ("BWR") during start up and shutdowns shows that oxygen concentrations vary with the change in temperature by more than an order of magnitude in comparison to oxygen levels during normal operating conditions.[10] Similar oxygen dependence on temperature can be expected in Pressurized Water Reactors (PWRs).

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[8] Exh. RIV000035 at 8.

[9] EPRI Materials Reliability Program: Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application, MRP-47 at 4-19. *See also id.* at 4-27. Exh. NYS000350.

[10] *See* John J. Taylor, *R&D Status Report*, Nuclear Power Division, EPRI Journal

The developers of the Fen equations (ANL) specifically instruct how users should account for transients where temperature varies, as described in the following NRC reports; NUREG/CR-6583 states "the values of temperature and DO may be conservatively taken as the maximum values for the transient."[11] NUREG/CR-6909 is even more specific and quantifies the appropriate level of DO a user should consider. "The DO value is obtained from each transient constituting the stress cycle. For carbon and low alloy steels, the dissolved oxygen content, DO, associated with a stress cycle is the highest oxygen level in the transient, and for austenitic stainless steels, it is the lowest oxygen level in the transient. A value of 0.4 ppm for carbon and low-alloy steels and 0.05 ppm for austenitic stainless steels can be used for the DO content to perform a conservative evaluation."[12]

Entergy's approach for estimating DO values during transients in calculating Fen is in direct contravention of the specifications provided by ANL and endorsed by NRC, as well as basic laws of physics, and was, therefore, inappropriate and misguided. Entergy has not provided an adequate rational for failing to abide by well-founded recommendations.

iii. The calculations do not accurately consider heat transfer coefficients in the calculation of CUFen values.

Heat transfer is a major factor in the determination of CUFen because it controls the cyclic thermal stresses during transients. Thermal stresses arise when there is a change in the local fluid temperature, such as during heat-ups or cool-downs or due to local mixing of hot and

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(Jan/Feb 1983), Exh. RIV000040.

[11] NUREG/CR-6583 at 78. Exh. NYS000356

[12] NUREG/CR-6909 at A-5. Exh. NYS000357

cold fluids. Failures result from either low stress at high cycle or high stress at low cycle. Most of the thermal stresses experienced at Indian Point are of the latter kind. Damage from such stresses is a serious concern. For example, such stress has caused through-the-wall-cracks in pipes at nuclear reactors.[13] As of 2007, at least thirteen cases of leakage from thermal fatigue have occurred at nuclear reactors.[14] Based on this experience, the International Atomic Energy Agency ("IAEA") has concluded that the frequency of leakage from thermal fatigue will increase with time.[15]

In order to calculate thermal stress and its impact on fatigue life, the temperature distribution of a component during a transient must be determined, or, in other words, the rate at which heat is transferred to the reactor component surface during the transient. Thermal-

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[13] NRC Bulletin No. 88-08: Thermal Stresses in Piping Connected to Reactor Coolant Systems (June 22, 1988) (discussing "circumferential crack extending through the wall of a short, unisolable section of emergency core cooling system (ECCS) piping that is connected to the cold leg of loop B in the RCS" at Farley 2), Exh. RIV000044; NRC Bulletin No. 88-08, Supplement 1: Thermal Stresses in Piping Connected to Reactor Coolant Systems (June 24, 1988) (discussing "crack extending through the wall of" a "section of emergency core cooling system (ECCS) piping that is connected to the hot leg of loop 1 of the RCS" at Tihange 1 in Belgium), Exh. RIV000045; NRC Bulletin No. 88-08, Supplement 2: Thermal Stresses in Piping Connected to Reactor Coolant Systems (August 4, 1988) (discussing the crack incidents at Farley 2 and Tihange 1), Exh. RIV000046; NRC Information Notice 97-46: Unisolable Crack in High-Pressure Injection Piping (July 9, 1997) (discussing "through-wall crack in the weld connecting the MU/HPI pipe and the safe-end of the 2A1 reactor coolant loop (RCL) nozzle" that was "caused by high-cycle fatigue due to a combination of thermal cycling and flow induced vibration" at Oconee Unit 2), Exh. RIV000047.

