September 25, 2008

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

SUBJECT: INDIAN POINT ENERGY CENTER – NRC EVALUATION OF CHANGES, TESTS, OR EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS TEAM INSPECTION REPORT - UNIT 2; AND OPEN ITEM CLOSEOUT - UNIT 3 COMBINED INSPECTION REPORT 05000247/2008012 AND 05000286/2008010

Dear Mr. Pollock:

On August 14, 2008, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Indian Point Energy Center (IPEC). The enclosed inspection report documents the inspection results, which were discussed on August 14, 2008, with Mr. T. Orlando, Director of Engineering, and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspection involved field walkdowns; examination of selected procedures, calculations and records; observation of activities; and interviews with station personnel.

This report documents one NRC identified finding which was of very low safety significance (Green). The finding was determined to involve a violation of NRC requirements. However, because of the very low safety significance of the violation, and because it was entered into your corrective action program, the NRC is treating it as a non-cited violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy. If you contest the NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U. S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001, with copies to the Regional Administrator, Region 1; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspectors at the IPEC.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of the NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Lawrence T. Doerflein, Chief
Engineering Branch 2
Division of Reactor Safety

Docket No: 50-247/286
License No: DPR-26, DPR-64

Enclosure: Combined Inspection Report 05000247/2008012 and 05000286/2008010
w/Attachment: Supplemental Information

cc w/encl:
Senior Vice President, Entergy Nuclear Operations
Vice President, Operations, Entergy Nuclear Operations
Vice President, Oversight, Entergy Nuclear Operations
Senior Manager, Nuclear Safety and Licensing, Entergy Nuclear Operations
Senior Vice President and COO, Entergy Nuclear Operations
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Chairman, Standing Committee on Energy, NYS Assembly
Chairman, Standing Committee on Environmental Conservation, NYS Assembly
Chairman, Committee on Corporations, Authorities, and Commissions
M. Slobodien, Director, Emergency Planning
P. Eddy, NYS Department of Public Service
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T. Seckerson, County Clerk, Westchester County Board of Legislators
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T. Judson, Central NY Citizens Awareness Network
M. Elie, Citizens Awareness Network
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M. Mariotte, Nuclear Information & Resources Service
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J. Riccio, Greenpeace
P. Musegaas, Riverkeeper, Inc.
M. Kaplowitz, Chairman of County Environment & Health Committee
A. Reynolds, Environmental Advocates
D. Katz, Executive Director, Citizens Awareness Network
K. Coplan, Pace Environmental Litigation Clinic
M. Jacobs, IPSEC
W. Little, Associate Attorney, NYSDEC
M. J. Greene, Clearwater, Inc.
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J. Spath, New York State Energy Research, SLO Designee
A. J. Kremer, New York Affordable Reliable Electricity Alliance (NY AREA)

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U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	50-247, 50-286
License No:	DPR-26, DPR-64
Report No:	05000247/2008012 and 05000286/2008010
Licensee:	Entergy Nuclear Northeast
Facility:	Indian Point Nuclear Generating Units 2 and 3
Location:	450 Broadway, GSB Buchanan, NY 10511-0308
Dates:	July 28, 2008 through August 14, 2008
Inspectors:	A. Ziedonis, Reactor Inspector (Team Leader) K. Mangan, Senior Reactor Inspector S. Smith, Reactor Inspector
Approved by:	Lawrence T. Doerflein, Chief Engineering Branch 2 Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000286/2008-010, 05000247/2008-012; 07/28/2008 - 08/14/2008; Indian Point Nuclear Generating Units 2 and 3; Followup of Events and Notices of Enforcement Discretion and Other Activities.

The report documents a two week (on-site) team inspection covering the Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications on Unit 2; open item closure on Unit 3; and, Followup of Events and Notices of Enforcement Discretion inspections on both units. The inspection was conducted by three region-based engineering inspectors. One finding of very low risk significance (Green) was identified, and was considered to be a non-cited violation. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. <u>NRC-Identified and Self-Revealing Findings</u>

Cornerstone: Mitigating Systems

- <u>Green</u>. The team identified a non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion III, Design Control, because Entergy did not verify the adequacy of the internal recirculation pump minimum flow rates. Specifically, Entergy did not verify the adequacy of the pump minimum flow rates for sustained operation under low flow rate conditions or for strong-pump to weak-pump interactions which could result in dead-heading the weaker pump during parallel pump operation. Following identification of the issue, Entergy revised the Emergency Operating Procedures (EOP) to not start a second internal recirculation pump during conditions of high head recirculation, submitted a licensee event report (LER) for each generating unit, and entered the issue into the corrective action program.

  The finding was determined to be more than minor because it is associated with the design control attribute of the Mitigating Systems (MS) Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. On Unit 2, the team determined the finding was of very low safety significance because it was a design or qualification deficiency confirmed not to result in loss of operability or functionality. On Unit 3, the finding was determined to be of very low safety significance based on a Significance Determination Process (SDP) Phase 3 risk assessment. Also, the Unit 3 finding had a crosscutting aspect in the area of Problem Identification and Resolution because Entergy did not implement operating experience information through changes to station processes, procedures, and equipment. (IMC 0305 aspect P.2 (b)) (Section 4OA5)

B. <u>Licensee-Identified Violations</u>

None.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R17 Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications (IP 71111.17)

.1 Evaluations of Changes, Tests, or Experiments (24 samples)

a. Inspection Scope

The team reviewed one safety evaluation to determine whether the changes to the facility or procedures, as described in the Updated Final Safety Analysis Report (UFSAR), had been reviewed and documented in accordance with 10 CFR 50.59. In addition, the team evaluated whether Entergy had been required to obtain NRC approval prior to implementing the change. The team interviewed plant staff and reviewed supporting information including calculations, analyses, design change documentation, procedures, the UFSAR, technical specifications (TS), and plant drawings, to assess the adequacy of the safety evaluation. The team compared the safety evaluation and supporting documents to the guidance and methods provided in Nuclear Energy Institute (NEI) 96-07, “Guidelines for 10 CFR 50.59 Evaluations,” as endorsed by NRC Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," to determine the adequacy of the safety evaluation.

