UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
2100 RENAISSANCE BLVD., SUITE 100
KING OF PRUSSIA, PA 19406-2713

July 28, 2014

Mr. John Ventosa
Site Vice President
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
450 Broadway, GSB
Buchanan, NY 10511-0249

SUBJECT: INDIAN POINT POWER STATION - NRC EVALUATION OF CHANGES, TESTS, OR EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS TEAM INSPECTION REPORT 05000247/2014007 AND 05000286/2014007

Dear Mr. Ventosa:

On June 20, 2014, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at the Indian Point Power Station, Units 2 and 3. The enclosed inspection report documents the inspection results, which were discussed on June 20, 2014, with Mr. John Dinelli, General Manager, Site Operations 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. 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 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, and because the issue has been entered into your corrective action program, the NRC is treating this 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, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at Indian Point. 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 for your disagreement, to the Regional Administrator, Region I, and the NRC Resident Inspector at Indian Point Power Station.

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,

/RA/

Christopher G. Cahill, Acting Chief
Engineering Branch 2
Division of Reactor Safety

Docket Nos. 50-247 and 50-286
License Nos. DPR-26 and DPR-64

Enclosure:
Inspection Report 05000247/2014007 and 05000286/2014007
w/Attachment: Supplemental Information

cc w/encl: Distribution via ListServ

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket Nos.	50-247 and 50-286
License Nos.	DPR-26 and DPR-64
Report Nos.	05000247/2014007 and 05000286/2014007
Licensee:	Entergy Nuclear Northeast (Entergy)
Facility:	Indian Point Power Station, Units 2 and 3
Location:	450 Broadway, GSB Buchanan, NY 10511-0249
Dates:	June 2 - 20, 2014
Inspectors:	D. Kern, Senior Reactor Inspector, Division of Reactor Safety (DRS), Team Leader T. Burns, Reactor Inspector, DRS N. Floyd, Reactor Inspector, DRS J. Rady, Reactor Inspector, DRS
Approved By:	Christopher G. Cahill, Acting Chief Engineering Branch 2 Division of Reactor Safety

SUMMARY

IR 05000247/2014007, 05000286/2014007; 06/02/2014 – 06/20/2014; Indian Point Power Station, Units 2 and 3; Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications.

This report covers a 3-week Evaluation of Changes, Tests, or Experiments and Permanent Plant Modifications inspection conducted by four region-based engineering inspectors. The inspectors identified one finding of very low safety significance (Green), which was a non-cited violation (NCV). The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process (SDP)," dated June 2, 2011. Cross-cutting aspects are determined using IMC 0310, "Aspects Within the Cross-Cutting Areas," dated December 19, 2013. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy dated July 9, 2013. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 5.

NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

Green. The team identified a Green non-cited violation of Title 10 *Code of Federal Regulations* (10 CFR) Part 50, Appendix B, Criterion III, Design Control, because Entergy did not ensure the control air pressure regulator (IA-PCV-1548) for Unit 3 auxiliary boiler feedwater (ABFW) flow control valve BFD-FCV-406B was suited and designed to perform its safety-related function. Specifically, IA-PCV-1548 was not designed or qualified for use in the harsh environment area where it was located. Immediate corrective actions included evaluation of IA-PCV-1582 and BFD-FCV-406B to verify component operability. The issue was entered into the corrective action program as condition report IP3-2014-1364, to further evaluate both the extent-of-condition and the station's processes for maintaining configuration control over mechanical components installed in harsh environment areas.

The finding was more than minor because the finding was associated with the Design Control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of assuring the reliability and capability of systems that respond to initiating events to prevent undesirable consequences. Additionally, the issue was similar to example 3.j in Appendix E of Inspection Manual Chapter 0612, in that the design control issue resulted in a reasonable doubt of operability. The team determined the finding was of very low safety significance (Green) because it was a design or qualification deficiency confirmed not to result in a loss of operability.

The finding had a cross-cutting aspect in the area of Human Performance, Design Margin (H.6), because Entergy did not maintain the operational temperature design margin for the control air pressure regulator to the ABFW flow control valve. The margin between the ABFW pump room peak environmental temperature and the design/qualified temperature of IA-PCV-1582 was not carefully guarded and changed only through a systematic and rigorous process. (Section 1R17.1)

Licensee-Identified Violations

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 (29 samples)

a. Inspection Scope

The team reviewed two safety evaluations 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 Title 10 of the Code of Federal Regulations (10 CFR) 50.59 requirements. In addition, the team evaluated whether Entergy had been required to obtain U.S. Nuclear Regulatory Commission (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, 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,” Revision 1, 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 twenty-seven 10 CFR 50.59 screenings for which Entergy had concluded that a safety evaluation was not 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 and screenings that Entergy had performed and approved during the time period covered by this inspection not previously reviewed by NRC inspectors. All safety evaluations since the last modifications inspection were reviewed, and the screenings and applicability determinations selected were 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 the procedures adequately implemented the requirements of 10 CFR 50.59. The reviewed safety evaluations and screenings are listed in the Attachment.

b. Findings

Commercial Grade Dedication of Control Air Pressure Regulator for Unit 3 Auxiliary Boiler Feedwater Flow Control Valve

Introduction: The team identified a Green non-cited violation (NCV) of Title 10 *Code of Federal Regulations* (CFR) Part 50, Appendix B, Criterion III, Design Control, because Entergy did not ensure the control air pressure regulator (IA-PCV-1548) for auxiliary boiler feedwater (ABFW) flow control valve BFD-FCV-406B was suited and designed to perform its safety-related function. Specifically, IA-PCV-1548 was not designed or qualified for use in harsh environment areas such as the Unit 3 ABFW room where it was located.

Description: Commercial grade dedication (CGD) package S26992 and procurement evaluation (PE) 106118 were developed to certify commercially procured Masoneilan Model 78-4 air pressure regulators for use in safety-related applications. Unit 3 safety-related applications for this component included the ABFW flow control valves (FCVs) and emergency diesel generator air start valves. The procurement evaluation was performed as an item equivalency evaluation which evaluated this component as a replacement for five previous air pressure regulators which had become obsolete. Procurement engineers reviewed CGD test report IP3 12N-0137 and dedicated 10 of the Masoneilan Model 78-4 air pressure regulators for safety-related use on June 18, 2012. On March 11, 2013, one of these dedicated air pressure regulators was installed as a replacement for IA-PCV-1548 during the BFD-FCV-406B 12-year overhaul/rebuild activity (work order (WO) 52255242).

