UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 4, 2008

Vice President, Operations
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
Buchanan, NY 10511-0249

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3 - ISSUANCE OF AMENDMENTS RE: PASSIVE FAILURE ANALYSIS (TAC NOS. MD8290 AND MD8291)

Dear Sir or Madam:

The Commission has issued the enclosed Amendment No. 257 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2 (IP2) and Amendment No. 238 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3 (IP3). The amendments consist of changes to the Updated Final Safety Analysis Reports (UFSARs) in response to your application dated March 13, 2008.

The amendments revise the UFSARs by allowing a 24 hour delay after a loss-of-coolant-accident before postulating a passive failure in the emergency core cooling systems for IP2 and IP3 or the component cooling water system for IP2.

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,

John P. Boska

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

Docket Nos. 50-247 and 50-286

Enclosures:
1. Amendment No. 257 to DPR-26
2. Amendment No. 238 to DPR-64
3. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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. 257
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 March 13, 2008, 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 Updated Final Safety Analysis Report as indicated in the safety evaluation attached 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. 257 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 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Mark G. Kowal

Mark G. Kowal, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the License

Date of Issuance: December 4, 2008

ATTACHMENT TO LICENSE AMENDMENT NO. 257

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

instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) ENO pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; Amdt. 42 10-17-78

(5) ENO pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility. Amdt. 220 09-06-01

C. This amended license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

ENO is authorized to operate the facility at steady state reactor core power levels not in excess of 3216 megawatts thermal. Amdt. 241 10-27-04

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 257, are hereby incorporated in the license. ENO shall operate the facility in accordance with the Technical Specifications.

(3) The following conditions relate to the amendment approving the conversion to Improved Standard Technical Specifications:

1. This amendment authorizes the relocation of certain Technical Specification requirements and detailed information to licensee-controlled documents as described in Table R, "Relocated Technical Specifications from the CTS," and Table LA, "Removed Details and Less Restrictive Administrative Changes to the CTS" attached to the NRC staff's Safety Evaluation enclosed with this amendment. The relocation of requirements and detailed information shall be completed on or before the implementation of this amendment.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENTERGY NUCLEAR INDIAN POINT 3, LLC

ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 238
License No. DPR-64

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

   A. The application for amendment by Entergy Nuclear Operations, Inc. (the licensee) dated March 13, 2008, 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 Updated Final Safety Analysis Report as indicated in the safety evaluation attached to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-64 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 238, 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 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Mark G. Kowal

Mark G. Kowal, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the License

Date of Issuance: December 4, 2008

ATTACHMENT TO LICENSE AMENDMENT NO. 238

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

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

(4) ENO pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; Amdt. 203 11/27/00

(5) ENO pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility. Amdt. 203 11/27/00

C. This amended license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

ENO is authorized to operate the facility at steady state reactor core power levels not in excess of 3216 megawatts thermal (100% of rated power).

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 238, are hereby incorporated in the License. ENO shall operate the facility in accordance with the Technical Specifications.

(3) (DELETED) Amdt. 205 2-27-01

(4) (DELETED) Amdt. 205 2-27-01

D. (DELETED) Amdt.46 2-16-83

E. (DELETED) Amdt.37 5-14-81

F. This amended license is also subject to appropriate conditions by the New York State Department of Environmental Conservation in its letter of May 2, 1975, to Consolidated Edison Company of New York, Inc., granting a Section 401 certification under the Federal Water Pollution Control Act Amendments of 1972.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 257 TO FACILITY OPERATING LICENSE NO. DPR-26

AND AMENDMENT NO. 238 TO FACILITY OPERATING LICENSE NO. DPR-64

ENTERGY NUCLEAR OPERATIONS, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3

DOCKET NOS. 50-247 AND 50-286

1.0 INTRODUCTION

By letter dated March 13, 2008, Agencywide Documents Access and Management System (ADAMS) Accession No. ML080800361, Entergy Nuclear Operations, Inc. (Entergy or the licensee) submitted a request for changes to the Indian Point Nuclear Generating Unit Nos. 2 and 3 (IP2 and IP3) Updated Final Safety Analysis Reports (UFSARs) to reflect a revised Emergency Core Cooling System (ECCS) and Component Cooling Water System (CCWS) single passive failure licensing basis for the loss-of-coolant accident (LOCA) recirculation phase. The proposed changes support Entergy's resolution of Nuclear Regulatory Commission (NRC) Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors," by establishing a licensing basis, consistent with approved regulatory positions regarding passive failure, that support meeting the regulatory requirements of GL 2004-02. There are no changes to actual plant equipment approved in this safety evaluation (SE), only changes to assumptions used in the analysis of LOCAs.

The purpose of the licensing basis change is to establish the following licensing basis for passive failures in fluid systems:

1. Revise the IP2 and IP3 LOCA recirculation phase single passive failure licensing basis such that a passive failure is assumed to occur 24 hours or greater following initiation of a LOCA. The current licensing basis assumes that a passive failure in the internal recirculation system at IP2 and IP3 could occur at any time after the start of the LOCA recirculation phase.

