July 24, 2006

Mr. Michael Kansler
President
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 - ISSUANCE OF AMENDMENT RE: LARGE BREAK LOSS-OF-COOLANT ACCIDENT (LBLOCA) ANALYSIS METHODOLOGY (TAC NO. MC8427)

Dear Mr. Kansler:

The Commission has issued the enclosed Amendment No. 248 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated September 26, 2005, as supplemented by letter dated April 11, 2006.

The amendment approves the use of Westinghouse Best Estimate Large Break Loss-of-Coolant Accident (BE-LBLOCA) methodology described in WCAP-16009-P-A, “Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM). It also revises Reference 6 of TS 5.6.5.b, Core Operating Limits Report, to reflect the change in the approved methodology.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

/RA/

John P. Boska, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-247

Enclosures:
1. Amendment No. 248 to DPR-26
2. Safety Evaluation

cc w/encls: See next page

Indian Point Nuclear Generating Unit No. 2

cc:

Mr. Gary J. Taylor
Chief Executive Officer
Entergy Operations, Inc.
1340 Echelon Parkway
Jackson, MS 39213

Mr. John T. Herron
Senior Vice President and
Chief Operating Officer
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. Fred R. Dacimo
Site Vice President
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
295 Broadway, Suite 1
P.O. Box 249
Buchanan, NY 10511-0249

Mr. Paul Rubin
General Manager, Plant Operations
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
295 Broadway, Suite 2
P.O. Box 249
Buchanan, NY 10511-0249

Mr. Oscar Limpias
Vice President Engineering
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. Christopher Schwartz
Vice President, Operations Support
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. John F. McCann
Director, Licensing
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Ms. Charlene D. Faison
Manager, Licensing
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. Michael J. Columb
Director of Oversight
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. James Comiotes
Director, Nuclear Safety Assurance
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
295 Broadway, Suite 1
P.O. Box 249
Buchanan, NY 10511-0249

Mr. Patric Conroy
Manager, Licensing
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
295 Broadway, Suite 1
P. O. Box 249
Buchanan, NY 10511-0249

Assistant General Counsel
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. Peter R. Smith, President
New York State Energy, Research, and
Development Authority
17 Columbia Circle
Albany, NY 12203-6399

Mr. Paul Eddy
New York State Department
of Public Service
3 Empire State Plaza
Albany, NY 12223

cc:

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Senior Resident Inspector's Office
Indian Point 2
U. S. Nuclear Regulatory Commission
P.O. Box 59
Buchanan, NY 10511

Mr. Charles Donaldson, Esquire
Assistant Attorney General
New York Department of Law
120 Broadway
New York, NY 10271

Mayor, Village of Buchanan
236 Tate Avenue
Buchanan, NY 10511

Mr. Raymond L. Albanese
Four County Coordinator
200 Bradhurst Avenue
Unit 4 Westchester County
Hawthorne, NY 10532

Mr. William DiProfio
PWR SRC Consultant
139 Depot Road
East Kingston, NH 03827

Mr. Garry Randolph
PWR SRC Consultant
1750 Ben Franklin Drive, 7E
Sarasota, FL 34236

Mr. William T. Russell
PWR SRC Consultant
400 Plantation Lane
Stevensville, MD 21666-3232

Mr. Jim Riccio
Greenpeace
702 H Street, NW
Suite 300
Washington, DC 20001

Mr. Phillip Musegaas
Riverkeeper, Inc.
828 South Broadway
Tarrytown, NY 10591

Mr. Mark Jacobs
IPSEC
46 Highland Drive
Garrison, NY 10524

July 24, 2006

Mr. Michael R. Kansler
President
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 - ISSUANCE OF AMENDMENT RE: LARGE BREAK LOSS-OF-COOLANT ACCIDENT (LBLOCA) ANALYSIS METHODOLOGY (TAC NO. MC8427)

Dear Mr. Kansler:

The Commission has issued the enclosed Amendment No. 248 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated September 26, 2005, as supplemented by letter dated April 11, 2006.