The team also reviewed a sample of twenty-three 10 CFR 50.59 screenings and applicability determinations for which Entergy had concluded that no safety evaluation was required. These reviews were performed to assess whether Entergy's threshold for performing safety evaluations was consistent with 10 CFR 50.59. The sample of issues inspected that had been screened out by Entergy included procedure changes, design changes, calculations, and set point changes.

The single safety evaluation reviewed was the only safety evaluation performed by Entergy during the time period covered under this inspection (i.e., since the last team inspection that evaluated changes, tests, or experiments). The screenings and applicability determinations were selected based on the risk significance of the associated structures, systems, and components (SSCs).

In addition, the team compared Entergy's administrative procedures, used to control the screening, preparation, review, and approval of safety evaluations, to the guidance in NEI 96-07 to determine whether those procedures adequately implemented the requirements of 10 CFR 50.59. The safety evaluations, screenings, and applicability determinations reviewed by the team are listed in the attachment.

b. Findings

No findings of significance were identified.

.2 Permanent Plant Modifications (8 samples)

.2.1 125 Volt Direct Current Circuit Breaker Replacements

a. Inspection Scope

The team reviewed a modification to replace the direct current (DC) HFB-model circuit breakers in panel 23 due to breaker age concerns. The review was performed to determine whether the design bases, licensing bases, and performance capability of the DC electrical distribution system had been degraded by the modification. Additionally, the 10 CFR 50.59 screen associated with this modification was reviewed as described in section 1.1 of this report.

The team assessed selected design attributes to determine whether they were consistent with the design and licensing bases. The attributes included component safety classification, breaker trip coordination requirements, and seismic qualification of the breaker and electrical panel. The team evaluated design assumptions in the supporting evaluations and analyses to determine whether they were technically appropriate and consistent with the Updated Final Safety Analysis Report (UFSAR). The team reviewed selected evaluations, drawings, analysis, procedures, and the UFSAR to determine whether they were properly updated with any revised design information. The team evaluated the post-modification tests to determine whether the breaker would function in accordance with design requirements. In addition, the team interviewed the responsible design and system engineers to discuss the circuit breaker replacements and design requirements. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2.2 Removal of Turbine Trip Protection for Uneven Expansion

a. Inspection Scope

The team reviewed a modification to remove the turbine trip feature protecting against uneven expansion of turbine rotational components with respect to the stationary components of the system. The review was performed to determine whether the design bases, licensing bases, and performance capability of the steam system or reactor protection system had been degraded by the modification. Additionally, the 10 CFR 50.59 screen associated with this modification was reviewed as described in section 1.1 of this report.

The team assessed selected design attributes to determine whether they were consistent with the design and licensing bases. These attributes included component safety classification, adequacy of operator indication for protection of the turbine, and the establishment of appropriate procedure guidance to manually trip the turbine in the event of uneven turbine expansion. The team evaluated design assumptions in the supporting evaluations and analyses to determine whether they were technically appropriate and consistent with the UFSAR. The team reviewed selected evaluations, drawings, analyses, procedures, and the UFSAR to determine whether they were properly updated with any revised design information. The team evaluated the post-modification test to verify that the trip function had been properly isolated. In addition, the team interviewed the responsible design and system engineers to discuss the modification and the design requirements. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2.3 Removal of Turbine Trip Protective Features

a. Inspection Scope

The team reviewed a modification to the main generator stator water cooling system. The modification removed single point vulnerabilities that could lead to an inadvertent main turbine trip, including main generator rectifier cooling flow and stator water cooling inlet flow. The review was performed to determine whether the design bases, licensing bases, and performance capability of the steam system or reactor protection system had been degraded by the modification. Additionally, the 10 CFR 50.59 screen associated with this modification was reviewed as described in section 1.1 of this report.

The team assessed selected attributes of the modification process to determine whether they were consistent with the design and licensing bases. These attributes included component safety classification, adequacy of operator indication for protection of the turbine, and the establishment of appropriate procedure guidance to manually trip the turbine based on alarms and other indications. Design assumptions were reviewed to evaluate whether they were technically appropriate and consistent with the UFSAR. The team reviewed selected calculations, drawings, analysis, procedures, and the UFSAR to determine whether they were properly updated with revised design information and operating guidance. The team evaluated the post-modification tests to verify that the safety related trip functions associated with the turbine were not degraded by the modification. In addition, the team interviewed the responsible design and system engineers to discuss the modification and the design requirements. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2.4 Internal Recirculation Pump Level Transmitter Modification

a. Inspection Scope

The team reviewed a modification to level transmitter LT-938, which is used for indication of internal recirculation pump suction level during inservice testing. The modification was performed to support changes in testing requirements of the internal recirculation pumps, due to changes in American Society of Mechanical Engineers (ASME) code acceptance criteria, which will require a higher suction water level to ensure adequate submergence during testing at higher flow rates. The review was performed to determine whether the design bases, licensing bases, and performance capability of the internal recirculation system had been degraded by the modification. Additionally, the 10 CFR 50.59 screen associated with this modification was reviewed as described in section 1.1 of this report.

The team assessed selected design attributes to determine whether they were consistent with the design and licensing bases. These attributes included component safety classification, instrument uncertainty, adequacy of level transmitter position, and adequacy of the water level for pump testing. The team evaluated design assumptions in the supporting evaluations and analyses to determine whether they were technically appropriate and consistent with the UFSAR. The team reviewed selected evaluations, drawings, analysis, procedures, and the UFSAR to determine whether they were properly updated with any revised design information. The team evaluated the post-modification test to determine whether the final installed set points were within the acceptance band to verify that the level transmitter would function in accordance with design assumptions. In addition, the team interviewed the responsible design and system engineers to discuss the modification and the design requirements. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2.5 Installation of ¾-inch Vent Line in Safety Injection System Piping

a. Inspection Scope

The team reviewed a modification to install a vent line on a relative high point in the safety injection discharge line to allow for venting gasses to ensure the safety injection piping remains full of water. The review was performed to determine whether the design bases, licensing bases, and performance capability of the safety injection system had been degraded by the modification. Additionally, the 10 CFR 50.59 screen associated with this modification was reviewed as described in section 1.1 of this report.

