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**UNITED STATES**
**NUCLEAR REGULATORY COMMISSION**

**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,<br>Entergy Nuclear Indian Point 3, LLC, and<br>Entergy Nuclear Operations, Inc. | DPR-26, DPR-64<br><br>December 21, 2011 |

# SUPPLEMENTAL REPORT OF DR. RICHARD T. LAHEY, JR. IN SUPPORT OF CONTENTIONS NYS-25 AND NYS-26B/RK-TC-1B

**Prepared for the State of New York**
**Office of the Attorney General**

**SUPPLEMENTAL REPORT OF DR. RICHARD T. LAHEY, JR.**

**Concerning the Use of the WESTEMS Computer Code and the Analysis of Cumulative Metal Fatigue of Critical Reactor Components in the Indian Point License Renewal Proceeding**

The purpose of this Supplemental Report is to document my technical concerns associated with the use of the WESTEMS computer code to obtain revised fatigue analyses for important components in the two operating nuclear reactors at the Indian Point site in Buchanan, New York (*i.e.*, Indian Point Unit 2 & Indian Point Unit 3 ). It should be noted that these fatigue analyses did not consider RPV internals.

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1. As discussed in my accompanying Report, in order to perform more mechanistic, but less conservative, fatigue evaluations, Entergy contracted with Westinghouse (W) to redo the ex-vessel fatigue analyses for some important reactor components at Indian Point Unit 2 and Indian Point Unit 3 (IP-2 and IP-3). These new results were reported separately in 2010 [Entergy NL-10-082 *discussing* WCAP-17199-P / WCAP-17200-P, "Environmental Fatigue Evaluation for Indian Point Unit 2/3" (June 2010)]. Rather than performing standard ASME code evaluations, as Entergy had done when it initially filed its application in 2007, these new calculations were done using WESTEMS, a proprietary computer code of

W. Following Entergy's submission of these refined calculations obtained via WESTEMS to the ASLB and some other interested parties, the ASLB directed Entergy to release the WESTEMS code manuals to the State of New York for review. The parts of the manuals transmitted to us contained assumptions and models (particularly for the thermal-hydraulics models) used by W in WESTEMS.

2. A critical measure of metal fatigue is the environmentally-adjusted cumulative usage factor (expressed as $CUF_{en}$) which must be < 1.0 during the period of extended plant operation.

3. In a number of instances, the 2010 refined WESTEMS results are extremely close to the critical CUFen criterion of 1.0 (*i.e.*, for the IP-2 RHR line, CUFen = 0.9434, and for the IP-3 RHR line, CUFen = 0.9961).

4. Unfortunately, an error analysis of the WESTEMS results was not made available to the State of 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.

5. What is clear, however, is that there are many possible sources of error in these WESTEMS results. For example**:**

(i) Stratification-induced thermal transients were developed for the IP-2 & 3 pressurizers based on plant-specific thermocouple (TC), or thermal sensor, data [WCAP-17149-P, Rev. 1, "Evaluation of

Pressurizer Insurge/Outsurge Transients for Indian Point Unit 2," IPECPROP00056663 (July 2010); WCAP-17162-P, Rev. 1, "Evaluation of Pressurizer Insurge/Outsurge Transients for Indian Point Unit 3," IPECPROP00056717 (July 2010)]. However, there was no discussion of how the so-called 'inverse problem' was done to account for the thermal inertia of the structures so that one could properly relate the transient fluid temperature to the transient thermal sensor data. As it turns out, this type of evaluation is not trivial and if done incorrectly it can lead to significant errors in the transient inner wall temperature and thus the transient thermal stresses and fatigue analysis. The same is true when one tries to account for the effect of thermal sleeves by simply modifying a heat transfer coefficient, as W has done. This rather crude analytical approach is not needed (*i.e.*, a more accurate 3-D thermal analysis can be done using various commercially available computational fluid dynamic (CFD) codes, such as *Fluent*) and it leads to errors in the transient thermal stresses and thus the resultant fatigue analysis.

(ii) A rather simple thermal-hydraulics model was also developed by W and is used in WESTEMS to calculate the thermal transients in conduits having significant flow area changes (e.g., near piping junctions and nozzles). In particular, this model was a 1-D nodal model. Such models are insufficient for the accurate evaluation of the transient heat transfer phenomena of interest. Indeed, detailed 3-D computational fluid dynamic (CFD) models are needed to accurately evaluate the transient developing boundary layers (both thermal and momentum) in non-uniform conduits where so-called entrance effects may occur. When doing 1-D evaluations the resulting temperature transients and induced stresses at the conduit walls will be incorrect and there will necessarily be errors in the computed $CUF_{en.}$. In addition, these errors typically lead to an under-prediction of the local transient fluid-to-wall heat transfer [*e.g.*, Cengel & Turner, "Fundamentals of Thermal-Fluid Sciences," McGraw-Hill, pp 759-760 (2001)], thus yielding non-conservative $CUF_{en}$ results (*i.e.*, the predicted $CUF_{en}$ will be too small).

(iii) There was obviously a lot of engineering judgment used in obtaining some of the fatigue analysis results presented by W. For example, quasi-steady-state results (*e.g.*, the stress-inducing moments created in the piping and fittings due to thermal

stratification in the pressurizer) were apparently combined with transient results from WESTEMS to yield the net thermally-induced transient stresses and thus the corresponding fatigue results (*i.e.*, $CUF_{en}$). Unfortunately, the uncertainty in these evaluations has not been quantified. Indeed, as noted previously, an error analysis was not part of the W reports.

