UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
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January 13, 2012

Mr. John Ventosa
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 NUCLEAR GENERATING UNITS 2 AND 3 – NRC EVALUATION OF CHANGES, TESTS, OR EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS TEAM INSPECTION REPORT 05000247/2011007 AND 05000286/2011007

Dear Mr. Ventosa:

On December 1, 2011, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Indian Point Nuclear Generating Units 2 and 3. The enclosed inspection report documents the inspection results, which were discussed on December 1, 2011, with Mr. L. Coyle and other members of your staff, and during a subsequent telephone call with Mr. P. Conroy on January 12, 2012.

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. In conducting the inspection, the team reviewed selected procedures, calculations and records, observed activities, and interviewed 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 the finding as a non-cited violation (NCV) consistent with Section 2.3.2 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 I; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Senior Resident Inspector at Indian Point Nuclear Generating Unit 2. In addition, if you disagree with the cross-cutting aspect assigned to the finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis of your disagreement, to the Regional Administrator, Region I, and the NRC Senior Resident Inspector at Indian Point Nuclear Generating Unit 2.

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, Agencywide Documents Access and Management 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,

Lawrence T. Doerflein

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

Docket No. 50-247, 50-286
License No. DPR-26, DPR-64

Enclosure:
Inspection Report 05000247/2011007; 05000286/2011007
w/Attachment: Supplemental Information

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Sincerely,

/RA/

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

Docket No. 50-247, 50-286
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Enclosure:
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REGION I

Docket Nos.:	50-247, 50-286
License Nos.:	DPR-26, DPR-64
Report No.:	05000247/2011007 and 05000286/2011007
Licensee:	Entergy Nuclear Northeast (Entergy)
Facility:	Indian Point Nuclear Generating Units 2 and 3
Location:	450 Broadway, GSB Buchanan, NY 10511-0249
Inspection Period:	November 14, 2011 – December 1, 2011
Inspectors:	E. Burket, Reactor Inspector, Division of Reactor Safety (DRS), Team Leader F. Arner, Senior Reactor Inspector, DRS D. Orr, Senior Reactor Inspector, DRS J. Richmond, Senior Reactor Inspector, DRS J. Brand, Reactor Inspector, DRS M. Orr, Reactor Inspector, DRS
Approved By:	Lawrence T. Doerflein, Chief Engineering Branch 2 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000247/2011007, 05000286/2011007; 11/14/2011-12/01/2011; Indian Point Nuclear Generating Units 2 and 3; Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications.

This report covers a two week on-site inspection period of the evaluations of changes, tests, or experiments and permanent plant modifications. The inspection was conducted by six region based engineering inspectors. One finding of very low risk significance (Green) was identified. The finding was also considered to be a non-cited violation (NCV). The significance of most findings is indicated by their color (Green, White, Yellow, Red) using NRC Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). The cross-cutting aspect was determined using IMC 0305, "Operating Reactor Assessment Program." 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.

Cornerstone: Barrier Integrity

- Green. The team identified a non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," in that Entergy did not ensure that design changes, including field changes, were subject to design control measures commensurate with those applied to the original design. Entergy implemented an instrument setpoint change, but delayed re-calibration of the in-field setpoint values and did not evaluate the adequacy of the in-field actual setpoints, which were later found outside the value required by the design basis. Specifically, Entergy revised surveillance procedures for the Unit 2 reactor protection system (RPS) over-power delta-temperature (OPdT) instrument to use a setpoint value specified in the Core Operating Limits Report (COLR). However, the procedures were not required to be performed until the next regularly scheduled surveillance period. Technical Specification 3.3.1 requires the allowable values to be set as specified by the COLR. Two of the four instrument channels had in-field values outside of the required allowable value. Entergy entered this issue into their corrective action program and performed an immediate operability evaluation and determined that the OPdT instrument was capable of performing its intended functions with the current in-field values.

  The team determined that the failure to ensure in-service instrument setpoint values satisfied design and licensing basis requirements was a performance deficiency. This issue was more than minor because it was associated with the design control attribute of the Barrier Integrity Cornerstone and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers (e.g., fuel cladding) protect the public from radionuclide releases caused by accidents or events. The team performed a Phase 1 Significance Determination Process screening, in accordance with NRC IMC 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," and determined the finding was of very low safety significance (Green) because it affected only fuel barrier portion of the barrier integrity cornerstone.

The team determined that this finding had a cross-cutting aspect in the area of Human Performance, Work Practices because Entergy did not ensure adequate supervisory or management oversight of a design change. [IMC 0310, Aspect H.4(c)] (Section 1R17.02.1)

Other Findings

None

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

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

.1 Evaluations of Changes, Tests, or Experiments (Unit 2: 22 samples; Unit 3: 20 samples)

a. Inspection Scope

The team reviewed four safety evaluations (two per unit) 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 requirements. In addition, the team evaluated whether Entergy had been required to obtain NRC approval prior to implementing the changes. The team interviewed plant staff and reviewed supporting information including calculations, analyses, design change documentation, procedures, the UFSAR, the Technical Specifications (TS), and plant drawings to assess the adequacy of the safety evaluations. The team compared the safety evaluations 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 evaluations.

The team also reviewed a sample of thirty-eight (20 for Unit 2 and 18 for Unit 3) 10 CFR 50.59 screenings 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 included design changes, calculations, and procedure changes.

The team reviewed the safety evaluations that Entergy had performed and approved during the time period covered by this inspection (i.e., since the last modifications inspection) not previously reviewed by NRC inspectors. The screenings and applicability determinations were selected based on the safety significance, risk significance, and complexity of the change to the facility.

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 reviewed safety evaluations and screenings are listed in the Attachment.

b. Findings

No findings were identified.

.2 Permanent Plant Modifications (9 samples per unit {18 total})

.2.1 Setpoint Change for Unit 2 Reactor Protection System Over-Power Delta-Temperature Instrument

a. Inspection Scope

The team reviewed a setpoint change associated with Engineering Change (EC) 5000034071 that revised the tolerance for a time constant in the Over-Power Delta-Temperature (OPdT) reactor protection system (RPS) instrument. The time constant tolerance was revised to make it consistent with the value specified in the Core Operating Limits Report (COLR). The setpoint change had been a corrective action previously identified in CR-IP2-2004-06713, to resolve inconsistencies between engineering calculations, surveillance tests and calibration procedures, and the COLR. Entergy determined that the actual field setpoint would be adjusted during the next routinely scheduled surveillance test.

The team assessed Entergy's technical evaluations, calculations, and design details, and interviewed engineering personnel to determine whether the revised time constant tolerance would allow the OPdT instrument to function in accordance with the design and licensing bases requirements. Calculations, analyses, and procedures affected by the setpoint change were reviewed to verify they had been properly updated. In addition, the team evaluated Entergy's determination that the setpoint change did not require immediate implementation. A review of condition reports (CR) was performed to determine whether there were any reliability or performance issues associated with the new time constant tolerance. Additionally, the 10 CFR 50.59 screening determination associated with this modification was reviewed as described in section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

Introduction: The team identified a finding of very low safety significance (Green), involving a non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," in that Entergy did not ensure that design changes, including field changes, were subject to design control measures commensurate with those applied to the original design. Specifically, Entergy implemented an OPdT instrument setpoint change which revised procedures but did not confirm that the in-field actual setpoint values were within the tolerance required by the design change. Entergy subsequently evaluated the adequacy of the in-field actual setpoint values after the team identified that those values were outside the value required by the design basis.

Description: In 2003, Unit 2 was transitioned to Improved Technical Specifications and adopted a COLR, which specified different values and tolerances for some RPS instrument setpoints than the previous custom technical specifications. In 2004, Entergy implemented a power up-rate at Unit 2. As part of the up-rate process, Westinghouse analyzed the setpoints for the OPdT and Over-Temperature Delta-Temperature (OTdT) RPS instruments. During an Entergy staff review of the Westinghouse analysis, Entergy identified that the OPdT time constant was not set as described in the COLR

(CR-IP2-2004-06713). In 2006, Setpoint Change Request SCR-06-2-009 was issued to resolve the discrepancy. In August 2011, Entergy identified that SCR-06-2-009 had not been completely implemented, and the value of the OPdT time constant had not been revised as intended. As a result, EC5000034071 was prepared and implemented to revise the setpoint of the OPdT time constant to a value consistent with the COLR.

