Entergy

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

Fred Dacimo
Vice President
License Renewal

January 17, 2008

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

NL-08-016

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

SUBJECT: **Reply to Request for Additional Information Regarding License Renewal Application – (Reactor Coolant Pump Flywheel and Leak Before Break Analyses)**

Reference: NRC letter dated December 21, 2007; "Request for Additional Information for the Review of the Indian Point Nuclear Generating Unit Nos. 2 And 3, License Renewal Application—Reactor Coolant Pump Flywheel and Leak Before Break Analyses "

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 regarding Reactor Coolant Pump Flywheel and Leak Before Break Analyses.

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 1-17-08.

Sincerely,

Patric W. Conroy for per telecon

Fred R. Dacimo
Vice President
License Renewal

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NRR

Attachment:

1. Reply to NRC Request for Additional Information Regarding License Renewal Application - (Reactor Coolant Pump Flywheel and Leak Before Break Analyses)

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
Mr. Mark Cox, NRC Senior Resident Inspector, IP2
Mr. Paul Cataldo, NRC Senior Resident Inspector, IP3
Mr. Paul D. Tonko, President, NYSERDA
Mr. Paul Eddy, New York State Dept. of Public Service

ATTACHMENT I TO NL-08-016

REPLY TO NRC REQUEST FOR ADDITIONAL INFORMATION

REGARDING

LICENSE RENEWAL APPLICATION

(Reactor Coolant Pump Flywheel and Leak Before Break Analyses)

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)

The U.S. Nuclear Regulatory Commission (NRC or staff) has reviewed the information related to Reactor Coolant Pump Flywheel And Leak Before Break Analyses provided by the applicant in the Indian Point Nuclear Generating Unit Nos. 2 and 3 (IP2 and IP3) LRA. The staff has identified that additional information is needed to complete the review as addressed below.

Reactor Coolant Pump Flywheel Analysis

RAI 4.7.1-1

On page 4.7-1 of the license renewal application (LRA), first paragraph, the applicant stated that the aging effect of concern is fatigue crack initiation and growth in the flywheel bore keyway from stresses due to starting the motor. Discuss whether stress corrosion cracking should also be considered as a degradation mechanism in the bore keyway considering the effect of the environment, stress conditions, and material.

Response for RAI 4.7.1-1

Section 4 of the LRA pertains to time limited aging analyses (TLAA). As there is no analysis of the reactor coolant pump flywheels dealing with stress corrosion cracking, there is no TLAA associated with stress corrosion cracking.

Cracking due to stress corrosion is an aging effect considered in the aging management review of those components that are in the scope of license renewal and are subject to aging management review. As the RCP flywheel (RCP motor) is an active component, it is not subject to aging management review and therefore is not addressed in the aging management review as indicated in Section 3. In any event, the flywheel is a carbon steel component exposed to indoor air. Since the flywheel operates at ambient temperature in a dry indoor air environment, cracking due to stress corrosion is not a plausible aging effect. This is in accordance with the EPRI Mechanical Tools for the identification of aging effects.

RAI 4.7.1-2

On page 4.7-1 of the LRA, second paragraph, the applicant stated that the Westinghouse report WCAP-15666-A used 6000 start/stop cycles of a reactor coolant pump in the analysis of the flywheel. However, as shown in Table 4.3-1 of the LRA, under the "Analyzed Number of Cycles" column, the reactor coolant pump start/stop condition has 10,000 cycles.

(a) Discuss why 10,000 cycles of the reactor coolant pump startup/stop condition were not used in the flywheel analysis.
(b) LRA Table 4.3-1 lists various normal, test, and abnormal conditions. Some of those conditions may affect flywheel operation and the structural integrity of the flywheel. However, the applicant only mentioned the reactor coolant pump start/stop condition in WCAP-15666-A. Discuss whether other normal, test and abnormal conditions in LRA Table 4.3-1 should be used in WCAP-15666-A.

Response for RAI 4.7.1-2

(a) The 10000 RCP starts shown in Table 4.3-1 are considered for their impact on the RCS. This value applies to starts from any one of the four RCPs and therefore is not an appropriate value to use in the analysis for a single RCP motor flywheel. Heatup and cooldown cycles are limited to 200. Even if ten starts and stops for the limiting pump occur during each heatup and cooldown cycle, only 2000 RCP cycles will result. This is well below 6000 cycles. Therefore, 6000 is an acceptable number of cycles for the RCP flywheel analysis.

