December 30, 2008

Vice President, Operations
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Buchanan, NY 10511-0249

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3, LICENSE RENEWAL APPLICATION – MISCELLANEOUS ITEMS

Dear Sir or Madam:

By letter dated April 23, 2007, as supplemented by letters dated May 3, 2007, and June 21, 2007, Entergy Nuclear Operations, Inc., submitted an application pursuant to Title 10 of the *Code of Federal Regulations Part 54*, to renew the operating licenses for Indian Point Nuclear Generating Unit Nos. 2 and 3, for review by the U.S. Nuclear Regulatory Commission (NRC or the staff). The staff is reviewing the information contained in the license renewal application and has identified, in the enclosure, areas where additional information is needed to complete the review. Further requests for additional information may be issued in the future.

Items in the enclosure were discussed with Mr. Robert Walpole, and a mutually agreeable date for the response is January 30, 2009. If you have any questions, please contact me at 301-415-1627, or via e-mail Kimberly.green@nrc.gov.

Sincerely,

/RA/

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

Docket Nos. 50-247 and 50-286

Enclosure:
As stated

cc w/encl: See next page

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ADAMS Accession No.: ML083640270

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INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3
LICENSE RENEWAL APPLICATION
REQUEST FOR ADDITIONAL INFORMATION (RAI)

RAI-2.3A.4.2-2 (Unit 2)

Indian Point Nuclear Generating Unit No. 2 (IP2) UFSAR Section 14.1.10, "Excessive Heat Removal Due To Feedwater System Malfunctions," explains in the case of excessive feedwater (FW) flow resulting from an accidental full opening of one FW control valve, the resulting transient is similar to, but less severe than the hypothetical steamline break transient described in Section 14.2.5. Therefore, the failure is bounded by the analysis presented in Section 14.2.5. UFSAR Section 14.2.5.6, Containment Peak Pressure for a Postulated Steam Line Break, specifically indicates that for IP2 the applicant takes credit for the main FW stop valves, BFD-5's, closing within 120 seconds, in the event of the failure of the main FW control valve.

In response to a telephone conference call the staff had with the applicant on March 7, 2008 (ADAMS Accession number ML080840568), the applicant revised its response to RAI 2.3A.4.2-1. In its amended response, dated March 24, 2008, the applicant reiterated that the FW valves credited for FW isolation are safety-related. This response did not specifically include FW isolation valves, BFD-5's, by name, and they are not included within the boundary flags for system scope, nor highlighted on license renewal drawings for having an intended function in accordance with 10 CFR 54.4(a)(1).

The staff requests the applicant to: a) justify the exclusion of these isolation valves, BFD-5's, from the scope of license renewal in accordance with 10 CFR 54.4(a)(1), and b) verify whether a similar issue to Unit 3 exists for Unit 2 as stated in the following RAI (RAI 2.3B.4.2-2) for Unit 3, which credits closure of BFD-5's and BFD-90's valves in the event of a main steam line break on a high steam flow safety injection logic, which would require not only the inclusion of BFD-5's but also BFD-90's within scope per 10 CFR 54.4(a)(1).

RAI 2.3B.4.2-2 (Unit 3)

Similar to the issue stated in the above RAI for Unit 2 (RAI-2.3A.4.2-2), a comparable issue applies for FW isolation valves for Unit 3. However, the Unit 3 analysis differs slightly from Unit 2 to include the FW isolation valves (BFD-90's) associated with the FW regulating bypass valves. UFSAR Section 14.2.5, "Rupture of a Steam Pipe," states that in the event of a main steam line break incident, the motor-operated valves (MOVs) associated with each of the FW regulating valves (FRVs) will also close. The mechanical stroke time of 120 seconds to close these associated MOVs has been analyzed and is acceptable. In addition, license renewal drawing 9321-20193 shows a "HIGH STEAM FLOW SI LOGIC" signal goes to these motor-operated isolation valves. UFSAR Section 14.2.5.1 states that redundant isolation of the main FW lines is necessary, because sustained high FW flow would cause additional cooldown. Therefore, in addition to the normal control action which will close the main FW valves, any safety injection signal will rapidly close all FW control valves (including the motor-operated block valves and low-flow bypass valves), trip the main FW pumps, and close the FW pump discharge valves.

The motor-operated block valves shown on the license renewal drawings are BFD-5's and BFD-90's for the main FRVs, and the low flow bypass regulating valves, respectively. The FW isolation valves, BFD-5's and BFD-90's, are not shown to be included within the system boundary flags for system scope, nor highlighted on LRA drawings for having an intended function in accordance with 10 CFR 54.4(a)(1). In its amended response, dated March 24, 2008, the applicant did not specifically include these FW isolation valves, BFD-5's and BFD-90's, by name.