The team assessed selected design attributes to determine whether they were consistent with the design and licensing bases. These attributes included component safety classification, ASME piping requirements, and procedural guidance for venting operations. The team evaluated design assumptions in the supporting evaluations and analyses to determine whether they were technically appropriate and consistent with the UFSAR. The team reviewed selected evaluations, drawings, analysis, procedures, and the UFSAR to determine whether they were properly updated with any revised design information. The team evaluated the post-modification test to determine whether the new piping and valve would function in accordance with design requirements. In addition, the team interviewed the responsible design and system engineers to discuss the installation of the vent line as well as design requirements. Finally, the team walked down the safety injection system vent line to detect any potentially abnormal installation conditions. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2.6 Modification to Replace Hydraulic Snubbers

a. Inspection Scope

The team reviewed documents regarding the replacement of Bergen-Patterson snubbers with Lisega snubbers of equivalent load rating and pin-to-pin dimension. The Bergen-Patterson snubbers were replaced due to age degradation and obsolescence. The new snubbers were selected based on equivalency of design. Additionally, the new snubbers enhanced design qualities related to inspection and preventive maintenance requirements. The review was performed to determine whether the design bases, licensing bases, and performance capability of systems and components supported by the snubbers had been degraded by the modification. Additionally, the 10 CFR 50.59 screen associated with this modification was reviewed as described in section 1.1 of this report.

The team assessed selected design attributes to determine whether they were consistent with the design and licensing bases. These attributes included component safety classification, load rating and load requirements, hydraulic fluid viscosity, allowable displacement, and snubber inspection requirements. The team evaluated design assumptions in the supporting evaluations and analyses to determine whether they were technically appropriate and consistent with the UFSAR. The team reviewed selected evaluations, drawings, analyses, procedures, and the UFSAR to determine whether they were properly updated with any revised design information. In addition, the team interviewed the responsible design and system engineers to discuss vendor acceptance testing of the snubbers, as well as snubber installation and post-installation inspection. Finally, the team walked down a sample of Lisega snubbers to detect any potentially abnormal installation conditions. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2.7 Main Boiler Feed Pump Temperature Control Valve Modifications

a. Inspection Scope

The team reviewed a modification to replace the temperature control valves (TCVs) on the seal water injection system for the main boiler feed pump. The modification was performed to increase the reliability of the automated temperature control feature, as well as provide more appropriately sized valves for temperature control of the seal water injection system. The review was performed to determine whether the design bases, licensing bases, and performance capability of the safety injection system had been degraded by the modification. Additionally, the 10 CFR 50.59 screen associated with this modification was reviewed as described in section 1.1 of this report.

The team assessed selected design attributes to determine whether they were consistent with the design and licensing bases. These attributes included component safety classification, automated set points, manual valve control features, and the ability to provide adequate seal water injection to ensure functionality of the main boiler feed pumps. The team evaluated design assumptions in the supporting evaluations and analyses to determine whether they were technically appropriate and consistent with the UFSAR. The team reviewed selected evaluations, drawings, work orders, procedures, and the UFSAR to determine whether they were properly updated with any revised design information. The team evaluated the post-modification tests to determine whether the new valves would function in accordance with design assumptions. In addition, the team interviewed the responsible design and system engineers to discuss the modification and the design requirements. Finally, the team walked down the new TCVs to detect any potentially abnormal installation conditions. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2.8 Modification to Install a Spacer Ring in Main Feedwater Valve

a. Inspection Scope

The team reviewed a modification to install a cage spacer in main feedwater flow control valve (FCV) 427, to prevent the valve cage from shifting in position while in service. The review was performed to determine whether the design bases, licensing bases, and performance capability of the safety injection system had been degraded by the modification. Additionally, the 10 CFR 50.59 screen associated with this modification was reviewed as described in section 1.1 of this report.

The team assessed selected design inputs and attributes to determine whether they were consistent with the design and licensing bases. These attributes included component safety classification, effect on valve flow coefficient and stroke time, material compatibility with feedwater chemistry, and evaluations for changes in piping stress. The team evaluated design assumptions in the supporting evaluations and analyses to determine whether they were technically appropriate and consistent with the UFSAR. The team reviewed selected evaluations, drawings, analysis, procedures, and the UFSAR to determine whether they were properly updated. The team evaluated the post-modification tests to verify that the valve's ability to stroke was not degraded by the modification. In addition, the team interviewed the responsible design and system engineers to discuss the modification and the design requirements. The team also walked down the main feedwater flow control valves to detect possible abnormal installation conditions. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (IP 71152)

a. Inspection Scope

The team reviewed a sample of condition reports associated with 10 CFR 50.59 issues and plant modification issues to determine whether Entergy was appropriately identifying, characterizing, and correcting problems associated with these areas, and whether the planned or completed corrective actions were appropriate. The condition reports reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

4OA3 Follow-up of Events and Notices of Enforcement Discretion (IP 71153 – 2 samples)

.a Inspection Scope

.1 (Closed) LER 05000247/2007005, Technical Specification Prohibited Condition Due to Exceeding the Allowed Completion Time for an Inoperable Recirculation Pump Caused by a Potential Strong Pump-Weak Pump Interaction During a Small Break Loss of Coolant Accident (SBLOCA)

On November 8, 2007, Unit 2 entered Technical Specification 3.5.2, "Emergency Core Cooling System," Condition A, for one or more Emergency Core Cooling (ECCS) trains inoperable. A condition was identified, during an NRC Component Design Bases Inspection, where a stronger internal recirculation pump could shut the discharge check valve of the weaker internal recirculation pump, causing the weaker pump to deadhead. This condition applied to certain accident scenarios with conditions of high pump head and low flow, such as during a SBLOCA. Immediate actions were taken to declare one train of the internal recirculation system inoperable, and revise Emergency Operating Procedures (EOPs) to eliminate the requirement to start a second internal recirculation pump. The team reviewed the LER, as well as the corrective actions to the EOPs to verify that the changes were adequate. The team also reviewed additional procedures, calculations, condition reports, corrective actions, and conducted interviews with engineering staff to verify that the condition was adequately corrected. The team determined that Entergy's failure to evaluate the internal recirculation pumps for adequate minimum flowrates was a finding of very low safety significance (Green) involving a non-cited violation (NCV) of 10 CFR 50, Appendix B, Design Control (see section 4OA5.1b below). This LER is closed.