The team reviewed the procurement evaluation and associated third party CGD test report to verify that the design bases, licensing bases, and performance capability of the air pressure regulator had not been degraded. Six critical characteristics were identified and properly verified though testing. The team also reviewed WO 52255242 to determine whether the air pressure regulator was properly installed and tested. TSP-011, Environmental Qualification (EQ) Program Harsh Areas and Service Conditions, Revision 10, classified the AFBP room as a harsh area, susceptible to a peak temperature of 244 degrees Fahrenheit (F) following a high energy line break (HELB) of the steam driven ABFW turbine steam supply line or exhaust line. The team noted PE 106118 and WO 52255242 specifically documented the following note to Planning and Maintenance: “This item is not approved for ‘EQ’ end-use applications.” The manufacturer specification sheet for Masoneilan mmodere regulators specified an ambient temperature range of -40 to +182F.

The team determined that, contrary to notations in PE 106118 and WO 52255242, the 78-4 air pressure regulator had been installed in a harsh environment area that it was not designed or qualified for. The team met with design engineers, procurement engineers, and the work planning supervisor to discuss the station’s processes for maintaining configuration control over mechanical components installed in harsh environment areas. The team also questioned the potential effect on IA-PCV-1548 and

BFD-FCV-406B operability. Engineers initiated condition report (CR) IP3-2014-1364 to perform a prompt operability determination (OD) and evaluate cause and extent-of-condition of this issue. The OD determined the air pressure regulator installed by WO 52255242 had not been designed and qualified for use in a harsh environment, such as the ABFW pump room. Four parts within the regulator could potentially be adversely affected by elevated room temperatures during a HELB in the ABFW pump room. Following further assessment of the ABFW room HELB temperature profile, material characteristics of the air pressure regulator parts, design of HELB isolation features within the ABFW pump room, and the associated short period of elevated temperature (< 1 minute) engineers concluded IA-PCV-1548 and BFD-FCV-406B would remain operable. In the unlikely event IA-PCV-1548 failed, engineers concluded the BFD-FCV-406B would fail open to its safety position and continuously feed the #32 steam generator (SG).

The team reviewed the OD and determined it was technically sound. The team further reviewed the resulting impact on operator response and on potential ABFP motor damage if the ABFP FCV failed open. The team concluded the FCV failure would complicate operator response during an event, but procedures provided adequate instruction for operators to safely maintain SG level. Additionally, procedures contained sufficient instruction to ensure operators did not stop and restart the ABFW motor too frequently and overheat the motor.

The team performed a plant walkdown and noted Masoneilan Model 78-4 or 78-40 air pressure regulators were installed on all four of the motor driven ABFW pump FCVs. Manufacturer specifications state an ambient temperature range of -40 to +182F for both of these air pressure regulator Models. Engineers performed a more in-depth OD as part of the issue apparent cause evaluation and determined each of the 4 FCVs remained operable. Engineers informed the team CR IP3-2014-1364 would evaluate both the extent-of-condition and the station's processes for maintaining configuration control over mechanical components installed in harsh environment areas.

Analysis: The team determined that Entergy's failure to ensure control air pressure regulator IA-PCV-1548 was designed and qualified to perform its safety-related function in the harsh environment of the Unit 3 ABFW room was a performance deficiency and a failure to meet the requirements of 10 CFR 50, Appendix B, Criterion III. Specifically, in March 2013 during a 12-year valve overhaul, Entergy installed a Masoneilan Model 78-4 air pressure regulator for safety related use in a harsh environment with a design peak temperature above the manufacturers' rating. Commercial grade dedication package S26992 did not include testing or evaluation for use in this harsh environment. The finding was more than minor because the finding was associated with the Design Control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of assuring the reliability and capability of systems that respond to initiating events to prevent undesirable consequences. Additionally, the issue was similar to example 3.j in Appendix E of Inspection Manual Chapter (IMC) 0612, in that the design control issue resulted in a reasonable doubt of operability.

The team evaluated the finding in accordance with IMC 0609, Significance Determination Process, Attachment 0609.04, Phase 1 - Initial Characterization of Findings, dated June 19, 2012, for the Mitigating Systems Cornerstone, and IMC 0609, Appendix A, The Significance Determination Process (SDP) for Findings At-Power, dated June 19, 2012. The team determined the finding was of very low safety significance (Green) because it was a design or qualification deficiency confirmed not to result in a loss of operability.

The team assigned a cross-cutting aspect associated with this finding, because the air pressure regulator was procured, commercial grade dedicated for safety-related use, and installed in the plant between June 2012 and March 2013 and was reflective of current performance. The finding had a cross-cutting aspect in the area of Human Performance, Design Margin, because Entergy did not maintain the operational temperature design margin for the ABFW FCV control air pressure regulator. Specifically, the margin between the AFFW pump room peak EQ temperature and the design/qualified temperature of IA-PCV-1582 was not carefully guarded and changed only through a systematic and rigorous process. (H.6)

Enforcement: Appendix B of 10 CFR 50, Criterion III, Design Control, requires in part, that design control measures shall be established to assure applicable design bases are correctly translated into specifications, procedures, and instructions. Additionally, measures shall be established for selection and review for suitability of application of materials, parts, and equipment that are essential to the safety-related functions of the structures, systems, and components. Contrary to the above, from March 11, 2013 to June 17, 2014, specifications, procedures, and work instructions were not sufficiently established to ensure control air pressure regulator IA-PCV-1582 was designed and suited for the harsh environment in which it was installed. Consequently, during performance of WO 52255242, 12-year overhaul of BFD-FCV-406B, a Masoneilan Model 78-4 air pressure regulator was installed in a harsh environment for which it was not designed or qualified. This created a reasonable doubt of the reliability and capability of BFD-FCV-406B to perform its safety function. Immediate corrective actions included evaluation of IA-PCV-1582 and BFD-FCV-406B to verify component operability. The issue was entered into the corrective action program as CR IP3-2014-1364, to further evaluate both the extent-of-condition and the station's processes for maintaining configuration control over mechanical components installed in harsh environment areas. This violation is being treated as a NCV consistent with Section 2.3.2 of the Enforcement Policy. **(NCV 05000286/2014007-01, Deficient Design Control Results in Non-Qualified Component Installed in Harsh Environment for Unit 3 BFD-FCV-406B Actuator)**

.2 Permanent Plant Modifications (14 samples)

.2.1 Replacement of No. 32 Emergency Diesel Generator GE Model CR120A262-41 Relays

a. Inspection Scope

The team reviewed nuclear engineering change EC-15966 that replaced General Electric (GE) Model CR120A262-41 relays with GE Model CR120AD series relays within the No. 32 emergency diesel generator (EDG) electrical control system. A total of six relays were replaced within the No. 32 EDG control panel PQ1. The types of relays replaced included two equipment start relays, two crank relays, one overcrank relay, and one run relay. The modification was performed because the existing relay Model was obsolete. The review was performed to verify that the design bases, licensing bases, and performance capability of the replacement relays had not been degraded by the modification. The 10 CFR 50.59 screening determination associated with this modification was reviewed as described in section 1R17.1 of this report.