The amendment approves the use of Westinghouse Best Estimate Large Break Loss-of-Coolant Accident (BE-LBLOCA) methodology described in WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM). It also revises Reference 6 of TS 5.6.5.b, Core Operating Limits Report, to reflect the change in the approved methodology.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

/RA/

John P. Boska, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-247

Enclosures:
1. Amendment No. 248 to DPR-26
2. Safety Evaluation

cc w/encls: See next page

Accession Number: ML061710291/ *see safety evaluation dated May 12, 2006

OFFICE	LPL1-1/PM	LPL1-1/LA	SPWB/BC	OGC	LPL1-1/BC
NAME	JBoska	SLittle	JNakoski*	MYoung	RLaufer
DATE	6/28/06	6/29/06	5/12/06	7/19/06	7/21/06

Official Record Copy

DATED: July 24, 2006

AMENDMENT NO. 248 TO FACILITY OPERATING LICENSE NO. DPR-26 INDIAN POINT UNIT 2

PUBLIC
LPL1-1 R/F
RidsNrrDorlLpla
RidsNrrLASLittle
RidsNrrPMJBoska
RidsOGCMailCenter
GHill (2)
RidsNrrDirsItsb
RidsAcrsAcnwMailCenter
ECobey, RI
FOrr

cc: Plant Mailing list

ENTERGY NUCLEAR INDIAN POINT 2, LLC

ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-247

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 248
License No. DPR-26

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Entergy Nuclear Operations, Inc. (the licensee) dated September 26, 2005, as supplemented on April 11, 2006, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (I) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-26 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 248, are hereby incorporated in the license. ENO shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Richard J. Laufer, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the License and
Technical Specifications

Date of Issuance: July 24, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 248

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Replace the following page of the License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove Page	Insert Page
3	3

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove Page	Insert Page
5.6-3	5.6-3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 248 TO FACILITY OPERATING LICENSE NO. DPR-26

ENTERGY NUCLEAR OPERATIONS, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

1.0 INTRODUCTION

By letter dated September 26, 2005, Agencywide Documents Access and Management System Accession No. ML052770536 (Reference 1), Entergy Nuclear Operations, Inc. (Entergy or the licensee), submitted a request for changes to the Indian Point Nuclear Generating Unit 2 (IP2) Technical Specifications (TSs). The proposed change would revise the methodology for large-break loss-of-coolant accident (LBLOCA) analyses for IP2. The licensee provided additional information by letter dated April 11, 2006, Accession No. ML061080451 (Reference 2). This supplement provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration.

The licensee requested approval to apply the NRC-approved Westinghouse Best Estimate Loss-of-Coolant Accident (BE-LBLOCA) methodology described in WCAP-16009-P-A, “Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM),” January 2005 (Reference 3), at IP2.

The NRC staff reviewed the licensee’s evaluation of the emergency core cooling system (ECCS) performance analyses for IP2 done in accordance with the ASTRUM methodology (Reference 3), operating at 102 percent of a licensed core power of 3216 megawatts thermal (MWt). For IP2, the LOCA analyses were conducted assuming the plant uses core loads of Westinghouse 15 x 15 Upgraded Fuel Assemblies (UFAs) and 422 Vantage^{+} fuel assemblies (422V^{+}).

IP2 is a four-loop, pressurized-water reactor (PWR) of the Westinghouse Electric design, enclosed within a large, dry containment. The ECCS consists of the residual heat removal system (RHR), used for low pressure injection (LPI) flow, high-head safety injection (HHSI) flow delivered to the cold legs, four accumulators with a cover gas pressure of 600 psia also injecting into the cold legs, and recirculation pumps inside the containment building. The shut-off head of the RHR low pressure injection pumps is about 140 psia.