[14] See Institute for Energy, Development of a European Procedure for Assessment of High Cycle Thermal Fatigue in Light Water Reactors: Final Report of the NESC-Thermal Fatigue Project, EUR22763, 2007, http://ie.jrc.ec.europa.eu/publications/scientific_publications/2007/EUR22763EN.pdf, at 53 (hereinafter "Assessment of High Cycle Thermal Fatigue, EUR22763"), Exh. RIV000048.

[15] *Id.* at 52.

hydraulic computer codes together with plant data are used to calculate such temperature distributions. Heat transfer coefficients (h), water temperature, cycling period, and interface motion are all important inputs to this heat transfer analysis, and the consequent determination of the CUFen values. The CUFen value will vary greatly depending on the inputs used to perform the heat transfer analysis. The heat transfer coefficient h is the most important parameter in this regard.[16] h is an experimental parameter, and has been measured under a range of conditions. It is known for well -defined, controlled conditions. However, the local flow at the surface of many reactor components during transients is not well defined and, therefore, approximations and assumptions are required in calculating the proper h for a given set of conditions. Such approximations lead to uncertainties in the CUFen because uncertainties in h directly impact the errors in the calculated stress. Typical variations in h could increase stress by a factor of 2. Increase in turbulence due to local discontinuities, and increase in the rate of the local temperature change increases h.[17] Increase in h increases the corresponding stress and reduces fatigue life. For example, h along nozzles and bends varies in intensity because of the large variation in turbulence along their surface. This leads to non-uniform heat loads and introduces larger uncertainty in the stress distribution in comparison to simpler flow configurations. Another factor that would lead to non-uniform stress distributions is preferential wall wear due to flow accelerate corrosion ("FAC") in low alloy steel components: Entergy ultrasonic

[16] Exh. RIV000035 at 14

[17] Notably, fatigue life reduction due to large temperature differences and temperature fluctuations was not considered during the initial design of PWRs. It was only after many reactors experienced severe cracking in the late 1980s due to stratification that the PWRs became

examination reports show that wall thickness may vary significantly circumferentially in bends and welds.[18] One diagram indicates that in components where flow is not fully developed, component wall thickness can vary by more than 400% at Indian Point.[19]

Dr. Hopenfeld's analysis of Entergy's heat transfer calculations concludes that Entergy employed unrealistically low heat transfer coefficients in the determination of CUFen.[20] A more realistic selection for this key parameter indicates that the CUFen is significantly larger than predicted by Entergy. More realistic heat transfer calculations alone could have increased the CUFen value by as much as a factor of two for several components. Only modest modifications in this value to account for the inherent uncertainties would cause corresponding CUFen for IP 2 to exceed unity.

iv. The calculations use an unjustified number of transients in the calculation of CUFen values.

The number of transients used in Entergy's "refined" calculations directly affects the resulting CUFen value. However, the actual number of transients during the proposed extended operation period is unknown. This casts further doubt on the CUFen methodology used by Entergy and, in turn, the accuracy of the new calculations.

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a concern. *See* NRC Bulletin No. 88-08, *supra* Note 16.

[18] See Entergy, Indian Point Unit 3 Flow Accelerated Corrosion, 3RF13 Outage, 2005 (Ultrasonic Examination Report of Main Steam/FAC-05-TD-03 (January 6, 2005); Ultrasonic Examination Report of HD/FAC-05-VCD-08(01-02) (March 23, 2005)), Exh. RIV000049.

[19] *See* Entergy, Indian Point Unit 3 Flow Accelerated Corrosion, 3RF13 Outage, 2005 (Ultrasonic Examination Report of Main Steam/FAC-05-TD-03 (January 6, 2005) (schematic indicating that in one section of piping, wall thickness varied from 0.059" to 0.257", which represents a difference of a factor of 4, or 400%), Exh. RIV000049.