(b) RG-1.14, Revision 1, Section C, Subsection 2 provides the regulatory position for flywheel design, and those guidelines were followed in the flywheel evaluation in WCAP-15666. The NRC reviewed and approved WCAP-15666 and agreed that it is an acceptable flywheel analysis. (NRC Letter, Herbert N. Berkow to Robert H. Bryan, Chairman, Westinghouse Owner's Group, "Safety Evaluation of Topical Report WCAP-15666, "Extension of Reactor Coolant Pump Motor Flywheel Examination" (TAC No. MB2819)," 5 May 2003)

Section 2 (page 2-19) of WCAP-15666 states "There are no significant mechanisms for inservice degradation of the flywheels, since they are isolated from the primary coolant environment." Since the flywheels are isolated from the primary coolant environment, the remaining transients in Table 4.3-1 have no effect on the flywheel operation and structural integrity.

RAI 4.7.1-3

On page 4.7-1 of the LRA, second paragraph, the applicant stated that the reactor coolant pump flywheel is inspected every 20 years.

(a) Discuss the inspection history, results, method used, area/volume, and coverage.
(b) Discuss future inspection plans including whether a volumetric inspection will be performed at the end of 40 years or during the extended period of operation. If not, discuss how the structural integrity of the flywheel can be ensured.
(c) Discuss whether the flywheel surface is painted. If the flywheel surface is painted, discuss the effectiveness of the surface or visual examination if these inspection methods were used in the past or will be used in the future.

Response for RAI 4.7.1-3

(a) At IPEC, the RCP motor flywheels are inspected using approved NDE methods. The flywheel NDE methods include volumetric, surface, and visual examination techniques.

Volumetric - The ultrasonic examinations performed include a keyway corner exam, a radial gage hole exam, and a periphery exam. In the gage hole exam, the full axial depth of the gage hole is traversed. The exam is performed at each of four gage holes. Additionally, an ultrasonic examination is performed from the periphery of the flywheel scanning toward the bore. Essentially 100% of the specified volume coverage is obtained.

Surface - The surface examination performed includes the bore and keyway surfaces of the flywheel using dye penetrant inspection techniques. Essentially 100% of the specified surface coverage is obtained.

Visual - The visual examination includes inspection of high stress areas on all surfaces. Essentially 100% of the specified surface coverage is obtained.

There have been no recordable indications identified from the IPEC RCP flywheel inspections.

(b) As a result of the NRC Staff's approval of WCAP-15666, IPEC extended the RCP flywheel inspection frequency from once every 10 years to once every 20 years. This change occurred in 2004. IPEC will continue to inspect the RCP flywheels as described above at a frequency of at least once every 20 years through the period of extended operation. Based on the evaluations provided in WCAP-15666, which has been approved by the NRC Staff, the above inspection methods and frequency are sufficient to ensure structural integrity of the RCP flywheels through the period of extended operation.

(c) Some of the surface areas of the RCP flywheels are painted. However, the areas that are subject to inspection via volumetric, surface, and visual examinations, as specified above, are not painted. Therefore, the effectiveness of the NDE examinations performed on the RCP flywheel is not compromised.

RAI 4.7.1-4

On page 4.7-1 of the LRA, third paragraph, the applicant stated that "As indicated in Tables 4.3-1 and 4.3-2, the allowable number of heatup and cooldown cycles for 60 years of operation is 200 for Units 2 and 3....Because the 6000 cycles assumed in the analysis far exceeds the expected cycles in 60 years..." It is not clear whether the applicant was comparing the 6000 cycles in the analysis to the 200 cycles of heatup/cooldown. If this was the applicant's intention, it should be noted that 6000 cycles are related to the pump startup/stop whereas the 200 cycles are related to the heatup and cooldown. During each heatup cycle, there may be multiple reactor coolant pump startups. The comparison should be between the projected/ expected cycles in 60 years vs. cycles used in the analysis for the reactor coolant pump start/stop event. Clarify the statements in the above quotes or provide further information in support of this conclusion.

Response for RAI 4.7.1-4

The third paragraph of Section 4.7.1 of the LRA is reproduced here.