2. Revise the IP2 CCWS single passive failure licensing basis such that a passive failure is assumed to occur 24 hours or greater following initiation of a LOCA. The current licensing basis assumes that a passive failure in the IP2 CCWS could occur at any time.

The current licensing basis requires the licensee to show that the LOCA recirculation phase will be successful assuming that 50% of the sump strainer surface area is blocked by debris. This is a typical licensing basis for operating pressurized-water reactors (PWRs), as NRC Regulatory

Guide (RG) 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems," Revision 0, dated June 1974, found this assumption to be acceptable.

Based on further NRC research, GL 2004-02 calls for all PWRs to reanalyze the LOCA recirculation phase using a mechanistic analysis, showing how much debris could be generated during a LOCA and how much could be transported to the sump strainers, and accounting for any chemical effects inside the containment building that could affect the recirculation flow. The demonstration that the recirculation sump and containment sump strainer designs at IP2 and IP3 are capable of accommodating the GL 2004-02 debris loads, including chemical effects, will not be addressed here. The NRC staff understands that the licensee intends to address these issues in its GL 2004-02 submittals and through the resolution of the GL 2004-02 NRC audit open items. The NRC audit open items are contained in the NRC's audit report, dated July 29, 2008, ADAMS Accession No. ML082050433. The GL 2004-02 submittals will become the new licensing basis if they are accepted by the NRC. The change to the current licensing basis addressed in this SE will become part of the assumptions for future GL 2004-02 submittals.

The strategy that Entergy has proposed as part of the new GL 2004-02 analyses requires certain changes to the current licensing bases, as noted above. Generally, this is because IP2 and IP3 have certain unique licensing bases which are more conservative than the licensing bases at many other plants. The typical licensing bases for a Westinghouse PWR during LOCA recirculation is to maintain the safety function in case of an active failure in the short term (less than 24 hours following LOCA initiation), or an active failure or a specified passive failure in the long term (greater then 24 hours following LOCA initiation). The specified passive failure is typically a leakage of fluid through a failed seal at a pump or valve. The assumed seal leakage is generally about 50 gallons per minute (gpm). In contrast, the IP2 and IP3 licensing bases currently assume, during the LOCA recirculation phase, that the safety function be maintained assuming an active failure or a specified passive failure at any time. The specified passive failure includes the rupture of a pipe in the recirculation flow path (subsequent to the rupture of the reactor coolant system), and the pipe leak rate can exceed 3000 gpm.

2.0 REGULATORY EVALUATION

Regulatory and licensing requirements pertaining to the requested amendment concerning the passive failure criteria for the LOCA recirculation phase include the following:

a. The *Code of Federal Regulations*

Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.46 (10 CFR 50.46), "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," delineates the criteria that must be met by the ECCS. IP2 and IP3 use some ECCS evaluation models developed under 10 CFR 50.46(a)(1)(ii), which states "Alternatively, an ECCS evaluation model may be developed in conformance with the required and acceptable features of Appendix K ECCS Evaluation Models." 10 CFR Part 50, Appendix K, "ECCS Evaluation Models," Section I.D.1, "Single Failure Criterion", states "An analysis of possible failure modes of ECCS equipment and of their effects on ECCS performance must be made. In carrying out the accident evaluation the combination of ECCS subsystems assumed to be operative shall be those available after the most damaging single failure of ECCS equipment has taken place."

b. General Design Criteria (GDC) for IP2 and IP3

The following explains the applicability of GDC for IP2 and IP3. The construction permits for IP2 and IP3 were issued by the Atomic Energy Commission (AEC) on October 14, 1966 and August 13, 1969, and the operating licenses were issued on September 28, 1973, and December 12, 1975. The plant GDC are listed in each plant's UFSAR, Chapter 1.3, "General Design Criteria," with more details given in the applicable UFSAR sections. The AEC published the final rule that added 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," in the *Federal Register* (36 FR 3255) on February 20, 1971, with the rule effective on May 21, 1971. In accordance with an NRC staff requirements memorandum from S. J. Chilk to J. M. Taylor, "SECY-92-223 - Resolution of Deviations Identified During the Systematic Evaluation Program," dated September 18, 1992 (ADAMS Accession No. ML003763736), the Commission decided not to apply the Appendix A GDC to plants with construction permits issued prior to May 21, 1971. Therefore, the GDC which constitute the licensing bases for IP2 and IP3 are those in the UFSARs.

As discussed in the UFSARs, the licensees for IP2 and IP3 have made some changes to the facilities over the life of the units that have committed them to some of the GDCs from 10 CFR Part 50, Appendix A. The extent to which the Appendix A GDC have been invoked can be found in specific sections of the UFSARs and in other IP2 and IP3 licensing basis documentation, such as license amendments.