(iv) There were no evaluations of the potential failure of highly fatigued structures and fittings external to the RPV due to a secondary side LOCA or any other thermal shock loads. Unfortunately, these transient loads can be much larger than what W has included in their fatigue failure evaluations for IP-2 & 3 and they can lead to the early failure of fatigue-weakened components. If so, this can lead to a primary side LOCA which, in turn, will challenge core cooling.

(v) In regions of the piping system where the flow area is constant and entrance effects are not significant, it is normal engineering practice to use single-phase heat transfer coefficients and 1-D thermal-hydraulic models, and apparently this is what W has done. However, it is well known [*e.g.*, Kreith, "Principles of Heat Transfer," Int. Text Book Co., pg. 353 (1961)] that the inherent uncertainty in single-phase heat transfer coefficients (in particular those associated with the Dittus-Boelter correlation, which is an option in WESTEMS) is at least +/- 25%, and the uncertainities can be even worse when constant heat transfer coefficients are assumed. Anyway, this uncertainty must be taken into account in an overall error analysis (which, unfortunately, W did not provide in their reports). Moreover, based on 'engineering judgment,' constant single-phase heat transfer coefficients were typically assumed and used by W (apparently because the use of more realistic variable coefficents leads to numerical instabiltites in WESTEMS [Yang, LTR-PAFM-03-42, Section 2.5a (2006)]) and their assumed magnitude often varied widely within a particular component and from component to component [*e.g.*, Roarty et al., WCAP-14173, Fig. D.4.1 (1995)]. In addition, the WESTEMS user had the ability to specify the number of spatial nodes, the time step, and "smooth" the results. As a consequence the code user can strongly influence the $CUF_{en}$ results.

6. In order to more fully explain my technical concerns associated with fatigue, let us now consider some of the Indian Point reactor components that were analyzed for Entergy by W. As noted in a SMiRT-19 paper [Cranford & Gary-W, 8/07] 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 W were for this critical component. That is, $CUF_{en}$ = **0.9961** for IP-3 [IPECPROP00057883, Table 5-36], and $CUF_{en}$ = **0.9434** for IP-2 [IPECPROP 00057881, Table 5-32], 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[1] 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).

7. Other reactor components analyzed using WESTEMS also had disturbingly large values for $CUF_{en}$. For example, the hot leg charging water nozzle had a $CUF_{en} > 1.0$ when the current licensing basis (CLB) transients were used for IP-2. Even when assumed (*i.e.*, extrapolated) plant-specific transients were used, and a constant heat transfer coefficient was assumed, the results were still large enough to be of concern (*i.e.*, for IP-3, $CUF_{en}$ = **0.722** [IPECPROP0057432, Table 5-27]), particularly since an error analysis has not been done by W.

8. Similarly, a thermal fatigue analysis of the IP-2 boron injection tank (BIT) nozzle was done using an assumed constant heat transfer coefficient and a hand calculation was apparently done to modify an ASME-approved model implicit

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[1] Even though the assumed single-phase convective heat transfer coefficient for the RHR and accumulator nozzle evaluations was rather large, it appears that WESTEMS does not allow for the onset of nucleate boiling during depressurization transients, in which the resultant boiling heat transfer coefficient can be even larger. Thus the transient thermal stress evaluations may be non-conservative.

in WESTEMS. Nevertheless, the final result was still quite large (*i.e.*, $CUF_{en}$ = **0.8553** [IPECPROP00057559, Table 5-10]).

9. As another example of how a WESTEMS code user can influence the results, the pressurizer surge line nozzles (which comprise a very important part of the primary system's pressure boundary) for IP-2 & 3 were analyzed using assumed constant heat transfer coefficients which were different in some spatial locations [IPECPROP00057917, Table 5-1]. When current licensing basis (CLB) transients were used, the result was $CUF_{en} > 1.0$, and even when the assumed plant-specific thermal transients were modified by the code user, IP-2 still had a very large $CUF_{en}$ (*i.e.*, **0.822**) [IPECPR00057276, Tables 5-18 & 5-19].

10. In my September 8, 2010 declaration, I raised concerns about the WESTEMS code and its reliance on user assumption, judgments, and interventions. The USNRC Staff confirmed in the SSER [NUREG-1930, Supplement 1 at 4-2], WESTEMS permits the code user to make assumptions and interventions that can affect the outcome. In addition, the USNRC Staff required Entergy to disclose and make clear those user interventions that will be used in <u>future</u> WESTEMS analysis of IP-2 and IP-3 [SSER, NUREG-1930, Supplement 1 at 4-2], but, for some reason, this disclosure was not required for the <u>previous</u> WESTEMS results that Entergy has already submitted to the ASLB and the parties for IP-2 and IP-3. Obviously, this information is needed in order to do a proper review of these 2010 results.

11. It is apparent that the WESTEMS code is based on rather simple models (particularly the thermal-hydraulic models) and that the thermal stress results for $CUF_{en}$ are strongly influenced by the code user's assumptions, manipulations and interventions. There is a lot of "engineering judgment" implicit in the $CUF_{en}$ results, and, since an error analysis has not been done to bound the uncertainty, and many results are disturbingly close to the $CUF_{en} = 1.0$ limit, I do not believe that one can trust these results to assure the safety of the 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 has <u>not</u> 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. Finally, it should be noted that the WESTEMS results presented by Entergy did not consider the synergistic effect of stress corrosion cracking, embrittlement and fatigue on RPV internals. As discussed at length in my accompanying Report, this is a serious omission, and one must be rectified in order to protect the health and safety of the American public during extended operations of the two Indian Point reactors.

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Dr. Richard T. Lahey, Jr.