EC5000034071 revised the OPdT time constant setpoint value in plant calculations and Instrumentation and Control (I&C) calibration and surveillance procedures to the value specified in the COLR. Specifically, the OPdT time constant was revised from a band of 9.7 to 10.3 seconds, to a band of 10.0 to 10.6 seconds in the applicable I&C and TS surveillance procedures. EC topic note detail 1.3 stated that immediate field implementation was not required (i.e., no post-modification surveillance test or re-calibration was needed). Based on interviews with Entergy staff, the team determined that Entergy design engineering had not reviewed the actual in-field setpoint values during the preparation or approval of the setpoint modification. Although the modification approved implementation of the setpoint change, it allowed the existing in-field actual setpoints to remain unchanged until the next regularly scheduled surveillance period. Specifically, the COLR documented the design basis requirements for the OPdT instrument setpoints, but the approved setpoint change did not evaluate a setpoint value different than specified in the COLR. The team identified that an actual in-field setting could have been as low as 9.7 seconds verses a required setting of greater than or equal to 10 seconds. Therefore, the team concluded that Entergy had not applied the same level of design control to the in-field actual setpoints that had been applied to the original design.

The 10 CFR 50.59 screen determined that no safety evaluation was required, in part, because the revised tolerance band for the time constant was equal to or conservative of the value provided in the COLR. The team concluded that the 10 CFR 59.59 screen did not evaluate the impact of allowing the existing setpoint to remain outside the design limit as required by the COLR.

The team identified that TS Table 3.3.1-1, Reactor Protection System Instrumentation Function 6, required the OPdT setpoints to be set as specified in the COLR. The COLR (i.e., 2-GRAPH-RPC-6, revision 13) Attachment 2, OPdT Allowable Value, specified that the OPdT time constant be equal to or greater than 10 seconds. The team reviewed the most recent as-left calibration results and identified that OPdT channels 2 and 4 were left set at 9.9 and 9.8 seconds, respectively; channels 1 and 3 were set at 10.2 and 10.3 seconds, respectively. In follow-up to NRC questions regarding extent-of-condition, Entergy determined that the Unit 3 surveillance procedures would allow the OPdT time constants to be set as low as 9.5 seconds; all four Unit 3 channels had as-left settings greater than 10 seconds. Entergy performed an immediate operability evaluation and determined that the OPdT instrument was capable of performing its intended functions with a time constant as short as 9.7 seconds. Entergy entered this issue into their corrective action program as CR-IP2-2011-06047 and CR-IP3-2011-05353.

The team reviewed a Westinghouse assessment, performed as part of Entergy's operability evaluation. Westinghouse determined that the conclusions of the Unit 2

UFSAR remained valid. To reach its conclusion, Westinghouse evaluated the Steamline Break Mass and Energy Release Analysis, Steam Generator Tube Rupture Analysis, and the Hot Full Power Steamline Break event. The team determined that Entergy's modification review of EC5000034071 and 10 CFR 50.59 screening only documented a review of the Hot Full Power Steamline Break event. In addition, the team noted Entergy's design group did not perform an interface review with either reactor engineering (responsible for assessing core thermal limits), or nuclear analysis (responsible for assessing affects on response time assumptions contained in design and licensing basis analysis). Entergy added these issues into the corrective action program (CAP) as CR-IP2-2011-06047.

Entergy subsequently identified that the I&C procedures for both the Unit 2 and Unit 3 OTdT RPS instruments also allowed the time constants to be set outside of the allowable values specified in the COLR. The most recent as-left calibration results for the Unit 2 OTdT time constants indicated that three of the four instrument channels were out of specification; the Unit 3 instrument channels were within allowable values. Entergy performed an immediate operability evaluation and determined that the Unit 2 OTdT instrument was capable of performing its intended functions with the as-left values of the time constants. Entergy entered this issue into their corrective action program as CR-IP2-2011-06070.

Analysis: The team determined that the failure to apply design control measures commensurate with those applied to the original design was a performance deficiency that was reasonably within Entergy's ability to foresee and prevent. Specifically, Entergy implemented a setpoint change, but allowed the existing in-field actual setpoints to remain unchanged until the next regularly scheduled surveillance period without evaluating the adequacy of the in-field actual setpoints, which were later found outside the value required by the design and licensing bases.

The finding is more than minor because it is associated with the Design Control attribute of the Barrier Integrity Cornerstone and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers (e.g., fuel cladding) protect the public from radionuclide releases caused by accidents or events. In addition, this issue was similar to example 3.k of NRC IMC 0612, Appendix E, "Examples of Minor Issues," which determined that calculation errors would be more than minor if, as a result of the errors, there was reasonable doubt on the operability of the component, or if significant programmatic deficiencies were identified that could lead to worse errors if uncorrected. For this issue, there was a reasonable doubt on the operability of the OPdT instrument, in that a knowledgeable engineer could not determine the acceptability of a shorter time constant without a detailed review of transient and accident analyses and analysis modeling assumptions.

The team performed a Phase 1 Significance Determination Process screening, in accordance with NRC IMC 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," and determined the finding was of very low safety significance (Green) because it only affected the fuel barrier portion of the barrier integrity cornerstone.

This finding had a cross-cutting aspect in the area of Human Performance, Work Practices, because Entergy did not ensure adequate supervisory or management oversight of a design change. Specifically, Entergy's design organization that performed this setpoint change activity considered the time constant value in the COLR to be a nominal value without appropriate tolerance. The design organization did not recognize that the COLR value was a design basis number required to be implemented by plant Technical Specifications. Entergy's oversight of this activity also did not identify the need for interfacing reviews by other organizations, such as reactor engineering or nuclear analysis. As a result, Entergy's oversight failed to ensure that potential non-compliances with design and licensing requirements were properly evaluated. [IMC 0310, Aspect H.4(c)]

Enforcement: 10 CFR 50, Appendix B, Criterion III, "Design Control," requires, in part, that design changes, including field changes, are subject to design control measures commensurate with those applied to the original design. Contrary to the above, from December 2004 until present, Entergy did not apply the same level of design control measures to the in-field actual setpoint values of the OPdT instrument that had been applied to the original design values. Specifically, Entergy implemented an OPdT instrument setpoint change but allowed the existing in-field actual setpoints to remain unchanged until the next regularly scheduled surveillance period and did not evaluate the adequacy of the in-field actual setpoints, which were later found outside the value required by the design basis. Technical Specification Table 3.3.1-1 Function 6 (i.e., OPdT) specified that the allowable value shall not exceed its computed trip setpoint by more than 2.4% of span, where the time constant value was greater than or equal to the value specified in the COLR. The COLR specified that the OPdT time constant be greater than or equal to 10 seconds. EC5000034071 revised the OPdT time constant from a band of 9.7 to 10.3 seconds to a band of 10.0 to 10.6 seconds in the applicable I&C and TS surveillance procedures. However, the EC specifically approved a delayed re-calibration of the in-field setpoint values, and did not evaluate the impact of a time constant value lower than allowed by the COLR. Specifically, the last performed calibration had left the time constant set at 9.9 seconds for OPdT channel 2 and 9.8 seconds for channel 4. Therefore, Entergy failed to verify the adequacy of design. Because this violation was of very low safety significance (Green) and was entered into Entergy's corrective action program (CR-IP2-2011-06047), this violation is being treated as a NCV consistent with Section 2.3.2 of the NRC Enforcement Policy. (**NCV 05000247/2011007-01, Failure to Correctly Implement an Approved Setpoint Change to Reactor Protection System Instruments**)

.2.2 Unit 2 Recirculation Pumps Flow Increase to Meet In-Service Testing Code Requirements

a. Inspection Scope

The team reviewed a modification (EC-34089) that replaced the two-inch test line for the internal safety injection (SI) recirculation pumps with a six-inch test line. The test line was replaced to increase the test flow rate to satisfy new American Society of Mechanical Engineers (ASME) requirements for pump testing. The new requirements include a comprehensive pump test to be performed at or above 80% of the internal recirculation pump design flow rate of 3000 gallons per minute. The modification included replacing a manual gate valve in the test line along with adding a flow orifice, a ball valve and a discharge sparger with a ring header to reduce the flow velocity to prevent hydraulic disturbance in the pump suction sump during testing.