The following GDCs, as found in the IP2 and IP3 UFSARs, pertain to this license amendment:

UFSAR GDC 37 states that engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. Such engineered safety features shall be designed to cope with any size reactor coolant piping break up to and including the equivalent of a circumferential rupture of any pipe in that boundary, assuming unobstructed discharge from both ends.

UFSAR GDC 38 states that all engineered safety features shall be designed to provide such functional reliability and ready testability as is necessary to avoid undue risk to the health and safety of the public.

UFSAR GDC 40 states that adequate protection for those engineered safety features, the failure of which could cause an undue risk to the health and safety of the public, shall be provided against dynamic effects and missiles that might result from plant equipment failures.

UFSAR GDC 41 states that engineered safety features, such as the emergency core cooling system and the containment heat removal system, shall provide sufficient performance capability to accommodate the failure of any single active component without resulting in undue risk to the health and safety of the public.

UFSAR GDC 42 states that engineered safety features shall be designed so that the capability of these features to perform their required function is not impaired by the effects of a LOCA to the extent of causing undue risk to the health and safety of the public.

UFSAR GDC 43 states that protection against any action of the engineered safety features, which would accentuate significantly the adverse after effects of a loss of normal cooling shall be provided.

UFSAR GDC 44 states that an ECCS with the capability for accomplishing adequate emergency core cooling shall be provided. This core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal water reaction to acceptable amounts for all sizes of breaks in the reactor coolant piping up to the equivalent of a double-ended rupture of the largest pipe. The performance of such ECCS shall be evaluated conservatively in each area of uncertainty.

UFSAR GDC 52 states that where an active heat removal system is needed under accident conditions to prevent exceeding containment design pressure, this system shall perform its required function, assuming failure of any single active component.

c. Single Failure Criteria

The NRC's application of single failure criteria for ECCS systems is discussed in SECY-77-439 (Reference 1).

The following definitions are taken from SECY-77-439:

Active Failure in a Fluid System:

An active failure in a fluid system means (1) the failure of a component which relies on mechanical movement for its operation to complete its intended function on demand, or (2) an unintended movement of the component. Examples include the failure of a motor- or air-operated valve to move or to assume its correct position on demand, spurious opening or closing of a motor- or air-operated valve, or the failure of a pump to start or to stop on demand. In some instances such failures can be induced by operator error.

Passive Failure in a Fluid System:

A passive failure in a fluid system means a breach in the fluid pressure boundary or a mechanical failure which adversely affects a flow path. Examples include the failure of a simple check valve to move to its correct position when required, the leakage of fluid from failed components, such as pipes and valves--particularly through a failed seal at a valve or pump--or line blockage. Motor-operated valves which have the source of power locked out are allowed to be treated as passive components. In the study of passive failures, it is current practice to assume fluid leakage owing to gross failure of a pump or valve seal during the long-term cooling mode following a LOCA (24 hours or greater after the event) but not pipe breaks. No other passive failures are required to be assumed because it is judged that compounding of probabilities associated with other types of passive failures, following the pipe break associated with a LOCA, results in probabilities sufficiently small that they can be reasonably discounted without substantially affecting overall systems reliability.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," in Section 6.3, "Emergency Core Cooling System," Revision 3, states that "the ECCS should retain its capability to cool the core in the event of a failure of any single active component during the short term immediately following an accident, or a single active or passive failure during the long-term recirculation cooling phase following an accident." Note that only the single worst failure need be considered over the course of the accident, not two failures.

3.0 TECHNICAL EVALUATION

3.1 Description of the ECCS

The ECCS at IP2 and IP3 are nearly identical. The description below is for IP2. To convert the designators to IP3 replace the 2 prefix with 3 (e.g, the 21 pump on IP2 will be the 31 pump on IP3). The function of the ECCS is to provide core cooling and negative reactivity to ensure that the reactor core is protected after any of the following accidents:

a. LOCA, for coolant leakage greater than the capability of the normal charging system,

b. Rod ejection accident, which due to the rupture of the rod housing pressure boundary is also a type of small-break LOCA,

c. Loss of secondary coolant accident, including uncontrolled steam release or loss of feedwater, and

d. Steam generator tube rupture (SGTR).

There are three phases of ECCS operation: injection, cold leg recirculation, and hot leg recirculation. In the injection phase, water is taken from the refueling water storage tank (RWST) and injected into the reactor coolant system (RCS) through the cold legs. When sufficient water is removed from the RWST to ensure that enough boron has been added to maintain the reactor subcritical and the recirculation sump and containment sump have enough water to supply the required net positive suction head to the ECCS pumps, suction is switched to the recirculation sump for cold leg recirculation. At about 6.5 hours following a LOCA, the ECCS flow is shifted to the hot leg recirculation phase to provide a backflush, which would reduce the boiling in the top of the core and any resulting boron precipitation. The hot leg recirculation flowpath sends some of the water to the RCS hot legs and some to the RCS cold legs.