[20] See Exh. RIV000035 at 17-18

To evaluate the remaining fatigue life of a given component, it is necessary to consider the past as well as future loading during all transients. It is apparent that Entergy has not adequately considered either past or future transients at Indian Point.

To assess the severity of past transients, each transient that has occurred must be described to determine its contribution to the CUFen. Historical records in such cases are incomplete and insufficient to provide adequate inputs for the number of heat-up and cool-down transients, stratification frequency, and system ΔT. These parameters are required for thermal hydraulic and stress calculations.

Data regarding events at the Indian Point reactors in the past is incomplete and unreliable. It would appear that Entergy has relied upon some undisclosed model in using other plant data in developing the number of transients for the new calculations. For example, stratification is a plant-specific phenomenon that generally requires 3D modeling; Entergy has failed to show how the number of transients concerning stratification was developed for the pressurized surge line by virtue of using other plant data. Entergy's methodology for calculating the likely future number of transients is also suspect and Entergy fails to provide a credible justification for the extensive engineering judgment upon which it relies for determining future transients.

Justification of an appropriate number of future transients is critical in light of the fact that Indian Point will be entering an extended period of operation. It is commonly accepted that the useful life of most engineering components and structures follow a "bathtub curve" trend.[21]

[21] WordIQ.com, Bathtub curve - Definition, http://www.wordiq.com/definition/Bathtub_curve (last visited Dec. 21, 2011).

A "bathtub curve" is defined as "the phenomenon that the fraction of products failing in a given timespan is usually high early in the lifecycle, low in the middle, and rising strongly towards the end. When plotted as a curve, this looks like the profile of a bathtub." This phenomenon dictates that at the beginning and at the end of life, component failure occurs at a very high frequency. In between these two extremes, the failure is relatively low and constant. For this reason Entergy must justify its use of the straight-line extrapolation for the number of transients it assumed in calculating the CUFen values. Even if Entergy's approach was justifiable, there is doubt they have used the approach correctly in the calculations relied upon.

2. Entergy failed to expand the scope of components to be assessed for metal fatigue as required by NRC regulations and industry guidance.

In order for Entergy to have an effective AMP to monitor for metal fatigue, it must expand the scope of the fatigue analysis beyond simply representative components, to identify other components whose CUFen may be greater than 1.0. This is necessary at Indian Point for several reasons. First, Entergy's initial CUFen findings presented in Tables 4.3-13 and 4.3-14 of the April 2007 LRA indicated that the CUFen values for various components exceeded the regulatory threshold of 1.0. Under these circumstances, regulatory and industry guidance requires Entergy to identify additional reactor locations for potential high susceptibility to metal fatigue in order to ensure that appropriate aging management measures are taken in a timely fashion. In particular, EPRI's MRP-47 indicates that when "plant-unique evaluations . . . show that some of the NUREG/CR-6260 locations do not remain within allowable limits for 60 years of plant operation when environmental effects are considered . . . plant specific evaluations

should *expand* the sampling of locations accordingly to include other locations where high usage factors might be a concern."[22] NUREG-1801, GALL, Revision 1 also contemplates such an expanded review when CUFen values have been found to exceed unity, stating that "[f]or programs that monitor a sample of high fatigue usage locations, corrective actions include a review of *additional* affected reactor coolant pressure boundary locations," and that sample locations identified in NUREG/CR-6260 are simply the "minimum" set of components to analyze.[23]

Entergy's "refined" calculations, which purport to show that for all components evaluated, the CUFen is now less than 1.0, does not alleviate Entergy of its obligation to expand the number of components reviewed. To begin with, the fact that Entergy's initial findings showed CUFen values over 1.0, by itself, triggered the responsibility to perform an expanded review, notwithstanding whether any additional calculations later show all CUFen values within regulatory acceptance criteria. Moreover, as discussed at length above, the accuracy of the "refined" evaluation is highly suspect, and, in actuality, the CUFen values for the evaluated components indeed may exceed unity during the proposed period of extended operation. Thus, an expanded review must still be conducted.[24]

In addition, the most recent version of the GALL Report, Revision 2, specifies that the

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[22] MRP-47 3-4 (2005)(emphasis added)(NYS000350).