The analyzed core power is 102% of the core power of 3216 MWt: 3280.3 MWt.

2.0 REGULATORY EVALUATION

The regulatory requirements and guidance which the NRC staff considered in its review of this requested action are as follows:

1. Section 50.46 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.46), "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," requires, in part, that ECCS cooling performance be calculated in accordance with an acceptable evaluation model and be calculated for a number of postulated LOCAs of different sizes, locations, and other properties. It also requires that "... uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (b) of this section, there is a high level of probability that the criteria would not be exceeded."

2. Section 50.46(b) of 10 CFR 50 also states detailed acceptance criteria for LOCA evaluations. These are as follows:

(1) The calculated maximum fuel element cladding temperature shall not exceed 2200 EF.
(2) The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.
(3) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.
(4) Calculated changes in core geometry shall be such that the core remains amenable to cooling.
(5) After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

Westinghouse submitted the final version of an evaluation model for the LBLOCA analysis to the NRC as topical report WCAP-16009-P-A (Reference 3). The NRC staff approved this methodology in a safety evaluation dated November 5, 2004, which is included in the first section of the final proprietary and non-proprietary versions of WCAP-16009. The ADAMS accession numbers are listed in Reference 3. The NRC safety evaluation states that the topical report is acceptable as a reference in license applications to the extent specified and under the limitations delineated in the topical report and in the NRC safety evaluation. The NRC staff reviews each licensee's application which applies this topical report to ensure that these conditions are met.

3.0 TECHNICAL EVALUATION

The LBLOCA analyses were performed for IP2 to demonstrate that the system design would provide sufficient ECCS flow to transfer the heat from the reactor core following a LOCA at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling would be prevented, and (2) the clad metal-water reaction would be limited so as not to compromise cladding ductility and not result in excessive hydrogen generation. The NRC staff reviewed the analyses to assure that the analyses reflected suitable redundancy in components and features; and suitable interconnections, leak detection, isolation, and containment capabilities were available such that the safety functions could be accomplished, assuming a single failure, for LOCAs considering the availability of onsite power (assuming offsite electric power is not available), or offsite electric power (assuming onsite electric power is not available).

In its April 11, 2006, submittal (Reference 2), the licensee stated, “Both Entergy Nuclear Operations, Inc., and Westinghouse Electric Company (analysis vendor) have ongoing processes such that the values and ranges of the BE-LBLOCA analysis inputs for peak cladding temperature and oxidation-sensitive parameters bound the values and ranges of the as-operated Indian Point 2 parameters, in accordance with the approved methodology (WCAP-16009-P-A).” The NRC staff finds that this statement, along with the generic NRC acceptance of the ASTRUM methodology (Reference 3), provides assurance that ASTRUM and its LBLOCA analyses apply to IP2 at the analyzed power of 3280.3 MWt.

In its submittal, the licensee provided the results for the IP2 BE-LBLOCA analyses at 3280.3 MWt (102 percent of the current licensed power of 3216 MWt) performed in accordance with the ASTRUM methodology. The licensee’s results for the calculated peak cladding temperatures (PCTs), the maximum cladding oxidation (local), and the maximum core-wide cladding oxidation are provided in the following table along with the acceptance criteria of 10 CFR 50.46(b).

Table 1 - LARGE BREAK LOCA ANALYSES RESULTS

Parameter	ASTRUM 422 V^+ “present” Results	ASTRUM 15 X 15 Upgraded Results	10 CFR 50.46 Limits
Limiting Break Size/Location	DEG/PD*	DEG/PD*	N/A
Cladding Material	Zirlo	Zirlo	(Cylindrical) Zircaloy or Zirlo
Peak Clad Temperature	1962 °F	1814 °F	2200 °F (10 CFR 50.46(b)(1))
Maximum Local Oxidation	2.39 %	2.5 %	17.0% (10 CFR 50.46(b)(2))
Maximum Total Core-Wide Oxidation (All Fuel)	0.35 %	0.30 %	1.0% (10 CFR 50.46(b)(3))

*DEG/PD is a double ended guillotine break at the pump discharge.