The team reviewed the modification to verify that the design basis, licensing basis and performance capability of the internal SI recirculation pumps had not been degraded by the modification. The team interviewed engineering staff and reviewed technical evaluations associated with the modification to determine if the test line sizing was appropriate to satisfy the required flow rate and whether the piping stress analyses evaluated both test and accident system operating conditions to ensure the piping stress would remain within allowable limits. This included a review of an engineering evaluation of the expected heatup rate within the system during full flow testing to ensure surveillance test temperature conditions would remain consistent with bounding engineering pipe stress analyses. The team reviewed affected drawings, procedures, and calculations to ensure that they were properly identified and revised. The associated post-modification test (PMT) results were reviewed to ensure appropriate acceptance criteria had been identified and satisfied. The team also reviewed condition reports to determine if there were reliability or performance issues that may have resulted from the modification. Additionally, the 10 CFR 50.59 screening determination associated with this modification was reviewed as described in section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.3 Unit 2 Service Water Pump Zurn Strainer Backwash Circuit Modification

a. Inspection Scope

The team reviewed a modification (EC-10675) to the Unit 2 service water (SW) pump Zurn strainer circuitry to provide timed operation of the Zurn strainer backwash arm motor. The service water pumps have basket type motor operated rotary strainers which include a washing system to eliminate fine debris that passes through the intake structure. Prior to this modification, the Unit 2 Zurn strainer wiper arm motors were operated continuously whenever their associated SW pump was operating. This created additional wear on the internal components which required replacement on a six-month interval. The new timers were installed to provide for optimum intermittent operation of the wiper arms and were set to operate for five minutes every two hours. Additionally, the modification removed the strainer differential pressure (DP) signal for the automatic wiper arm motor start logic but maintained the high strainer DP alarm for each pump.

The team reviewed the modification to verify that the design basis, licensing basis and performance capability of the service water pumps had not been degraded by the modification. The team interviewed engineering staff and reviewed technical evaluations to determine whether the service water pump strainers would function in accordance with the design assumptions, including limiting the strainer DP consistent with assumptions in the service water engineering hydraulic analyses. The team walked down the Zurn strainer circuitry to ensure the engineering change was installed in accordance with design drawings and instructions. The associated PMT results were reviewed to ensure appropriate acceptance criteria had been met. The team also reviewed affected procedure revisions to ensure consistency with the design change. The team discussed the change with engineering and operations personnel to determine if there were reliability or performance issues that may have resulted from the modification. Additionally, the 10 CFR 50.59 screening determination associated with this modification was reviewed as described in Section 1R17.1 of this report. The documents reviewed in association with this modification are listed in the Attachment.

b. Findings

No findings were identified.

.2.4 Evaluation of Procedure Changes to Address Pressure Locking Conditions for Residual Heat Removal Valve MOV-744

a. Inspection Scope

The team reviewed procedure revisions associated with EC-24005. The revision involved the normal post accident positioning of the residual heat removal (RHR) system motor operated common discharge isolation valve, MOV-744, on both Unit 2 and Unit 3. The change maintained MOV-744 in a de-energized normally open position during the recirculation phase of an accident and provided for closing the valve if required by operating procedures. The valve is normally de-energized open per Technical Specification requirements during plant operation and had previously remained open during the injection phase of a loss-of-coolant-accident (LOCA) but was manually re-energized and closed after shutdown of the RHR pumps during the establishment of the recirculation phase of operation. The revision to the emergency operating procedures reduced the number of potential active position changes of the valve and precluded the potential for pressure locking issues associated with the valve.

The team reviewed the procedure revisions to verify that the design basis, licensing basis and performance capability of equipment during the recirculation phase of operation had not been degraded by the operational change. The team interviewed Entergy personnel, and reviewed calculations and technical evaluations to verify that the internal recirculation pumps and RHR system recirculation capability would still be able to perform their function in accordance with design assumptions. The team reviewed various operating procedures to ensure they appropriately reflected the revised strategy. This included a review of the containment leakage rate testing program document and in-service testing (IST) program documents to ensure they were revised to reflect the new position of MOV-744. Additionally, the Unit 2 post-LOCA dose calculations were reviewed to ensure the change had not affected the conclusions within the analyses. The procedure changes were performed on both Unit 2 and Unit 3. The team walked down the location of the valve and the valve motor control center in Unit 3, which was representative of Unit 2, in order to gain a relative understanding of the path required by an operator to gain access to the components. This included discussions with plant operators on their understanding of the purpose of the change and their familiarity with the revised procedures. Additionally, the 10 CFR 50.59 screening determination associated with this modification was reviewed as described in Section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.5 Installation of High Point Vent in the Unit 2 Residual Heat Removal System

a. Inspection Scope

The team reviewed a modification (EC-16920) to the Unit 2 RHR system that installed a new high point vent on the 14-inch suction line from the reactor coolant system 22 hot leg, downstream of primary containment isolation valve AC-732. During monthly surveillance checks on Unit 3, the presence of a gas void in the same location had occasionally been detected. The potential existed for a similar gas void in Unit 2 because the Unit 2 piping arrangement and source for potential gas in-leakage was similar to that in Unit 3. Therefore, this modification added a vent connection downstream of AC-732, similar to Unit 3, to mitigate a potential gas void and ensure the RHR suction piping would remain full of water. Installation of the vent connection was performed without depressurizing or draining the RHR system by utilizing a hot tap process.

The team evaluated the modification to determine whether the design basis, licensing basis, or performance capability of the RHR system had been degraded by the modification. The team assessed Entergy's technical evaluations and design details, including installation specifications and foreign material exclusion (FME) control, and interviewed engineering personnel to determine whether the RHR system would function in accordance with the design assumptions and whether the specified FME controls were adequate to prevent foreign material, such as drill shavings, from entering the reactor coolant system. Drawings and procedures were reviewed to verify they were properly updated. The team also reviewed the completed work order and interviewed maintenance personnel to assess whether installation activities were performed as specified by the design. Post-modification test results were reviewed to verify the acceptance criteria had been met. In addition, the team walked down the new RHR vent valve to independently evaluate material condition and to ensure it was installed in accordance with design instructions. A review of condition reports was performed to determine whether there were any reliability or performance issues associated with installation of the new vent valve. Additionally, the 10 CFR 50.59 screening determination associated with this modification was reviewed as described in section 1R17.1 of this report. The documents reviewed are listed in the Attachment to this report.

b. Findings
No findings were identified.

.2.6 Unit 2 Residual Heat Removal Pump Motor Flood Protection

a. Inspection Scope

The team reviewed a modification (EC5000034211) that installed a float operated drain valve and drain line in the primary auxiliary building (PAB). The drain line penetrated the PAB exterior wall and was routed below grade to a nearby storm drain manhole in the transformer yard. The drain line was designed to protect the RHR pump motors from damage due to a postulated internal flooding event in the PAB. Entergy performed an evaluation of current offsite dose calculation requirements and determined the extra flood protection would not create a new effluents pathway nor significantly increase the dose consequences of existing pathways due to a postulated internal flooding event.

The team evaluated the modification to determine whether the design basis, licensing basis, or performance capability of systems, structures, or components (SSCs) located in the PAB, including the RHR system, had been degraded by the modification. The team assessed Entergy's calculations, radiological dose assessment analysis, and engineering evaluations for the sizing and placement of the float valve and associated drain piping to verify the adequacy of the design. Drawings, procedures, and preventive maintenance tasks were reviewed to verify they had been properly updated. The team also reviewed the completed work order and interviewed Entergy personnel to assess whether installation activities were performed as specified by the design. PMT results were evaluated to determine whether the float valve and drain line would function in accordance with the design assumptions. In addition, the team walked down the float valve and drain line penetration into the yard manhole to independently evaluate material condition and to verify it was installed in accordance with design instructions. A review of condition reports was performed to determine whether there were any reliability or performance issues associated with installation of the new drain line and float valve. Additionally, the 10 CFR 50.59 screening determination associated with this modification was reviewed as described in section 1R17.1 of this report. The documents reviewed are listed in the Attachment to this report.

b. Findings

No findings were identified.