The ECCS Function is provided by three separate ECCS systems: High-Head Safety Injection (HHSI), Residual Heat Removal (RHR) injection, and Containment Recirculation. The ECCS accumulators and the RWST are also part of the ECCS. Each ECCS is divided into subsystems as follows:

a. The HHSI System is divided into three 50% capacity subsystems (i.e., HHSI pumps 21, 22, and 23) which share two pump discharge headers (i.e., 21 and 23). Each HHSI subsystem consists of one pump as well as associated piping and valves to transfer water from the suction source to the core. HHSI subsystem 22 is aligned to inject using the flow path associated with both HHSI subsystem 21 and 23. If all three HHSI pumps start, flow from HHSI pump 22 will be divided between header 21 and 23. If either HHSI pump 21 or 23 fails to start, either valve 851A or 851B will close automatically so that HHSI pump 22 will inject via the header associated with the failed pump. For IP3, orifices are used in place of valves 851A and 851B, which ensure sufficient injection flow via the header associated with the failed pump. The HHSI pumps have a shutoff head of approximately 1500 pounds per square inch gage (psig). Since this is below the RCS normal operating pressure of 2235 psig, IP2 and IP3 are classified as low-head safety injection plants.

b. The RHR Injection System is divided into two 100% capacity subsystems (i.e., RHR pumps 21 and 22). Each ECCS RHR subsystem consists of one RHR pump and one RHR heat exchanger as well as associated piping and valves to transfer water from the suction source to the core. Although either RHR heat exchanger may be credited for either RHR subsystem, one RHR heat exchanger must be operable for each operable RHR injection subsystem.

c. The Containment Recirculation System is divided into two 100% capacity subsystems (recirculation pumps 21 and 22) located within the containment building. Each subsystem consists of one Containment Recirculation pump and one RHR heat exchanger (the RHR heat exchangers are located inside the containment) as well as associated piping and valves to transfer water from the recirculation sump to the core. Although either RHR heat exchanger may be credited for either Recirculation subsystem, one RHR heat exchanger must be operable for each operable Containment Recirculation subsystem.

The three ECCS systems (3 HHSI, 2 RHR, and 2 Recirculation) are grouped into three trains (5A, 2A/3A, and 6A) such that any 2 of the 3 trains are capable of meeting all ECCS capability assumed in the accident analysis. Technical Specifications (TSs) require that all three trains be operable during power operation. Each ECCS train consists of the following:

1. ECCS Train 5A includes subsystems HHSI 21 and containment recirculation 21;

2. ECCS Train 2A/3A includes subsystems HHSI 22 and RHR 21; and,

3. ECCS Train 6A includes subsystems HHSI 23, RHR 22, and containment recirculation 22.

The ECCS trains use the same designation as the Safeguards Power Trains required by TS limiting condition for operation (LCO) 3.8.9, Distribution Systems - Operating, with Safeguards Power Train 5A supported by diesel generator (DG) 21 (DG-33 for IP3), Safeguards Power Train 2A/3A supported by DG 22 (DG-31 for IP3), and Safeguards Power Train 6A supported by DG 23 (DG-32 for IP3). Each of the subsystems (3 HHSI, 2 RHR and 2 Recirculation) are interconnected and redundant such that any combination of 2 HHSI pumps, 1 RHR pump and 1 recirculation pump is capable of supplying 100% of the flow required to mitigate the accident consequences. This interconnecting and redundant subsystem design provides the operators with the ability to utilize components from different trains to achieve the required 100% flow to the core. The design intent is that any two of the three safeguards power trains is capable of providing 100% of the required ECCS flow; however, any combination of the minimum number of pumps is capable of providing 100% of the required ECCS flow. This allows the safety function to be maintained if there is the failure of one DG, in combination with the loss of offsite power.

During the injection phase of LOCA recovery, a suction header supplies water from the RWST to the HHSI and RHR pumps. The discharge from the HHSI and RHR pumps feeds injection lines to each of the RCS cold legs. For LOCAs that are too small to depressurize the RCS below the shutoff head of the HHSI pumps, the charging pumps supply water until the RCS pressure decreases below the HHSI pump shutoff head. During this period, the steam generators are used to provide part of the core cooling function.

During the recirculation phase of LOCA recovery, the containment recirculation pumps take suction from the containment recirculation sump and direct flow through the RHR heat exchangers to the cold legs. The RHR heat exchangers are cooled by the CCWS, which is cooled by the service water system. The RHR pumps can also be used to provide a backup method of recirculation, in which case the RHR pump suction is transferred from the RWST to the containment sump. The RHR pumps can then operate in place of the recirculation pumps to supply recirculation flow directly or supply the suction of the HHSI pumps. Initially, recirculation is through the same paths as the injection phase. Subsequently, recirculation flow is split between the hot and cold legs.