> "As indicated in Tables 4.3-1 and 4.3-2, the allowable number of heatup and cooldown cycles for 60 years of operation is 200 for Units 2 and 3. The analyzed number of cycles is far greater than the expected number, even if multiple reactor coolant pump starts are assumed in each startup shutdown cycle. Because the 6000 cycles assumed in the analysis far exceeds the expected cycles in 60 years, and because the analysis is based on 60 years rather than 40 years, this analysis does not meet the 10CFR54(3)(a)(3) criteria for a TLAA. It does not involve time-limited assumptions defined by the current operating term or by an operating term less than the current operating term plus the period of extended operation requested in the license renewal application."

The second sentence acknowledges that there may be multiple starts/stops per heatup; however, even if a conservative number of starts/stops of the limiting motor is assumed, the value is still well below 6000. (You would have to assume an unrealistic 30 starts/stops per each and every heatup to get to 6000 starts for the limiting flywheel. Ten starts per heatup is a conservative estimate, and that only results in 2000 starts for 200 heatups.)

RAI 4.7.1-5

Discuss why in LRA Section A.2.2, *Evaluation of Time-Limited Aging Analyses,* of Appendix A, *Updated Final Safety Analysis Report Supplement,* there is no discussion for the reactor coolant pump flywheel. If the time-limited aging analysis is applicable for the reactor coolant pump flywheel, a discussion should be included in Appendix A. Revise Appendices A.2 and A.3 of the LRA for Units 2 and 3, respectively, as necessary.

Response for RAI 4.7.1-5

Section 4.0 of the IPEC LRA evaluates potential time-limited aging analyses. Section 4.7.1 of the IPEC LRA concludes as follows.

> *Evaluation*
>
> Evaluation is not applicable since the flywheel analysis is not a time-limited aging analysis as defined by 10 CFR 54.3. The analysis does not meet Part (3) of the definition in 10 CFR 54.3, Definitions.

Since the flywheel is not susceptible to stress corrosion cracking and the number of start/stop cycles bound the projected number of cycles for 60 years, the analysis is not a TLAA.

As this analysis is not a time-limited aging analysis, it is not included in Appendix A of the license renewal application.

Leak Before Break

RAI 4.7.2-1

On page 4.7-2 of the LRA, first paragraph, the applicant stated that for Unit 2, leak before break (LBB) analyses are documented in WCAP-10977, WCAP-10977, Supplement 1, and WCAP-10931. By letter dated February 23, 1989, the NRC staff issued its safety evaluation approving the applicant's LBB application. In its safety evaluation, the NRC staff granted LBB for selected Unit 2 piping systems based on the technical basis of WCAP-10977, Revision 2; WCAP-10977, Supplement 1; and WCAP-10931, Revision 1.

(a) Confirm that Revision 2 to WCAP-10977 and Revision 1 to WCAP-10931 are the correct revisions that were used for the Unit 2 LBB application.
(b) Provide a list of piping systems in Units 2 and 3 that have been granted for LBB.

Response for RAI 4.7.2-1

(a) Revision 2 and Supplement 1 to WCAP-10977 and Revision 1 to WCAP-10931 are the correct revisions that were used for the Unit 2 LBB application.

(b) The primary coolant system loop piping (the hot leg from the reactor vessel to the reactor coolant pumps, the intermediate crossover pipe, and the cold leg from the steam generators to the reactor vessel) in Unit 2 and 3 were evaluated for LBB.

RAI 4.7.2-2

On page 4.7-2 of the LRA, second paragraph, the applicant stated that Unit 3 LBB analyses have been documented in the Westinghouse report, WCAP-8228. However, other LBB analyses prepared for Unit 3 have been reviewed by the NRC. Please confirm whether there are other applicable LBB analyses of record for Unit 3, and provide a history and summary description of all these analyses (including the cited WCAP-8228), including the parameters that were evaluated and conclusions reached for each analysis.

Response for RAI 4.7.2-2

Between 1981 and 1984, IPEC performed LBB analyses for the IP3 primary loop piping. These analyses took into account thermal aging effects on cast stainless steel components in the IP3 primary loop. The results of the 1984 LBB analyses were documented in Fracture Proof Design Corporation Report 80-121, Revision 1. IPEC requested authorization to apply LBB methodology to the primary loop piping based on these analyses.