.2 (Closed) LER 05000286/2007003, Technical Specification Prohibited Condition Due to Exceeding the Allowed Completion Time for an Inoperable Recirculation Pump Caused by a Potential Strong Pump-Weak Pump Interaction During a Small Break Loss of Coolant Accident (SBLOCA)

On November 8, 2007, the Unit 3 internal recirculation pump no. 31 was declared inoperable and Technical Specification 3.5.2, "Emergency Core Cooling System," Condition A, was entered for one or more Emergency Core Cooling (ECCS) trains inoperable. A condition was identified, during an NRC Component Design Bases Inspection, where a stronger internal recirculation pump could shut the discharge check valve of the weaker internal recirculation pump, causing the weaker pump to deadhead. This condition applied to certain accident scenarios with conditions of high pump head and low flow, such as during a SBLOCA. Immediate actions were taken to declare one train of the internal recirculation system inoperable, and revise Emergency Operating Procedures (EOPs) to eliminate the requirement to start a second internal recirculation pump. The team reviewed the LER, as well as the corrective actions to the EOPs to verify that the changes were adequate. The team also reviewed additional procedures, calculations, condition reports, corrective actions, and conducted interviews with engineering staff to verify that the condition was adequately corrected. Also see section 4OA5.1a below for additional inspection activity related to this Unit 3 LER. The team determined that Entergy's failure to evaluate the internal recirculation pumps for adequate minimum flowrates was a finding of very low safety significance (Green) involving an NCV of 10 CFR 50, Appendix B, Design Control. (see section 40A5.1b below) This LER is closed.

b. Findings

See section 4OA5.1b for the finding related to LERs 05000247/2007005 and 05000286/2007003.

4OA5 Other Activities

.1 (Closed) URI 05000286/2007006-02: Inadequate Design Control of Recirculation Pumps

a. Inspection Scope

During the Unit 3 Component Design Bases Inspection (CDBI) performed in 2007, the team identified an unresolved item (URI) concerning the adequacy of design control associated with a modification that replaced both internal recirculation pumps (low pressure recirculation (LPR) pumps 31 and 32, or 31 LPR pump and 32 LPR pump) in March 2007. Specifically, Entergy did not assess two critical design parameters associated with design basis requirements for the pumps: minimum flow requirements for sustained pump operation under low flow conditions, which involved design flow rates for small break loss-of-coolant accidents (SBLOCA) that were potentially below the vendor recommended flow rates for sustained operation of the pumps; and strong-pump to weak-pump interactions that could result in parallel pump dead-heading of the weaker pump. With respect to conditions of parallel pump operation that result in a strong-pump to weak-pump interaction, the weaker pump will become dead-headed without an adequately sized minimum flow line. As a result of the NRC-identified issue, Entergy determined that the weaker pump was only susceptible to dead-heading during SBLOCA scenarios involving high head recirculation. Immediate corrective actions were taken by Entergy to address this performance deficiency. URI 2007006-02 was opened to allow an integrated NRC review of the LPR pump's prior operability with respect to pump dead-heading, and also with respect to Entergy's evaluation of the LPR pumps sustained minimum flow requirements, which was still ongoing at the completion of the CDBI inspection in December 2007.

During this inspection, the team completed the integrated review of both the sustained minimum flow and the dead-heading issues. The team reviewed procedures, design basis documents, calculations, condition reports, corrective actions, and conducted interviews with engineering staff to verify measures were established to maintain design basis requirements with respect to:

- the sustained minimum flow issue. The team reviewed recirculation system hydraulic models performed by Entergy for SBLOCA scenarios to determine the expected minimum core flows and individual pump flows. The team also reviewed evaluations performed by the pump vendor, Flowserve, to evaluate the sustained minimum flow requirements of the new internal recirculation pumps during SBLOCA scenarios. Based on review of Entergy's analyses and Flowserve's evaluations, the team was able to verify that individual pump flows during SBLOCA scenarios would be sufficient to meet the sustained minimum flow requirements for the pumps to operate successfully. The team noted the analysis for LPR pump sustained minimum flow was performed on both units.
- the LPR pump dead-heading issue. The team reviewed completed surveillance test data and vendor pump curve data. See the discussion under "Description" in section 4OA5.1.b.

Based on the team's review of the Entergy analysis of the sustained minimum flow issue and the corrective actions taken to address the dead-heading issue, this unresolved item is closed.

b. Findings

Introduction: The team identified a finding of very low safety significance (Green) involving a non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion III, "Design Control," at both Unit 2 and Unit 3, because Entergy did not verify the adequacy of the internal recirculation pump minimum flow rates. Specifically, Entergy did not verify the adequacy of the pump minimum flow rates for sustained operation under low flow rate conditions or for strong-pump to weak-pump interactions.

Description: For both units, the internal recirculation portion of the low-head safety injection system consists of two low pressure recirculation (LPR) pumps, located in primary containment, that take suction from a containment sump and discharge into a common header. Each LPR pump has a 3/4-inch minimum flow line upstream of the pump discharge check valve, and the two pumps share a 2-inch minimum flow line on the common discharge header. All three minimum flow lines return to the containment sump. With respect to system operation, prior to December 2007, the EOPs directed operators to sequentially start both recirculation pumps during the recirculation phase of any loss-of-coolant accident (LOCA).