There are two ECCS sumps located in the containment building, the recirculation sump and the containment sump, which are located in different quadrants of the containment building. The recirculation sump is the only suction source for the recirculation pumps. The RHR pumps have multiple suction alignments. The RHR pump suctions are aligned to the RWST during power operation but can be aligned to the containment sump after the completion of the injection phase of a LOCA. The RHR pumps are not needed during the LOCA recirculation phase unless the recirculation pump subsystems suffer a failure.

For the CCWS, there is a difference in design between IP2 and IP3. IP3 was designed with two main headers and two surge tanks, and valves that can split the headers into two trains. This allows IP3 to cope with a passive failure in the CCWS and still maintain one header in operation. IP2 has a single main header and a single surge tank. The following describes the CCWS for IP2. The CCWS provides a heat sink for the removal of process and operating heat from safety-related components during a design basis accident. During normal operation, the CCWS also provides this function for various nonessential components, as well as the spent fuel storage pool. The CCWS serves as an intermediate system to prevent the release of radioactive byproducts from radioactive systems to the service water system, which discharges into the Hudson River.

The CCWS consists of three pumps and two heat exchangers. The CCW pumps are connected to a common discharge header that is arranged so that any of the three pumps will supply either CCW heat exchanger and the heat exchangers are connected to a common discharge header so that both heat exchangers supply all CCWS heat loads. The CCW pumps share a common suction header that is supported by a single surge tank. Any one of the three CCW pumps in conjunction with any one of the two CCW heat exchangers is sufficient to accommodate the normal and post accident heat load. Therefore, the CCWS is considered to consist of two, 100% capacity trains. A CCW train consists of any of the three CCW pumps in conjunction with a CCW heat exchanger. Each of the three CCW pumps is powered from a separate safeguards power train. A vented surge tank in the system ensures that sufficient net positive suction head is available. CCW pumps continue to operate following a safety injection signal without loss of offsite power (LOOP); however, CCW pumps must be manually started as needed following a safety injection signal that includes a LOOP. The CCW pumps are not re-started immediately during the injection phase; therefore, the water volume of the CCWS must act as a heat sink during the injection phase when the CCW pumps are not running. This is acceptable even though the HHSI pump bearings are cooled by CCW because the cooling water is circulated by a booster pump directly connected to the HHSI pump motor shaft. The design basis of the CCWS is for one CCW train to remove the post-LOCA heat load from the recirculated water passing through the RHR heat exchangers during the recirculation phase. At least one CCW pump must be in operation during the recirculation phase, or alternate cooling supplies must be aligned. The CCWS is designed to perform its function with a single failure of any active component, assuming a LOOP.

3.2 NRC GL 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors."

The NRC issued GL 2004-02 on September 14, 2004, to request that all commercial PWRs perform an evaluation of the ECCS recirculation functions in light of the information provided in the letter and, if appropriate, take additional actions to ensure system function. Additionally, addressees were requested to submit the information specified in this letter to the NRC. This request was based on the identified potential susceptibility of PWR recirculation sump screens to debris blockage during design-basis accidents requiring recirculation operation of ECCS and on the potential for additional adverse effects due to debris blockage of flowpaths necessary for ECCS recirculation and containment drainage.

Entergy's supplemental response to GL 2004-02, dated February 28, 2008 (ADAMS Accession No. ML080670135) details the progress Entergy has made at IP2 and IP3. In this letter, Entergy reports the following progress:

a. Miscellaneous
   1. Containment walkdowns to identify and quantify the types and locations of potential debris sources
   2. Programmatic and procedural enhancements

b. Hardware
   1. Installation of replacement recirculation sump and containment sump passive strainers
   2. Installation of flow channeling barriers to enhance debris settlement
   3. Installation of a debris trash rack on the fuel transfer canal drain

c. Testing
   1. Dissolution/erosion measurements of plant-specific calcium silicate (IP2).
   2. Strainer head loss testing - debris only
   3. Chemical effects testing at Vuez

d. Analysis
   1. Debris generation analyses
   2. Debris transport analyses (subject to revision)
   3. Strainer head loss qualification - debris only (IP2 and IP3 (preliminary))
   4. Clean screen head loss evaluation
   5. Post-accident containment water level calculations (subject to revision)
   6. Net positive suction head available (NPSHA) analysis (subject to revision)
   7. Component (excluding pumps) downstream effects evaluations
   8. Fuel blockage downstream effects evaluations

e. Licensing

1. Submittal of buffer replacement license amendment request (IP2)

Since receiving Entergy's supplemental response letter, the NRC has issued the license amendments converting to the sodium tetraborate pH buffer for both IP2 and IP3. This post-LOCA buffer reduces chemical effects in the recirculated water which reduces head-loss across the sump strainers. Entergy has installed the new chemical buffer at IP2 and IP3.