[23] NUREG-1801, Rev. 1, § X.M1, 5, 7, pp. X M-1 to X M-2 (emphasis added) (NYS000146A-C).

[24] *See* ASLB Memorandum and Order (Ruling on Motion for Summary Disposition of NYS-26/26A/Riverkeeper TC-1/1A (Metal Fatigue of Reactor Components) and Motion for Leave to File New Contention NYS-26B/Riverkeeper TC-1B) (November 4, 2010), at 20.

sample set for fatigue calculations that consider the effects of the reactor water environment "should include the locations identified in NUREG/CR-6260 and *additional plant-specific component locations* in the reactor coolant pressure boundary if they may be *more limiting* than those considered in NUREG/CR-6260."[25] Entergy has, to date, not provided any analysis that would support a conclusion that that the CUFen values in LRA Tables 4.3-13 and 4.3-14 (which represent the same locations as those identified in NUREG/CR-6260) bound all other components at the plant. To the contrary, Entergy's initial findings indicated that the CUFen value for several components would exceed unity, demonstrating that such components were not necessarily the most limiting.[26] Furthermore, Dr. Hopenfeld's analysis of Entergy's "refined" calculations clearly shows that the revised CUFen values for the components evaluated are likely under-predicted and that, if properly evaluated, many could exceed unity.

In fact, NRC Staff has now conceded that there may be more limiting components, and that the CUFen values in LRA Tables 4.3-13 and 4.3-14 may not be bounding. Specifically, on February 10, 2011, NRC Staff issued a Request for Additional Information ("RAI") to Entergy, asking that Entergy "[c]onfirm and justify that the plant-specific locations listed in LRA Tables 4.3-13 and 4.3-14 are bounding for the generic NUREG/CR-6260 components" and "[c]onfirm and justify that the locations selected for environmentally-assisted fatigue analyses in LRA Tables 4.3-13 and 4.3-14 consist of the most limiting locations for the plant (beyond the generic

[25] NUREG-1801, Rev. 2, § X.M1, p. X M1-2 (emphasis added)(NYS000147A-D).
[26] Exh. RIV000035 at 23.

components identified in the NUREG/CR-6260 guidance)."[27]

In response, Entergy only indicated that the locations in LRA Tables 4.3-13 and 4.3-14 are the same as those locations provided in NUREG/CR-6260, and, in Commitment 43, promised to determine at some future point in time, though before entering the period of extended operation, whether the NUREG/CR-6260 locations evaluated are the limiting locations for the Indian Point plant configurations. Entergy's apparent reason for requiring additional time was the need to review "design basis ASME Code Class 1 fatigue evaluations."[28] In August 2011, NRC Staff issued Supplement 1 to the Safety Evaluation Report pertaining to the Indian Point license renewal proceeding, memorializing this exchange, and accepting Entergy's responses and new commitment.[29]

However, NRC Staff's acceptance of a vague commitment to perform necessary metal fatigue investigation and analyses in the future, and Entergy's failure to actually "confirm and justify" bounding and limiting locations for Indian Point renders Entergy's AMP insufficient to comply with 10 C.F.R. § 54.21(c), or the regulatory guidance ofNUREG-1801, Generic Aging Lessons Learned (GALL) Report, as it does not demonstrate that metal fatigue will be

[27] Request for Additional Information for the Review of the Indian Point Nuclear Generating Unit Numbers 2 and 3, License Renewal Application (Feb. 10, 2011), at 13, ADAMS Accession No. ML110190809, Exh. RIV000057.