In analyses for the IP2 2004 power uprate, the licensee also addressed the concern that present fuel (422 V^+) may have pre-existing oxidation that must be considered in its LOCA analyses. In a previous letter dated August 12, 2004 (ADAMS accession number ML042380253), the licensee indicated that it considered that the zircaloy clad fuel has both pre-existing oxidation and oxidation resulting from the LOCA (pre- and post-LOCA oxidation both on the inside and outside cladding surfaces). This licensee position was approved in an NRC SE dated October 27, 2004 (Reference 4). The licensee had also noted that the fuel with the highest LOCA oxidation will likely not be the same fuel that has the highest pre-LOCA oxidation.

The licensee indicated that when the calculated pre-LOCA oxidation was factored into the licensee's BE-LBLOCA analyses for the zircaloy clad fuel, consistent with the previous Westinghouse methodology for IP2, that even during a fuel pin's final cycle in the core the sum of the calculated pre- and post-LOCA oxidation was sufficiently small that the total local oxidation remained less than the acceptance criterion of 10 CFR 50.46(b)(2) as noted above. The NRC staff finds this appropriately addressed the issue with pre-LOCA oxidation because the computer code (WCOBRA-TRAC) used in the previous methodology is the same code that is used in the ASTRUM methodology.

The concern with core-wide oxidation relates to the amount of hydrogen generated during a LOCA. Because hydrogen that may have been generated pre-LOCA (during normal operation) will be removed from the reactor coolant system throughout the operating cycle, the NRC staff noted that pre-existing oxidation does not contribute to the amount of hydrogen generated post-LOCA and therefore, it does not need to be addressed when determining whether the calculated total core-wide oxidation meets the criterion of 10 CFR 50.46(b)(3).

As discussed previously, the licensee requested Westinghouse to conduct the BE-LBLOCA analyses for IP2 at about 102 percent of the current licensed power level of 3216 MWt using an NRC-approved Westinghouse methodology (ASTRUM) (Reference 3). The NRC staff concluded that the results of these analyses (see Table 1) demonstrated compliance with 10 CFR 50.46(b)(1) through (b)(3) for licensed power levels of up to 3216 MWt. Meeting these criteria provides reasonable assurance that, at the current licensed power level, the IP2 core will remain amenable to cooling as required by 10 CFR 50.46(b)(4). The capability of IP2 to satisfy the long-term cooling requirements of 10 CFR 50.46(b)(5) was addressed in Amendment No. 241 as approved in an NRC SE dated October 27, 2004 (Reference 4).

The NRC staff finds that the discussions in the previous SE (Reference 4) referred to above also apply to the present analyses addressed in this SE. In the previous SE, the NRC staff found that the licensee complied with the requirements of 10 CFR 50.46(b) regarding peak cladding temperature, maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long-term cooling during a LBLOCA.

3.1 LBLOCA Conclusions

Based on its review as discussed above, the NRC staff concluded that the Westinghouse ASTRUM methodology, as described in WCAP-16009-P-A, is acceptable for use for IP2 in demonstrating compliance with the requirements of 10 CFR 50.46(b). The NRC staff's conclusion was based on the licensed core power level of 3216 MWt (plus 2.0 percent measurement uncertainty or 3280.3 MWt).

The NRC staff's review of the acceptability of the ASTRUM methodology for IP2 focused on assuring that the IP2 specific input parameters or bounding values and ranges (where appropriate) were used to conduct the analyses, that the analyses were conducted within the conditions and limitations of the NRC-approved Westinghouse ASTRUM methodology, and that the results satisfied the requirement of 10 CFR 50.46(b) based on a licensed power level of 3216 MWt.