.2.7 Modification of Piping Connection and Support Associated with Unit 2, 23 Charging Pump Suction Stabilizer Vent Line

a. Inspection Scope

The team reviewed a modification (EC-2259) in Unit 2 that modified the existing 3/4 inch vent connection on the 23 charging pump suction stabilizer. Entergy performed this modification to address repeated leakage at the threaded coupling. The suction stabilizer and corresponding pulsation dampener were designed to prevent vibration induced pipe failure and charging pump cavitation. The modification removed the existing screwed coupling welded to the stabilizer and replaced it with a fully welded joint. An additional support was installed on the vent piping to reduce vibration.

The team reviewed the modification to verify that the design basis, licensing basis and performance capability of the charging system had not been degraded by the modification. The team interviewed engineering staff and reviewed technical evaluations associated with the modification to determine if the charging pump would function in accordance with the design assumptions. The team reviewed drawings and procedures to ensure that they were properly updated. In addition, the inspectors performed field inspections and system walk downs to evaluate the installation and material condition. The associated PMT results were reviewed to ensure appropriate acceptance criteria had been met. The team also reviewed condition reports to determine if there were reliability or performance issues that may have resulted from the modification. Additionally, the 10 CFR 50.59 screening determination associated with this modification was reviewed as described in Section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.8 Unit 2 Service Water Bay Water Level Indicators

a. Inspection Scope

The team reviewed a modification (EC-5828) that upgraded the service water level indicators from a manual method of measuring SW level to an electronic system. The modification used the existing level transmitters, LE-7607-2 and LE-7608-2, and installed new digital level indicators. The modification was implemented to alleviate operator burden and to eliminate personnel safety concerns associated with taking manual readings within the SW bay area. In addition, the modification provided local SW bay water level indication to the control panels for traveling water screens 27 and 28, an alarm on the annunciator panel EPR 10 located in the control building, and an alarm in the main control room. The modification also corrected a non-conservative six-inch water level error due to improper installation of the existing SW bay level indicators.

The team reviewed the modification to verify that the design basis, licensing basis and performance capability of the safety-related SW system had not been degraded by the modification. The team interviewed engineering staff and reviewed technical evaluations to determine if the new SW level electronic indicators would function in accordance with the design assumptions. The associated work order instructions and documentation were reviewed to verify that maintenance personnel implemented the modification as designed. The team walked down the accessible portions of the Unit 2 SW system and new level indicators to assess their material condition and to ensure the level indicators were installed in accordance with design instructions. The team reviewed the PMT results to ensure the appropriate acceptance criteria had been met and that the tests demonstrated the adequacy of the new design. A review of the condition reports associated with the SW system was also performed to determine if there were reliability or performance issues that may have resulted from the modification. Additionally, the 10 CFR 50.59 screening determination associated with this modification was reviewed as described in Section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.9 Revise Setpoint for Unit 2 Emergency Diesel Generator Jacket Water Temperature Controllers

a. Inspection Scope

The team reviewed a modification (EC-2603) that changed the setpoint for the jacket water temperature controllers, TC-5001, -5002, and -5003, for each of the emergency diesel generators (EDG) in Unit 2. The setpoint for each controller was changed 10 degrees Fahrenheit (°F) from 125°F +/- 3°F to 135°F +/- 3°F. Entergy performed this modification to prevent jacket water temperature readings below the administrative low temperature limit of 120°F. Specifically, the 23 EDG jacket water temperature indicator (TI-5046) was found to read approximately 112°F during the winter seasons due to lower ambient temperatures. Similar concerns were also identified on the 21 and 22 EDGs. Entergy determined there were no operability concerns with the EDGs because the temperatures never dropped below the recommended vendor minimum limit of 90°F. By increasing the setpoint by 10 degrees, the jacket water heater temperature control switch would maintain the heaters energized longer, resulting in higher jacket water temperatures.

The team reviewed the modification to verify that the design basis, licensing basis and performance capability of the EDGs had not been degraded by the jacket water controller setpoint change. The team interviewed engineers, and reviewed applicable technical evaluations and design drawings to verify that the jacket water system would function in accordance with design assumptions. The team reviewed the associated PMT results to ensure the appropriate acceptance criteria had been met. In addition, the team walked down the EDGs and jacket water system components to assess their material condition. The team also confirmed that surveillance tests, operational procedures, and drawings had been appropriately updated to reflect the change. Additionally, the 10 CFR 50.59 screening determination associated with this modification was reviewed as described in Section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.10 Unit 3 Refueling Water Storage Tank Low Low Level Alarm Switch Modification

a. Inspection Scope

The team reviewed a modification (EC5000041964) on Unit 3 to replace an existing refueling water storage tank (RWST) low low level alarm switch (LIC-921) with a smaller range level alarm switch (LC-923). The smaller range dedicated LC-923 alarm switch provides better resolution and more accuracy compared to the original alarm switch. The alarm function of LIC-921 was removed but the local indicating dial of LIC-921 was retained at the instrument panel at the RWST. LC-923 was supplied by a different vendor and utilized a different sensor technology than the original LIC-921. The LC-923 alarm switch provides an input to one of two annunciators in the control room that cue operators to align the safety injection system to recirculation mode prior to the RWST reaching a level too low to support safety injection pump operation.

The team reviewed the modification to verify that the design basis, licensing basis and performance capability of the RWST level indication and alarms had not been degraded by the modification. Specifically, the team verified that seismic and environmental qualification, instrument accuracy and setpoint drift, instrument redundancy and diversity, and operating characteristics and requirements were equivalent or improved. The team interviewed design engineers and reviewed drawings, calculations, evaluations, vendor data, PMT results, and associated maintenance work orders to verify that the LC-923 installation and LIC-921 modification was appropriately implemented. Finally, the team walked down the RWST level and alarm instruments with the design engineer to verify they had been installed in accordance with design instructions. The 10 CFR 50.59 screening determination associated with this modification was also reviewed as described in section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.11 32 Emergency Diesel Generator Control Relay Modification

a. Inspection Scope

The team reviewed a Unit 3 modification (98-3-040-EDG) that replaced several 125 volt direct current (VDC) control relays on the 32 emergency diesel generator. The replacement control relays included the engine start relays, run relay, cranking relays, and the shutdown relay. The 32 EDG control relay replacements were intended to improve reliability of the EDG and were replacements for an obsolete relay model.

The team reviewed the modification to verify that the design basis, licensing basis and performance capability of the 32 EDG had not been degraded by the modification. Specifically, the team reviewed attributes such as minimum and maximum operating voltages, amperage requirements, relay response timing, and seismic qualification to verify the new relays were equivalent or improved when compared to the previous relays. The team interviewed design engineers, and reviewed evaluations, calculations, vendor information, wiring and schematic diagrams, PMT results, and associated maintenance work orders to determine whether the 32 EDG control relay replacements were appropriately implemented. Finally, the team walked down the 32 EDG control cabinet with the design engineer to evaluate material condition. The 10 CFR 50.59 screening determination associated with this modification was also reviewed as described in section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.12 Unit 3 Pressurizer Backup Heater Group 32 Bank 8.9.31 Retire-In-Place Modification

a. Inspection Scope

The team reviewed a Unit 3 modification (EC-14246) to retire-in-place pressurizer backup heater group 32 bank 8.9.31. Backup heater bank 8.9.31 was determined to be grounded during a troubleshooting maintenance activity using work order 51485917. Pressurizer heaters maintain a constant pressure for the reactor coolant system and a minimum required available capacity is required by Technical Specification 3.4.9.b.

The team reviewed the modification to verify that the design basis, licensing basis and performance capability of the Unit 3 pressurizer heaters was maintained within Technical Specification requirements. The team verified that operators administratively tracked the available capacity of pressurizer heaters and that the simulator was appropriately modeled for the loss of pressurizer backup heater group 32 bank 8.9.31. The team interviewed design engineers, and reviewed evaluations, wiring and schematic diagrams, and the associated maintenance work order to ensure all aspects of the modification and its potential impact on plant and electrical system operation were appropriately considered and implemented. Additionally, the 10 CFR 50.59 screening determination associated with this modification was reviewed as described in section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.13 Replacement of Unit 3 Control Room Air Conditioner Service Water Temperature Control Valves

a. Inspection Scope

The team reviewed a modification (EC-7854) that replaced the 1-1/4 inch temperature control valves, SWN-TCV-1310, -1311, -1312, and -1313, installed in the service water outlet lines from each control room air conditioning system (CRACS) condenser with smaller 3/4 inch control valves. Entergy replaced these valves to address excessive seat wear associated with the larger valves which was experienced during periods of low flow and high pressure such as during winter months. The seat wear resulted in leakage that allowed cold service water to flow through and cool the condenser and the refrigerant. The cool, low-pressure refrigerant was attributed to low suction pressure trips of the compressor upon system startup. Entergy determined that the use of smaller valves would minimize seat wear because they would operate at increased seat openings.