In performing the analyses noted above, Entergy has determined, based on the assumptions in its testing and analyses, that in the worst-case scenario, the debris-generation and chemical effects may possibly approach the design limits for head-loss for the containment sump strainers, which are smaller than the recirculation sump strainers. The original recirculation sump strainers had a surface area of about 50 square feet. They have been replaced by larger sump strainers with a surface area of about 3200 square feet. The original containment sump strainers had a surface area of about 30 square feet. They have been replaced by larger sump strainers with a surface area of about 1000 square feet. The containment sump would not normally be used during the LOCA, as the recirculation pumps and the recirculation sump would normally fulfill the recirculation safety function. However, under the current licensing basis Entergy is required to also analyze an active or passive failure of the recirculation system during the recirculation phase of LOCA response. A failure may require that the RHR pumps, with a suction from the containment sump, provide the recirculation flow. Entergy explored the possibility of further increasing the size of the containment sump strainers, but size and location limitations make this impractical. Entergy's proposed solution is to modify the current licensing basis so that the passive failure of the recirculation system is assumed to occur in the long-term (after 24 hours). As the containment sump would not be needed until 24 hours after the LOCA, this analysis assumption would provide additional opportunity for debris to settle out in low-flow areas of containment or at the recirculation sump strainers and reduces the potential debris loading for the strainers at the containment sump.

3.3 Compliance with NRC Regulations

The applicable regulation is 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," which delineates the criteria that must be met by the ECCS. IP2 and IP3 use some ECCS evaluation models developed under 10 CFR 50.46(a)(1)(ii), which states "Alternatively, an ECCS evaluation model may be developed in conformance with the required and acceptable features of Appendix K ECCS Evaluation Models." 10 CFR Part 50, Appendix K, "ECCS Evaluation Models," section I.D.1, "Single Failure Criterion", states "An analysis of possible failure modes of ECCS equipment and of their effects on ECCS performance must be made. In carrying out the accident evaluation the combination of ECCS subsystems assumed to be operative shall be those available after the most damaging single failure of ECCS equipment has taken place." Footnote 2 to 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," states "The conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are under development."

Because the issue is still under development, the NRC staff relies on the guidance in NUREG - 0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," and SECY-77-439 (Reference 1) to identify "the most damaging single failure of ECCS equipment." NUREG-0800, Section 6.3, "Emergency Core Cooling System," Revision 3, provides that "the ECCS should retain its capability to cool the core in the event of a failure of any single active component during the short term immediately following an accident, or a single active or passive failure during the long-term recirculation cooling phase following an accident." Thus, either the failure may be a failure of an active component in the short term or an active or passive component in the long term. SECY-77-439 provides that the long-term cooling phase following a LOCA starts 24 hours after the LOCA, and identifies the passive failure to be considered as the failure of a pump seal or a valve seal, not a pipe break.

The NRC did not intend 10 CFR 50.46 to require that all passive failures be considered. The NRC staff relies on the criteria of SECY-77-439 to select an appropriate passive failure for the licensing basis. Since the revised licensing basis meets the criteria of 10 CFR 50.46, the NRC staff finds that the proposed revisions to the UFSAR are in compliance with NRC regulations.

3.4 Compliance with IP2 and IP3 GDC

The relevant IP2 and IP3 GDCs are listed in Section 2.0 of this SE. None of the GDCs require the consideration of a passive failure, but they do require the ECCS to accommodate the failure of any single active component. The NRC staff's review did not reveal any noncompliances with the IP2 and IP3 GDCs. There are no changes needed to the IP2 and IP3 GDCs as a result of this revision to the licensing basis.

3.5 ECCS LOCA Recirculation Phase Passive Failure Criteria

The current licensing basis for the IP2 and IP3 ECCS requires an assumption that an active or passive failure in the internal recirculation system at IP2 and IP3 could occur at any time after the start of the LOCA recirculation phase. The passive failure specified in the UFSAR includes the rupture of a pipe in the recirculation flow path (subsequent to the rupture of the reactor coolant system), and the pipe leak rate can exceed 3000 gpm. Entergy's request is to revise the IP2 and IP3 LOCA recirculation phase single passive failure licensing basis such that a passive failure is assumed to occur 24 hours or greater after the LOCA. The specific changes that Entergy will make to the IP2 and IP3 UFSAR are detailed in their application, in Section 2.0. The NRC staff finds these changes to the UFSAR to be acceptable, as they are consistent with current NRC staff guidance on passive failures, they are consistent with the plant's GDC, they are consistent with licensing bases the NRC has approved for other nuclear plants, they do not violate any NRC regulations, and the licensee has already completed significant enhancements to plant equipment to improve the LOCA recirculation system. Note that the assumption that the passive failure will not occur for 24 hours may be used in the analyses, however operational procedures must not assume that as a fact. In the current accident response procedures, in the case of insufficient water flow from the recirculation pumps to the reactor core, the operators are directed to achieve the required core cooling using the RHR pumps in place of the recirculation pumps. That action will remain the same even if 24 hours have not elapsed. In summary, recirculation using RHR pumps will continue to be the backup system to the recirculation pumps and is to be used whenever the recirculation pumps fail to perform their safety function.