NRC Bulletin 88-04, "Safety-Related Pump Loss," documented industry operating experience regarding design deficiencies involving a weaker pump (i.e., low discharge head at a given flow rate) that could be dead-headed when operated in parallel with a stronger pump (i.e., higher discharge head at the equivalent flow rate), under low flow conditions, for system configurations where both pumps share a common minimum flow line. Letter IP3-89-036, dated May 12, 1989, provided the licensee's Bulletin 88-04 response to the NRC. The licensee stated that although the recirculation pumps shared a common minimum flow line, the potential for a stronger pump to dead-head a weaker pump did not exist. The basis, in part, was that having the individual pump minimum flow lines upstream of the pump discharge check valve would ensure flow through the pump even if the stronger pump would cause the discharge check valve on the weaker pump to close. The licensee also credited the EOPs with preventing the weak pump from becoming dead-headed, based on an assumption that by the time the EOPs directed starting of the second pump, flow to the reactor core would be sufficient to allow both pumps to operate at a point on their performance curves where there was adequate flow for both pumps.

In December 2007, following NRC identification of potential parallel pump dead-heading of the LPR pumps at Unit 3, Entergy took actions to prevent the parallel operation of the internal LPR pumps. Subsequent action was taken by Entergy at Unit 2 upon confirmation of a similar configuration. Entergy entered this issue into their corrective action program as CR-IP2-2007-04558 and CR-IP3-2007-04212. As an immediate corrective action, Entergy revised EOPs 2-ES-1.2 and 2-ES-1.3, "Transfer to Cold Leg Recirculation," and also 2-ES-1.4 and 3-ES-1.4, "Transfer to Hot Leg Recirculation," so that the second internal recirculation pump would not be started during conditions of high head recirculation on either unit.

The team concluded that Entergy, as part of the Unit 3 modification in 2007 and the Unit 2 modification in 2000 which installed two new LPR pumps on each unit, had not evaluated the design for strong-pump to weak-pump interaction. Regarding Unit 3, the team determined, based on a review of vendor supplied pump performance curves and pump surveillance data, that the 31 LPR pump was susceptible to dead-heading if both the 31 and 32 LPR pumps were operated in parallel during certain SBLOCA scenarios involving high head recirculation, as required by EOPs. The team's review of the new recirculation pump performance curves identified that the 32 LPR pump had approximately 10 pounds-per-square-inch (psi) greater discharge pressure, under low flow conditions, than the 31 LPR pump. The team noted that the installed 3/4 inch minimum flow valve was throttled to 1.5 turns open, resulting in an as-found 0.1 gallons-per-minute (gpm) flow. This low flow rate would not have been sufficient to prevent pump damage if the 31 LPR pump discharge check valve closed due to the higher discharge pressure for the 32 LPR pump.

In addition, the previous engineering evaluation for potential strong-pump to weak-pump interaction of the recirculation pumps appeared to be inconsistent with Entergy's most current SBLOCA accident analysis performed as a result of the NRC-identified issue, and also inconsistent with the current throttled configuration of the 3/4 inch minimum flow line.

Regarding Unit 2, the team determined that it was unlikely that the 21 and 22 LPR pumps were susceptible to parallel pump dead-heading, based on vendor pump curves and surveillance test data, which showed that the current pump discharge pressures were nearly equivalent for low flow conditions.

As noted in section 40A5.1a, Entergy performed an analysis for both units which determined the individual LPR pump flows during SBLOCA scenarios would be sufficient to meet the sustained minimum flow requirements for the pumps.

<u>Analysis:</u> The team determined that Entergy's failure to evaluate the LPR pumps for suitability of application to the internal recirculation system configuration at Unit 2 and Unit 3 constituted a performance deficiency and a finding. Absent the 2007 NRC CDBI identification of the issue at Unit 3, the similar issue at Unit 2 would likely have remained undiscovered. The finding is greater than minor because it is associated with the design control attribute of the Mitigating Systems (MS) Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage).

Unit 3: Using Phases 1 and 3 of the NRC's Significance Determination Process, the team determined the significance of the 31 LPR pump susceptibility to parallel pump dead-heading, between March 2007 and December 2007. The team evaluated this finding using NRC Inspection Manual Chapter (IMC) 0609.04, "Phase 1 – Initial Screening and Characterization of Findings." Using the Table 4a characterization worksheet for the MS Cornerstone, the finding was determined to represent an actual loss of a safety function for a single LPR train for greater than the Technical Specification allowed outage time because of the differences in pump performance, during certain SBLOCA scenarios that required high pressure recirculation (HPR). Accordingly, this issue required evaluation under Appendix A to IMC 0609.

A Region I Senior Reactor Analyst (SRA) completed a Phase 3 risk assessment determining that this issue was of very low safety significance (Green). The Phase 3 assessment was conducted because the issue was not suitable to a Phase 2 analysis. The 31 LPR pump was assumed to fail internally, due to insufficient minimum pump flow (pump damage), if the 32 LPR pump also was started in SBLOCA initiating events when entering high pressure recirculation. The operation of the 31 LPR pump would not have been affected if the 32 LPR pump failed to start independently or because it did not have electrical power. The SRA used the IP3 Standardized Plant Analysis Review (SPAR) model version 3.45 to complete an internal events review. As a bounding case, the SRA assumed that the 31 internal LPR pump would fail to run in all SBLOCA initiating events. The SRA then reviewed the increase in core damage probability for sequences where HPR was assumed to fail. The dominate core damage sequence was a SBLOCA with: success of AFW and high pressure injection, failure to cooldown, and subsequent failure of HPR. The estimated increase in core damage probability, given the nine month exposure period (March to December 2007), was very small: four-orders of magnitude below the 1E-6 per year Green-White risk significance threshold (E-10 per year).

The cause of this finding had a cross-cutting aspect in the area of Problem Identification and Resolution because Entergy did not implement operating experience information through changes to station processes, procedures, and equipment (P.2.(b)). Specifically, during the recent modification to the internal recirculation pumps, Entergy did not sufficiently review their original response to NRC Bulletin 88-04 regarding the potential dead-heading of safety related pumps. Additionally, previous Licensee Event Reports (LERs) from other stations document that the same strong-pump to weak-pump interaction has occurred at other power reactor plants within the industry.

Unit 2: The team determined that both LPR pumps (21 and 22) were not likely susceptible to parallel pump dead-heading during certain SBLOCA scenarios, based on vendor pump curves and current surveillance test data, and therefore would have delivered adequate coolant flow to the reactor core as required by Emergency Operating Procedures. The team evaluated this finding using NRC Inspection Manual Chapter (IMC) 0609.04, "Phase 1 – Initial Screening and Characterization of Findings." Using the Table 4a characterization worksheet for the MS Cornerstone, the finding was determined to be of very low safety significance (Green) because it was a design or qualification deficiency confirmed not to result in loss of operability or functionality.