The team reviewed the modification to verify that the design basis, licensing basis and performance capability of the control room air conditioning system had not been degraded by the modification. The team interviewed engineering staff, and reviewed technical evaluations associated with the modification to determine if the new temperature control valves would function in accordance with the design assumptions. The team reviewed drawings, procedures, and calculations to ensure that they were properly updated. The associated PMT results were reviewed to ensure appropriate acceptance criteria had been met. The team also reviewed condition reports to determine if there were reliability or performance issues that may have resulted from the modification. Additionally, the team walked down the temperature control valves to assess their material condition and to verify they were installed in accordance with design instructions. The 10 CFR 50.59 screening determination associated with this modification was also reviewed as described in section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.14 Installation of Vortex Suppressors in the Unit 3 Vapor Containment

a. Inspection Scope

The team reviewed a modification (EC-14974) that installed vortex suppressors within the Unit 3 vapor containment (VC) above the internal recirculation sump strainer and above the VC sump strainer. Each vortex suppressor consists of a stainless steel frame structure that supports sections of grating that provide the vortex suppression function. Entergy personnel implemented this change to address the generic industry concern associated with the potential for vortex formation at the strainer inlet during certain loss-of-coolant accident scenarios. Specifically, engineering personnel determined that the vortex suppressors were required to mitigate potential reliability and operational concerns associated with air ingestion during post-LOCA recirculation operation.

The inspectors reviewed the modification to verify that the design basis, licensing basis, and performance capability of the internal recirculation and VC sumps had not been degraded by the modification. The inspectors reviewed several related calculations associated with sump strainer performance and post-LOCA debris loading to ensure that Entergy used conservative assumptions and appropriate inputs to adequately evaluate the modification. The inspectors reviewed Entergy's completed installation work orders including the associated drawings, weld specification sheets, weld maps, and completed weld data sheets. The inspectors also reviewed condition reports to determine if there were reliability or performance issues that may have resulted from the modification. The 10 CFR 50.59 screening determination associated with this modification was also reviewed as described in section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.15 Unit 3 Residual Heat Removal Pump Cooling Evaluation

a. Inspection Scope

The team reviewed an evaluation (EC-25886) that analyzed a statement in the UFSAR which asserted cooling was not needed to the Unit 3 residual heat removal pumps for 24 hours in an accident condition. During research performed for a calculation of cooling flow to the Unit 2 RHR pumps, Entergy identified that this statement in the Unit 3 UFSAR may not be valid. Entergy performed this evaluation based on results of seal testing conducted by the RHR pump seal manufacturer which showed that some cooling would be required prior to 24 hours to ensure adequate seal performance in an accident scenario. Specifically, the results concluded the RHR pump seals require a cooling water recirculation flush provided by the component cooling water (CCW) system be maintained at 150°F or below to ensure no seal degradation occurs. There is a potential that during some scenarios which involve a loss of CCW, the RHR pump seal temperature may exceed 150°F less than 24 hours into the event. The evaluation analyzed RHR pump function during a large break LOCA, a small break LOCA, a main steam line break, a steam generator tube rupture, an Appendix R event, and normal plant operating modes. Entergy concluded that seal performance during these conditions would be either unaffected or impacted negligibly, and therefore, the RHR pumps would maintain capability to perform their intended design functions.

The team reviewed the evaluation and associated calculations to verify that assumptions and parameters used were valid and considered bounding scenarios. The parameters included RHR pump suction temperature, availability of cooling water, and the duration that cooling was unavailable to the pumps. Additionally, the team verified that affected procedures were updated to include caution statements associated with operating RHR pumps without CCW available. The 10 CFR 50.59 screening determination associated with this modification was also reviewed as described in section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.16 31 EDG Jacket Water Pressure Switch Setpoint Change

a. Inspection Scope

The team reviewed a modification (EC-7202) that changed the emergency diesel generator jacket water (JW) pressure switch setpoint associated with starting and stopping the EDG pre-lubrication (pre-lube) oil pump. Specifically, the design objective of the change was to ensure the pre-lube oil pump starts, as required, after the EDG is stopped. The modification was implemented because Entergy determined that the JW pressure switch trip setpoint had not accounted for the static head of the JW system. The modification installed a like-in-kind switch with a 12 +/- 0.5 psig decreasing setpoint, a conservative change from the original setpoint of 8 +/- 0.5 psig. The switch also performs a secondary function to stop the pre-lube oil pump on increasing pressure of the JW system after the EDG is started.

The team reviewed the modification to verify that the design basis, licensing basis, and performance capability of the EDG had not been degraded by the modification. The team assessed whether the modification was consistent with requirements and assumptions in the design and licensing bases. The team reviewed calculations and technical evaluations to assess whether the modification was consistent with design assumptions. Post-modification testing data was reviewed to verify the operating jacket water pressure range of the EDGs. The team performed walk downs of the 31, 32 and 33 EDGs, and observed the 32 EDG while running, to assess operation of the equipment while in service and the material condition. The team conducted interviews with engineering staff to determine if the EDGs would function in accordance with the design assumptions. Additionally, the 10 CFR 50.59 screening determination associated with this modification was also reviewed as described in section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.17 High Point Vent Valve Installation on Unit 3 Containment Spray System Piping

a. Inspection Scope

The team reviewed a modification (EC-17096) that installed a system high point vent valve (SI-208) on the Unit 3 containment spray system (CSS) piping. The new valve provides the capability to vent and eliminate potential gas accumulation within the 12-inch common suction header to the CSS pumps. In response to NRC Generic Letter 2008-01, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems," Entergy identified the location as having the potential for gas accumulation and subsequent gas binding or damage to the pumps, with no provision to eliminate the void.

The team reviewed the modification to verify that the design basis, licensing basis, and performance capability of the CSS had not been degraded by the modification. The team assessed if the modification was consistent with requirements in the design and licensing bases. The team reviewed the stress analysis calculation and technical evaluations to assess whether the modification was consistent with design assumptions. Components and materials were reviewed to ensure that the modification conformed to the design specifications and to verify that the system was seismically qualified. Design assumptions were reviewed to evaluate whether they were technically appropriate and consistent with the UFSAR. The team also verified that selected drawings, calculations and procedures were properly updated based on the new configuration. The team reviewed the installation details and work order process to ensure control of FME during installation. The team reviewed the PMT results and data sheets to verify the system would function in accordance with design requirements. The team performed a walk down of the system to verify the vent was installed in accordance with design instructions. Additionally, the team conducted interviews with engineering staff to determine whether the affected SSCs functioned in accordance with the design assumptions and to verify if the modification corrected the previously identified problem. Additionally, the 10 CFR 50.59 screening determination associated with this modification was also reviewed as described in section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.18 Installation of Internal Pipe Mechanical Seals at Pipe Weld Connections within Underground Service Water Line

a. Inspection Scope

The team reviewed a modification (EC-24032) on Unit 3 that installed internal pipe mechanical seals to prevent further corrosion at the weld seams in the underground 24-inch service water system line from the SW pump discharge to the primary auxiliary building. The 24-inch SW pipe line is a butt welded, cement-lined carbon steel pipe with gaps at the circumferential weld locations. The modification was designed to act as waterproof barriers to protect the internal welded seam surface from further corrosion and erosion due to exposure of the welds to brackish river water. The seals are a vendor supplied and installed product manufactured of ethylene propylene rubber with corrosion resistant stainless steel retaining bands.