The team assessed whether the modification was consistent with requirements in the design and licensing bases. The team reviewed technical evaluations to assess whether the modification was consistent with design assumptions. Power requirements were reviewed to verify that the relays met the manufacturer's specifications. Replacement components were reviewed to ensure that they were seismically qualified. Design assumptions were reviewed to evaluate whether they were technically appropriate and consistent with the UFSAR. The team also verified selected drawings and preventive maintenance procedures were properly updated based on the installation of the replacement relays. The team reviewed the post modification testing to verify proper operation of the equipment. The team performed a walkdown of the No. 32 EDG control panel to identify any abnormal conditions and to verify proper operation of the equipment while in-service. Finally, the team conducted interviews with engineering staff to determine if the affected Structures, Systems, and Components (SSCs) would function in accordance with the design assumptions. The documents reviewed are listed in the attachment.

b. Findings

No findings were identified.

.2.2 Replacement of Westinghouse Type HFB Circuit Breakers with Cutler Hammer Type HFD Circuit Breakers in Unit 3 Power Panel 31

a. Inspection Scope

The team reviewed equivalent engineering change EC-24714 that replaced Westinghouse Type HFB circuit breakers with Cutler Hammer Type HFD circuit breakers. A total of 14 circuit breakers were replaced within Power Panel 31.

The modification was performed because the existing circuit breaker Model became obsolete. The review was performed to verify that the design bases, licensing bases, and performance capability of the replacement circuit breakers had not been degraded by the modification. The 10 CFR 50.59 screening determination associated with this modification was reviewed as described in section 1R17.1 of this report.

The team assessed whether the modification was consistent with requirements in the design and licensing bases. The team reviewed calculations and technical evaluations to assess whether the modification was consistent with design assumptions. Time-Current Characteristic Curves were reviewed to ensure that selective coordination was adequate. Replacement components were reviewed to ensure that they were seismically qualified. Design assumptions were reviewed to evaluate whether they were technically appropriate and consistent with the UFSAR. The team also verified whether selected drawings and calculations were properly updated based on the installation of the replacement circuit breakers. The team reviewed the post modification testing to verify proper operation of the equipment. The team also interviewed engineering staff to determine whether the affected SSCs would function in accordance with the design assumptions. The documents reviewed are listed in the attachment.

b. <u>Findings</u>

No findings were identified.

.2.3 <u>Replacement of No. 33 Emergency Diesel Generator GE Model CR120A262-41 Relay</u>

a. <u>Inspection Scope</u>

The team reviewed nuclear engineering change EC-26647 that replaced a GE Model CR120A262-41 relay with a GE Model CR120AD series relay within the No. 33 EDG electrical control system. The shutdown relay was replaced within the No. 33 EDG control panel PQ2. The modification was performed because the existing relay Model was obsolete. The review was performed to verify that the design bases, licensing bases, and performance capability of the replacement relay had not been degraded by the modification. The 10 CFR 50.59 screening determination associated with this modification was reviewed as described in section 1R17.1 of this report.

The team reviewed technical evaluations to assess whether the modification was consistent with design assumptions. Power requirements were reviewed to verify that the relay met the manufacturer’s specifications. Replacement components were reviewed to ensure that the modification conformed to the design specifications. Design assumptions were reviewed to evaluate whether they were technically appropriate and consistent with the UFSAR. The team also verified selected drawings and preventive maintenance procedures were properly updated based on the installation of the replacement relay. The team reviewed the post modification testing to verify proper operation of the equipment. The team performed a walkdown of the No. 33 EDG control panel to identify any abnormal conditions and to verify proper operation of the equipment while in-service. The team also conducted interviews with engineering staff to determine whether the affected SSCs would function in accordance with the design assumptions.

The documents reviewed are listed in the attachment.

b. Findings

No findings were identified.

.2.4 Replacement of Unit 3 Reactor Protection System and Engineered Safeguards System Westinghouse Type BFD Relays

a. Inspection Scope

The team reviewed nuclear engineering change EC-22663 that replaced Westinghouse Type BFD relays with Cutler Hammer Type NBFD relays. A total of 12 relays within the Reactor Protection System and the Engineered Safeguards System were replaced with new Model relays. The modification was performed because the existing relay Model was obsolete. The review was performed to verify that the design bases, licensing bases, and performance capability of the replacement relays had not been degraded by the modification. The 10 CFR 50.59 screening determination associated with this modification was reviewed as described in section 1R17.1 of this report.

The team assessed whether the modification was consistent with requirements in the design and licensing bases. The team reviewed technical evaluations to verify the modification was consistent with design assumptions. Power requirements were reviewed to verify that the relays met the manufacturer's specifications. Replacement components were reviewed to ensure that the modification conformed to the design specifications and that they were seismically qualified. Design assumptions were reviewed to evaluate whether they were technically appropriate and consistent with the UFSAR. The team also verified whether selected drawings and preventive maintenance procedures were properly updated based on the installation of the replacement relays. The team reviewed the post modification testing to verify proper operation of the equipment. The team also interviewed engineers to determine whether the affected SSCs would function in accordance with the design assumptions. The documents reviewed are listed in the attachment.

b. Findings

No findings were identified.

.2.5 Unit 3 Safety Injection Pumps No. 31 – 33 Mechanical Seal Upgrade

a. Inspection Scope

The team reviewed nuclear engineering change EC-5000037179 that replaced the No. 32 safety injection (SI) pump mechanical seal. The modification package scope changed from replacing all three SI pump seals to only the 32 SI pump seal. The modification included a redesign of the original John Crane Type 1 mechanical seal into a cartridge type seal, and a new pump gland plate. Entergy implemented this modification to address leakage issues at the seals and to facilitate proper installation during maintenance activities.

The team reviewed the modification to verify that the design bases, licensing bases, and performance capability of the SI pump had not been degraded by the modification. The team verified that the design specifications of the new seal were equivalent or improved from original seal specifications. The team interviewed design engineers and reviewed maintenance procedures, completed work orders, and the vendor technical manual to verify that the seal replacement modification was adequately implemented. In addition, the team reviewed corrective action documents 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 reviewed as described in section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.6 Install Cathodic Protection System on Unit 2 Condensate Storage Tank Supply and Return Piping in the Vicinity of the Auxiliary Boiler Feedwater Pump Building

a. Inspection Scope

The team reviewed nuclear engineering change EC-25391 that installed an impressed current cathodic protection system (CPS) on the Unit 2 condensate storage tank (CST) supply and return piping buried near the ABFW pump building. The modification included an air cooled rectifier, a series of direct buried anodes, a direct buried reference cell, and connections to the CST piping. Entergy implemented this modification to reduce the corrosion potential of the buried CST piping, after finding a through-wall leak in the 8-inch condensate return line to the Unit 2 CST.