The staff requests the applicant to justify the exclusion of these isolation valves, BFD-5's and BFD-90's, from scope of license renewal in accordance with 10 CFR 54.4(a)(1).

RAI-2.3A.4.5-2 (Unit 2)

In LRA Section 2.3.4.5 the applicant describes systems not described elsewhere in the application credited for mitigating the consequences of a Unit 2 fire event in the auxiliary feedwater (AFW) room. Each system listed has the following intended function: to support safe shutdown in the event of a fire in the auxiliary feed pump room (10 CFR 50.48) function in accordance 10 CFR 54.4(a)(3). The applicant states "no LRA drawings are provided based on the intended function of supporting safe shutdown in the event of a fire in the auxiliary feed pump room." However, the applicant states in LRA Section 2.2 that "[c]omponents subject to aging management review are highlighted on license renewal drawings, with the exception of components in scope for 10 CFR 54.4(a)(2)." Since the structures and components that support mitigating the consequences of a fire event are in scope in accordance with 10 CFR 54.4(a)(3) and subject to an AMR in accordance with 10 CFR 54.21(a)(1), then the components should have been highlighted on license renewal drawings. However, the applicant did not highlight the components or flowpaths needed to support this event. In addition, the applicant did not, in accordance with 10 CFR 54.21(a)(1), identify and list the structures and components that are subject to an AMR. Therefore, based upon the information provided in the LRA, the staff was not able to verify which components are included in scope to perform the stated function and are subject to an AMR.

For each system identified in LRA Section 2.3.4.5, the staff requests the applicant to a) identify the system support function for the AFW pump room fire event, b) clearly identify the portions of the systems' flow paths that support these functions that are subject to an AMR, and c) identify the portions of these flow paths that are not already in scope for 10 CFR 54.4(a)(1) or (a)(2).

RAI 3.4.2-1

In LRA Section 3.4.2, the applicant summarizes its AMR results for the IP2 auxiliary feedwater pump room fire event. In the LRA, the applicant states that:

> The components in the systems required to supply feedwater to the steam generators during the short duration of the fire event are in service at the time the event occurs or their availability is checked daily. Therefore, integrity of the systems and components required to perform post-fire intended functions for at least one hour is continuously confirmed by normal plant operation. During the event these systems and components must continue to perform their intended functions to supply feedwater to the steam generators for a minimum of one hour. Significant degradation that could threaten the performance of the intended functions will be apparent in the period immediately preceding the event and corrective action will be required to sustain continued operation. For the minimal one hour period that these systems would be required to provide make up to the steam generators, further aging degradation that would not have been apparent

prior to the event is negligible. Therefore, no aging effects are identified, and no Summary of Aging Management Review table is provided.

Section 54.21(a)(1) of 10 CFR requires that for those systems, structures, and components within the scope of license renewal, as delineated in § 54.4, applicants must identify and list those structures and components subject to an aging management review. Additionally, Section 54.21(a)(3), requires that for each structure and component identified in paragraph 54.21(a)(1), applicants must demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation. Based on the information contained in the LRA, Entergy has not demonstrated that the effects of aging for passive, long-lived components within the systems credited for providing flow to the steam generators during the fire event will be adequately managed.

For those systems, or portions thereof, that are identified in response to RAI 2.3.4.5-2, part c, the staff requests that the applicant provide a list of passive, long-lived component types, material, environment, and aging effect combinations, and the programs that will be used to manage the aging effects.

RAI 3.0.3.3.3-1

In the LRA, the applicant stated that the existing program will be enhanced to include the minimum wall thickness for the new heat exchangers added to the scope of the program, and to specify that if visual examination is performed, the acceptance criterion is "no unacceptable signs of degradation." These acceptance criteria for visual examination are not clear and appear to be subjective. Therefore, the staff requests that Entergy clarify, in quantitative terms, what acceptance criteria are used for the visual examination of the heat exchanger tubes.

RAI 3.0.3.3.4-1

LRA Table B-2 identifies AMP B.1.18, Inservice Inspection Program, as a plant-specific condition monitoring program for the applications. The staff notes that Entergy has committed to enhance the "detection of aging effects" program element of the Inservice Inspection Program to revise the AMP to provide for periodic visual inspections of lubrite sliding supports used in the steam generator supports and reactor coolant pump (RCP) supports in order to confirm the absence of aging effects. Please specify (1) which aging effects and parameters will be monitored for by the visual examinations, (2) the types of visual examinations (e.g., VT-1, EVT-1, VT-2, or VT-3), (3) inspection frequency and sample size for the visual examination method that will be used to monitor for aging, (4) the acceptance criteria that will be used to evaluate the examination results, and (5) the corrective action or actions that will be implemented if the inspection results do not conform to the acceptance standard(s) for these components.