[28] NL-11-32, Response to Request for Additional Information (RAI), Aging Management Programs, Indian Point Nuclear Generating Unit Nos. 2 & 3, Docket Nos. 50-247 and 50-286, License Nos. DPR-26 and DPR-64 (March 28, 2011), at 26, ADAMS Accession No. ML110960360, Exh. RIV000058.

[29] Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3, Supplement 1 (August 2011), available at, http://pbadupws.nrc.gov/docs/ML1124/ML11242A215.pdf.

appropriately monitored, managed and corrected during the period of extended operation. No party now disputes that Entergy's LRA Tables 4.3-13 and 4.3-14, which lists the same components as those identified in NUREG/CR-6260, may not represent limiting locations for the entire plant. As such, according to the guidance of the GALL Report, in order to have an adequate aging management program, Entergy must identify the locations that may be more limiting, and which will be the subject of CUFen calculations, now, and not just articulate a plan to determine such locations later. See Contention NYS-38/RK-TC-5.

Instead of conducting the analysis that would be required to confirm and justify that the components in LRA Tables 4.3-13 and 4.3-14 are bounding, Entergy only agreed to perform such an assessment in the future. Such an analysis would require an assessment of experience at Indian Point as well as other pressurized water reactor (PWR) plants; and identification and ranking of all components that are susceptible to thermal fatigue (including but not limited to nozzles, reducers, mixing tees and bends in feed water lines, surge lines, spray lines, and volume control system lines), in terms of numerous key parameters that are known to effect fatigue life (including the ratios of the local heat transfer coefficient and the local material conductivity, wall thickness, fluid temperature, ΔT, dissolved oxygen levels, flow velocities, number of transients, magnitude and cycling frequency of surface temperatures and loads, and surface discontinuities, and flow discontinuities in each component). Moreover, thermal striping during stratification should be generally considered as these effect fatigue life.[30] Entergy has demonstrably failed to

[30] NUREG-1801 requires that environmental effects be included in the calculations and

provide any information about how the analysis to determine the most limiting locations as Indian Point will be performed, such that Intervenors could meaningfully comment upon the adequacy of the analysis.

An analysis to determine the most limiting locations for which Entergy will have to perform CUFen evaluations must be performed before a determination is made about license renewal. Accordingly, NRC Staff's acceptance of Entergy's commitment in SER Supplement 1 to determine whether the NUREG/CR-6260 locations evaluated are the limiting locations for the Indian Point plant configurations at some time in the future is not warranted or acceptable.
In light of Entergy's failure to expand the scope of review and identify further components to be subject to CUFen analyses, Entergy has not demonstrated that it has a program to monitor, manage, and correct metal fatigue related degradation sufficient to comply with 10 C.F.R. § 54.21(c), or the regulatory guidance of NUREG-1801, Generic Aging Lessons Learned (GALL) Report.

3. Entergy has otherwise not provided sufficient details concerning an Aging Management Program ("AMP") to ensure that the degradation effects of metal fatigue would be adequately handled during the proposed period of extended operation.

The lack of a reliable, transparent, complete assessment of CUFen values for susceptible plant components at Indian Point fails to comply with the "Scope of Program" articulated in the GALL Report, which specifies that a program for managing metal fatigue must include adequate "*preventative measures* to mitigate fatigue cracking of metal components of the reactor coolant

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it does not exclude thermal striping from such requirements (NYS000146A-C).

pressure boundary caused by anticipated cyclic strains in the material."[31]

However, Entergy's plans for correcting metal fatigue related degradation depend initially upon calculating the vulnerability of plant components. Indeed, Entergy intends to rely upon future CUFen calculations throughout the period of extended operation to manage metal fatigue. Entergy's calculations are meant to signify when components require inspection, monitoring, repair, or replacement, and, according to Entergy, will determine when such actions are taken. Accordingly, the validity of Entergy's monitoring program depends upon the accuracy of the calculations of the CUFen. When a fatigue monitoring program is entirely based on a predictive analysis and not on actual measurements, and the analysis is flawed, the monitoring program is invalid. Thus, Entergy's flawed methodology for calculating CUFen, as discussed above, which Entergy ostensibly intends to employ throughout the period of extended operation, as well as Entergy's failure to expand the scope of components to be assessed, renders Entergy's vague commitments to inspect, repair, and replace affected locations insufficient to ensure proper management of metal fatigue during the proposed period of extended operation.

