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

October 29, 2009

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 NO. 2 - ISSUANCE OF AMENDMENT RE: REVISION TO EMERGENCY CORE COOLING SYSTEM (ECCS) VALVE SURVEILLANCE REQUIREMENTS (TAC NO. ME0917)

Dear Sir or Madam:

The Commission has issued the enclosed Amendment No. 263 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2 (IP2). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated March 25, 2009.

The amendment added two ECCS valves to TS Surveillance Requirement 3.5.2.1 for checking valve position every 7 days.

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 No. 50-247

Enclosures:
1. Amendment No. 263 to DPR-26
2. Safety Evaluation

cc w/encls: Distribution via Listserv

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. 263
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 25, 2009, 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. 263, 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 prior to entering Mode 4 during startup from refueling outage 2R19.

FOR THE NUCLEAR REGULATORY COMMISSION

Nancy L. Salgado

Nancy L. Salgado, 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: October 29, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 263

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
3.5.2-2	3.5.2-2

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. 263, 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.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify the following valves are in the listed position with power to the valve operator removed. Number / Position / Function 842 / Open / SI Test Line Stop 843 / Open / SI Test Line Stop 883 / Closed / RHR Return to RWST 743 / Open / RHR Miniflow Line 1870 / Open / RHR Miniflow Line 856B / Closed / HH Branch Line Stop Valve 23 Loop Hot Leg 856F / Closed / HH Branch Line Stop Valve 21 Loop Hot Leg 1810 / Open / Common RWST Suction Isolation for HHSI Pumps 882 / Open / Common Suction Isolation for RHR Pumps 744 / Open / Common discharge isolation for RHR pumps 745A / Open / 22 RHR Hx Inlet 745B / Open / 22 RHR Hx Inlet	7 days
SR 3.5.2.2	Verify each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.5.2.3	Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.5.2.4	Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	24 months

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 263 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 March 25, 2009, Agencywide Documents Access and Management System (ADAMS) Accession No. ML090900511 (Reference 1), Entergy Nuclear Operations, Inc. (Entergy or the licensee) submitted a request for changes to the Indian Point Nuclear Generating Unit No. 2 (IP2) Technical Specifications (TSs).

The proposed change would add two Emergency Core Cooling System (ECCS) motor-operated valves (MOVs), 745A and 745B, to TS Surveillance Requirement (SR) 3.5.2.1 for checking valve position every 7 days. These two valves are located in series on the inlet pipe to the number 22 residual heat removal heat exchanger. The TS SR is designed to verify that ECCS valves whose single failure could cause loss of the ECCS function are in the required position with ac power removed so that misalignment or single failure cannot prevent completion of the ECCS function. The TS SR revision supports Entergy's resolution to Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors" (Reference 2), by establishing a licensing basis in conformance with the licensee's response to GL 2004-02.

1.1 Background

In GL 2004-02, the Nuclear Regulatory Commission (NRC) requested that licensees for all pressurized-water reactors (PWRs) show by an evaluation of the ECCS recirculation system function that the system function would be met following a loss-of-coolant accident (LOCA). There are two separate ECCS sumps in the containment building at IP2, the recirculation sump and the containment sump. The recirculation sump is larger than the containment sump, and has two recirculation pumps mounted on it, which are used to recirculate water from the sump into the reactor coolant system and thereby to the reactor vessel for long-term core cooling following the ECCS injection phase of a LOCA. The containment sump is a backup to the recirculation sump for the provision of long-term core cooling, and the containment sump is the only sump from which the residual heat removal (RHR) pumps take a suction. Under the current licensing basis, in order to provide the required protection against equipment failures, the ECCS is arranged to allow either of the RHR pumps, drawing water from the containment sump, to take over the recirculation function from the recirculation pumps at any time after the start of the recirculation phase. For the revised licensing basis, Entergy plans to show that the recirculation sump system by itself is fully qualified to perform the recirculation safety function for the first 24 hours, including any single active failure, if MOVs 745A and 745B are open with their ac power deenergized. After 24 hours, Entergy would have to consider a passive failure in the recirculation sump system, which may make it necessary to switch to the containment sump and the RHR pumps.

ECCS valves whose single failure could cause loss of the ECCS function must be in the required position with power removed so that the single failure could not occur. Currently, when normal ac power is removed from MOVs 745A and 745B, valve position indication is lost. Entergy plans to provide the required position indication by installing an alternate source of power to the valve position indication during the spring 2010 refueling outage (2R19).

