Entergy

Entergy Nuclear Northeast
Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
Buchanan, NY 10511-0249
Tel (914) 788-2055

Fred R. Dacimo
Vice President
License Renewal

May 16, 2008

Re: Indian Point Units 2 & 3
Docket Nos. 50-247 & 50-286

NL-08-084

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

SUBJECT: **Reply to Request for Additional Information Regarding License Renewal Application – Time-Limited Aging Analyses and Boraflex**

Reference: NRC letter dated April 18, 2008; "Request for Additional Information for the Review of the Indian Point Nuclear Generating Unit Nos. 2 and 3, License Renewal Application – Time-Limited Aging Analyses and Boraflex"

Dear Sir or Madam:

Entergy Nuclear Operations, Inc is providing, in Attachment I, the additional information requested in the referenced letter pertaining to NRC review of the License Renewal Application for Indian Point 2 and Indian Point 3. The additional information provided in this transmittal addresses staff questions for Time-Limited Aging Analyses and Boraflex.

There are no new commitments identified in this submittal. If you have any questions or require additional information, please contact Mr. R. Walpole, Manager, Licensing at (914) 734-6710.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 5-16-08.

Sincerely,

Peter W. Conroy for
Fred R. Dacimo
Vice President
License Renewal

A128
NRR

Attachment:

1. Reply to NRC Request for Additional Information Regarding License Renewal Application – Time-Limited Aging Analyses and Boraflex

cc: Mr. Bo M. Pham, NRC Environmental Project Manager
Ms. Kimberly Green, NRC Safety Project Manager
Mr. John P. Boska, NRC NRR Senior Project Manager
Mr. Samuel J. Collins, Regional Administrator, NRC Region I
Mr. Sherwin E. Turk, NRC Office of General Counsel, Special Counsel
IPEC NRC Senior Resident Inspectors Office
Mr. Paul D. Tonko, President, NYSERDA
Mr. Paul Eddy, New York State Dept. of Public Service

ATTACHMENT I TO NL-08-084

REPLY TO NRC REQUEST FOR ADDITIONAL INFORMATION

REGARDING

LICENSE RENEWAL APPLICATION

Time-Limited Aging Analyses and Boraflex

ENTERGY NUCLEAR OPERATIONS, INC
INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 and 3
DOCKETS 50-247 and 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3
LICENSE RENEWAL APPLICATION (LRA)
REQUESTS FOR ADDITIONAL INFORMATION (RAI)
REGARDING TIME-LIMITED AGING ANALYSES AND BORAFLEX

Time-Limited Aging Analyses

RAI 4.3.1.8-1

License renewal application (LRA) Section 4.3.1 states "[c]urrent design basis fatigue evaluations calculate cumulative usage factors (CUFs) for components or sub-components based on design transient cycles." For CUF values listed in LRA Tables 4.3-13 and 4.3-14, please describe the details of how various environmental effects are factored into the calculation of the CUF using F_{en} values.

Response to RAI 4.3.1.8-1

For CUF values listed in LRA Tables 4.3-13 and 4.3-14, the F_{en} values were determined as described below.

NUREG-1801 calls for using formulas provided in NUREG/CR-5704 for austenitic stainless steel and NUREG/CR-6583 for carbon steel and low-alloy steel to calculate environmentally assisted fatigue correction factors (F_{en}). For IPEC, none of the locations identified in Tables 4.3-13 and 4.3-14 (NUREG/CR-6260 locations) are made of carbon steel, so calculation of F_{en} for carbon steel was not required.

The environmentally assisted fatigue correction factor (F_{en}) for **low alloy steel** was calculated as follows.

F_{en} = exp(0.929-0.00124T-0.101 $S^*T^*O^*\varepsilon^*$) based on NUREG/CR-6583, Eq. 6.5b

T =	25°C	Reference temperature for original fatigue curves	
S^* =	S	(0 < S (Sulfur) ≤ 0.015 wt.%)	
S^* =	0.015	(S ≥ 0.015 wt.%)	NUREG/CR-6583, Eq. 5.5a
T^* =	0	(T (Temperature) < 150°C)	
T^* =	T-150	(T = 150-350°C)	NUREG/CR-6583, Eq. 5.5b
O^* =	0	(DO (Dissolved Oxygen) < 0.05ppm)	
O^* =	ln(DO/0.04)	(0.05 ppm ≤ DO ≤ 0.5 ppm)	
O^* =	ln(12.5) = 2.53	(DO > 0.5 ppm)	NUREG/CR-6583, Eq. 5.5c
$\dot{\varepsilon}^*$ =	0	($\dot{\varepsilon}$ (strain rate) > 1%/s)	
$\dot{\varepsilon}^*$ =	ln($\dot{\varepsilon}$)	(0.001 ≤ $\dot{\varepsilon}$ ≤ 1%/s)	
$\dot{\varepsilon}^*$ =	ln(0.001)	($\dot{\varepsilon}$ < 0.001%/s)	NUREG/CR-6583, Eq. 5.5d

There are four low alloy steel subcomponents for each unit in Tables 4.3-13 and 4.3-14 for the NUREG-6260 locations at IPEC. The F_{en} was calculated for each location on each unit as shown below.