The team reviewed the modification to verify that the design bases, licensing bases, and performance capability of the CST and the auxiliary feedwater (AFW) system had not been degraded by the modification. The team interviewed engineering staff, walked down accessible portions of the CPS, and reviewed the post-modification test (PMT) results to verify that the modification was adequately implemented. In addition, the team reviewed procedures and design basis documentation to verify that they were adequately updated. 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.

.2.7 Install Vortex Suppressors over the Unit 3 Vapor Containment and Internal Recirculation Sumps

a. Inspection Scope

The team reviewed nuclear engineering change EC-14974 that installed vortex suppression structures above the internal recirculation (IR) and vapor containment (VC) sump strainers inside of containment to eliminate the potential for air ingestion into the emergency core cooling system (ECCS) during post-accident recirculation operation. The modification included a support frame structure and sections of grating, where the grating provides the vortex suppression function. Entergy implemented this modification to resolve concerns regarding NRC Generic Safety Issue 191, where Entergy Report IP-RPT-09-00046 concluded a potential existed for vortex formation and subsequent air ingestion into the ECCS.

The team reviewed the modification to verify that the design bases, licensing bases and performance capability of the IR and VC sumps, including the pumps and strainers, had not been degraded by the installation of the vortex suppressors. The team reviewed calculations and technical evaluations to verify that the suppressors would not adversely impact the sumps, and that the sumps would function in accordance with design assumptions. The team interviewed engineering staff and reviewed the PMT results, which consisted of a visual inspection of the installation, to verify that the modification was adequately implemented. In addition, the team reviewed procedures and design basis documentation to verify that they were adequately updated. 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 Replacement of Unit 3 Degraded Cement-Lined Carbon Steel Service Water Piping and Valves Associated with Containment Fan Cooler Units with Six Percent Moly Stainless Steel Piping

a. Inspection Scope

The team reviewed nuclear engineering change EC-17891 that replaced degraded cement-lined carbon steel piping in a portion of the service water (SW) return lines of the containment fan cooler units (FCUs). The degraded piping was replaced with six (6) percent molybdenum austenitic stainless steel material to improve the corrosion and erosion resistance. Entergy implemented this modification to address service water leaks at welded connections and to replace the piping that was identified as degraded.

The team reviewed the modification to verify that the design bases, licensing bases and performance capability of the containment FCUs had not been degraded by the replacement of the SW piping with a different material. The team reviewed technical evaluations and verified that the design specifications of the replacement piping were equivalent or improved. The team interviewed engineering staff, walked down all portions of the new piping to assess material condition, and reviewed the PMT results to verify that the modification was adequately implemented. In addition, the team reviewed procedures and design basis documentation to verify that they were properly updated. 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 Modification of Unit 3 Emergency Diesel Generator Air Start Tanks and Application of Protective Coating to Internal Surfaces

a. Inspection Scope

The team reviewed nuclear engineering change EC-6665 developed to eliminate corrosion identified in the No. 31, 32 and 33 EDG starting air receivers, which had begun to foul air lines. The team also reviewed nuclear engineering change EC-9465, developed to implement the modification on the No. 33 EDG air start tank. The modification included blasting the inner tank surface to remove existing corrosion and applying a fiber-reinforced polymer coating to the internal tank surfaces, to ensure long-term integrity and corrosion protection. Application of the protective coating required the installation of a man-way in each tank to allow access to effectively apply the coating system to the internal tank surface.

The team reviewed the modification of the No. 33 EDG air start tank to determine whether the design bases, licensing bases, and performance capability of the EDG starting air system is not compromised. In addition, the team evaluated whether Entergy had been required to obtain NRC approval prior to implementing the change. The team reviewed drawing changes prepared and issued with the modification package for the tank, the support structure, and interfacing systems. The team reviewed the air tank volume calculation (IP-CALC-08-00068) to verify tank volume had not inadvertently been reduced below design requirements when the protective coating was added to the inside of the tank. The team also reviewed the seismic qualification anchorage evaluation (IP-CALC-08-00067) to verify the tank remained seismically qualified following addition of the new man way. The team interviewed engineers, reviewed maintenance work orders that installed and tested the modification, and performed a post-installation field walkdown to verify the modification was properly implemented in accordance with American Society of Mechanical Engineers Section VIII Code for Unfired Pressure Vessels. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.10 Install Internal Pipe Mechanical Seals in the Unit 3 Service Water System Line #409

a. Inspection Scope

The team reviewed a nuclear engineering change EC-24032 that installed internal pipe mechanical seals at pipe weld joints in the 24 inch service water system line #409. The mechanical seals were installed in the 409 line from the service water pumps discharge manifold to the weld-o-let (installed under modification EC-19340) on the 18 foot elevation of the primary auxiliary building. The mechanical seals were designed to provide protection for the weld seams from corrosion and protect the cement lining from erosion as a result of Service Water flow. This modification, EC-24032, was worked in tandem with modification EC-19340 since EC-19340 installed two access points at a horizontal section of pipe over the discharge canal and in a vertical section of pipe in the "room with the rock." The new access points provided the means to access the buried portion of Service Water pipe line #409 for installation of approximately 57 internal pipe mechanical seals at circumferential welds. The modification identified that in order to maintain adequate SW hydraulic performance, the maximum number of installed mechanical seals was 115.

The team reviewed the modification to verify that the design basis, licensing bases and performance capability on the service water system had not been degraded by the modification. The team interviewed engineering staff and reviewed drawings, technical evaluations, and calculations associated with the modification to confirm that the system would function in accordance with the design assumptions. In addition, the team examined representative samples of the mechanical seal including the rubber sleeve and seal assembly (retaining band and wedges) and determined the seal was not part of the service water system pressure boundary. The sole function of the seal assembly was to provide a waterproof barrier to protect the weld seam locations inside of the pipe from further degradation. The team also reviewed WO 249668, which installed the modification (37 seals) in March 2011. No welding was required to install this modification. In addition, the team reviewed procedures and design basis documentation to verify that they were properly updated. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.11 Replace Unit 2, No. 23 125 Vdc Battery Charger Float and Equalizing Potentiometers

a. Inspection Scope

The team reviewed commercial grade dedication C1825884 and procurement evaluation 122196, which certified commercially procured Clarostat Model 43C2-20K potentiometers for use in safety-related applications. The manufacturer of the safety-related 125 Vdc battery chargers had declared the Clarostat potentiometer obsolete and

no longer provided replacement parts. Entergy located commercial grade potentiometers of the same Model and performed this CGD to support planned periodic maintenance (10-year overhaul) for the 125 Vdc battery chargers. The procurement evaluation identified seven critical characteristics to be verified for acceptance of the potentiometers. An Entergy, 10 CFR 50, Appendix B certified test facility performed testing and evaluation to verify the potentiometers met the specified critical characteristics. Procurement engineers reviewed the test report and dedicated the nine potentiometers on October 2, 2013. Two of the potentiometers were issued to replace the existing float potentiometer and equalizing potentiometer under work order 52263369, No. 23 Battery Charger Maintenance Procedure 10 year preventive maintenance, performed on March 1, 2014.