3.2 Slot Breaks at the Top and Side of the Pipe

The NRC staff also requested that the licensee address slot breaks at the top and side of a reactor pump discharge cold leg pipe, which could, under some circumstances, lead to greatly extended periods of core uncovery, resulting in fuel cladding oxidation in excess of the 10 CFR 50.46 (b)(2) limit, and also possibly in excess of the total hydrogen limit of 10 CFR 50.46 (b)(3). In its April 11, 2006 letter (Reference 2), the licensee discussed information that is included in a generic Westinghouse report written to address this issue. In its response to an NRC staff request for additional information (RAI), the licensee stated that the Emergency Operating Procedures (EOPs) at IP2 were based on approved Westinghouse Owners Group (WOG) EOP guidelines and direct timely operator actions that would avoid the conditions for extended core uncovery. In its letter, the licensee indicated that the operator procedures and actions would be effective in LBLOCA scenarios because extended core uncovery would take a significant amount of time to develop. The licensee concluded that the existing provisions continue to apply to the upcoming cycle of operation, because the extended core uncovery issue of concern is fuel-independent.

Based on its review of the information provided by the licensee, and as set forth above, the NRC staff concludes that the licensee's analyses have addressed the issue of slot breaks at the top and side of the cold leg pipe. The use of procedures based on the WOG EOP guidelines addresses this issue for the current IP2 licensing basis, but does not resolve the generic issue of slot breaks at the top and side of the pipe for any vendor methodology.

3.3 TS 5.6.5 Core Operating Limits Report (COLR)

The revisions to TS 5.6.5.b are described below:

Replace reference 5.6.5.b.6 relating to the previous IP2 LBLOCA methodology with:

> 6. WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty (ASTRUM)," M.E. Nissley, et al., January 2005.

The NRC staff has determined that topical report WCAP-16009-P-A is an acceptable methodology to apply to IP2 in that it ensures that the core operating limits are determined such that the applicable limits of the safety analysis, such as ECCS limits and accident analysis limits, are met. Therefore, WCAP-16009-P-A is an appropriate reference for the IP2 LBLOCA analyses.

3.4 Conclusion

In summary, the licensee has performed LBLOCA analyses for IP2 using an NRC-approved Westinghouse methodology. The NRC staff concluded that the licensee's LBLOCA analyses were performed using an NRC-approved Westinghouse methodology that applies to IP2. The licensee's LBLOCA calculations demonstrate the following:

- The calculated LBLOCA values for PCT, oxidation, and core-wide hydrogen generation are less than the limits specified in 10 CFR 50.46(b)(1) through (3), respectively.

- Compliance with 10 CFR 50.46(b)(1) through (3) and (5) assures that the core will remain amenable to cooling as required by 10 CFR 50.46(b)(4).

Therefore, the NRC staff finds the licensee's LOCA analyses acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (70 FR 67747). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Fred Dacimo (Entergy), to USNRC Document Control Desk, Subject: Proposed Change to Indian Point 2 Technical Specifications Regarding Loss of Coolant Accident Analysis Methodology," September 26, 2005 (ADAMS Accession No. ML052770536)

2. Fred Dacimo (Entergy), to USNRC Document Control Desk, Subject: Additional Information Regarding Proposed change to Indian Point 2 Technical Specifications for LBLOCA Analysis Methodology (TAC MC8427), April 11, 2006 (ADAMS Accession Nos. 061080451 and ML061080453)

3. WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," January 31, 2005 (ADAMS Accession Nos. ML050910159/61(NP) and ML050910162/63(P))

4. Patrick D. Milano, NRC, to Michael R. Kansler, Entergy, Subject: Indian Point Nuclear Generating Unit No. 2 - Issuance of Amendment re: 3.26 Percent Power Uprate (TAC No. MC1865) October 27, 2004 (ADAMS Accession Nos. ML042960007 and ML043020362)

Principal Contributor: F. Orr

Date: July 24, 2006