The NRC approved the application of LBB methods for Unit 3 primary loop piping through an SE issued in 1986. Updated LBB analyses were performed in 1997 in support of the Steam Generator Snubbers Deactivation Program at IP3. The results of the 1997 analysis were documented in Appendix A of WCAP-8228, Revision 1. As part of the stretch power uprate (SPU) for IP3, WCAP-16212 was prepared including updated LBB analyses to ensure that the elimination of the primary loop pipe breaks continues to be justified at the uprated operating conditions defined in Section 5.4.2. WCAP-16212 evaluated the effects of the stretch power uprate on the acceptability of the LBB status of the primary loop piping. This report in Section 5.4.2.5 documented the determination that the LBB conclusions shown in the 1984 report 80-121 and Appendix A of WCAP-8228, Revision 1 remain valid. The NRC concurred with this conclusion in Section 3.6.6.1 of the SER for the IP3 SPU. The potential time-limited assumptions in Appendix A of WCAP-8228, revision 1 and WCAP-16212 involve the thermal aging of cast austenitic stainless steel and the fatigue crack growth analysis. These two assumptions are addressed below.

Thermal Aging of CASS

The first analysis consideration in WCAP-10977 and WCAP-8228, Appendix A that could be influenced by time is the material properties of cast austenitic stainless steel used in the pipe fittings. Thermal aging causes an elevation in the yield strength of CASS and a decrease in fracture toughness, the decrease being proportional to the level of ferrite in the material. Thermal aging in these stainless steels will continue until a saturation, or fully aged, point is reached. WCAP-10977 and Appendix A of WCAP-8228 address the fracture toughness properties of statically cast CF8M stainless steel. Specifically, fully aged, bounding fracture toughness values were used to conservatively calculate the J value for the cast fittings. The IP3 LBB analysis uses the methodology of NUREG/CR-4513 and WCAP-10931 to determine saturation (fully aged) toughness values. The IP2 LBB analysis uses the methodology of WCAP-10931 to determine saturation (fully aged) toughness values. As the LBB evaluations for both units use saturated (fully aged) fracture toughness properties, these analyses do not have a material property time-dependency and are not considered TLAA.

Fatigue Crack Growth

The second analysis consideration that could be influenced by time is the accumulation of actual fatigue transient cycles used in WCAP-10977, Rev. 2, and Supplement 1 and WCAP-8228, Appendix A. A fatigue crack growth analysis of the reactor vessel inlet nozzle to safe-end region was performed to determine its sensitivity to the presence of small cracks. The nozzle to safe-end connection was selected because crack growth calculated at this location is representative of the entire primary loop. The nozzle to safe-end connection configuration includes an SA-508 Class 2 or Class 3 stainless steel clad nozzle connected to a stainless steel safe end by a nickel-based alloy weld. A fatigue crack growth rate law for the stainless steel clad low-alloy steel nozzle was obtained from ASME Section XI. Fatigue crack growth rate laws for stainless steel and Alloy 600 in a PWR environment were developed based on available industry literature. These crack growth rate laws were applied based on all normal, upset, and test reactor vessel fatigue transients resulting in projected rates of crack growth calculated in units of inches/cycle for ferritic steel, stainless steel and Alloy 600.

Section 6.0 of the WCAP-10977 and Appendix A, Section 8.0 of WCAP-8228 evaluate fatigue crack growth at the inlet nozzle to safe-end region. In Table 6-1 and Table 8-2 of the respective WCAPs, the fatigue crack growth for 40 years was found to be very small regardless of the material evaluated based on the projected transients and stress intensity factors. The projected 60-year transient cycles for IP2 and IP3 are essentially the same as the number of analyzed cycles for the original 40-year licensing period. The 60-year projections for IP2 show that none of the transients that affect the nozzle inlet to safe end fatigue analysis will exceed the analyzed cycles. The 60-year projections for IP3 show that no transient will exceed the number of analyzed cycles prior to the end of the period of extended operation. As a result there is reasonable assurance that the fatigue crack growth analyses presented in WCAP-10977 (IP2) and Appendix A of WCAP-8228, Revision 1, (IP3) will remain valid during the period of extended operation. Therefore, the LBB TLAA remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

RAI 4.7.2-3

On page 4.7-2 of the LRA, third paragraph, the applicant stated that the fully-aged fracture toughness values (i.e., bounding values) to address thermal aging of cast austenitic stainless steel (CASS) were used in the LBB analyses.