The team reviewed CGD C1825884 to verify that the design bases, licensing bases, and performance capability of the No. 23 125 Vdc battery charger had not been degraded. The team also reviewed CGD processes, instructions for test sample selection, and guidance for onsite review and acceptance of completed CGD test reports (condition report (CR) HQN-2014-00519 and WTHQN-2014-00207). Additionally, the team interviewed procurement engineering staff and reviewed technical evaluations, industry test standards, and test results associated with the CGD to determine whether the battery charger and its support systems would function in accordance with design assumptions (CR IP2-2014-03546). The team reviewed the associated work order to verify that maintenance personnel implemented the preventive maintenance consistent with battery charger design. The team reviewed the associated post-modification test results and performed a field verification of potentiometer installation in the No. 23 125 Vdc battery charger cabinet to verify the two potentiometers were properly installed and no abnormal visual characteristics were present. The documents reviewed are listed in the attachment.

b. Findings

No findings were identified.

.2.12 Replace No. 32 Emergency Diesel Generator Critical Control Relays

a. Inspection Scope

The team reviewed nuclear engineering change EC-5000039294, developed to replace critical control relays used in the No. 32 EDG shutdown control circuit. The existing GE Model CR120A262-41 relays had become obsolete and were replaced by GE Model 120AD03041AA relays. Nuclear engineering change EC-30340, was developed to support installation, post-maintenance testing, and return-to-service of five of these relays on the No. 32 EDG.

Commercial grade dedication 32072472 and PE 87997, certified commercially procured GE Model 120AD03041AA relays for use in safety-related applications. The procurement evaluation identified seven critical characteristics to be verified for acceptance of the relays. A third party, 10 CFR 50, Appendix B certified test facility performed testing and evaluation to verify the relays met the specified critical characteristics. Procurement engineers reviewed the test report and dedicated fifty relays on April 27, 2011. Five of these relays were issued under work orders 52309613 and 202651, to implement EC-30340. The work orders were performed August 10, 2011 and February 1, 2012 respectively.

The team reviewed EC-30340 and CGD 32072472 to verify that the design bases, licensing bases, and performance capability of the No. 32 EDG had not been degraded. The team interviewed procurement engineering staff and reviewed procurement evaluations, industry test standards, and test results associated with the CGD to determine whether the No. 32 EDG would function in accordance with design assumptions. Additionally, the team reviewed the Entergy Qualified Supplier List and associated industry quality audit of the third party test facility to verify 10 CFR 50, Appendix B program requirements had been properly certified. The team reviewed the work orders 52309613 and 202651 to verify the replacement relays were properly installed, consistent with EDG design. The team also reviewed the associated post-modification test results and performed a field verification of the relay installation in the No. 32 EDG control cabinet to verify the five relays were properly installed and no abnormal visual characteristics were present. The documents reviewed are listed in the attachment.

b. Findings

No findings were identified.

.2.13 Replace Pilot Valve Positioner for Unit 2 PCV-1139, Steam Supply to No. 22 Auxiliary Boiler Feedwater Pump

a. Inspection Scope

The team reviewed commercial grade dedication I7466265 and PE 99139, which certified commercially procured pilot valve assemblies for bailey positioners used in safety-related applications. Common applications for the baily positioners included main feed water regulation valves, pressurizer spray valves, and steam supply valves to the steam driven ABFW pump. The procurement evaluation identified three critical characteristics to be verified for acceptance of the potentiometers. A third party, 10 CFR 50, Appendix B certified test facility performed testing and evaluation to verify the pilot valve assemblies met the specified critical characteristics. Procurement engineers reviewed test report #11N8230 and dedicated the nine pilot valve assemblies on November 23, 2011. One of the pilot valve assemblies was issued to replace the existing leaking pilot valve assembly and calibrate the PCV-1139 positioner as a post-maintenance test using work order 279583 performed on February 1, 2012.

The team reviewed CGD I7466265 to verify that the design bases, licensing bases, and performance capability of PCV-1139 and the ABFW system had not been degraded. Additionally, the team interviewed procurement engineering staff and reviewed CGD processes, technical evaluations, industry test standards, and test results associated

with the CGD to determine whether PCV-1139 would function in accordance with design assumptions. The team also reviewed the Entergy Qualified Supplier List and associated industry quality audit of the third party test facility to verify 10 CFR 50, Appendix B program requirements had been properly certified. The team reviewed the associated work order to verify that maintenance personnel performed the corrective maintenance and post-maintenance testing consistent with the valve's design. The team reviewed the associated post-maintenance test results and performed a field verification of the PCV-1139 positioner pilot valve assembly installation to verify the pilot valve assembly was properly installed and no abnormal visual characteristics were present. The documents reviewed are listed in the attachment.

b. Findings

No findings were identified.

.2.14 Unit 2 Main Steam Safety Valves Modification

a. Inspection Scope

Unit 2 has 20 main steam safety valves (MSSVs), designed to lift and relieve steam to mitigate main steam system pressure increases during transients such as a turbine load reject or a reactor trip. Three MSSVs failed periodic in-service testing between March 2010 and March 2012. Failure analysis identified spindle wear and formation of wearsteps on the valve spindle as the most likely cause of MSSV binding, which caused the valves to lift above the maximum pressure permitted by TS. The manufacturer recommended machining the valve sleeve and installing bronze wear-sleeves in the inner diameter of the spring-washers (upper & lower) and adjusting bolts. Station engineers developed a nuclear engineering change (EC-39376 and EC-39377) consistent with the manufacturer-recommended corrective action address MSSV spindle wear. WOs 338180 and 326508 were performed to implement and test the modification on 7 of 20 MSSVs during the March 2014 refueling outage (2R21). Modification of the remaining 13 MSSVs was planned for performance during normal scheduled preventive maintenance opportunities in the next two refueling outages (2R22 and 2R23).

The team reviewed the design modification to verify that the design bases, licensing bases and performance capability of the MSSVs and main steam system had not been degraded by the modification. The team verified that the design specifications of the modified MSSV were equivalent or improved from the original MSSVs. The team interviewed design engineers and reviewed technical evaluations, purchase specifications, drawings, maintenance records, maintenance procedures, and post-modification testing results to verify that the MSSV modification was appropriately implemented. In addition, the team reviewed corrective action documents to determine if there were reliability or performance issues that may have resulted from the modification. Finally, the team walked down main steam piping and the seven modified MSSVs to assess the material condition and standby configuration. 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 (CRs) 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 and/or completed corrective actions were appropriate. In addition, the team reviewed CRs written on issues identified during the inspection to verify adequate problem identification and incorporation of the problem into the corrective action system. The CRs reviewed are listed in the Attachment.

b. Findings

No findings were identified.