In light of the absence of comprehensive, accurate metal fatigue calculations to properly guide Entergy's aging management efforts, Entergy has failed to define specific criteria to assure that susceptible components are inspected, monitored, repaired, or replaced in a timely manner. Once components with high CUFen values have been properly identified, Entergy must describe

[31] NUREG-1801, Rev. 1 § X.M1, 1, p. X M-1 (emphasis added) (NYS000146A-C); see also NUREG-1801, Rev. 2, § X.M1, 1 p. X M1-2 (NYS000147A-D)("The program ensures the fatigue usage remaining within the allowable limit, thus minimizing fatigue cracking of metal components caused by anticipated cyclic strains in the material.").

a fatigue management plan for each such component that should, at a minimum, rank components with respect to their consequences of failure, establish criteria for repair versus defect monitoring, and establish criteria for the frequency of the inspection (considering, for example defect size changes and uncertainties in the stress analysis and instrumentation), and allow for independent and impartial reviews of scope and frequency of inspection. Entergy has failed to do this.

PROPOSED FINDINGS

Based on the evidence provided in support of NYS-26B/RK-TC-1B the Board is able to conclude that:

1. Entergy's AMP for metal fatigue must be judged by whether Entergy meets the requirements of 10 C.F.R. § 54.21(c)(1)(iii).

2. In order to meet that regulatory requirement Entergy must demonstrate that the AMP upon which it relies will assure that no component within the reactor coolant pressure boundary will "exceed the design limit on fatigue usage". GALL, Rev. 1, Vol. 2 at X M.1

3. Entergy has no AMP for metal fatigue of reactor pressure vessel internals.

4. Entergy has developed its AMP for metal fatigue without considering the synergistic impact on components within the reactor coolant pressure boundary of metal fatigue, embrittlement and stress corrosion cracking.

5. Entergy has done one CUFen calculation, reported in its initial LRA, that identify a number of components with CUFen values <1.

6. Entergy has not expanded the scope of the components for which it will do CUFen

calculations to include “the most limiting locations” nor has it provided an analysis to justify a conclusion that the locations already selected are the most limiting.

7. Entergy’s AMP does not include specific criteria for determining when corrective action is to be taken and what corrective action will be taken.

7. Entergy relies on the WESTEMS program to determine whether any component has a CUFen <1.

8. WESTEMS does not include parametric runs and a detailed error analysis for the calculation results it produces.

9. WESTEMS uses numerous simplified analyses, particularly the thermal-hydraulic model, that create substantial uncertainties in the results achieved.

10. WESTEMS allows for substantial user interventions to modify input parameters and thus is susceptible to considerable uncertainty in the results reached depending upon the input parameters used.

11. Entergy does not provide any criteria for when and how user modifications to WESTEMS will be implemented.

12. Entergy’s analysis of the stresses to which metal-fatigued components may be subjected does not consider the stresses associated with DBA LOCA and other similar accidents.

13. Entergy’s development of its AMP for metal fatigue, including determining acceptance criteria, preventive action and corrective action criteria, does not consider the synergistic impact of multiple degradation mechanisms including metal fatigue, embrittlement and stress corrosion cracking.

14. Fen formulae have been determined based on laboratory experiments which fail to adequately represent in reactor environments.

15. Entergy has failed to justify the values it uses for critical input values that have a substantial impact on the final CUFen calculation including temperature variations, dissolved oxygen, heat transfer coefficients, number of transients in the past and projected for the future, all of which create a wide uncertainty range for the CUFen calculations and which uncertainties are not addressed by Entergy.