Entergy has performed a single failure analysis and has determined that, except for a passive failure (which is only postulated after 24 hours, see Reference 5), the only single failure that could require the backup capability of the containment sump is the spurious closure of MOVs 745A or 745B, which would affect the high-head recirculation mode of operation. The high-head recirculation mode of operation is used during hot-leg recirculation, since the high-head ECCS pumps are part of the flow path to the reactor coolant system hot legs. The high-head recirculation mode of operation is also used for the recirculation phase during small-break LOCAs, when the RCS pressure remains above the discharge pressure of the recirculation pumps. The high-head ECCS pumps at IP2 are the safety injection pumps, which have a shutoff head (maximum discharge pressure) of about 1500 pounds per square inch gage (psig), with suction aligned to the refueling water storage tank (RWST). The recirculation pumps have a shutoff head of about 200 psig (assuming atmospheric pressure in the containment building), and the RHR pumps have a shutoff head of about 165 psig (with suction aligned to the containment sump, assuming atmospheric pressure in the containment building).

Entergy's evaluation in its response to GL 2004-02 does not credit the containment sump for a period of 24 hours following a large-break LOCA. This allows time for debris generated by the LOCA to settle out in the containment building, or to collect on the recirculation sump strainers. The recirculation sump strainers are designed to accommodate the full debris load resulting from the damage done by the high-pressure leakage from the LOCA. Entergy states that by precluding the spurious closure of MOV 745A and MOV 745B, there will be no single active failure that would require switching from the recirculation sump to the containment sump. In order to preclude the spurious closure of MOVs 745A or 745B, and the consequential inability to perform high head recirculation using the recirculation sump, it is proposed that these valves be deenergized in the open position. The valves would be reenergized 24 hours following a design-basis accident (DBA).

The proposed amendment would add MOVs 745A and 745B to SR 3.5.2.1. The purpose of the SR is to verify that ECCS valves whose single active failure could cause loss of the ECCS function are in the required position with power removed so that the single active failure could not occur.

2.0 REGULATORY EVALUATION

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

2.1 The *Code of Federal Regulations*

2.1.1 10 CFR 50.46

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."

2.1.2 10 CFR 50.36

Section 182a of the Atomic Energy Act (Act) requires applicants for nuclear power plant operating licenses to include technical specifications as part of the license. The licensee provides technical specifications in order to maintain the operational capability of structures, systems and components that are required to protect the health and safety of the public. The NRC's regulatory requirements that are related to the content of the TS are contained in 10 CFR 50.36. The TS requirements in 10 CFR 50.36 include the following categories: (1) safety limits, limiting safety systems settings and limiting control settings; (2) limiting conditions for operation (LCO); (3) surveillance requirements; (4) design features; and (5) administrative controls.

Section 50.59 (c)(1)(i) requires a licensee to submit a license amendment pursuant to 10 CFR 50.90 if a change to the TS is required. Furthermore, the requirements of 10 CFR 50.59 necessitate that NRC approve the TS changes before the TS changes are implemented.

Section 50.36(c)(2)(ii) requires that a TS LCO be established for each item meeting one or more of the following criteria:

Criterion 1 Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2 A process variable, design feature, or operating restriction that is an initial condition for a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of the fission product barrier.

Criterion 3 A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of the fission product barrier.

Criterion 4 A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

The NRC staff reviewed the proposed changes using Criterion 3 criteria above.

2.2 General Design Criteria (GDC) for IP2

The following explains the applicability of GDC for IP2. The construction permit for IP2 was issued by the Atomic Energy Commission (AEC) on October 14, 1966, and the operating license was issued on September 28, 1973. The GDC that IP2 was designed and licensed to are listed in IP2's Updated Final Safety Analysis Report (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 are those in the IP2 UFSAR, and will be referred to as UFSAR GDC to distinguish them from the Appendix A GDC.

As discussed in the UFSAR, the licensee for IP2 has made some changes to the facility over the life of the unit that committed 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 IP2 UFSAR and in other IP2 licensing basis documentation, such as license amendments.

The following plant-specific licensing basis GDCs, as found in the IP2 UFSAR, pertain to this license amendment:

> UFSAR GDC 37: 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: 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: 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: 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: 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 loss-of-coolant accident to the extent of causing undue risk to the health and safety of the public.