(a) Provide a list of LBB piping systems that contain CASS components and identify the components.
(b) Provide the fully-aged fracture toughness values of the CASS materials used in the LBB analyses and the normal fracture toughness values.

(c) The applicant has an Aging Management Program B.1.37, *Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)*, to manage CASS components. However, Section 4.7.2 of the LRA did not mention this aging management program (AMP) to manage the LBB piping systems. Discuss whether AMP B.1.37 or some other AMP (and if so, which AMP) will be used to monitor aging effects the CASS components in the LBB piping systems.

Response for RAI 4.7.2-3

(a) As stated in response to RAI 4.7.2-1 above, LBB has only been applied to the main RCS piping and it has not been applied to any other systems or branch lines. The RCS piping material for both IP2 and IP3 is SA 376 Type 316 forged austenitic stainless steel while the fitting (i.e. elbows) material is SA 351 Type CF8M cast austenitic stainless steel (CASS).

(b) The pre-service (normal) and the fully aged fracture toughness values (i.e. J_{IC}, T_{mat} and J_{max}) for IP2 were taken from WCAP-10977 revision 2 (previously submitted to the NRC) as the lower bound values at 600^0 F. These IP2 fracture toughness values also bound the IP3 locations evaluated in WCAP-8228, revision 1.

(c) Section 3 of the LRA provides aging management review results for systems within the scope of license renewal. For those systems with CASS components and an environment that can result in thermal aging embrittlement, the Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) program is specified to manage aging effects consistent with Section IV of NUREG-1801. This program is specified in the tables in Section 3.1 that include LBB piping.

Section 4 of the LRA deals with time limited aging analyses. In accordance with 10 CFR 54.21(c)(1), aging management comes into play only in subparagraph (iii) if there is a TLAA and that TLAA does not remain valid or cannot be projected through the period of extended operation. As the IP2 and IP3 analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i) (see response to RAI 4.7.2-2 above), there is no aging management program specified in Section 4 to satisfy 10 CFR 54.21(c)(1)(iii) for these components.

RAI 4.7.2-4

In the third paragraph on page 4.7-2 of the LRA, several analyses are briefly mentioned. Provide a list and summary (e.g., purpose, parameters evaluated, and conclusions) of each analysis used for thermal aging of CASS.

Response for RAI 4.7.2-4

The analyses referred to in the third and fourth paragraphs on page 4.7-2 are the analyses listed in the first and second paragraphs on that page for Units 2 and 3.

WCAP- 10977, *Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Indian Point Unit 2*, Original - November 1985, Revision 1 - March 1986, Revision 2 - December 1986.

WCAP- 10977, Supplement 1, *Additional Information in Support of the Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Indian Point Unit 2*, January, 1989

WCAP-10931, Revision 1, *Toughness Criteria for Thermally Aged Cast Stainless Steel*, July 1986

Westinghouse WCAP-8228, Volume I, Revision 1, *Structural Evaluation of Reactor Coolant Loop/Support System for Indian Point Nuclear Generation Station Unit 3*, April 1997, Appendix A

These analyses are explained in the responses to RAIs 4.7.2-1, 4.7.2-2, 4.7.2-3 and 4.7.2-7 on this section of the application.

RAI 4.7.2-5

By letter dated May 19, 2000, Christopher I. Grimes of the NRC staff forwarded to Douglas J. Walters of the Nuclear Energy Institute an evaluation of thermal aging embrittlement of CASS components (ADAMS Accession No. ML003717179). In the NRC staff's evaluation, the staff provided its positions on how to manage CASS components. Discuss whether and how the CASS components in the LBB piping satisfy the staff positions in its evaluation dated May 19, 2000.

Response for RAI 4.7.2-5

The aging management review results for CASS components are provided in Section 3 of the LRA. The aging management review results for CASS components in Section 3 agree with the Staff position expressed in the May 19, 2000 letter from Christopher I. Grimes.
The response to RAI 4.7.2-3(a) identifies that the only CASS components to which LBB has been applied are pipe fittings (elbows). These fittings will be screened based upon the molybdenum content, casting method, and ferrite content; then inspected as appropriate, in accordance with the Grimes letter. This will be performed under the IPEC Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program. This program is credited in multiple line items in Tables 3.1.2-3-IP2 and 3.1.2-3-IP3 and, as described in LRA Section B.1.37, will be consistent with the program described in NUREG-1801, XI.M12. The description of the program in LRA Appendix B references the Grimes letter in question.

