From: Kimberly Green
To: <dtyner@entergy.com>,"MICHAEL D STROUD" <MSTROUD@entergy.com>
Date: 12/10/2007 5:40:49 PM
Subject: Draft Telecon Summary — RCP and LBB
cc: <IPNonPublicHearingFile@nrc.gov>

Mike,

Please look over the draft telecon summary and let me know if there are any errors or discrepancies.

Thanks,

Hearing Identifier: IndianPointUnits2and3NonPublic
Email Number: 288

Mail Envelope Properties (477228E6.HQGWDO01.OWGWPO04.200.2000008.1.15A8E9.1)

Subject: Draft Telecon Summary — RCP and LBB
Creation Date: 12/10/2007 5:40:49 PM
From: Kimberly Green

Created By: KJG1@nrc.gov

Recipients
<IPNonPublicHearingFile@nrc.gov>
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Reply Requested: No
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None

Concealed Subject: No
Security: Standard

LICENSEE: Entergy Nuclear Operations, Inc.

FACILITY: Indian Point Nuclear Generating Unit Nos. 2 and 3

SUBJECT: SUMMARY OF TELEPHONE CONFERENCE CALL HELD ON DECEMBER 4, 2007, BETWEEN THE U.S. NUCLEAR REGULATORY COMMISSION AND ENTERGY NUCLEAR OPERATIONS, INC., CONCERNING DRAFT REQUESTS FOR ADDITIONAL INFORMATION PERTAINING TO THE INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3, LICENSE RENEWAL APPLICATION

The U.S. Nuclear Regulatory Commission (NRC or the staff) and representatives of Entergy Nuclear Operations, Inc. held a telephone conference call on December 4, 2007, to discuss and clarify the staff's draft requests for additional information (D-RAIs) concerning the Indian Point Nuclear Generating Unit Nos. 2 and 3, license renewal application. The telephone conference call was useful in clarifying the intent of the staff's D-RAIs.

Enclosure 1 provides a listing of the participants and Enclosure 2 contains a listing of the D-RAIs discussed with the applicant, including a brief description on the status of the items.

The applicant had an opportunity to comment on this summary.

Kimberly Green, Safety Project Manager
Projects Branch 2
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket Nos. 50-247 and 50-286

Enclosures:
1. List of Participants
2. List of Draft Requests for Additional Information

cc w/encls: See next page

LICENSEE: Entergy Nuclear Operations, Inc.

FACILITY: Indian Point Nuclear Generating Unit Nos. 2 and 3

SUBJECT: SUMMARY OF TELEPHONE CONFERENCE CALL HELD ON DECEMBER 4, 2007, BETWEEN THE U.S. NUCLEAR REGULATORY COMMISSION AND ENTERGY NUCLEAR OPERATIONS, INC., CONCERNING DRAFT REQUESTS FOR ADDITIONAL INFORMATION PERTAINING TO THE INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3, LICENSE RENEWAL APPLICATION

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Kimberly Green, Safety Project Manager
Projects Branch 2
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Docket Nos. 50-247 and 50-286

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cc w/encls: See next page

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ADAMS Accession No.:

OFFICE	LA:DLR	PM:RPB2:DLR	OGC	BC:RPB2:DLR
NAME		KGreen	STurk	RFranovich
DATE	12/ /07	12/ /07	12/ /07	12/ /07

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TELEPHONE CONFERENCE CALL
INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3
LICENSE RENEWAL APPLICATION

LIST OF PARTICIPANTS
DECEMBER 4, 2007

PARTICIPANTS	AFFILIATIONS
Kim Green	U.S. Nuclear Regulatory Commission (NRC)
John Tsao	NRC
Mike Stroud	Entergy Nuclear Operations, Inc. (Entergy)
Alan Cox	Entergy
Stan Batch	Entergy
Nelson Azevedo	Entergy

DRAFT REQUESTS FOR ADDITIONAL INFORMATION
INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3
LICENSE RENEWAL APPLICATION

DECEMBER 4, 2007

The U.S. Nuclear Regulatory Commission (NRC or the staff) and representatives of Entergy Nuclear Operations, Inc. held a telephone conference call on December 15, 2007, to discuss and clarify the following draft requests for additional information (D-RAIs) concerning the Indian Point Nuclear Generating Unit Nos. 2 and 3 license renewal application (LRA).

Reactor Coolant Pump Flywheel Analysis

D-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.