4OA6 Meetings, including Exit

The team presented the inspection results to Mr. J. Dinelli, General Manager, Site Operations, and other members of Entergy's staff at an exit meeting on June 20, 2014. 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

<u>Entergy Personnel</u>
K. Alfieri, Mechanical Design Engineer
V. Andreozzi, Manager, Design and Program Engineering
M. Anicetti, Maintenance Planning Supervisor
F. Bauer, Senior Mechanical Design Engineer
F. Bloise, Senior Lead Electrical Design Engineer
J. Bubniak, Senior Lead Mechanical Engineer
T. Chan, NSSS Engineering Supervisor
F. Dahl, Senior Licensing Engineer
J. Dinelli, General Manager, Site Operations
E. Ginzburg, Mechanical Design Engineer
R. Gioggia, System Engineer
J. Hill, Instrumentation and Control Design Engineering Supervisor
A. Kaczmarek, Senior Instrumentation and Controls Design Engineer
D. Musiyenko, Electrical Design Engineer
D. Morse, Cathodic Protection System Engineer
J. Raffaele, Electrical Design Engineering Supervisor
H. Robinson, Senior Lead Engineer
J. Ruch, Civil Design Engineer
R. Schimpf, Senior Lead Instrumentation and Control Design Engineer
R. Thoms, Senior Lead Procurement Engineer

<u>NRC</u>

S. Stewart	NRC, Senior Resident Inspector
A. Patel	NRC, Resident Inspector
G. Newman	NRC, Resident Inspector

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

05000286/2014007-01	NCV	Deficient Design Control Results in Non-Qualified Component Installed in Harsh Environment for Unit 3, BFD-FCV-406B Actuator (Section 1R17.1)

LIST OF DOCUMENTS REVIEWED

10 CFR 50.59 Evaluations

12-3001-00-EVAL, LBDCR #U3 2013 09-06-005 / Unit 3 UFSAR Revision of Service Water Hydraulic Analysis Computer Program from Pipeflow Model (Calculation 6604.266-8-SW-021) to PROTO-FLO Model (Calculation IP-CALC-06-00017), Revision 0
14-2001-00-EVAL, Declare Unit 2 RM-45 (SJAE Radiation Monitor) Operable on Basis of Relying on Manual Operator Actions in Place of Automatic Functions (EC 48178), Revision 0

10 CFR 50.59 or Procurement Screened-out Evaluations

DRN 12-01481, Unit 2 2-ES-1.3, Transfer to Cold Leg Recirculation, Revision 8
DRN 13-00080, Unit 3 3-ES-0.2, Natural Circulation Cooldown, Revision 2
EC-08648, Replace Jacket Water Pressure Switches JWPS-1 and -2 on EDG 31/32/33, Revision 1
EC-08889, Replace Jacket Water Pressure Switches on EDG 31, Revision 1
EC-15012, Provide Narrow Range Level Indication for IP2 EDG Fuel Oil Storage Tanks, Revision 0
EC-17096, Install HPV Valve on U3 CSS Piping from RWST to Suction of CSS Pump, Revision 0
EC-18294, Modification to Increase the Recirculation Flow for Motor Driven Auxiliary Feedwater Pumps, Revision 0
EC-19339, Installation of Thermal Relief Valves on the Tube Side of Jacket Water and Aftercooler HX for SBO and App R Diesel Generators, Revision 0
EC-21507, Rescaling of the Main Stack Vent Flow Transmitter and Indication, Revision 0
EC-22313, Replacement of Bergen-Patterson Snubbers with Lisega Snubbers Phase IV, Revision 0
EC-24913, Upgrade Obsolete LEFM Cards, Hard Drive, and Monitor, Revision 0
EC-31065, Replace Foxboro Summing Module with NUS Module in Overpressurization System, Revision 0
EC-40699, Add Local Narrow Range VC Pressure Indication to Support Revised Accident Analysis SIPD 1674, Revision 0
EC-43184, Replace Control Power Transformer for the Station Auxiliary Transformer, Revision 0
EC-43586, Replace 33 ABFP Suction Flow Alarm/Trip Switch FC-1136-AS with Different Model Barton Switch, Revision 0
EC-44012, Revise SW Piping Specification to Use Enecon Ceramalloy, Revision 0
EC-47124, Installation of Temporary Clamp on 10" SW Piping Line No. 1093 in 32 Main Transformer Moat, Revision 1
EC-5000039539, Unit Auxiliary Station Auxiliary Transformer Coordination Upgrade, Revision 0
PQE-EQ-SE-08.16.01, Commercial Grade Dedication Procurement Evaluation for ITT Barton Differential Pressure Switch Installed as #33 ABFP Suction Flow Switch, Revision 2
Item Equivalency Evaluation 106118, Air Filter Pressure Regulator, Revision 0

Audits and Self-Assessments
LO-IP3LO-00013-CA3, Plant Modifications and 50.59 Evaluations dated 5/02/11
LO-IP3LO-2013-00158, Plant Modifications and 50.59 Evaluations dated 1/14/14
QA-04-2012-IP-1, Engineering Audit dated 7/6/12
NUPIC Audit #22931, Argo Turboserve Corporation Nuclear, Schenectady, NY dated 11/02/11

Calculations and Engineering Evaluations
97-007147, Procurement Engineering Technical Evaluation, Revision 9
IP-CALC-08-00075, Emergency Diesel Generators Jacket Water Pressure Switches/Setpoints, Revision 1
IP-CALC-13-00002, Evaluation of Support to Mount Pressure Indicators PI-948-D and PI-948-E, Revision 0
IP3-CALC-EL-00118, 125Vdc System Short Circuit Calculation for Battery 31, Battery Charger 31, Power Panel 31 and Distribution Panels 31, 31A, and 33, Revision 3
IP3-CALC-VC-02654, Containment Building Water Volume Calculation, Revision 0
IP-RPT-10-00078, Emergency Diesel Generator Starting Air System Testing Data Evaluation, Revision 0
IP-RPT-12-00017, Cathodic Protection Survey of the Unit 2 Condensate Storage Tank Piping located at the Entergy Nuclear Operations Indian Point Nuclear Power Plant on April 26, 2012, Revision 0
IP3-RPT-08-00042, Auxiliary Feed Water Pump – Minimum Flow Evaluation, Revision 0
SECL-91-029, Indian Point Unit 3 Safety Evaluation of the Auxiliary Feedwater Pumps Deadheading Potential and Minimum Flow Adequacy dated 2/27/91