RAI 4.7.2-6

On page 4.7-2 of the LRA, third paragraph, the applicant stated that thermal aging causes an increase in the yield strength of CASS. However, the applicant did not discuss this parameter further with respect to the LBB analyses on page 4.7-2 of the LRA.

(a) Discuss whether the limiting yield strength was used in the LBB analyses.
(b) Discuss whether the LBB analyses approved for 40 years are applicable for 60 years in terms of yield strength used.

Response for RAI 4.7.2-6

(a) The yield and ultimate strength used in the LBB analysis for both IP2 and for IP3 were taken as the lower bound values since this results in the most limiting conditions. The increase in the material yield strength was not credited in the analysis since the lower bound values bound the aged values.

(b) As discussed in RAI 4.7.2-3 above, the mechanical properties and fracture toughness values used in the LBB analyses for both IP2 and for IP3 included the most limiting values for both the pre-service conditions as well as the fully aged conditions. Since no additional drop in fracture toughness properties is expected once fully aged conditions are reached, these analyses are time independent and therefore bound the 60 year operating life.

RAI 4.7.2-7

On page 4.7-2 of the LRA, fourth paragraph, the applicant discussed the fatigue crack growth analysis of the reactor vessel inlet nozzle to safe-end without any details.

(a) Discuss the results of the fatigue crack growth analysis.
(b) Discuss whether the final crack size satisfies the acceptance criteria and discuss the acceptance criteria.
(c) Discuss the postulated initial flaw size and location for the fatigue growth calculation.
(d) Discuss the material specification/ identification of the nickel-based alloy weld.
(e) The applicant stated that "...The nozzle to safe-end connection was selected because crack growth calculated at this location is representative of the entire primary loop..." Clarify whether the nozzle to safe-end connection is the limiting/bounding location in terms of the fatigue crack growth; if it is not, identify the limiting/bounding location and explain why it is sufficient to evaluate the representative location.
(f) If the above information can be found in a technical report, identify the report(s) and provide a copy of the report(s).

Response for RAI 4.7.2-7

(a) The results of the fatigue crack growth analyses used in WCAP-10977, Rev. 2, and Supplement 1 and WCAP-8228, Appendix A demonstrate that fatigue crack growth is small. (See Table 6-1 of WCAP 10977, Rev 2).

(b) The analyses in WCAP-10977, Rev. 2, and Supplement 1 and WCAP-8228, Appendix A show that the final crack size for small stable flaws varies by location within the primary coolant system loop piping. For each limiting location the final crack size satisfies LBB acceptance criteria since adequate margin exists between the calculated leak rate and the 1 gpm criterion in regulatory Guide 1.45 and there is sufficient margin between detectable leaks and large stable flaws.

(c) The fatigue crack growth analyses in WCAP-10977, Rev. 2, and Supplement 1 and WCAP-8228, Appendix A, postulate four initial flaw sizes ranging from 0.292 inches to 0.425 inches and locations with three different materials; Stainless Steel, SA 508 Low Alloy Steel, and an Inconel weld cross-section.

(d) The weld cross section is nickel-based alloy (Alloy 600 SA 82/182).

(e) The junction of the hot leg and the reactor vessel outlet nozzle is the load limiting/bounding location and the toughness critical locations associated with thermal aging occur in several pipe fittings. These are where the hot leg meets steam generators, the intermediate crossover leg, and where the cold leg meets the reactor vessel inlet nozzle.

(f) The above information can be found in WCAP-10977, Rev. 2, and Supplement 1 which were provided for Staff review in Con Edison letters to the USNRC dated May 23, 1988 and January 12, 1989

RAI 4.7.2-8

Pressurized water reactors have experienced primary water stress corrosion cracking in Alloy 600/82/182 weld material.

(a) Provide a list of Alloy 82/182 weld material in any of the piping systems that have been approved for LBB. In this list, include the name of the corresponding piping system, pipe size, and weld identification number.
(b) Discuss whether the Alloy 600/82/182 welds have been inspected and the inspection results.