This deficiency was not indicative of current performance because the modification on Unit 2 was performed in May of 2000. Therefore, there was no cross-cutting aspect associated with this finding.

Enforcement: 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that measures be established for verifying or checking the adequacy of design such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. Contrary to the above, Entergy replaced the internal recirculation pumps during modifications on Unit 3 in March of 2007 and on Unit 2 in May 2000, and did not verify the design adequacy of the pump minimum flow rates for sustained operation under low flow rate conditions or for strong-pump to weak pump interactions which could result in dead-heading the weaker pump during parallel pump operation. This condition existed until identified by the NRC in December of 2007, resulting in subsequent corrective actions by Entergy to revise the EOPs, as described above. Because this finding was of very low safety significance and was entered into the corrective action program as CR-IP2-2007-4558, and as CR-IP3-2007-4212, this violation is being treated as an NCV, consistent with section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000247/2008012-01, and NCV 05000286/2008010-01, Inadequate Design Control of Internal Recirculation Pumps)**

.2 (Closed) URI 05000247/2007007-03: Use of Motor Control Center (MCC) Methodology for Periodic Verification of the Design Basis Capability of Safety-Related Motor Operated Valves (MOVs)

a. Inspection Scope

During a Component Design Bases Inspection (CDBI) performed in 2007, the team identified an unresolved item (URI) concerning the adequacy of MCC testing methodology for MOVs. Specifically, Entergy did not use the testing methodology approved by the NRC as part of the Generic Letter (GL) 96-05 reviews, which required direct measurements of stem thrust and torque to be recorded at-the-valve. The URI was opened to determine if the results from the MCC testing methodology could adequately show that the design basis of the MOVs was maintained. The team interviewed the system engineer and found that following NRC-identification of the issue, Entergy suspended the MCC testing program, and subsequently re-tested all valves that had been previously tested using the MCC testing methodology. The re-test used the GL 96-05 testing methodology, and the team verified that the MOVs had maintained their design basis capability.

Additionally, the team reviewed the licensee's commitments as described in their response to GL 96-05 and determined that Entergy had committed to the at-the-valve testing methodology. The team concluded that prior to implementing the MCC testing methodology, Entergy was required to submit a change to the GL commitment. The team found that because the testing methodology did not conform to all the requirements outlined in the methodology basis documents, and the testing had not been previously approved by NRC, a violation of NRC requirements had occurred. However, because the retest determined that the valves had maintained their design basis capability, the team concluded that the associated finding was of minor significance that was not subject to enforcement action per section IV.B of the Enforcement Policy. This URI is closed.

b. Findings

No findings of significance were identified.

4OA6 Meetings, including Exit

The team presented the inspection results to Mr. T. Orlando, Director of Engineering, and other members of Entergy's staff at an exit meeting on August 14, 2008. The team verified that this report does not contain proprietary information.

ATTACHMENT

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

H. Anderson	Licensing Specialist
F. Bloise	Senior Design Engineer
G. Dahl	Licensing Specialist
J. Hill	Design Engineering Supervisor, I&C
T. McCaffrey	Design Engineering Manager
V. Myers	Design Engineering Supervisor, Mechanical
T. Orlando	Director of Engineering
A. Vitale	General Manager of Plant Operations
R. Walpole	Licensing Manager
A. Williams	Managers of Operations
J. Bencivenga	Senior Design Engineer

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Open and Closed

05000247/2008012-01	NCV	Inadequate Design Control of Internal Recirculation Pumps (Section 4OA5.1)
05000286/2008010-01	NCV	Inadequate Design Control of Internal Recirculation Pumps (Section 4OA5.1)

Closed

05000247/2007005	LER	Technical Specification Prohibited Condition Due to Exceeding the Allowed Completion Time for an Inoperable Recirculation Pump Caused by a Potential Strong Pump-Weak Pump Interaction During a Small Break Loss of Coolant Accident (Sections 4OA3.1)
05000286/2007003	LER	Technical Specification Prohibited Condition Due to Exceeding the Allowed Completion Time for an Inoperable Recirculation Pump Caused by a Potential Strong Pump-Weak Pump Interaction During a Small Break Loss of Coolant Accident (Section 4OA3.2)

05000247/2007007-03	URI	Use of Motor Control Center Methodology for Periodic Verification of the Design Basis Capability of Safety-Related MOVs (Section 4OA5.2)
05000286/2007006-02	URI	Inadequate Design Control of Internal Recirculation Pumps (Section 4OA5.1)

LIST OF DOCUMENTS REVIEWED

Section 1R017: Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications

10 CFR 50.59 Evaluations
07-2002-01-Eval, 10 CFR 72.212 Report Appendix F: New Licensing Basis Document for IPEC ISFSI, Rev. 1