Completed Surveillance Test Procedures
2-PT-R006, Main Steam Safety Valve Setpoint Determination, Revision 30 performed 3/16/14

Condition Reports

HQN-2014-00519*	IP3-2002-03882	IP3-2014-01377*
IP2-2007-01442	IP3-2011-05016	IP3-2014-01385*
IP2-2012-02238	IP3-2012-00363	IP3-2014-01388*
IP2-2014-03423*	IP3-2012-00364	IP3-2014-01387*
IP2-2014-03546*	IP3-2014-01037	IP3-2014-01389*
IP2-2014-03423*	IP3-2014-01264*	IP3-2014-01414*
IP2-2014-03423*	IP3-2014-01364*	LO-WTHQN-2011-01366
IP2-2014-03423*	IP3-2014-01372*	WT-WTHZN-2014-00207*

* CR written as a result of this inspection

Design & Licensing Bases
IP3-DBD-303, IP3 Design Basis Document for Auxiliary Feedwater System, Revision 4
IP3-DBD-304, IP3 Design Basis Document for Service Water System, Revision 4
IP3-DBD-306, Safety Injection System, Revision 4
IP3-DBD-317, IP3 Design Basis Document for Feedwater System, Revision 2
IP3-DBD-324, Emergency Diesel Generators, Revision 1

Indian Point Unit 2, Updated Final Safety Analysis Report, Revision 24
Indian Point Unit 2, Technical Specifications
Indian Point Unit 3, Updated Final Safety Analysis Report, Revision 5
Indian Point Unit 3, Technical Specifications

Drawings
49-A-73527, Wiring Diagram 31, 32, and 33 EDG, Revision 2
113E303, Sht. 4, IP3 Safeguards Actuation Schematics, Revision 30
113E303, Sht. 5, IP3 Safeguards Actuation Schematics, Revision 14
113E303, Sht. 6, IP3 Safeguards Actuation Schematics, Revision 22
113E303, Sht. 7, IP3 Safeguards Actuation Schematics, Revision 24
617F644, IP3 480V One Line Diagram, Revision 35
9321-F-20183, Sheet 1, IP3 Condensate & Boiler Feed Pump Suction, Revision 62
9321-F-20193, IP3 Boiler Feedwater, Revision 61
9321-F-27023, Sheet 1, Physical Diagram Service Water Piping, Revision 21
9321-F-27223, Flow Diagram Service Water System, Revision 46
9321-F-27503, Sheet 2, Flow Diagram Safety Injection System, Revision 49
9321-F-30083, Single Line Diagram DC System, Revision 60
9321-F-33853, Electrical Distribution and Transmission System, Revision 19
9321-F-70533, Sheet 1, IP3 Auxiliary Boiler Feed Pump Room Instrument Piping, Revision 25
9321-F-70563, IP3 Nuclear Plant Control Valve Hook-Up Details Instrumentation, Revision 33
9321-LI-30412, Sht. 16, 125Vdc Power Panel 31, Revision 2
9321-LL-3131-19, Unit 2 Condenser Air Removal System – Steam Supply Valves Penetration Isolation Valves, Instrument Air Isolation Valve PCV-1228 dated 10/4/01
9321-LL-3122-17, Unit 2 Rotary Blower SJAE Vent dated 4/19/00
400882, Station Blackout and Appendix R Diesel Generator Set PI&D Cooling Water System, Revision 0
502408, Containment Building IR Sump Vortex Suppressor Plan & Sections, Revision 0
502410, Containment Building VC Sump Vortex Suppressor Plan & Sections, Revision 0
503304, Cathodic Protection on CST Piping A.B.F.P. Building Conduit Mounting Detail, Revision 0
503438, CST Piping Cathodic Protection A.B.F.P. BLDG Block Diagram, Revision 0
A209762-72, Sheet 2, Flow Diagram for Service Water, Revision 72
DW-30493-01, Cathodic Protection for Condensate Piping Unit 2, Revision 7
IP3V-13-0002, Breaker Control Schematic, Revision 20
IP3V-0514-005, John Crane Type 1 Mechanical Seal (Cartridge Style), Revision 0

Maintenance Work Orders

00215665	00237683	00269898	52213870
00237677	00237684	00326508	52255242
00237678	00237685	00338180	52309358
00237679	00237686	00345904	52486978
00237680	00237687	00370347	52540359
00237681	00237688	51451678	
00237682	00249668	51451678	

Miscellaneous
Entergy Purchase Order 10304296, Acceptance Testing and Dedication of Auxiliary/Control Relays in accordance with #GP0036, Revision 0
Entergy Purchase Order 10323867, Acceptance Testing and Dedication of Air Pressure Regulator in accordance with Procurement Evaluation 106118, Revision 2
EPRI CGIRL01, Joint Utility Task Group Commercial Grade Item Evaluation for Electromagnetic Relays-Control, Revision 0
EPRI NP-5652, Guideline for the Utilization of Commercial Grade Items in Nuclear Safety-related Applications (NCIG-07) dated June 1988

EPRI Topical Report 017218-R1, Guideline for Sampling in the Commercial-Grade Item Acceptance Process dated January 1999
EPRI Topical Report 102260, Supplemental Guidance for the Application of EPRI Report NP-5652 on the Utilization of Commercial Grade Items dated March 1994
IEEE Std 323-1974, IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations
Indian Point Energy Center Offsite Dose Calculation Manual, Revision 4
Indian Point Unit 2 RM-45 Chemistry Grab-Sample Analysis for Period 2/11/14 - 2/24/14
Indian Point Unit 2 Technical Specifications through Amendment 238
Indian Point Unit 2 Updated Final Safety Analysis Report, Revision 24
IP3-13N-0222, CGB Lab Material Test Report dated 6/5/13
IP3-SCP-MULT-03276, Proto-Flo Software Control Plan, Revision 1
IP3-SPEC-MULT-02727, NBF/NBFD Reactor Protection Grade Relays, Revision 1
NEI 96-07, Guidelines for 10 CFR 50.59 Implementation, Revision 1
NL-13-003, Proposed Technical Specification Changes Regarding RWST Temperature and Containment Pressure, dated January 28, 2013
NL-14-046, Response to Request for Additional Information Regarding Containment Integrity Analysis, dated April 22, 2014
NRC Regulatory Guide 1.187, Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments, dated November 2000
NRC Regulatory Guide 1.186, Guidance and Examples for Identifying 10 CFR 50.2 Design
QR99-02, Environmental and Seismic Qualification Test Report for Eaton Cutler-Hammer NBF/NBFD Relays, Revision 1
RCM Technologies System 1000, Arrhenius File Report, Revision 17.0.c
Specification 9321-01-248-4, Steam Generator Safety Relief Valves, Revision 1
TR63550-11N, Seismic Qualification and Dedication Test Report, Revision 0
TS-MS-027, Specification for Service Water and Piping Components, Revision 4