RAI 3.0.3.3.4-2

The staff notes that the "corrective actions" program element for AMP B.1.18, Inservice Inspection Program, credits only the corrective actions in the ASME Code Section XI, Articles IWA-4000 and IWA-7000 as the corrective action criteria for the program. The ASME Code Section XI editions of record for IP units are the 2001 Edition of the ASME Code Section XI inclusive of the 2003 Addenda for IP2 and the 1989 Edition of the ASME Code Section XI, with no addenda for IP3. The staff noted that Entergy did not credit component-specific corrective action criteria in ASME Section XI, Article IWB-4000/7000 for Class 1 components, Article IWC-4000/7000 for Class 2 components, Article IWD-4000/7000 Class 3 components, or Article IWF-

4000/7000 for ASME Code Class component supports as being within the scope of the "corrective action" program element for this AMP. Clarify whether the content of the "corrective actions" program element was intended to mean that Entergy will implement the corrective action provisions in the ASME Code Section XI, Subsections IWA, IWB, IWC, IWD, and IWF that are applicable to the component Code Class in the applicable ASME Code Section XI edition of record.

RAI 3.0.3.3.7-1

LRA Table B-2 identifies AMP B.1.29, Periodic Surveillance and Preventative Maintenance Program, as an existing, plant-specific condition monitoring program for the LRA. NUREG-1800, Revision 1, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants" (SRP-LR), Appendix A, Section A.1.2.2 states that aging management programs for license renewal applications (LRAs) are to be defined in terms of the 10 program elements that are provided in Table A-1 of the same appendix.

The staff notes that the applicant plans to enhance the "scope of program," "parameters monitored," "detection of aging effects," and "acceptance criteria" program elements for this AMP to develop the program activities in the future (Commitment 21). Technical Specification (TS) 5.5.2 for IP2 and TS 5.5.2 for IP3 provide TS preventative maintenance and surveillance requirements.

1. The staff requests that Entergy supplement the "scope of program" program element to clearly identify the systems and components that are within the scope of this AMP.

2. The "detection of aging effects" program element did not clearly establish which type of non-visual NDE method (volumetric examination by UT, RT, or ET or surface examination by MT or PT) or visual method (i.e., EVT-1, VT-1, VT-2, or VT-3) would be credited for each of the aging effects that the program monitors for. In addition, it is not clear whether flexing of the elastomeric components in the circulating water system and emergency diesel generator exhaust system would be coupled to the visual examinations of the components, as it was proposed for other elastomeric components within the scope of the AMP.

   The staff requests a more definitive discussion on the specific inspection methods (UT, RT, ET, EVT-1, VT-1, VT-2, or VT-3) to detect the parameters that are associated with the aging effects the AMP monitors for. The staff also requests that Entergy clarify whether physical manipulation (i.e., flexing) of the elastomeric components in the circulating water system and emergency diesel generator exhaust system are credited under the AMP.

3. The "monitoring and trending" program element for the AMP did not provide any discussion on how the data from the inspections performed under the "detection of aging effects" program element would be collected, quantified, or evaluated against applicable acceptance criteria, and used to make predictions related to degradation growth or to schedule re-inspections of the components. The staff requests a more definitive discussion on how the inspection results or physical manipulation (flexing) results (for elastomers) will be collected and quantified, or evaluated against applicable acceptance criteria, and used to make predictions related to degradation growth or to schedule re-inspections or repairs of the components.

4. The "acceptance criteria" program element for the AMP only states that "acceptance criteria are defined in specific inspection and testing procedures and that these acceptance criteria include appropriate temperature, no significant wear, corrosion, cracking, change in material properties (for elastomers), and significant fouling based on applicable intended functions established by plant design basis. The program element discuss does not clearly identify the quantitative or qualitative criteria that will be used to assess the inspection results or reference the regulatory-based documents or standards that contain these criteria. The staff requests a clarification on the specific quantitative or qualitative acceptance criteria that will be used to evaluate the results of the specific inspection methods or physical manipulation methods (for elastomers) that are implemented under this AMP.

5. With respect to operating experience and the discussion on the NaOH tanks and recirculation pumps, clarify whether the term "no deficiencies" means that no evidence of loss of material (by corrosion, erosion, wear, or other mechanisms) was detected in the components or whether the meaning is that some amount of age-related degradation had been detected in the components and the amount of cracking or loss of material (wall loss) was found to be acceptable when compared to appropriate acceptance standards.

   With respect to the discussion on the IP2 and IP3 emergency diesel generators, the security diesel generator, and the IP3 Appendix R fire protection diesel generator, clarify whether the statements "no unacceptable loss of material" and "no significant corrosion or wear" mean that no loss of material (by corrosion, erosion, wear, or other mechanisms) was detected in the components or that some loss of material was detected in the components and the amount of loss of material was found to be acceptable when compared to appropriate acceptance standards. If some degradation was detected in these components and the amount of degradation was in conformance with the applicable acceptance criteria, clarify whether the scope of the AMP included appropriate reinspections of the components in order to account for potential degradation of the components.