10 CFR 50.59 Screened-out Evaluations
0-AOP-SEC-2, Aircraft Threat, Rev. 4
2-PT-M021A, Emergency Diesel Generator 21 Load Test, Rev. 17
2-PT-M108R04, RHR/SI System Venting, dated 4/19/08
2-PT-Q024B, 22 EDG Fuel Oil Transfer Pump, Rev. 10
2-PT-Q033A, 21 Charging Pump, Rev. 13
2-PT-R007AR20, Motor Driven AF Pump Full Flow, dated 1/22/08
2-SOP-27.3.1.1 21 Emergency Diesel Generator Manual Operation, Rev. 21
EC 5456, Revision to the 22 AFP Turbine Overspeed Set Point Lower Tolerance, Rev. 0
EOPs E-0 through ES-3.2, Westinghouse Owners Group Changes to Revision Number 2 of the EOPs (All procedures are Rev. 0)
ER-04-2-072, Main Boiler Feed Pump Seal Injection System Upgrade, Rev. 0
ER-05-2-137, Increase Reliability of the Stator Water Cooling Generator, Rev. 0
ER-06-2-027, Increase Recirculation Pump flows to meet IST Code Requirements by 2008, dated 4/22/08
ER-06-2-031, 118V AC/ 118V AC Electrical (Replacement of 2 Pole HFB Bkrs in IP2 125V DC Power Panel 23), Rev. 0
ER-06-2-048, Installation of ¾” Vent Valve Downstream of SI-MOV-888A/B, Rev. 0
ER-06-2-058, Hydraulic Snubber Replacements, Rev. 0
ER-06-2-115, Install Surge Suppressors on Relays to Defeat 21 and 22 MBFP, Rev. 0
ER-06-2-141, DC/ 125 DC System (Removing Delta Expansion Turbine Trip), Rev. 0
ER-07-2-047, FCV-427 Anti-Rotation Device, Rev. 0
IP2-03-24983, Power Uprate: Setpoint Changes, dated 1/3/07
IP-CALC-06-00218, AST Analysis for a Design-Basis Stem Generator Tube Rupture Analysis, Rev. 0
IP-SMM-AD-102, IPEC Implementing Procedure Preparation, Review, and Approval – Attachment 10.2: Core Operation Limits Report (COLR), Rev. 5
SCR-07-2-058, Set Point Change Number 07-2-058, Internal Recirculation Pump Level Transmitter Modification, Rev. 0
SPDDF-PC-439AR01, ESFAS Actuation on High Differential Steam line Pressure, dated 11/27/06

Modification Packages
ER-04-2-072, Main Boiler Feed Pump Seal Injection System Upgrade, Rev. 0
ER-05-2-137, Increase Reliability of the Stator Water Cooling Generator, Rev. 0
ER-06-2-048, ¾-inch Vent Line Install, Rev. 0
ER-06-2-058, Hydraulic Snubber Replacements, Rev. 0
ER-06-2-031, Replacement of 2 Pole HFB Bkrs in IP2 125V DC Power Panel 23, Rev. 0
ER-06-2-141, Removing Delta Expansion Turbine Trip, Rev. 0
ER-07-2-047, FCV-427 Anti-Rotation Device, Rev. 0
SCR-07-2-058, Set Point Change Number 07-2-058, Internal Recirculation Pump Level Transmitter Modification, Rev. 0

Calculations & Analysis
IP-CALC-07-00184, SIS Valve Operation Inside the Vapor Containment, Rev. 0
IP-CALC-06-00218, AST Analysis for a Design-Basis Steam Generator Tube Rupture Accident, Rev. 0
FIX-00046, Calibration of Turbine Inlet Pressure and High Steam Flow (SF)/ Safety Injection Components for Stretch Power Uprate, Rev. 03P
FIX-00129, Turbine Inlet Pressure Transmitter Static Head Sealing and Calibrations, Rev. 5
GMS-00035, Stress Analysis of Line 60 Due to Addition of Vent Valve Downstream of 888A and 888B, Rev. 0

Drawings
A225105, Logic Diagram – Safeguards Actuation Signals, Rev. 10
A225106, Logic Diagram – Feedwater Isolation, Rev. 7
ISI-2735, In-Service Inspection Program – Safety Injection System, Rev. 1
220619, Instrument and Control Loop Diagram Safety Injection System Loop 938 and 939, Rev. 2
9321-F-2019-114, Flow Diagram – Boiler Feedwater, 12/16/87

Drawing Change Notice (DCN)
EC-7052, Model D-1008-160-2 Valve Assembly (FCV-427), 04/04/08

Surveillance and Modifications Acceptance Tests
2-PT-Q62, High Steam Flow and Turbine First Stage Pressure Bistables, Rev. 14
2-PC-R19, Turbine First Stage Pressure, Rev. 21
PC-R19, Turbine First Stage Pressure, Rev. 19
PT-Q62, High Steam Flow and Turbine First Stage Pressure Bistables, Rev. 13

Audits and Self-Assessments
QA-04-2008-IP-1, Engineering Design Control, Rev. 0

Procedures
0-CY-1640, Chemistry Shutdown Plan, Rev. 17
0-CY-1645, Chemistry Response to Plant Causalities, Rev. 5
0-CY-2350, Primary System Zinc Injection, Rev. 2
0-RES-401-GEN, Lisega Snubber Installation and Removal, Rev. 1
2-ARP-SEF, Turbine and GE Generator Start-up, Rev. 55
2-PI-V001A, Inaccessible Snubber Inspections, Rev. 15
2-PI-V001B, Accessible Snubber Inspections, Rev. 14

2-PT-M108, RHR/SI System Venting, Rev. 4
2-PT-R002B, Recirculation Sump Level, Rev. 18.
2-PT-R016, Recirculation Pumps, Rev. 20
2-PT-Q033A, 21 Charging Pump, Rev. 13
2-PT-Q62, High Steam Flow and Turbine First State Pressure Bistables, Rev. 14
2-SOP-3.1, Charging Seal Water and Letdown Control, Rev. 61
2-SOP-3.5, Placing CVCS Demineralizers in or out of Service, Rev. 22
EN-DC-117, Post Modification Testing and Special Instructions, Rev. 1
EN-LI-100, Process Applicability Determination, Rev. 7
EN-LI-101, 10 CFR 50.59 Review Program, Rev. 4
PT-V11A-4, Recalibration of NIS and OT/OP Delta T Parameters Channel IV, Rev. 14

Work Orders
51229162
51326377
00144204

Work Requests
128436
128439

Vendor Manuals
IB 56-352-400, TURBO-GRAF – Turbine Supervisory Instruments Differential Expansion
IP 56-352-340A, TURBO-GRAF –Turbine Supervisory Instruments Casing Expansion / Differential Expansion