The team reviewed the modification to verify that the design basis, licensing basis and performance capability of the SW System had not been degraded by the modification. Specifically, the team verified that the installation of the seals would have negligible effect on the margin for flow requirements and the seismic qualification of the piping. The team interviewed design and system engineers, and reviewed evaluations, vendor information and procedures, post-modification testing results, and associated maintenance work orders to verify that the modification was appropriately implemented. The 10 CFR 50.59

screening determination associated with this modification was also reviewed as described in section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings 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 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. In addition, the team reviewed condition reports written on issues identified during the inspection to verify adequate problem identification and incorporation of the problem into the corrective action system. The condition reports reviewed are listed in the Attachment.

b. Findings

No findings were identified.

4OA6 Meetings, including Exit

On December 1, 2011, the team presented the preliminary inspection results to Mr. L. Coyle, and other members of Entergy's staff. The final inspection results were discussed with Mr. P. Conroy in a telephone call on January 12, 2012. The team returned the proprietary information reviewed during the inspection and verified that this report does not contain proprietary information.

ATTACHMENT

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Entergy Personnel
E. Bauer, Design Engineer
J. Bencivenga, Design Engineer
F. Bloise, Design Engineer
C. Bristol, Design Engineer
J. Bubniak, Design Engineer
T. Chan, System Engineering Supervisor
P. Conroy, Director of Nuclear Safety Assurance
L. Coyle, General Manager, Plant Operations
G. Dahl, Specialist, Licensing
K. Elliott, Fire Protection Engineer
T. Galati, Design Engineer
J. Hill, I&C Design Engineering Supervisor
A. Kaczmarek, Design Engineer
J. Kaczor, Design Engineer
A. King, Design Engineer
L. Liberatori, Design Engineer
T. McCaffrey, Manager, Design Engineering
R. Motko, Reactor Engineer
J. Raffaele, Supervisor, Design Engineering
R. Sergi, Design Engineer
J. Whitney, System Engineer

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

05000247/2011007-01	NCV	Failure to Correctly Implement an Approved Setpoint Change to Reactor Protection System Instruments (Section 1R17.02.1)

LIST OF DOCUMENTS REVIEWED

10 CFR 50.59 Evaluations
09-2001-00-Eval, Temporary Operating Procedure 2-TOP-014, Contingency Actions for Degraded Recirculation Line to CST, Rev. 0
10-3001-00 EVAL, Procedure Changes to Address Pressure Locking Conditions for RHR Valve MOV-744 (Engineering Change Evaluation EC-24005), Rev. 0
10-2001-00 EVAL, Procedure Changes to Address Pressure Locking Conditions for RHR Valve MOV-744 (Engineering Change Evaluation EC-24005), Rev. 0
11-3001-00 EVAL, Allow Testing of the Manipulator Crane with a Dummy Fuel Assembly in the Reactor, Rev. 0

<u>10 CFR 50.59 Screened-out Evaluations</u>
3-AOP-RHR-1, Loss of RHR, Rev. 9
3-E-0, Reactor Trip or Safety Injection, Rev. 2
3-E-3, Steam Generator Tube Rupture, Rev. 2
3-ECA-1.1, Loss of Emergency Coolant Recirculation, Rev. 2
EC-09063, Accept Flowserve Report EC 1180 and Drawing with Minimum Submergence Level for SI Recirculation Pumps, Rev. 0
EC-09743, Delete Containment Sump Level Transmitters LT-3304 & LT-941 from the List of Credited RCS Leakage Detection Instruments, Rev. 0
EC-12166, Refurbishment of River Water Pumps, Rev. 0
EC-14686, IP2 RHR System Valve 745 A&B Redundant Valve Position Indication, Rev. 0
EC-15086, Equivalent Change Evaluation for Replacement of GT-2 Battery, Rev. 0
EC-15137, Evaluate and Accept Repairs on Valve FP-900 Gland Follower, Rev.0
EC-15308, SQUG Evaluation for CRACS Seismic Capability, Rev. 0
EC-15883, Install New High Point Vent Valve in SI System, Rev. 0
EC-16001, Equivalent Change Evaluation for Replacement of Electronic Circuit Boards in the Leading Edge Flow Measurement System, Rev. 0
EC-16284, Evaluation of Freeze Seal for Repair of Valve 816, Rev. 0
EC-16701, Flood Gates on Doors DR-224 and DR-226, Rev. 0
EC-17784, Repair of Unit 2 Refueling Cavity Liner Plates, Rev. 0
EC-18904, Vortex and Critical Submergence Evaluations for RWST and CST, Rev. 0
EC-19083, Evaluation of Tornado Passage Effect on the EDG HVAC System, Rev. 0
EC-19489, Flowserve Report to Evaluate Minimum Flows and Associated Expected Service Life for Auxiliary Component Cooling Pumps, Rev. 0
EC-19779, Revision of IP3-CALC-RPC-00298 – Inst. Loop Accuracy and Setpoint Calculation for IP3 RCS Loop Low Flow for Inclusion of DCP 01-3-058 RCS, DC 97-3-039, and Changes Previously Made to Associated Surveillance Procedures, Rev. 0
EC-19816, Service Water System Refurbishment Project, Rev. 0
EC-19959, Modify the Required Action Statement for TRO 3.8.B Condition A to Include an OR Statement to Establish an Independent Power Supply if the Appendix R Diesel Generator Cannot be Restored to OPERABLE Status Within 30 Days, Rev. 0
EC-20030, Engineering Evaluation of Ultrasonic Thickness Measurements on Various Piping Systems for Flow Accelerated Corrosion Program, Rev. 0
EC-21406, Evaluation of Stainless Steel Bolting Material for 2" Flanges in SW System, Rev. 0
EC-22784, Equivalent Change Evaluation for Replacement of a Resistance to Current Converter (R/I Action Pak) Module, Rev. 0
EC-22830, Evaluate the IPEC Unit 2 & 3 RHR Systems per Westinghouse NSAL-09-8 Regarding Potential Voiding Under Postulated Modes 3 & 4 LOCA Conditions, Rev. 9
EC-23379, Clarification of IP3 UFSAR Regarding Service Water System Component Realignment during Post-LOCA Recirculation, Rev. 0
EC-24491, Evaluate Use of Enecon MetalClad CeramAlloy and Coating Compounds for Use on Stainless Steel Piping and Fittings in the Service Water System, Rev. 0
EC-24608, 3R16 Replacement of Valve SWT-235-2, Rev. 0
EC-25145, Blocking Open of CIVs During Electrical Testing of #31 DC Bus During Plant Modes 5 and/or 6, Rev. 1
EC-25423, Main Steam Isolation Valve Limit Switch Equivalency Change, Rev. 0
EC-25920, Alternate Pole Arrangement on 23 EDG JWPS-6-2, Rev. 0

EC-28201, Revise Procedure ONOP-CVCS-3, Emergency Boration, to Increase the Boron Requirements for Shutdown Margin in the Event of a Cooldown, Rev. 0
EC-28424, 33 EDG Cables Equivalency Change, Rev. 0
EC-28546, Provide Temporary Source for Cooling Seal Oil Unit in lieu of Service Water, Rev. 0
EC-30396, IRPI System Fuse Replacement with Time Delay Fuses, Rev. 0
EC-31238, SW Strainer Blowdown Setting Change, Rev. 0
EC-32102, Replacing Two Service Water Strainer Supports, Rev. 0

Modification Packages
98-3-040-EDG, Replacement of EDG GE Model CR120A262-41 Relays, Rev. 1
EC-10675, Unit 2 Service Water Pump Zurn Strainer Backwash Circuit Modification, Rev. 0
EC-14246, Retire-In-Place Pressurizer Backup Heater Group 32 Heater Bank 8.9.31, Rev. 0
EC-14974, Install Vortex Suppressors Over the VC and Recirc Sumps at Unit 3, Rev. 0
EC-16920, Install New High Point Vent Valve in RHR System Downstream of Valve 732, Rev. 0
EC-17096, Install a High Point Vent Valve on the Unit 3 Containment Spray System Piping, Rev. 0
EC-2259, Modification of Piping Connection and Support Associated With Unit 2 Charging Pump 23 Suction Stabilizer Vent Line, Rev. 0
EC-24005, Evaluation of Procedure Changes to Address Pressure Locking Conditions for RHR Valve MOV-744, Rev. 0
EC-24032, Installation of Internal Pipe Mechanical Seals at Pipe Weld Connections within Underground Service Water Line, Rev. 0
EC-25886, RHR Pump Cooling Evaluation, Rev. 0
EC-2603, Revise Setpoint for Unit 2 EDGs Jacket Water Temperature Controllers, Rev. 0
EC-34089, Unit 2 Recirculation Pumps Flow Increase to Meet In-Service Testing Code Requirements, Rev. 0
EC5000034071, Over-Power Delta-Temperature Tolerance Changes, Rev. 0
EC5000034211, Design Permanent Solution to Protect RHR Pump Motors from Internal Flooding, Rev. 0
EC5000041964, Replace LIC-921 RWST Lo Lo Level Alarm Switch with a New RWST Lo Lo Level Alarm Switch LC 923, Rev. 0
EC-5828, Modification to the Service Water Bay Water Level Indicators, Rev. 0
EC-7202, EDG Jacket Water Pressure Switch Associated with the Pre-Lube Pump Setpoint Change, Rev. 0
EC-7854, Replace 1 ¼" valves SWN-TCV-1310, 1311, 1312, and 1313 with Smaller (3/4") Valves, Rev. 0