RAI 3.1.2.2.7.2-1

Part A - The staff notes that the Inservice Inspection Program (as given in LRA Section B.1.18) is credited, in part, as an acceptable plant-specific condition monitoring program for the management of cracking in ASME Code Class 1 components, including ASME Code Class 1 cast austenitic stainless steel (CASS) components. However, the staff also notes that the inspections credited under this program might be either ultrasonic test (UT) examinations or enhanced VT-1 visual examinations. If UT examinations are credited for aging management of reduction of fracture toughness in the CASS components, clarify how the UT technique selected for the examination will be capable of differentiating between UT signals that derive from flaws or cracks in the CASS materials from those that derived from UT background noise signals as a result of the complexity of the CASS microstructure or component geometry.

Part B - The staff determined that Entergy credits its Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program in part to manage cracking in the non-ASME Code Class CASS pressurizer spray head. The staff also notes that the applicant's program includes a flaw evaluation methodology for CASS components that are susceptible to thermal aging embrittlement, and that alternatively, this AMP may credit UT or enhanced VT-1 visual examinations as an indirect basis for managing loss/reduction of fracture toughness as a result of thermal aging. However, the staff notes that the applicant's program is not specifically credited for the management of cracking in CASS components. The staff requests that Entergy justify its basis for crediting AMP B.1.37, Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program, to manage and detect cracking in the CASS pressurizer spray heads at IP2 and IP3, particularly when GALL AMP XI.M12 only credits this type of program for management of reduction of fracture toughness in CASS components and when the program may not actually be performing inspections of these components (i.e., the program has the option only to do the flaw tolerance evaluation without implementation of either a UT or EVT-1 examination).

RAI 3.1.2-1 Nickel Alloy AMRs

Part A - Clarify whether the following components at IP2 or IP3 are fabricated from Alloy 600 base metal materials or welded with Alloy 182 or Alloy 82 filler metal materials: (1) control rod drive (CRD) housing-CRD nozzle welds, (2) upper reactor vessel closure head (RVCH) head vent nozzle-to-RVCH welds, and (3) CRD housing penetration core exit thermocouple nozzle assembly (CETNA™) components.

Part B - The staff notes that in the applicant's response to Audit Question 208, dated December 18, 2007, the applicant stated that the LRA Tables 3.1.2-1-IP2 through 3.1.2-4-IP2 and LRA Tables 3.1.2-1-IP3 through 3.1.2-4-IP3 include numerous AMR items for nickel-alloy components. The applicant stated that these AMR items are compared to GALL Report items IV.A2-18 and IV.A2-19, which correspond to LRA table entries 3.1.1-31 and 3.1.1-65. The applicant stated that the AMR in LRA AMR 3.1.1-69 is only for management of cracking in the RV inlet and outlet nozzle safe-ends and the RV bottom head drain safe-ends. With respect to the AMRs on cracking of nickel alloy bottom mounted instrumentation (BMI) nozzle components, the staff notes that the response to Question 208 stated that the RV bottom head safe-ends at IP2 and IP3 are those for the RV bottom head drains. Yet the staff notes that LRA Tables 3.1.2-1-IP2 and 3.1.2-1-IP3 do not include any AMR entries for RV bottom head drains. Since your response to Audit Question 208 implies that the RV includes passive, long-lived bottom head drains, provide your basis on whether LRA Tables 3.1.2-1-IP2 and 3.1.2-1-IP3 need to be amended to include new AMRs for RV bottom head drains and their associated drain-to-bottom head welds, and if so clarify whether the bottom head drains are fabricated from Alloy 600 base metal materials or are weld to the bottom RV heads using Alloy 82 or 182 nickel alloy filler metal materials.

Part C - AMRs of LRA Tables 3.1.2-4-IP2 and 3.1.2-4-IP3, which pertain to the management of cracking in the steam generator (SG) primary nozzle closure rings, credit only the Water Chemistry Control Program to manage cracking of the components. GALL Report Table IV.D1, line item D1-1 for these components recommends, in part, that the Inservice Inspection Program be credited for aging management of this effect in addition to Water Chemistry Control Program – Primary and Secondary. Given the information requested in Part A of this RAI, provide a basis for why the AMRs on cracking of the nickel alloy SG primary nozzle closure rings were aligned to GALL AMR Table VI.D1, line item D1-6, and why the Inservice Inspection Program is not also credited.

Letter to Entergy Nuclear Operations, Inc. from K. Green, dated December 30, 2008

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