3.6 IP2 CCWS Passive Failure Criteria

The current licensing basis for the IP2 CCWS requires that methods be available to provide the safety function of maintaining core cooling assuming an active or passive failure following a LOCA. For the CCWS, there is a difference in design between IP2 and IP3. IP3 was designed with two main headers and two surge tanks, and valves that can split the headers into two trains. This allows IP3 to cope with a passive failure in the CCWS and still maintain one header in operation. IP2 has a single main header and a single surge tank. The ECCS recirculation pump motors are cooled by CCW, and there is no alternate cooling available. If a passive failure beyond the capability of makeup water occurs on the IP2 CCWS and cannot be isolated, and CCW can no longer supply the recirculation pump coolers, the recirculation pumps will have to be stopped and the RHR pumps started on the containment sump to supply recirculation flow. Although the RHR and HHSI pumps are normally cooled by the CCWS, there are alternate methods available to supply cooling water to the RHR and HHSI pump coolers, such as the use of the primary water system or the city water system. Therefore, IP2 requests a modification to the licensing basis to only postulate the CCWS passive failure after 24 hours, so that LOCA recirculation will be initially supplied by the recirculation pumps.

The IP2 UFSAR, Section 9.3.3.3.1, currently states that:

“In the unlikely event of a pipe severance in the component cooling loop, backup is provided for postaccident heat removal by the containment fan coolers.”

This sentence discusses core decay heat removal. As the CCWS also cools the RHR heat exchangers, the use of the RHR heat exchangers for removing core decay heat will be lost if there is a passive failure of the CCWS at IP2 during LOCA recirculation. However, the containment fan coolers are safety-related equipment cooled by service water, and will be able to provide the decay heat removal function. The sentence does not mention the effect on the IP2 recirculation pumps, but they would be affected by this passive failure as noted above.

Entergy proposes to revise the IP2 UFSAR as follows:

“In the unlikely event of a pipe severance in the component cooling loop, backup is provided for postaccident heat removal by the containment fan coolers. Pipe severance is a passive failure and is assumed to occur 24 hours or greater after event initiation.”

Entergy also proposes to add the following to the IP2 UFSAR:

“Should the break occur inside containment and the leak cannot be isolated the residual heat removal pumps and safety injection pumps, if required, are employed to recirculate uncooled spilled water to the core. Heat is removed from the core by boil-off of the water to the containment with the fan coolers being used to condense the resulting steam.”

IP2 UFSAR section 9.3.1.1.1 states that active components of the CCWS which are relied upon to perform the cooling function are redundant. This means the CCWS can perform its safety function in the event of a single active failure. The NRC staff notes that operating experience has shown that the probability of passive failures in low-temperature, low-pressure safety-related systems such as the CCWS is extremely low. The maximum temperature expected in the CCWS during LOCA recirculation is about 120 degrees F, and the maximum pressure about

100 psig. The CCWS design temperature is 200 degrees F, and the design pressure is 150 psig. During a LOCA, the only additional environmental stressors to the CCWS occur inside the containment building. The CCWS piping will be exposed to elevated external temperatures up to 270 degrees F, elevated external pressure up to 46 psig, and containment spray water containing boric acid on the outside of the piping. The CCWS piping is constructed of carbon steel, and has been designed to withstand these conditions. Although boric acid does cause corrosion on carbon steel piping, the corrosion rates are low under these conditions. The chemical pH buffer, which is stored in containment and dissolves into the recirculated water, will raise the pH and reduce corrosion rates further. There is no known mechanism which would lead to accelerated piping failures (other than consequential failures due to the LOCA, which are discussed below) in the first 24 hours. The NRC staff judges this licensing basis change to be acceptable, as it does not violate any NRC regulations, and the probability of a passive failure is very low. The NRC staff reminds the licensee that consequential failures which result from the LOCA must be considered in the analysis. For example, non-missile protected CCW pipes inside containment affected by the LOCA must be assumed to rupture at the initiation of the LOCA. The NRC staff notes that the non-missile protected pipes are small diameter, and have isolation valves outside containment. Leakage will result in a low-level alarm for the CCW surge tank, and the operators will have time to initiate makeup water and isolate the non-missile protected lines. This action is part of the current licensing basis and is not affected by this SE.

3.7 Health and Safety of the Public

Under case law, reasonable assurance of adequate protection of public health and safety is, as a general matter, defined by the NRC's health and safety regulations themselves. That is, unless otherwise provided, there is reasonable assurance of adequate protection of public health and safety when the licensee demonstrates compliance with the NRC's regulations. The regulations were established using defense-in-depth principles and conservative practices that provide a degree of margin to unsafe levels. Therefore, as this change is in compliance with NRC's regulations, the NRC staff finds that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner.