Discussion: The applicant indicated that the question is clear. This D-RAI will be sent as a formal RAI.

D-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.

Discussion: The applicant indicated that the question is clear. This D-RAI will be sent as a formal RAI.

D-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.

Discussion: The applicant indicated that the question is clear. This D-RAI will be sent as a formal RAI.

<u>D-RAI 4.7.1-4</u>

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.

<u>Discussion</u>: The applicant indicated that the question is clear. This D-RAI will be sent as a formal RAI.

<u>D-RAI 4.7.1-5</u>

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 TLAA 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.

<u>Discussion</u>: The applicant indicated that the question is clear. This D-RAI will be sent as a formal RAI.

Leak Before Break

<u>D-RAI 4.7.2-1</u>

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.

<u>Discussion</u>: The applicant indicated that the question is clear. This D-RAI will be sent as a formal RAI.

<u>D-RAI 4.7.2-2</u>

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.

Discussion: The applicant indicated that the question is clear. This D-RAI will be sent as a formal RAI.

D-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. (b) 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 the CASS components in the LBB piping systems.

Discussion: The applicant stated that only the reactor coolant system has been approved for LBB, and that the list of CASS components within the reactor coolant system that are subject to an AMR are in LRA Tables 3.1.2-3-IP2 and 3.1.2-3-IP3. After further review, the staff determined that the RAI should be sent since the list of "piping systems" approved for LBB is not given in the LRA.

The applicant also asked for clarification on the last sentence regarding which AMP will be used to monitor the CASS components. The staff clarified that the sentence should read, "…to monitor the aging effects of CASS components…" Therefore, this question will be reworded as stated above.

This D-RAI will be sent as a formal RAI.

D-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.

Discussion: The applicant indicated that the question is clear. This D-RAI will be sent as a formal RAI.

D-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 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.

Discussion: The applicant indicated that the question is clear. This D-RAI will be sent as a formal RAI.

D-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.

Discussion: The applicant indicated that the question is clear. This D-RAI will be sent as a formal RAI.

D-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).

Discussion: The applicant indicated that the question is clear. This D-RAI will be sent as a formal RAI.

D-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. (c) Discuss plans to monitor and mitigate primary water stress corrosion cracking of alloy 600/82/182 weld material in the LBB piping in the future.

Discussion: The applicant stated that LRA Tables 3.1.2-3-IP2 and 3.1.2-3-IP3 list component types whose material is nickel alloy. After further review, the staff determined that the RAI should be sent since the list of component types does not list specifically list alloy 82/182 .

The applicant stated that for part c of the question, the Nickel Alloy AMP manages the effects of primary water stress corrosion cracking, and that the applicant follows industry guidelines (i.e., MRP-126) for nickel alloy components. After further review, the staff determined that commitment in the LRA for alloy 82/182 (i.e., in the Nickel Alloy AMP) addresses the staff's concern. Therefore, this portion of the question will not be sent as a formal RAI

The remainder of this D-RAI will be sent as a formal RAI.

D-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.

Discussion: The applicant indicated that the question is clear. This D-RAI will be sent as a formal RAI.

D-RAI 4.7.2-10

The NRC has approved two power uprate applications (measurement uncertainty and stretch power uprates) for Unit 2 (ADAMS Accession numbers ML031420375 and ML023290636) and Unit 3 (ADAMS Accession numbers 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 Standard Review Plan 3.6.3. This question covers all piping systems that have been approved for LBB for both units.

Discussion: The applicant stated that their LBB methodology was approved prior to the finalization of Standard Review Plan 3.6.3. Therefore, this question will be reworded as follows:

> The NRC has approved two power uprate applications (measurement uncertainty and stretch power uprates) for Unit 2 (ADAMS Accession numbers ML031420375 and ML023290636) and Unit 3 (ADAMS Accession numbers 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.

The revised question will be sent as a formal RAI.

D-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) safety margins specified in SRP 3.6.3, (2) piping loads, (3) effects of power uprates, (4) leakage calculations as part of LBB analyses, (5) thermal stratification of certain piping systems, (6) crack stability, and (7) capability of the reactor coolant leakage detection system which is a part of overall LBB technology.

Discussion: The applicant stated that portions of the D-RAI are covered by previous questions. Upon further review, the staff agrees and the question will be reworded as follows:

> 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.

The revised question will be sent as a formal RAI.