Modifications (** designates Modification and 10 CFR 50.59 screen sample)
Commercial Grade Dedication C1825884, Replace Unit 2, No. 23 125 Vdc Battery Charger Float and Equalizing Potentiometers dated October 2, 2013
Commercial Grade Dedication 32072472, 125 VDC coil, 4 Pole Industrial Relay, Revision 0
Commercial Grade Dedication I7466265, Kit, Pilot Valve Assembly, for Bailey Positioner AV1, Revision 0

EC-9465, Modify No. 33 Emergency Diesel Generator Air Start Tank and Apply Protective Coating to Internal Surfaces, Revision 0
**EC-14974, Install Vortex Suppressors Over the VC and Recirc Sumps at Unit 3, Revision 0
**EC-17891, Replacement of Degraded Cement Lined Carbon Steel Service Water Piping and Valves Associated With Boiler Feed Pump Coolers and Fan Cooler Unit with 6% Moly Stainless Steel Piping and New Valves, Revision 0
EC-20432, Install Internal Pipe Mechanical Seals in the Unit 3 Service Water System Line #409, Revision 0
**EC-24714, Replacement of Westinghouse Type HFB Circuit Breakers with Cutler Hammer Type HFD Circuit Breakers in Power Panel 31, Revision 0
**EC-25391, Install Cathodic Protection System on IP2 CST Supply and Return Piping in the Vicinity of the AFW Pump Building, Revision 0

EC-30340, Child EC for Replacement of Second Set of Critical (Non-UPR) Relays on 32 EDG, Revision 0
EC-39376, Modification to IP2 MSSVs to Install Sacrificial Bronze Wear-Sleeves in the Inner Diameter of the Spring-Washers (Upper & Lower) and Adjusting Bolts (Base EC), Revision 0
EC-39377, Modification of MSSVs; Child EC to Base EC-39376, Revision 0
EC-48178, Temporary Modification for Unit 2 R-45 to Provide Automatic Actions from Monitor Skid, Revision 0
**EC-5000037179, SI Pumps 31-33 Mechanical Seal Upgrade, Revision 0
**EC-5000039293; EC-15966, Replacement of 32 EDG GE Model CR120A262-41 Relays, Revision 0
**EC-5000039293; EC-26647, Replacement of 33 EDG GE Model CR120A262-41 Relay, Revision 0
EC-5000039294, Replacement of Low SW Flow and Unit Parallel Relays for EDGs, Revision 0
**EC-5000039543; EC-22663, Replacement of Westinghouse Type BFD Relays, Revision 0

Normal and Special (Abnormal) Operations Procedures
2-SOP-30.1, Electric Heat Tracing, Revision 27
3-AOP-UC-1, Uncontrolled Cooldown, Revision 3
3-ARP-007, Panel SDF – Turbine Recorder, Revision 31
3-SOP-CB-003, Containment Pressure Relief and Purge Systems Operation, Revision 34
3-SOP-CSS-001, Containment Spray System Operation, Revision 3
3-SOP-ESP-001, Local Equipment Operation and Contingency Actions, Revision 24

Procedures
0-SYS-409-GEN, Belzona and Enecon Metal Repair Applications, Revision 2
2-ARP-SAF-1, Process Radiation Monitors, Revision 36
2-COL-30.1, Electric Heat Tracing, Revision 27
2-PC-EM28, Effluent Radiation Monitor R-45 Calibration, Revision 13
2-PT-Q80, Effluent Radiation Monitor R-45 Channel Operational Test, Revision 16
2-VLV-087-VSR, Crosby Style HA – Main Steam Safety Relief Valve, Revision 2
3-COL-CS-001, Containment Spray System, Revision 15
3-COL-CSV-001, Containment Spray Verification, Revision 7

3-IC-PC-I-F-1136S, Auxiliary Boiler Feed Pump No. 33 Recirculation Flow Control, Revision 15
3-IC-PC-I-F-1136S, Auxiliary Boiler Feed Pump No. 33 Recirculation Flow Control, Revision 16
3-PMP-030-SIS, Disassembly, Inspection, and/or Replacement of the Safety Injection System Pump, Revision 9
3-PT-Q120C, 33 Auxiliary Feedwater Pump, Revision 18
3-PT-R035N, Leakage Test for Containment Pressure Instrumentation Lines, Revision 4
EN-DC-115, Engineering Design Process, Revision 16
EN-DC-117, Post Modification Testing and Special Instructions, Revision 6
EN-DC-132, Control of Engineering Documents, Revision 6
EN-DC-134, Design Verification, Revision 5
EN-DC-136, Temporary Modifications, Revision 10
EN-DC-306, Commercial Grade Item Evaluation, Revision 4
EN-DC-313, Procurement Engineering Process, Revision 10
EN-LI-100, Process Applicability Determination, Revision 14
EN-LI-101, 10 CFR 50.59 Evaluations, Revision 11
EN-MP-112, Shelf Life Program, Revision 5
EN-MP-120, Material Receipt, Revision 7
EN-TQ-202, Simulator Configuration control, Revision 8
EN-WM-105, Planning, Revision 12
OAP-007, Containment Entry and Egress, Revision 23
SEP-CATH-IP-001, IPEC Cathodic Protection Monitoring Program, Revision 0

<u>Vendor Technical Manuals</u>
NYPA 54-100048888, ITT Barton Model 581A-0 Differential Pressure Indicating Switch, Revision 0

LIST OF ACRONYMS

ABFW	Auxiliary Boiler Feedwater
AC	Alternating Current
ADAMS	Document Management System
CFR	Code of Federal Regulations
CGD	Commercial Grade Dedication
CPS	Cathodic Protection System
CR	Condition Report
CST	Condensate Storage Tank
DBD	Design Basis Document
DRS	Division of Reactor Safety
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
Entergy	Entergy Nuclear Northeast
EQ	Environmental Qualification
F	Fahrenheit
FCVs	Flow Control Valves
FCUs	Fan Cooler Units
GE	General Electric
HELB	High Energy Line Break
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Internal Recirculation
MSSVs	Main Steam Safety Valves
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	U. S. Nuclear Regulatory Commission
OD	Operability Determination
PARS	Publicly Available Records
PE	Procurement Evaluation
PMT	Post-Modification Test
SDP	Significance Determination Process
SG	Steam Generator
SI	Safety Injection
SSCs	Structures, Systems, and Components
SW	Service Water
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
VC	Vapor Containment
Vdc	Volts, Direct Current
WO	Work Order