Response for RAI 4.7.2-8

Below is a list of Alloy 82/182 weld material used in the primary coolant system loop piping that has been approved for LBB. The list includes the name of the corresponding piping system, pipe size, and weld identification number.

Unit	Piping Identification	Pipe Size	Weld ID Number
IP2	Primary Coolant Loop 21 Rx Vessel Outlet Nozzle	29" I.D.	RPVS-21-1A
IP2	Primary Coolant Loop 21 Rx Vessel Outlet Nozzle	27 ½" I.D.	RPVS-21-14A
IP2	Primary Coolant Loop 22 Rx Vessel Outlet Nozzle	29" I.D.	RPVS-22-1A
IP2	Primary Coolant Loop 22 Rx Vessel Outlet Nozzle	27 ½" I.D.	RPVS-22-14A
IP2	Primary Coolant Loop 23 Rx Vessel Outlet Nozzle	29" I.D.	RPVS-23-1A
IP2	Primary Coolant Loop 23 Rx Vessel Outlet Nozzle	27 ½" I.D.	RPVS-23-14A
IP2	Primary Coolant Loop 24 Rx Vessel Outlet Nozzle	29" I.D.	RPVS-24-1A
IP2	Primary Coolant Loop 24 Rx Vessel Outlet Nozzle	27 ½" I.D.	RPVS-24-14A
IP3	Primary Coolant Loop 31 Rx Vessel Outlet Nozzle	29" I.D.	INT-1-4100-1(DM)
IP3	Primary Coolant Loop 31 Rx Vessel Inlet Nozzle	27 ½" I.D.	INT-1-4100-16(DM)

Unit	Piping Identification	Pipe Size	Weld ID Number
IP3	Primary Coolant Loop 32 Rx Vessel Outlet Nozzle	29" I.D.	INT-1-4200-1(DM)
IP3	Primary Coolant Loop 32 Rx Vessel Inlet Nozzle	27 ½" I.D.	INT-1-4200-16(DM)
IP3	Primary Coolant Loop 33 Rx Vessel Outlet Nozzle	29" I.D.	INT-1-4300-1(DM)
IP3	Primary Coolant Loop 33 Rx Vessel Inlet Nozzle	27 ½" I.D.	INT-1-4300-16(DM)
IP3	Primary Coolant Loop 34 Rx Vessel Outlet Nozzle	29" I.D.	INT-1-4400-1(DM)
IP3	Primary Coolant Loop 34 Rx Vessel Inlet Nozzle	27 ½" I.D.	INT-1-4400-16(DM)

These welds are routinely inspected as part of the Inservice Inspection Program. The welds were volumetrically inspected on Unit 2 in Spring 2006 and on Unit 3 in Fall 1999 with no unacceptable indications.

RAI 4.7.2-9

On page 4.7-2 of the LRA, fourth paragraph, the applicant stated that the crack growth due to fatigue was evaluated assuming the reactor vessel experienced the total allowable numbers of normal, upset, and test transients.

(a) Provide a list of the transient conditions and associated number of cycles (for 40 years) used in the fatigue crack growth analysis (e.g., 200 cycles of heatup) and the number of cycles for those transient conditions projected to 60 years.

(b) Clarify whether the fatigue crack growth calculation discussed in the fourth paragraph on page 4.7-2 is for 40 years or 60 years.

(c) Clarify whether a fatigue crack growth calculation was performed for 60 years.

Response for RAI 4.7.2-9

(a) The list of the transient conditions and associated number of cycles (for 40 years) used in the fatigue crack growth analysis are the design transients originally defined in the plant's equipment specifications and analyzed in the original component stress reports. Table 4.1-8 of IP2 UFSAR revision 20, (2006) and Table 4.1-8 of IP3 UFSAR revision 1, (2005) lists the design transient cycles. The projected numbers of transient cycles for 60 years remain within these analyzed values. The design transients in UFSAR Table 4.1-8 are included in LRA Tables 4.3-1 and 4.3-2.

(b) The fatigue crack growth calculation discussed in the fourth paragraph on page 4.7-2 documents that transient induced fatigue growth is for the numbers of cycles discussed in response to the part (a) of this RAI. The time to experience those transients is not a factor in the analysis.