16. Entergy has failed to demonstrate that it has an AMP that will assure that the component within the reactor coolant pressure boundary will not "exceed the design limit on fatigue usage". GALL, Rev. 1, Vol. 2 at X M.1.

ARGUMENT

The legal issues raised by this contention have essentially been brief and resolved by the Board's ruling in rejecting Entergy's Motion for Summary Disposition and admitting NYS-26B/RK-TC-1B. Summary Disposition Order.

Arguments advanced by Entergy and Staff that Entergy need not provide greater detail than a bare assertion that it will comply with GALL, were rejected by the Board, placing principal reliance on the Commission's *Vermont Yankee* decision. Summary Disposition Order at 16 ("Only by describing the parameters of the CUF calculations, demonstrating their methodology, and addressing all of the elements in the Gall Report would an applicant satisfy the goals of the GALL Report, which are to prevent the applicant from exceeding the design limit for fatigue with an AMP that 'monitors and tracks the number of critical thermal and pressure

transients for the selected reactor coolant system components' using analyses that include 'the effects of the coolant environment on component fatigue life.'" (Citation omitted)).

Similarly, when Entergy and NRC Staff argued, relying on the Commission's *Vermont Yankee* decision, it was improper to question the adequacy of calculations done to demonstrate the existence of an effective AMP, the Board held "once an applicant has chosen to include EAFs in its CUF calculations, there is nothing in NRC regulations or in the Commission's recent decision in Vermont Yankee that prohibits an intervenor from questioning the adequacy, reliability, and breadth of these calculations when applied to Entergy's AMP under Section 54.21(c)(1)(iii), as New York and Riverkeeper have done." Summary Disposition Order at 22-3.

Finally, the Board rejected the assertion that a challenge to the CUFen calculation in this case is prohibited by the Commission ruling in *Vermont Yankee*:

> NYS-26B/RK-TC-1B further differs from the contention that was held inadmissible in Vermont Yankee by challenging whether Entergy's AMP is adequate enough to meet Subsection (iii), and whether it meets the recommendations of the GALL Report. As the Commission noted:
>
> > An applicant may commit to implement an AMP that is consistent with the GALL Report and that will adequately manage aging. But such a commitment does not absolve the applicant from demonstrating, prior to issuance of a renewed license, that its AMP is indeed consistent with the GALL Report. We do not simply take the applicant at its word.
>
> New York and Riverkeeper have provided just such a challenge to the adequacy of Entergy's AMP and specifically to its FMP that has been designated by Entergy to serve as its AMP. Because Entergy calculated CUF analyses as part of its efforts to meet Subsection (iii), the methodology and breadth of these calculations may come under scrutiny.

Summary Disposition Order at 24 (citation and footnote omitted).

Thus, the principal remaining issues for resolution are not legal questions, but factual questions. New York and Riverkeeper have presented a substantial body of technical expertise and documents to demonstrate that the AMP offered by Entergy to demonstrate compliance with 10 C.F.R. § 54.21(c)(iii) fails to achieve the goals set forth in GALL to which Entergy has committed compliance. The documents on which New York and Riverkeeper rely and the analyses and opinions of their experts have been well-known to Entergy and NRC Staff for at least a year. Nothing in the documents produced in this case by Entergy or NRC Staff provides substantial support for contrary positions by Entergy and NRC Staff. Pursuant to 10 C.F.R. § 2.336(e)(2) any attempt by Entergy or NRC Staff to now rely on pre-existing documents not produced as required by 10 C.F.R. § 2.336(a) and (b) or to offer analyses not previously included in documents developed and produced in this case is sanctionable by exclusion of the document or the testimony or both. Thus, Entergy will not be able to defend its incomplete and imprecise metal fatigue calculations or its AMP which essentially consists of those calculations. New York and Riverkeeper urge the Board to reject Entergy's license renewal application for its failure to demonstrate that its AMP for metal fatigue meets the requirements of

CONCLUSION

For all the reasons stated, Entergy's license renewal application should be denied.

Respectfully submitted,

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