UFSAR GDC 43: 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: An emergency core cooling system 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 emergency core cooling system shall be evaluated conservatively in each area of uncertainty.

UFSAR GDC 52: 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.

2.3 Single Failure Criteria

The NRC's application of single failure criteria for ECCS systems is discussed in SECY-77-439 (Reference 6).

The following definition is 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.

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.

2.4 NRC's Position on Removing Power From MOVs

Refer to Branch Technical Position (BTP) 8-4 (formerly BTP EICSB-18, see References 4 and 7), which states, in part, that where a single failure in an electrical system could impede the system's ability to perform its intended safety function, the effect on plant safety must be evaluated. The BTP addresses the strategy of disconnecting power to fluid system components such as MOVs as a way to deal with the possibility of such single failures. EICSB-18 was made part of IP2's licensing basis in an NRC letter to Consolidated Edison Company dated June 18, 1975 (Reference 7).

3.0 TECHNICAL EVALUATION

3.1 Proposed Change

The licensee proposes to add the following RHR system valves to the IP2 TS SR 3.5.2.1:

Number	Position	Function
745A	Open	22 RHR Heat Exchanger (Hx) Inlet
745B	Open	22 RHR Hx Inlet

The safety function of the 745A and 745B valves is to:

1. Remain open to ensure the availability of safety injection flow paths to the reactor coolant system cold and hot legs during both the injection and recirculation phases of a DBA.

2. Close to isolate a passive failure involving portions of the RHR subsystem and the piping connecting the RHR pump and recirculation pump discharges to the high-head safety injection pumps (line #60).

Under the proposed amendment, the 745A and 745B valves would continue to perform these functions. The ECCS passive failure licensing basis (Reference 5) does not require the 745A and 745B valve isolation function until 24 hours following a DBA. Therefore, in order to perform the isolation function if a passive failure were to occur, the 745A and 745B valves would need to be reenergized at that time.

3.2 Description of the ECCS

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 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 phase 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), 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 RCS. 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. The HHSI pumps have a shutoff head of approximately 1500 psig. Since this is below the RCS normal operating pressure of 2235 psig, IP2 is classified as a low-head safety injection plant.

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 RCS. 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 RCS. 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. 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, Safeguards Power Train 2A/3A supported by DG 22, and Safeguards Power Train 6A supported by DG 23. 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 RCS cold legs. The RHR heat exchangers are cooled by the component cooling water system, 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, during the hot-leg recirculation phase, 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.

3.3 NRC Staff Evaluation

The licensee's proposal to add MOVs 745A and 745B to SR 3.5.2.1 refers to BTP EICSB-18, "Application of the Single Failure Criterion to Manually-Controlled Electrically-Operated Valves" (Reference 7). EICSB-18 states, "Where it is determined that failure of an electrical system component can cause undesired mechanical motion of a valve or other fluid system component, and this motion results in loss of the system safety function, it is acceptable, in lieu of design changes that also may be acceptable, to disconnect power to the electric systems of the valve or other fluid system component." It is assumed, in EICSB-18, that no single failure can both restore power to the electrical system and cause mechanical motion of the components served by the electrical system. The power disconnection strategy for MOVs was originally addressed in 1976, in Amendment 20 to IP2's Facility Operating License (Reference 8), and was further revised in Amendment 256 (Reference 9).

SR 3.5.2.1 applies to valves that, when improperly positioned, could cause more than one ECCS train to become inoperable (i.e., result in loss of the system safety function, since the ECCS can fulfill its safety function with the loss of a single train). The positions of such valves are verified every 7 days.

The NRC staff reviewed the IP2 ECCS for single active failures which could result in a loss of the recirculation system function when using the Containment Recirculation subsystem (the recirculation pumps taking suction from the recirculation sump, and the flowpaths needed to deliver the water to the RCS). Other than MOVs 745A and 745B, the NRC staff did not identify any additional single active failures which could result in a loss of the recirculation system function when using the Containment Recirculation subsystem. For this single failure review, the NRC staff relied on the criteria given in SECY-77-439 (Reference 6). Therefore, the NRC staff concludes that following the addition of MOVs 745A and 745B to SR 3.5.2.1, the IP2 ECCS Containment Recirculation subsystem will meet the single active failure criteria, without reliance on the RHR recirculation subsystem (the RHR pumps taking suction from the containment sump). Note that under the current licensing basis, the RHR recirculation subsystem is required to function for the high-head recirculation mode of operation if an active failure (inadvertent closure) is experienced by MOV 745A or 745B. After the plant modification to deenergize MOVs 745A and 745B in the open position, there will be no need to rely on the RHR recirculation subsystem to function in the event of single active failures.