Calculations, Analysis, and Evaluations
97-126-SP, EDG Jacket Water Pressure Switches Setpoint Change, Rev. 0
IP3-CALC-STR-03334, Qualification of Internal Mechanical Seal Assembly, Rev. 1
IP3-CALC-SWS-03312, Evaluation of Hydraulic Impact of Mechanical (Internal) Seals Installed in 24" SWS Line 408 – Non-Essential Header Operation, Rev. 0
IP-CALC-04-01806, Over-Power Delta-Temperature & Over-Temperature Delta-Temperature Loop Accuracy, Rev. 0
IP3-RPT-ED-00922, Appendix R Diesel Generator System Evaluation, Rev. 4
IP3-CALC-SI-00725, Instrument Loop Accuracy / Setpoint Calc / RWST Level (IP3), Rev. 3

FFX-00706-00, Analysis to Demonstrate the Pressure Boundary Integrity of the Fire Protection Piping in the PAB, Rev. 0
GCC-569-001-0, Analysis of HP Motor Mounting and Panel Support in Zurn Strainer Pit, Rev. 0
IP3-CALC-RPC-00298, Indian Point 3 – Instrument Loop Accuracy / Setpoint Calculation / RCS Loop Low Flow, Rev. 1
IP3-CALC-268, Emergency Diesel Generator/Jacket Cooling Water Tank Static Head Calculation, Rev. 0
IP-CALC-09-00235, Stress Analysis for the Addition of Vent Valve SI-208 Load on Line 181, Rev. 0
IP-CALC-09-00179, Indian Point ECCS Sump Strainer Certification Calculation Based on NPSH, Minimum Flow, Structural Limit and Void Fraction Requirements, Rev. 3
IP3-CALC-EL-00184, 125 VDC Component Sizing, Rev. 3
IP-CALC-07-00153, Evaluation of Pipe Support SIH-365 through SIH-371 Recirculation Pump Test Line, Rev. 0
IP-CALC-07-00123, Stress Analysis of SI Line #293, Rev. 1
FIX-00099-00, Emergency Diesel Generator Accuracy of Lube Oil & Jacket Water Temperature Switches, Rev. 0
00-086, Hydraulic Impact of Installing Mechanical Seals in the Essential Service Water Header at Indian Point 3, Rev. 1
CN-CRA-03-55, IP2 LOCA Doses for Stretch Power Uprate, Rev. 0
IP-RPT-09-00046, Indian Point Units 2 and 3 Design and Evaluation of Vortex Suppression Grating, Rev. 1
CN-TA-03-041, Unit-2 OTDT/OPDT Setpoint Analysis for 4.7% Power Up-rate Program, Rev. 2
IP-CALC-08-00031, Misc. Structural Evaluation for IP2 & IP3 RHR Pump Motor Flood Protection, Rev. 0
IP-CALC-08-00024, Sizing Calculation for RHR Pump Flooding Line, Rev. 0
89-03-084 EDG, 480 Volt Emergency Diesel Generators Control Circuit Timing Relays and Pressure Switch Setpoint Determination, Rev. 1

<u>Condition Reports</u>

CR-IP2-1999-07141
CR-IP2-2000-05387
CR-IP2-2001-01133
CR-IP2-2001-00301
CR-IP2-2004-04889
CR-IP2-2004-06713
CR-IP3-2006-03692
CR-IP2-2007-02965
CR-IP2-2007-02986
CR-IP2-2008-00013
CR-IP2-2008-03676
CR-IP2-2008-03705
CR-IP2-2008-04020
CR-IP2-2008-04622
CR-IP2-2008-04653
CR-IP3-2008-00909
CR-IP2-2009-05399
CR-IP2-2009-00567
CR-IP2-2009-00817
CR-IP3-2009-04226
CR-IP2-2010-02570
CR-IP2-2010-04322
CR-IP2-2010-06861
CR-IP3-2010-00504
CR-IP3-2010-00588
CR-IP3-2010-02142
CR-IP3-2010-00045
CR-IP2-2011-02937
CR-IP2-2011-05785*
CR-IP2-2011-05787*
CR-IP2-2011-05792*
CR-IP2-2011-05852*
CR-IP2-2011-05855*
CR-IP2-2011-05862*
CR-IP2-2011-06041*
CR-IP2-2011-06043*
CR-IP2-2011-06047*
CR-IP2-2011-06070*
CR-IP2-2011-06087*
CR-IP2-2011-06071*
CR-IP3-2011-00989
CR-IP3-2011-01255
CR-IP3-2011-01546
CR-IP3-2011-02783
CR-IP3-2011-03776
CR-IP3-2011-04993
CR-IP3-2011-05136*
CR-IP3-2011-05302*
CR-IP3-2011-05309*
CR-IP3-2011-05327*
CR-IP3-2011-05353*

CR-IP3-2011-05502* CR-IP3-2011-10869 CR-IP3-2011-05297
(* denotes NRC identified during this inspection)

Drawings
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IP3V-91-0068, General Plan 40'-0 DIA X 42' – 3 High Dome Roof Tank – Refueling Water Storage Tank, Rev. 2
IP3V-13-0006, Diagram of Conn. for Diesel Gen #31, 32 & 33 DC Wiring Panel 32, Rev. 7
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B228014, Fire Barrier Penetrations, Charging Pump Rooms, Rev. 5
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9321-F-1461, Diesel Generator Building Concrete Foundation Plan, Rev.10
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9321-F-4046, Diesel Generator Building Floor Drains & Vent. Control Air Piping Plans and Sections, Rev. 19
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9321-F-20333, Flow Diagram Service Water System, Unit 3, Sh. 2, Rev. 28
9321-F-20343, Flow Diagram City Water, Unit 3, Sh. 1, Rev. 36
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9321-F-20983, Turbine Building and Heater Bay Service & Cooling Water Piping, River Water System, Unit 3, Sh. 2, Rev. 12
9321-F-27223, Flow Diagram Service Water System, Rev. 45
9321-F-27503, Flow Diagram Safety Injection System, Sh. 1, Rev. 42
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9321-F-27513, Flow Diagram Auxiliary Coolant System, Sh. 1, Rev. 31
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0-NF-203, Internal Transfer of Fuel Assemblies and Inserts, Rev. 2
2-AOP-FLOOD-1, Flooding, Rev. 7
2-ARP-003, High Jacket Water Temperature, Rev. 9
2-ARP-004, Waste Disposal Panel, Rev. 3
2-ARP-014, PAB Flooding, Rev. 2
2-ARP-SJF, Cooling Water and Air, Rev. 39
2-COL-10.1.1, Safety Injection System, Rev. 34
2-COL-3.1, Chemical and Volume Control System, Rev. 40
2-COL-4.2.1, Residual Heat Removal System Check Off List, Rev. 28
2-ECA-1.3, Loss of Emergency Coolant Recirculation Caused By Sump Blockage, Rev. 2
2-ES-1.3, Transfer to Cold Leg Recirculation, Rev. 6
2-ES-1.4, Transfer to Hot Leg Recirculation, Rev. 3
2-IC-PC-I-T-21 EDG, Emergency Diesel Generator No. 21 Temperature Instruments Calibration, Rev, 8
2-PC-EM37, Over-Temperature Delta-T and Over-Power Delta-T Setpoint Generators, Rev. 11
2-PT-2Y017, Penetration Fire Barrier Seal Inspection, Rev. 1
2-PT-A048, Rollup Fire Doors, Rev. 0
2-PT-M108, RHR/SI/CS System Venting, Rev. 10
2-PT-R016, Recirculation Pumps, Rev. 21
2-PT-V11A-1, Recalibration of NIS and OT/OP T Parameters - Channel 1, Rev. 39
2-PT-V11A-2, Recalibration of NIS and OT/OP T Parameters - Channel 2, Rev. 39
2-PT-V11A-3, Recalibration of NIS and OT/OP T Parameters - Channel 3, Rev. 34
2-PT-V11A-4, Recalibration of NIS and OT/OP T Parameters - Channel 4, Rev. 40
2-SOP-24.1, Service Water System Operation, Rev. 59
2-SOP-4.3.1, Residual Heat Removal System Operation, Rev. 63
2-TOP-008, Contingency Actions for PAB Flooding, Rev. 1
2-TOP-014, Compensatory Actions for Repairs to the Recirculation Line to CST, Rev. 0
3-AOP-ANNUN-1, Failure of Flight or Supervisory Panel Annunciators, Rev. 4
3-AOP-SSD-1, Control Room Inaccessibility Safe Shutdown Control, Rev. 13
3-ARP-005, Panel SBF-2-SAFEGUARDS, Rev. 35
3-COL-CS-001, Containment Spray System, Rev. 15
3-COL-CSV-001, Containment Spray Verification, Rev. 7
3-COL-RCS-001, Reactor Coolant System, Rev. 30
3-COL-SI-001, Safety Injection System, Rev. 42
3-ES-1.3, Transfer to Cold Leg Recirculation, Rev. 7
3-FR-C.1, Response to Inadequate Core Cooling, Rev. 1
3-FR-C.2, Response to Degraded Core Cooling, Rev. 2
3-FR-H.1, Response to Loss of Secondary Heat Sink, Rev. 4
3-GRAPH-EL-4, EOP Equipment Load List, Rev. 3
3-GRAPH-RPC-16, Core Operating Limits Report, Rev. 4
ONOP-CVCS-3, Emergency Boration, Rev. 11