IP2

$T_{(\text{reference temperature °C})}$	=	25	Reference temperature for original fatigue curves
$O^*_{(RCS)}$	=	0.0	RCS dissolved oxygen is ≤ 50 ppb

Since $O^*_{(RCS)}$ equals 0.0, S*, T*, and ε^* terms are eliminated.

F_{en} = exp(0.929-(0.00124)(T))

$F_{en\ (\text{bottom head to shell})}$ = exp(0.929-(0.00124)(25)) = 2.45
$F_{en\ (\text{inlet nozzles})}$ = exp(0.929-(0.00124)(25)) = 2.45
$F_{en\ (\text{outlet nozzles})}$ = exp(0.929-(0.00124)(25)) = 2.45
$F_{en\ (\text{surge line nozzles})}$ = exp(0.929-(0.00124)(25)) = 2.45

IP3

$T_{(\text{reference temperature °C})}$ = 25 Reference temperature for original fatigue curves

Since $O^*_{(RCS)}$ equals 0.0, S*, T*, and ε^* terms are eliminated.

F_{en} = exp(0.929-(0.00124)(T))
$F_{en\ (\text{bottom head to shell})}$ = exp(0.929-(0.00124)(25)) = 2.45
$F_{en\ (\text{inlet nozzles})}$ = exp(0.929-(0.00124)(25)) = 2.45
$F_{en\ (\text{outlet nozzles})}$ = exp(0.929-(0.00124)(25)) = 2.45
$F_{en\ (\text{surge line nozzles})}$ = exp(0.929-(0.00124)(25)) = 2.45

The environmentally assisted fatigue correction factor (F_{en}) for **austenitic stainless steel** was calculated as follows.

F_{en} =	exp(0.935-T′O′ε′)		NUREG/CR-5704, Eq. 13
T′ =	0	(T < 200ºC)	
T′ =	1	(T ≥ 200ºC)	NUREG/CR-5704, Eq. 8a
O′ =	0.260	(DO < 0.05 ppm)	
O′ =	0.172	(DO ≥ 0.05 ppm)	NUREG/CR-5704, Eq. 8b
$\dot{\varepsilon}'$ =	0	($\dot{\varepsilon}$ > 0.4%/s)	
$\dot{\varepsilon}'$ =	ln($\dot{\varepsilon}$ /0.4)	(0.0004 ≤ $\dot{\varepsilon}$ ≤ 0.4%/s)	
$\dot{\varepsilon}'$ =	ln(0.0004/0.4)	($\dot{\varepsilon}$ < 0.0004%/s)	NUREG/CR-5704, Eq. 8c

There are four stainless steel subcomponents for each unit in Tables 4.3-13 and 4.3-14 for the NUREG-6260 locations at IPEC. The F_{en} will be calculated for each location on each unit as shown below.

IP2

$T'_{(\text{Surge line})}$	=	1.0	(T ≥ 200ºC)
$T'_{(\text{Charging nozzle})}$	=	1.0	(T ≥ 200ºC)
$T'_{(\text{SI nozzle})}$	=	1.0	(T ≥ 200ºC)
$T'_{(\text{RHR piping})}$	=	1.0	(T ≥ 200ºC)

$O'_{(All)}$	= 0.260	RCS dissolved oxygen is ≤ 50 ppb
$\dot{\varepsilon}'_{(all)}$	= -6.91	Assume bounding strain rate
$F_{en\ (Surge\ line)}$	= exp(0.935-(1.0)(0.260)(-6.91)) = 15.35	
$F_{en\ (Charging\ nozzle)}$	= exp(0.935-(1.0)(0.260)(-6.91)) = 15.35	
$F_{en\ (SI\ nozzle)}$	= exp(0.935-(1.0)(0.260)(-6.91)) = 15.35	
$F_{en\ (RHR\ piping)}$	= exp(0.935-(1.0)(0.260)(-6.91)) = 15.35	

<u>IP3</u>

$T'_{(Surge\ line)}$	= 1.0	(T ≥ 200ºC)
$T'_{(Charging\ nozzle)}$	= 1.0	(T ≥ 200ºC)
$T'_{(SI\ nozzle)}$	= 1.0	(T ≥ 200ºC)
$T'_{(RHR\ piping)}$	= 1.0	(T ≥ 200ºC)
$O'_{(All)}$	= 0.260	RCS dissolved oxygen is ≤ 50 ppb
$\dot{\varepsilon}'_{(all)}$	= -6.91	Assume bounding strain rate
$F_{en\ (Surge\ line)}$	= exp(0.935-(1.0)(0.260)(-6.91)) = 15.35	
$F_{en\ (Charging\ nozzle)}$	= exp(0.935-(1.0)(0.260)(-6.91)) = 15.35	
$F_{en\ (SI\ nozzle)}$	= exp(0.935-(1.0)(0.260)(-6.91)) = 15.35	
$F_{en\ (RHR\ piping)}$	= exp(0.935-(1.0)(0.260)(-6.91)) = 15.35	