4.0 PRECEDENT

The NRC staff has previously approved changes in licensing basis associated with GL 2004-02. For example, Duke Energy submitted a license amendment for the Catawba Nuclear Station, Units 1 and 2, dated March 29, 2007, which requested a revision to the licensing bases to combine two containment sumps into one large containment sump which would be more resistant to clogging from debris, and to eliminate the use of trash racks at the sump in favor of sump strainers which are more resistant to debris clogging. The NRC approved this license amendment on November 8, 2007, ADAMS Accession No. ML073020585.

5.0 PUBLIC COMMENTS

Comments on this license amendment request were received from Mr. Lochbaum of the Union of Concerned Scientists in a letter dated July 2, 2008 (ADAMS Accession No. ML081960599), in which Mr. Lochbaum described the license amendment as flawed and urged the NRC not to grant it. Mr. Lochbaum did not contend that the proposed change involves a significant hazard. Mr. Lochbaum provided supporting information in which he argued that SECY-77-439 is flawed in allowing the consideration of passive failures to be delayed for 24 hours following the LOCA.

One of Mr. Lochbaum's points is that the stresses on piping and valves from elevated temperatures and pressures during the first day of a LOCA event are significantly higher than the stresses from lower temperature and pressures during the remainder of the LOCA period, and therefore a passive failure may be more likely to occur within the first 24 hours of a LOCA.

The overall effect of the licensee's actions to resolve the concerns of GL 2004-02 has increased the safety aspects of the LOCA recirculation system. Most of the plants impacted by the concerns of GL 2004-02 were licensed in the 1970's, yet the NRC waited until 2004 to address the concern with recirculation sump clogging because it is a very low probability event due to the low frequency of large-break LOCAs and extensive research was needed to decide if action was justified. The actions taken by PWR licensees, including IP2 and IP3, since the issuance of GL 2004-02 have served to improve the safety posture of the plants by improving the LOCA recirculation system.

IP2 and IP3 have a more conservative licensing basis than certain other PWRs. Other PWRs are only required to consider a passive failure of a pump seal or valve seal, with a leak rate of 50 gpm. Under these conditions, IP2 and IP3 would meet the LOCA recirculation criteria, assuming that passive failure occurs when LOCA recirculation is initiated, with just the use of the internal recirculation system (recirculation pumps). In that case, the use of the RHR pumps for LOCA recirculation would truly be just a backup system. Other PWRs typically do not have a backup LOCA recirculation system. It is the fact that IP2 and IP3 have a conservative licensing basis that leads them to request this license amendment. They could have achieved the same relief by asking the NRC to reduce their passive failure criteria to the more typical criteria of a failed pump seal or valve seal, but instead they asked to be allowed the 24-hour delay, consistent with the NRC staff's current position as stated in SECY 77-439.

The NRC staff acknowledges that pressures and temperatures are highest during the first 24 hours following a LOCA. The maximum temperature expected in the LOCA recirculation system is about 270 degrees F, and the maximum pressure would be the shutoff head of the recirculation pumps, about 220 psig. However, the NRC staff notes that operating experience has shown that the probability of passive failures in these relatively low-temperature, low-pressure, safety-related systems such as the LOCA recirculation system is extremely low. For these reasons and the reasons stated in this SE, the NRC staff has decided it is appropriate to approve this license amendment.

6.0 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.

7.0 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

(73 FR 37503). 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.

8.0 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.

9.0 REFERENCES

U.S. Nuclear Regulatory Commission, "Information Report by the Office of Nuclear Reactor Regulation on the Single Failure Criterion," Commission Paper SECY 77-439, August 17, 1977 (ADAMS Accession No. ML060260236).

Principal Contributor: J. Boska

Date: December 4, 2008

December 4, 2008

Vice President, Operations
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
Buchanan, NY 10511-0249

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3 - ISSUANCE OF AMENDMENTS RE: PASSIVE FAILURE ANALYSIS (TAC NOS. MD8290 AND MD8291)

Dear Sir or Madam:

The Commission has issued the enclosed Amendment No. 257 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2 (IP2) and Amendment No. 238 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3 (IP3). The amendments consist of changes to the Updated Final Safety Analysis Reports (UFSARs) in response to your application dated March 13, 2008.

The amendments revise the UFSARs by allowing a 24 hour delay after a loss-of-coolant-accident before postulating a passive failure in the emergency core cooling systems for IP2 and IP3 or the component cooling water system for IP2.

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 Nos. 50-247 and 50-286
Enclosures:
1. Amendment No. 257 to DPR-26
2. Amendment No. 238 to DPR-64
3. Safety Evaluation

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Amendment No.: ML082240194

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AMENDMENT NO. 257 TO FACILITY OPERATING LICENSE NO. DPR-26 INDIAN POINT UNIT 2 AND AMENDMENT NO. 238 TO FACILITY OPERATING LICENSE NO. DPR-64 INDIAN POINT UNIT 3

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