Miscellaneous
05-0299-MD-00-RE, 50.59 Evaluation for IP3 Cycle 14 Core Reload Design, Rev. 1
ER 03-2-217, Setpoints, Rev. 0
Final Report, Control Room Envelope In-leakage Testing at Indian Point 2 Nuclear Generating Station, dated 02/00
Indian Point Nuclear Generating Unit No. 2 – Issuance of Amendment RE: 3.36 percent Power Uprate (TAC No. MC 1865), dated 10/27/04
Indian Point 2 Improved Technical Specifications
Indian Point 2 Improved Technical Specifications
IPEC Top 10 Technical Issue: IPEC Power Supply PM's, Rev. 2
IP2-FW/SGL DBD, Feedwater System / Steam Generator Control System Design Basis Document, Rev. 1
Letter from Consolidated Edison Company to NRC, NEI Pilot Program for use of NURGEG-1465, dated 04/13/00
Letter from NRR to Entergy, Indian Point Nuclear Generating Unit No. 2 – Relief Request P-2 on Testing of Recirculation Pumps, dated 04/01/08
Lisega: Shock Absorbers Rigid Struts '93, April 1996 Edition
Letter, Lake Engineering Co. to Entergy, Seal Life Evaluation of Bergen-Paterson Snubbers Entergy Nuclear Contract No. 4500543558 – Change 1 Lake Engineering Company Project No. 948, dated 12/28/05
Letter, USNRC to Consolidated Edison Company: Issuance of Amendment Number 173 for Indian Point Nuclear Generating Unit 2, dated 07/26/94
NF-IP-07-25, Indian Point Unit 2 Cycle Core 19 Loading Plan, 03/24/08
PFP-212, General Floor Plan – Primary Auxiliary Building, Rev. 7

QA-04-2008-IP-1, Quality Assurance Audit Report: Engineering Design Control
Updated Final Safety Analysis Report: Indian Point Unit 2, Rev. 20
WCAP-16157-P, Indian Point Nuclear Generating Unit No. 2 Stretch Power Uprate NSSS and BOP Licensing Report, Rev. 0
Westinghouse Certification of Conformance for Breaker RHFA3100Y, dated 03/28/08

Section 4OA2: Identification and Resolution of Problems

Condition Reports (* denotes NRC identified during this inspection)

IP2-2003-00231	IP2-2007-01208	IP2-2007-02208	IP2-2008-01056
IP2-2008-01414	IP2-2008-01581	IP2-2008-01822*	IP2-2008-02011
IP2-2008-02509	IP2-2008-03778*	IP2-2008-03801*	

Section 4OA3: Event Followup

IP 2 LER 2007-005-00: Technical Specification Prohibited Condition due to Exceeding the Allowed Completion Time for an Inoperable Recirculation Pump caused by a Potential Strong Pump-Weak Pump Interaction During a Small Break LOCA, 01/07/08
IP 3 LER 2007-003-00: Technical Specification Prohibited Condition due to Exceeding the Allowed Completion Time for an Inoperable Recirculation Pump caused by a Potential Strong Pump-Weak Pump Interaction During a Small Break LOCA, 01/07/08

Section 4A05: Other Activities

10 CFR 50.59 Screened-out Evaluations
EC 5682, Revision of Procedures EOP ES-1.3 and ES-1.4, 02/12/08

Condition Reports

IP2-2007-04212	IP2-2007-04296	IP2-2007-04411	IP2-2007-04558
IP2-2007-04670	IP2-2007-04905	IP3-2007-04411	

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2-ES-1.3, Transfer to Cold Leg Recirculation, Rev. 1
2-ES-1.4, Transfer to Hot Leg Recirculation, Rev. 1
2-PT-R016, Recirculation Pumps, Rev. 20
3-ES-1.3, Transfer to Cold Leg Recirculation, Rev. 1
3-ES-1.3, Transfer to Hot Leg Recirculation, Rev. 2
3PT-R013, Recirculation Pumps In-Service Test, Rev. 19
EN-DC-313, Procurement Engineering Process, Rev. 2
EN-DC-141, Design Inputs, 07/24/06
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EN-MP-101, Materials, Purchasing, and Contracts Process, Rev. 2
EN-MP-121, Materials, Purchasing and Contracts Training, Qualification and Certification, Rev. 1
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280-RLCA02848-02A, Unit 3 Internal Recirculation Pump Curves, 01/16/07
IP-CALC-04-00809, Attachment 10, Unit 2 Internal Recirculation Pump Curves, 01/11/00

IP-RPT-04-00890, Technical Basis for Using MC2 Technology for Periodic Verification Testing at Indian Point 2 and Indian Point 3, Rev. 02
IP-RPT-08-00009, Engineering Study for Pump Minimum Flow Evaluation – Safety Injection Recirculation Pumps, 01/29/08
IPEC Licensed Operator Requalification Training Program: E-1 and FR-P Series EOPs, 06/25/08
Letter from Consolidated Edison Company to NRC, Completion of Licensing Action for Generic Letter 96-05 Regarding Capability of Motor-Operated Valves, Indian Point Nuclear Generating Unit No. 2 (TAC No. M97057), dated 03/05/01
NRC Bulletin 88-04: Potential Safety-Related Pump Loss, 05/05/88
NRC Inspection Report 05000286/2007006, Indian Point Unit 3 Component Design Bases Inspection (CDBI), 02/01/08
NRC Regulatory Issue summary 2000-17, Managing Regulatory Commitments Made by Power Reactor Licensees to the NRC Staff
PS 98-002, Procurement Specification for Replacement of Two Containment Recirculation Pumps, 04/08/99
SAO 270, Indian Point Station Procurement Program, 06/19/99
STR-27, Indian Point Energy Center MC2 Program Questions, Rev. 0

LIST OF ACRONYMS

ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
DBA	Design Basis Accident
DC	Direct Current
ECCS	Emergency Core Cooling System
EOP	Emergency Operating Procedure
FCV	Flow Control Valve
gpm	Gallons per Minute
HPR	High Pressure Recirculation
IMC	Inspection Manual Chapter
IPEC	Indian Point Energy Center
IR	Inspection Report
LER	Licensee Event Report
LOCA	Loss-of-Coolant Accident
LPR	Low Pressure Recirculation
MCC	Motor Control Center
MOV	Motor Operated Valve
MS	Mitigating System
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
SBLOCA	Small Break Loss-of-Coolant Accident
SDP	Significance Determination Process
SPAR	Standardized Plant Analysis Review
SRA	Senior Reactor Analyst
SSC	Structures, Systems and Components
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item