RAI 4.3.1.8-2

Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants (SRP-LR) Section 4.3.2.1.1.3 provides the basis for the staff acceptance of an aging management program to address environmental fatigue. It states, "[t]he staff has evaluated a program for monitoring and tracking the number of critical thermal and pressure transients for the selected reactor coolant system components. The staff has determined that this program is an acceptable aging management program to address metal fatigue of the reactor coolant system components according to 10 CFR 54.21(c)(1)(iii)." The staff is unable to determine if the Fatigue Monitoring Program of Indian Point 2 and Indian Point 3 contain sufficient details to satisfy this criterion. Please provide adequate details of the Fatigue Monitoring Program such that the staff can make a determination based on the criterion set forth in SRP-LR Section 4.3.2.1.1.3. Also, please explain in detail the corrective actions and the frequency that such actions will be taken so the acceptance criteria will not be exceeded in the period of extended operation.

Response to RAI 4.3.1.8-2

The IPEC Fatigue Monitoring Program was compared to the program described in NUREG-1801 (GALL), Section X.M1, Metal Fatigue of Reactor Coolant Pressure Boundary. The program description in the GALL report is directly applicable to the IPEC units. As indicated in LRA Section B.1.12, during the period of extended operation the IPEC program will be consistent with the GALL program with one exception. The exception to GALL is that rather than performing periodic updates of CUF calculations, IPEC periodically assesses the number of transient cycles compared to calculation assumptions and updates the CUF calculations, if necessary. Based on this comparison including evaluation of the exception, the approvals set forth in the GALL report apply to the IPEC Fatigue Monitoring Program.

The program description was modified per letter NL-08-21, Indian Point Nuclear Generating Units Nos. 2 & 3 License Renewal Application Amendment 2, dated January 22, 2008. This letter commits to complete CUF calculations for all areas identified in NUREG-6260 (LRA Table 4.3-13 for IP2 and LRA Table 4.3-14 for IP3), incorporating the effect of the reactor coolant environment, for IP2 and IP3. Once these CUF calculations are complete (at least 2 years prior to the period of extended operation), IPEC will ensure that the cycles analyzed in the new or updated calculations are included in the Fatigue Monitoring Program. IPEC will continue to manage the effects of fatigue throughout the period of extended operation by monitoring cycles incurred and assuring they do not exceed the analyzed numbers of cycles.

As required by IPEC technical specifications, the Fatigue Monitoring Program tracks actual plant transients and evaluates these against the design transients. The plant transient counts are updated at least once each operating cycle. This frequency is acceptable since the evaluation during each update determines if the number of design transients could be exceeded prior to the next update. The Fatigue Monitoring Program ensures that the numbers of transient cycles experienced by the plant remain within the analyzed numbers of cycles and hence, the component CUF calculations remain valid.

The program requires corrective actions before exceeding the analyzed number of transient cycles. The corrective actions are implemented in accordance with the IPEC corrective action program. IPEC may perform further reanalysis if cycle counts approach analyzed numbers. These calculation updates will be governed by Entergy's 10 CFR 50 Appendix B Quality Assurance (QA) program and include design input verification and independent reviews ensuring that valid assumptions, transients, cycles, external loadings, analysis methods, and environmental fatigue life correction factors will be used in the fatigue analyses. Repair or replacement of the affected component(s), if necessary, will be done prior to exceeding the allowable CUF in accordance with established plant procedures governing repair and replacement activities. These established procedures are governed by Entergy's 10 CFR 50 Appendix B QA program and meet the applicable repair or replacement requirements of the ASME Code Section XI.

RAI 3.0.3.2.3-1

Indian Point 2 Updated Final Safety Analysis Report, Revision 20, dated 2006, Section 14.2.1 on page 55 of 218, states in part that:

> "Northeast Technology Corporation report NET-173-01 and NET-173-02 are based on conservative projections of amount of boraflex absorber panel degradation assumed in each sub-region. These projections are valid through the end of the year 2006."

Please confirm that the Boraflex neutron absorber panels in the Indian Point Unit 2 spent fuel pool have been re-evaluated for service through the end of the current licensing period. Also, please discuss the plans for updating the Boraflex analysis during the period of extended operation.

Response to RAI 3.0.3.2.3-1

Boron-10 areal density gage for evaluating racks (BADGER) testing was performed in February 2000, July 2003, and again in July 2006. Using the latest test data and RACKLIFE code projections, the Boraflex neutron absorber panels in the Indian Point Unit 2 spent fuel pool will meet the Technical Specification requirements through the end of the current licensing period. The next BADGER test will be performed prior to the end of calendar year 2009. As required by the Boraflex Monitoring Program (LRA Section B.1.3), periodic BADGER testing and RACKLIFE code projections will continue through the period of extended operation to confirm acceptable Boraflex condition.

The referenced section of the Indian Point 2 Updated Final Safety Analysis Report will be updated in the next revision.