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3-OSP-TG-002, City Water Cooling of Seal Oil Unit during an Outage, Rev. 0
3-PT-M108, RHR/SI/CS System Venting, Rev. 14
3-PT-Q83, RWST Level Instrument Check and Calibration (LIC-921 and LC-923), Rev. 34
3-PT-R010A, Residual Heat Removal System Leakage Test, Rev. 14
3-PT-V11A, Calibration of OTdT & OPdT Dynamic Setpoint Compensators, Rev. 13
3-SOP-CSS-001, Containment Spray System Operation, Rev. 3
3-SOP-EL-015, Operation of Non-Safeguards Equipment during Use of EOPs, Rev. 20
3-SOP-ESP-001, Local Equipment Operation and Contingency Actions, Rev. 21
3-SOP-RHR-001, Residual Heat Removal System Operation, Rev. 42
3-SOP-SI-002, Filling the Refueling Water Storage Tank, Rev. 11
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EN-DC-115, Engineering Change Process, Rev. 12
EN-DC-117, Post Modification Testing and Special Instructions, Rev. 4
EN-DC-134, Design Verification, Rev. 4
EN-DC-315, Flow Accelerated Corrosion Program, Rev. 6
EN-DC-319, Inspections and Evaluation of Boric Acid Leaks, Rev. 7
EN-LI-100, Process Applicability Determination, Rev. 10
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LARP-028, Unit 2 Service Water Screen Trouble Bldg., Rev. 5
OAP-007, Containment Entry and Egress, Rev. 23
OAP-017, Plant Surveillance and Operator Rounds, Rev. 6
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131377	32112	52193256
147007	51324292	52214688
168963-09	51451678	52214810
206242	51479747	52261358
213490	51485917	52267012
213491	51565373	52267017
226647	51794538	52309613
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Form 654, Nuclear-Qualified SOR Pressure Switches, (02.11) dated 2011
UEC Cat 54-B-02, United Electric Controls 54 Series Pressure, Vacuum and Temperature Switches, Rev. 2

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QA-04-2010-IP-1, Engineering Design Control Audit, dated 5/27/10

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2-GRAPH-RPC-6, Cycle 20 Core Operating Limits Report (COLR), Rev. 13
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ENN-IC-G-003, Instrument Loop Accuracy and Setpoint Calculation Methodology, Rev. 0
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Fire Hazards Analysis IP2-RPT-03-00015, Rev. 2
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IP3-DBD-314, Design Basis Document for the Reactor Coolant System, Rev. 2
IP3-DBD-323, Design Basis Document for the Containment Spray System, Rev. 1
IP3-DBD-324, Design Basis Document for the Emergency Diesel Generators and Appendix R Diesel Generator, Rev. 1
Letter, NRC to Consolidated Edison, Safety Evaluation for Indian Point 2 Susceptibility of Safety-related Equipment to Flooding from Non-seismic Systems Outside Containment (ML100321278), dated 12-18-1980
NL-72-B13, IP-2 Review of Non-Seismic Equipment Failures, dated 12/18/72
NL-73-A45, Investigation on Effects of Postulated Break in a Main Steam or Feedwater Line on the Auxiliary Feedwater System, dated 4/9/73
PA-80-C04, Safety Evaluation Report Susceptibility of Safety-Related Systems to Flooding from Failure of Non-Category 1 Systems for IP 2, dated 12/18/80
PD-95-034, IP-2 Individual Plant Examination of External Events, dated 12/95
PI-V17, Penetration Fire Barrier Seal Inspections, Rev. 6, dated 4/11/03
PMT Plan for EC 10675, Test of Zurn Strainer Circuits, Rev. 0
SEP-AP-J-006, Primary Containment Leakage Rate Program, Rev. 1
TEAR IPEC 2011-24, Training Evaluation for EC-26647 Replacement of 33 EDG Shutdown Relay
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Technical Specifications
TS-MS-027, Specification for Service Water Piping & Piping Components, Rev. 3
Updated Final Safety Analysis Report, Indian Point Unit 2, Rev. 22
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Westinghouse letter to Mr. J.F. Conway, Manager Nuclear Fuel Supply Consolidated Edison Co. of New York, Inc., Indian Point Unit 3 Dummy Fuel Assembly Offer, dated 07/25/73

Surveillance and Modification Acceptance Tests

2-IC-PC-I-T-22 EDG, Emergency Diesel Generator No. 22 Temperature Instruments Calibration, performed 3/30/11and 9/20/11

2-IC-PC-I-P-23DJW, Diesel Generator No. 23 Jacket Water Pressure, performed 11/9/10

3-PT-CS032A, Flow Test of SW HDR CK VLVS and Flow Test of Underground Portions of Line 409, performed 4/3/11

3-PT-M079A, 31 EDG Functional Test, performed 4/11/08

LIST OF ACRONYMS

ADAMS	Agencywide Documents Access and Management System
ASME	American Society of Mechanical Engineers
CAP	Corrective Action Program
CCW	Component Cooling Water
CFR	Code of Federal Regulations
COLR	Core Operating Limits Report
CR	Condition Report
CRACS	Control Room Air Conditioning System
CSS	Containment Spray System
DBA	Design Basis Accident
DBD	Design Basis Document
DP	Differential Pressure
DRS	Division of Reactor Safety
EC	Engineering Change
EDG	Emergency Diesel Generator
FME	Foreign Material Exclusion
I&C	Instrumentation and Control
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IST	In-service Testing
JW	Jacket Water
LOCA	Loss-of-Coolant Accident
MOV	Motor Operated Valve
NCV	Non-cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
OPdT	Over-Power Delta-Temperature
OTdT	Over-Temperature Delta-Temperature
PAB	Primary Auxiliary Building
PARS	Publicly Available Records
PMT	Post-Modification Test
RHR	Residual Heat Removal
RPS	Reactor Protection System
RWST	Refueling Water Storage Tank
SDP	Significance Determination Process
SI	Safety Injection
SSC	Structure, System and Component
SW	Service Water
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
VC	Vapor Containment
VDC	Volts Direct Current