The RHR recirculation subsystem remains as the backup system to the Containment Recirculation subsystem, and is needed to cope with certain passive failures. In the plant's emergency operating procedures, the operators are directed to use the RHR recirculation subsystem at any time, if there is a functional failure of the Containment Recirculation subsystem. The role of the RHR recirculation subsystem, as a backup to the Containment Recirculation subsystem, remains in effect.

An additional condition given in BTP EICSB-18 is that "...these valves should have redundant position indication in the main control room and the position indication system should, itself, meet the single failure criterion." Although the current position indication for 745A and 745B does not satisfy this requirement, the licensee will install a modification to the position indication that meets the guidance in EICSB-18, by providing continuous and redundant valve position indication at all times, even when the valves are deenergized.

Therefore, the NRC staff finds the licensee's proposal to add the normally open valves, 745A and 745B, to SR 3.5.2.1 is acceptable because:

(a) The IP2 ECCS remains in compliance with NRC regulations, particularly 10 CFR 50.46 and Appendix K to 10 CFR Part 50.

(b) The performance of the IP2 ECCS will continue to satisfy the criteria given in the UFSAR GDC for IP2.

(c) The proposal is consistent with BTP EICSB-18, "Application of the Single Failure Criterion to Manually-Controlled Electrically-Operated Valves" (Reference 7).

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 and changes surveillance requirements. 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 (74 FR 23444). 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. Entergy Letter NL-09-004, "License Amendment Request Regarding Surveillance Requirements For ECCS Valves Required to be in Position with Power to the Valve Operator Removed," dated March 25, 2009 (ADAMS Accession No. ML090900511).

2. NRC Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," dated September 13, 2004.

3. Entergy Letter NL-08-025, "Supplemental Response to NRC Generic Letter 2004-02, Potential Impact Of Debris Blockage On Emergency Recirculation During Design Basis Accidents At Pressurized-Water Reactors," dated February 28, 2008 (ADAMS Accession No. ML080670135).

4. BTP 8-4 (formerly EICSB-18), "Application of the Single Failure Criterion to Manually-Controlled Electrically-Operated Valves," Rev. 3, March 2007. (Attached to Standard Review Plan, NUREG-0800, Section 8).

5. NRC Letter, “Indian Point Nuclear Generating Unit Nos. 2 and 3 - Issuance of Amendments Re: Passive Failure Analysis (TAC Nos. MD8290 and MD8291),” dated December 4, 2008 (ADAMS Accession No. ML082240194).

6. 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).

7. NRC Letter on the December 27, 1974, Order for Modification of Facility License, dated June 18, 1975 (ADAMS Legacy Library Accession No. 8111100247).

8. NRC Letter issuing Amendment 20 to Facility Operating License DPR-26 for Indian Point Nuclear Generating Unit No. 2,” dated September 6, 1976 (ADAMS Accession No. ML003774952).

9. NRC Letter, “Indian Point Nuclear Generating Unit No. 2 - Issuance Of Amendment Re: Revision To ECCS Valve Surveillance Requirements (TAC No. MD7501),” dated October 29, 2008 (ADAMS Accession No. ML082700099).

Principal Contributor: John Boska

Date: October 29, 2009

October 29, 2009

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

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 - ISSUANCE OF AMENDMENT RE: REVISION TO EMERGENCY CORE COOLING SYSTEM (ECCS) VALVE SURVEILLANCE REQUIREMENTS (TAC NO. ME0917)

Dear Sir or Madam:

The Commission has issued the enclosed Amendment No. 263 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2 (IP2). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated March 25, 2009.

The amendment added two ECCS valves to TS Surveillance Requirement 3.5.2.1 for checking valve position every 7 days.

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. 263 to DPR-26
2. Safety Evaluation

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AMENDMENT NO. 263 TO FACILITY OPERATING LICENSE NO. DPR-26 INDIAN POINT UNIT 2

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