(c) The fatigue crack growth calculation in WCAP-10977 was performed for the number of cycles discussed above. No new calculation was performed for 60 years since the projected number of cycles for 60 years is less than the numbers of cycles used in the analysis. As indicated in LRA section 4.7.2, IP2 and IP3 analyses will remain valid during the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

RAI 4.7.2-10

The NRC has approved two power uprate applications (measurement uncertainty and stretch power uprates) for Unit 2 (ADAMS Accession Nos. ML031420375 and ML023290636) and Unit 3 (ADAMS Accession Nos. ML042960007 and ML050600380). Please discuss whether the results of the 40-year LBB analyses are bounding for conditions at the end of 60 years in light of the power uprates. The discussion should include assessments of piping loads, stresses, and safety margins as specified in NUREG-1061, Vol. 3. This question covers all piping systems that have been approved for LBB for both units.

Response for RAI 4.7.2-10

As discussed in LRA Section 4.7.2, the leak before break analyses for IP2 and IP3 will remain valid during the period of extended operation. This is because they are not "40-year" analyses, rather they are analyses based on saturated material properties and numbers of design transients that will not be exceeded in 60 years.

The effects of the power uprate on the LBB analyses were addressed in the power uprate analyses (WCAP-16156 for IP2 and WCAP-16211 for IP3). The conclusion of both WCAPs (in Sections 5.4.2) is that the dynamic effects of RCS primary loop pipe breaks need not be considered in the structural design basis at the SPU conditions. This was approved by the NRC in their acceptance of those submittals (See SERs for Unit 2 (ADAMS Accession Nos. ML031420375 and ML023290636) and Unit 3 (ADAMS Accession Nos. ML042960007 and ML050600380). Full references for the two WCAPs are as follows.

WCAP-16156-P, "Indian Point Nuclear Generating Unit No. 2, Stretch Power Uprate NSSS Engineering Report," February 2004

WCAP-16211-P, "Power Uprate Project, Indian Point Unit 3 Power Plant, NSSS Engineering Report, June 2004

The LBB analysis for both IP2 and for IP3 used a Westinghouse-proprietary methodology which has been previously reviewed and approved by the NRC Staff. Although the Westinghouse methodology is consistent with the methodology provided in NUREG-1061, volume 3, these two methodologies are not exactly the same. In some cases the Westinghouse methodology is more conservative while in other cases the NUREG methodology is slightly more conservative. However, both of these methods provide sufficient margins of safety to ensure that leakage from a crack under normal operating loads would be detected by the existing leak detection systems, but that this same crack would not result in pipe failure under postulated accident loads.

RAI 4.7.2-11

On page 4.7-2 of the LRA, fifth paragraph, the applicant concluded that "...Thus, the IP2 and IP3 analyses will remain valid during the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i)..." The applicant's conclusion was based on the evaluation of fatigue crack growth and thermal aging of CASS. For each LBB piping system included in the TLAA evaluation, discuss the applicability of the 40-year LBB analyses for the period of extended operation based on the following considerations and parameters,

(1) leakage calculations as part of LBB analyses,
(2) crack stability, and
(3) capability of the reactor coolant leakage detection system which is a part of overall LBB technology.

Response for RAI 4.7.2-11

(1) The leakage calculations for both IP2 and IP3 were based on the operating loads, the material properties and the through-wall crack length for each of the bounding locations. Since the number of fatigue cycles analyzed bounds the period of extended operation (i.e. 60 years) and since the evaluations used the fully aged fracture toughness values, the bounding flaw size and the material properties are also bounding for 60 years. The normal operating loads are unaffected by the additional 20 years of operation since the operating conditions are not changed for the period of extended operating. Therefore, the leakage calculations performed in support of the LBB for 40 years of operation remain valid for the additional 20 years of operation.

(2) As discussed above, the crack stability analyses were performed using the most limiting fracture toughness values considering both the pre-service conditions as well as the fully aged conditions. Since no additional drop in fracture toughness properties is expected once fully aged conditions are reached, these analyses are time independent and therefore bound the 60 year operating life.

(3) The leak detection systems for both IP2 and IP3 are based on the following instrumentation.

1. Containment air radioactive particulate monitor
2. Containment air radioactive gas monitor (sensitivity variable depending on the amount of fuel clad leakage to provide radiogas to the coolant)
3. Containment sump monitor
4. Fan cooler unit condensate flow rate monitor

The capability of the leak detection system components remains unchanged from that represented in the LBB NRC SERs for each unit.